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NUCLEAR ENERGY AGENCY

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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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The mission of the NEA is:

– to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
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THE COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

"The Committee on the Safety of Nuclear Installations (CSNI) shall be responsible for the activities of the Agency that support maintaining and advancing the scientific and technical knowledge base of the safety of nuclear installations, with the aim of implementing the NEA Strategic Plan for 2011-2016 and the Joint CSNI/CNRA Strategic Plan and Mandates for 2011-2016 in its field of competence.

The Committee shall constitute a forum for the exchange of technical information and for collaboration between organisations, which can contribute, from their respective backgrounds in research, development and engineering, to its activities. It shall have regard to the exchange of information between member countries and safety R&D programmes of various sizes in order to keep all member countries involved in and abreast of developments in technical safety matters.

The Committee shall review the state of knowledge on important topics of nuclear safety science and techniques and of safety assessments, and ensure that operating experience is appropriately accounted for in its activities. It shall initiate and conduct programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach consensus on technical issues of common interest. It shall promote the co-ordination of work in different member countries that serve to maintain and enhance competence in nuclear safety matters, including the establishment of joint undertakings, and shall assist in the feedback of the results to participating organisations. The Committee shall ensure that valuable end-products of the technical reviews and analyses are produced and available to members in a timely manner.

The Committee shall focus primarily on the safety aspects of existing power reactors, other nuclear installations and the construction of new power reactors; it shall also consider the safety implications of scientific and technical developments of future reactor designs.

The Committee shall organise its own activities. Furthermore, it shall examine any other matters referred to it by the Steering Committee. It may sponsor specialist meetings and technical working groups to further its objectives. In implementing its programme the Committee shall establish co-operative mechanisms with the Committee on Nuclear Regulatory Activities in order to work with that Committee on matters of common interest, avoiding unnecessary duplications.

The Committee shall also co-operate with the Committee on Radiation Protection and Public Health, the Radioactive Waste Management Committee, the Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle and the Nuclear Science Committee on matters of common interest."

Summary of existing uncertainty methods

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Abstract

A summary of existing and most used uncertainty methods is presented, and the main features are compared. One of these methods is the order statistics method based on Wilks' formula. It is applied in safety research as well as in licensing. This method has been first proposed by GRS for use in deterministic safety analysis, and is now used by many organisations world-wide. Its advantage is that the number of potential uncertain input and output parameters is not limited to a small number. Such a limitation was necessary for the first demonstration of the Code Scaling Applicability Uncertainty Method (CSAU) by the United States Regulatory Commission (USNRC). They did not apply Wilks' formula in their statistical method propagating input uncertainties to obtain the uncertainty of a single output variable, like peak cladding temperature. A Phenomena Identification and Ranking Table (PIRT) was set up in order to limit the number of uncertain input parameters, and consequently, the number of calculations to be performed. Another purpose of such a PIRT process is to identify the most important physical phenomena which a computer code should be suitable to calculate. The validation of the code should be focussed on the identified phenomena. Response surfaces are used in some applications replacing the computer code for performing a high number of calculations.

The second well known uncertainty method is the Uncertainty Methodology Based on Accuracy Extrapolation (UMAE) and the follow-up method "Code with the Capability of Internal Assessment of Uncertainty (CIAU)" developed by the University Pisa. Unlike the statistical approaches, the CIAU does compare experimental data with calculation results. It does not consider uncertain input parameters. Therefore, the CIAU is highly dependent on the experimental database. The accuracy gained from the comparison between experimental data and calculated results are extrapolated to obtain the uncertainty of the system code predictions for a nuclear power plant. A high effort is needed to provide the data base for deviations between experiment and calculation results in CIAU. That time and resource consuming process has been performed only by University of Pisa for the codes CATHARE and RELAP5 up to now. The data base is available only there. That is the reason why this method is only used by University of Pisa.

1 Introduction

The safety analysis used to justify the design and construction of water-cooled nuclear power plants demonstrate that the plants are designed to respond safely to various postulated accidents in a deterministic thermal-hydraulic safety evaluation. These postulated accidents include loss of coolant accidents (LOCAs) in which the pressure boundary containing the water, which takes heat away from the reactor core, is breaking and a range of other transients in which conditions deviate from those in normal operation.

The equations of state and heat transfer properties of water (including relevant two-phase phenomena) vary very substantially over the conditions that would occur in such accidents. This variation includes major non-linearity and discontinuities. They can therefore only be modelled by computer codes.

The models of the thermal-hydraulic computer codes approximate the physical behaviour, and the solution methods in the codes are approximate. Therefore, the code predictions are not exact but uncertain.

Uncertainty analysis methods therefore had to be developed to estimate safety margins if best estimate codes are used to justify reactor operation. A particular stimulus was the USNRC's revision of its regulations in 1989 to permit the use of realistic models with quantification of uncertainty in licensing submissions for Emergency Core Cooling Systems (ECCS) [1].

2 Common methods

Within the uncertainty methods considered, uncertainties are evaluated using either

(a) propagation of input uncertainties (CSAU and GRS) or

(b) extrapolation of output uncertainties (UMAE, CIAU and Siemens method for model uncertainties).

For the 'propagation of input uncertainties', uncertainty is obtained following the identification of 'uncertain' input parameters with specified ranges or/and probability distributions of these parameters, and performing calculations varying these parameters. The propagation of input uncertainties can be performed by either deterministic or statistical methods. For the 'extrapolation of output uncertainty' approach, uncertainty is obtained from the (output) uncertainty based on comparison between calculation results and significant experimental data.

A short description of these methods is provided in the following sections. More detailed information can be obtained from the IAEA Safety Series No. 52 with the title: "Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation" [2]. The following descriptions are taken from this report.

2.1 CSAU method

The aim of the CSAU methodology is to investigate the uncertainty of safety related output parameters. In the demonstration cases these were only single valued parameters, such as Peak Cladding Temperature (PCT) or minimum water inventory, with no time dependent values. Prior to this, a procedure is used to evaluate the code applicability to a selected plant scenario. Experts identify all the relevant phenomena. Following this step, the most important phenomena are identified and are listed as 'highly ranked' phenomena, based on an examination of experimental data and code predictions of the scenario under investigation ranking them as highly important. In the resulting phenomena identification and ranking table (PIRT), ranking is accomplished by expert judgement. The PIRT and code documentation are evaluated and it is decided whether the code is applicable to the plant scenario. The CSAU methodology is described in detail by Boyack et al. [3]. Applications have been performed for a large break (LB) LOCA and a small break (SB) LOCA for a PWR [3, 4].

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

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

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

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

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

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

2.2 Statistical (GRS) method

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

(a) The uncertainty space of input parameters (defined by their uncertainty ranges) is sampled at random according to the combined probability distribution of the uncertain parameters. Code calculations are performed by sampled sets of parameters.

(b) The number of code calculations is determined by the requirement to estimate a tolerance and confidence interval for the quantity of interest (such as PCT or cladding temperature versus time). Following a proposal by GRS, Wilks' formula [5, 6] is used to determine the number of calculations to obtain the uncertainty bands.

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

The smallest number *n* of code runs to be performed is according to Wilks' formula:

$$(1-\alpha/100)^n \geq \beta/100$$

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

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

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

The minimum number of calculations can be found in Table 1.

Table 1. Minimum number of calculations *n* for one-sided and two-sided statistical tolerance limits

	One-sided statistical tolerance limit			Two-sided statistical tolerance limit		
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

Upper statistical tolerance limits are the upper *b* confidence for the chosen *a* fractile. The fractile indicates the probability content of the probability distributions of the code results (e.g. *a* 95% means that PCT is below the tolerance limit with at least *a* = 95% probability). One can be *b* % confident that at least *a* % 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.

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

As a consequence, the number *n* of code runs is independent of the number of selected input uncertain parameters, only depending on the percentage of the fractile and on the desired confidence level percentage. The number of code runs for obtaining sensitivity measures is also independent of the number of parameters. As an example, 100 runs were carried out in the analysis of a reference reactor, using 50 parameters. More description can be found by references of Hofer [7] and Glaeser [8].

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

Experts identify significant uncertainties to be considered in the analysis, including the modelling uncertainties and the related parameters, and identify and quantify dependencies between uncertain parameters. Probability density functions (PDFs) are used to quantify the state of knowledge of uncertain parameters for the specific scenario. Uncertainties of code model parameters are derived based on validation experience.

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

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

Statistical uncertainty and sensitivity analysis provides statements on:

- Uncertainty range of code results that enables to determine the margin between the bound of uncertainty range closest to an acceptance criterion and the acceptance criterion. The 95% fractile, 95% confidence limit for output parameters versus time are provided.
- Sensitivity measures about the influence of input parameters on calculation results, i.e. a ranking of importance versus time are provided which
  - allows a ranking of input parameters on output uncertainty as result of the analysis,
  - guides further code development,
  - prioritizes experimental investigations to obtain more detailed information.

The sensitivity or importance measures give useful information about those input parameters influencing the uncertainty of computer code results most. That information can be used to find out which ranges and distributions of input uncertainties should potentially be determined more accurately. These sensitivity measures, using regression or correlation techniques from the sets of input parameters and from the corresponding output values, allow ranking of the uncertain input parameters in relation to their contribution to output uncertainty. The ranking of parameters is therefore a result of the analysis, not of prior expert judgement.

The same ordered statistical method using Wilks' formula for setting up the number of code calculations followed later the AREVA-Method [9], ASTRUM (Automatic Statistical TReatment of Uncertainty) - Method of Westinghouse [10], KREM in Korea, ESM-3D in France and several more. The AREVA method has been licensed by USNRC in the year 2003 and the ASTRUM Method in 2004.

2.3 CIAU method

The CIAU method is based on the principle that it is reasonable to extrapolate code output errors observed for relevant experimental tests to real plants, as described in Ref. [11, 12]. The development of the method implies the availability of qualified experimental data. A first step is to check the quality of code results with respect to experimental data by using a procedure based on Fast Fourier Transform Method. Then a method (UMAE) is applied to determine both Quantity Accuracy Matrix (QAM) and Time Accuracy Matrix (TAM).

Considering Integral Test Facilities (ITFs) of a reference light water reactor (LWR) and qualified computer codes based on advanced models, the method relies on code capability qualified by application to facilities of increasing scale. Direct data extrapolation from small scale experiments to the reactor scale is difficult due to the imperfect scaling criteria adopted in the design of each scaled down facility. Only the accuracy (i.e. the difference between measured and calculated quantities) is therefore extrapolated. Experimental and calculated data in differently scaled facilities are used to demonstrate that physical phenomena and code predictive capabilities of important phenomena do not change when increasing the dimensions of the facilities; however, available IT facility scales are far from reactor scale.

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

The method uses a database from similar tests and counterpart tests performed in ITFs that are representative of plant conditions. The quantification of code accuracy is carried out by using a procedure based on fast Fourier transform (FFT), characterizing the discrepancies between code calculations and experimental data in the frequency domain and defining figures of merit for the accuracy of each calculation. FFT is used for the acceptability check of the calculations. In the UMAE, the ratio of the experimental to calculated value is used for the extrapolation. The use of this procedure intends to avoid the influence of engineering judgement in evaluating the adequacy of the code results. Different requirements have to be fulfilled in order to extrapolate the accuracy.

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

It is not possible to establish a correspondence between each input and each output parameter without performing additional specific calculations, which, however, are beyond the scope of the UMAE. The process starts with the experimental and calculated database. Following the identification (e.g. from the CSNI validation matrix) of the physical phenomena involved in the selected transient scenario, relevant thermal-hydraulic aspects (RTAs) are used to evaluate the acceptability of code calculations, the similarity among experimental data and the similarity between plant calculation results and available data. Statistical treatment is pursued in order to process accuracy values calculated for the various test facilities and to obtain uncertainty ranges with a 95% confidence level. These are superimposed as uncertainty bands bracketing the ASM calculation.

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

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

The basic idea of the CIAU can be summarized in two parts [12]:

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

(ii) Association of an uncertainty to each plant state.

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

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

The idea at the basis of the CIAU may be described more specifically as follows: The uncertainty in code prediction is the same for each plant state. A quantity uncertainty matrix (QUM) and a time uncertainty vector (TUV) can be set up including values of Uq and Ut obtained by means of an uncertainty methodology.

Possible compensating errors of the used computer code are not taken into account by UMAE and CIAU, and error propagation from input uncertainties through output uncertainties is not performed. A high effort is needed to provide the data base for deviations between experiment and calculation results in CIAU. That time and resource consuming process has been performed only by University of Pisa for the codes CATHARE and RELAP5 up to now. The data base is available only there. That is the reason why this method is only used by University of Pisa.

2.4 Siemens method

The Siemens method considered code model uncertainties, nuclear power plant uncertainties and fuel uncertainties separately. The model uncertainties were determined by comparison of relevant experimental data with calculation results. However, nuclear power plant conditions and fuel conditions are treated statistically [13]. The Siemens method was applied to evaluate model uncertainties in the nuclear power plant (NPP) Angra-2, Brazil for the LBLOCA licensing analysis.

Siemens applied their method to be licensed in the USA. The USNRC, however, had a concern of compensating errors of the computer codes and asked for propagation from input uncertainties through output uncertainties. There is also a concern of scaling distortions of integral experiments and different time scales in the experiments compared with reactor scale and possible influence on the deviation between calculations and data. The derived uncertainties are also dependent on the selection of integral experiments. Additional uncertainties of plant conditions and fuel related parameters should be quantified either by additional bias or statistical evaluation.

2.5 Comparison of main methods

A comparison of the relevant features of the main uncertainty methods are listed in Table 2.

Table 2. Comparison of relevant features of uncertainty methods

	Feature	CSAU	Statistical/ GRS	UMAE
1	Determination of uncertain input parameters and of input uncertainty ranges	Experts	Experts	[a]
2	Selection of uncertain parameter values within the determined range for code calculations	Experts	Random selection	Not necessary
3	Support of identification and ranking of main parameter and modelling uncertainties	Yes	No	No
4	Accounting for state of knowledge of uncertain parameters (distribution of input uncertainties)	Yes	Yes	No
5	Probabilistic uncertainty statement	Yes	Yes	Yes

	Feature	CSAU	Statistical/ GRS	UMAE
6	Statistical rigour	No	Yes	No
7	Knowledge of code specifics may reduce resources necessary to the analysis	Yes	No	No
8	Number of code runs independent of number of input and output parameters	No	Yes	Yes
9	Typical number of code runs	LB: 8 SB: 34	59 PWR: 93-300[c] LOFT[b]: 59-150[c] LSTF: 59-100[d]	n.a.[e]
10	Number of uncertain input parameters	LB: 7 (+5) SB: 8	LOFT: 13-64[c] PWR: 17-55[c] LSTF: 25-48[d]	n.a.
11	Quantitative information about influence of a limited number of code runs	No	Yes	No
12	Use of response surface to approximate result	Yes	No	No
13	Use of biases on results	Yes	No	For non-model uncertainties
14	Continuous valued output parameters	No	Yes	Yes
15	Sensitivity measures of input parameters on output parameters	No	Yes	No

[a] The differences between experimental and used input data constitute one of the sources for uncertainty.
[b] LOBI, LOFT and LSTF are test facilities.
[c] Numbers of different participants in the OECD BEMUSE (Best Estimate Method – Uncertainty and Sensitivity Evaluation) comparison project.
[d] Numbers of different participants in the OECD UMS (Uncertainty Methods Study) comparison project.
[e] This depends on the stage of the analysis. The first application to the analysis of the SB LOCA counterpart test in a PWR required roughly 20 code runs; the analysis of a similar nuclear power plant scenario would require a few additional code runs.
n.a.: not applicable

3 Number of calculations for statistical methods to meet more than one regulatory limit

A very controversial international discussion took place about the number of calculations to be performed using ordered statistics methods [14-18]. That issue was mainly brought up when more than one regulatory acceptance criterion or limit has to be met. Wald [19] extended Wilks' formula for multi-dimensional joint/ simultaneous tolerance limits or intervals. However, it seems that a direct and satisfactory extension of the concept of tolerance limits for safety-relevant applications in nuclear safety is difficult, and even not necessary. A slightly modified concept has therefore been proposed by Krzykacz-Hausmann from GRS, introducing a **lower confidence limit** [20]. The lower confidence limit according to Clopper-Pearson [21] for the binomial parameter is now the unknown probability that a result is lower than a regulatory acceptance limit. Instead of direct joint tolerance limits for the outputs of interest, one considers the lower confidence limit for the probability of "complying with the safety limits for **all** outputs", i.e. "meeting the regulatory acceptance criteria". Basis is that both of the following statements are equivalent:

1. The Wilks' (probability $a = 95\%$ and confidence $b = 95\%$) limit for the results is below the regulatory acceptance limit.
2. The **lower b = 95% confidence limit** for the probability that the value of the result stays below the regulatory acceptance limit is greater or equal $a = 95\%$.

The regulatory acceptance limits are incorporated into the probabilistic statements. It turns out that (1) in the one-dimensional case, i.e. for a single output parameter, is this concept equivalent to the one-sided upper tolerance limit concept, and (2) the necessary number of model runs is also the same in the general case, i.e. independent of the number of outputs or criteria involved and of the type of interrelationships between these outputs or criteria. Therefore, the number of necessary model runs is the same as in the one-dimensional tolerance limit case, even if several output parameters are involved. In the one-dimensional case the lower 95%-confidence interval for the probability of "complying with the regulatory limit" corresponds to the two step procedure: (1) compute the tolerance limit as usual and (2) compare this tolerance limit with the given regulatory limit. In other words: The statement "there is a 95% confidence that the probability of "complying with the regulatory limit x_{reg} exceeds 95%" is equivalent to the statement "the computed 95%/ 95% tolerance limit x_{TL} lies below the regulatory limit x_{reg}". In the multi-dimensional case there is no such direct correspondence or equivalence.

The principal advantage of the confidence interval or limit (or "sign-test") approach seems to be that it can directly be used in the multi-dimensional case, i.e. multiple output or several output variables, too, while the multidimensional extensions of the tolerance limit approach suffer from (1) not being unique because the runs with the highest value for checking the first limit has to be eliminated for comparison with the next limit, and so on, and (2) require substantially increased calculation runs, (3) are in most cases not necessary since functions of several variables can be reduced to the one-dimensional case.

Much more influence on the uncertainty range of computational results has the specified input uncertainty ranges. Less important is the distribution of these input uncertainties. Therefore, high requirements are on the specification and the justification for these ranges. 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. The selection of suitable experiments is important for the UMAE/ CIAU method as well.

4 Applications

Best estimate analysis plus uncertainty evaluation is used in licensing up to now in the following countries: Argentina (CIAU), Belgium (ordered statistics), Brazil (output uncertainties and statistical method as well as CIAU), China, Korea (ordered statistics), Lithuania (ordered statistics), Netherlands (statistical method), Spain (ordered statistics), Taiwan (ordered statistics) and USA (ordered statistics).

Significant activities for use in licensing are performed in these countries: Canada, Czech Republic, France, Germany, Hungary, Japan, Russia, Slovak Republic and Ukraine.

5 Conclusions

The safety demonstration method "uncertainty analysis" is becoming common practice world-wide, mainly based on ordered statistics. Basis for applications of statistical uncertainty evaluation methods is the development of the GRS-method.

Several activities are carried out on an international world-wide level, like in OECD and IAEA. Comparison of applications of existing uncertainty methods have been performed in the frame of OECD/ CSNI Programmes. Differences were observed in the results of uncertainty analysis to the same task. These differences are sometimes reason for arguing about the suitability of the applied methods. Differences of results may come from different methods. When statistical methods are used, differences may be due to different input uncertainties, their ranges and distributions. However, differences are already seen in the basic or reference calculations.

When a conservative method is used, it is claimed that all uncertainties which are considered by an uncertainty analysis are bounded by conservative assumptions. That may only be right when conservative code models are used in addition to conservative initial and boundary conditions. In many countries, however, a best estimate code plus conservative initial and boundary conditions are accepted for conservative analysis in licensing. Differences in calculation results of best estimate and even conservative codes would also be seen, comparing results of different users of the same computer code due to different nodalisations and code options the user is selecting. That was observed in all International Standard Problems where applicants calculated the same experiment or a reactor event. The main reason is that the user of a computer code has a big influence in applying a code. A user effect can also be seen in applications of uncertainty methods.

Another international discussion took place about the number of calculations to be performed using ordered statistics methods. That issue was mainly brought up when more than one acceptance criterion has to be met. However, much more influence on the results is by the specification of uncertainty ranges of these input parameters. Therefore, high requirements are on their specification and the justification for these ranges. The international OECD PREMIUM (**P**ost BEMUSE **RE**flood **M**odels **I**nput **U**ncertainty **M**ethods) Project will deal with this issue.

6 References

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Generic Application of Wilks' Tolerance Limit Evaluation Approach to Nuclear Safety

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Abstract

More than a decade has passed since the German regulatory body, e.g., GRS (Gesellschaft für Anlagen- und Reaktorsicherheit) introduced the so known as Wilks' formula (or GRS method) in the nuclear safety field and got a world-wide attention. The introduction of Wilks' tolerance limit evaluation approach which is based on a pure statistical method might be estimated as one of the most ground-breaking ideas. The beneficiary might be not only nuclear industries but also regulatory bodies due to its versatility and theoretical robustness leading to a very efficient tool now as a part of the best estimate plus uncertainty methodology, which is an alternative to the evaluation method in Appendix K to the 10 CFR 50.

We have watched how the above approach evolves through one of the most active international-cooperation activity of the BEMUSE projects under OECD/NEA, CSNI lead. Related to the application of the Wilks' formula, during the past years, it was one of very interesting findings to see that there still exists some openness depending on the expert judgments such as in the selection of the order of Wilks' formula or the number of random samples to deal with the meaning of the numerical outputs.

Motivated by the above finding, the authors exert to have an in-depth understanding of the possible meanings of the Wilks' set of formulas; a) one-sided and b) two-sided, and suggest some considerations to be taken for a more appropriate application to the nuclear safety field.

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1. Introduction

A tolerance limit evaluation approach, which is internally known as GRS (Gesellschaft für Anlagen- und Reaktorsicherheit) method[1] or Wilks' formula has been widely used mostly to identify the meaningful maximum code-output parameter value like peak cladding temperature from a limited number of code simulations. The most referred probability level to cut the upper limit as a way to set the safety limit has been accepted as the 95 % probability level, e.g., 95th percentile combined with the confidence level of 95 %. Historically, this practice of the 95th percentile in the nuclear field seems to stem from the standard engineering practice and be based on the US NRC's acceptance as stated in the regulatory guide 1.157, which guides the best-estimate and uncertainty approach to the emergency core cooling system analysis. More bounding statement about the uncertainty related approach is included in the acceptance criteria of the 10 CFR 50.46 which allows an approach of the Best Estimate Plus Uncertainty (BEPU) method as an alternative to the previous Evaluation Method (EM) in Appendix K.

Various methods have been suggested from 1990s to address the safety limit of major parameters including uncertainty and the GRS method seems to have become the most popular one among them. The introduction of Wilks' tolerance limit evaluation approach by GRS might be estimated as one of the most ground-breaking ideas. The beneficiary might be not only nuclear industries but also regulatory bodies due to its versatility and theoretical robustness leading to a very efficient tool now as a part of the Best Estimate Plus Uncertainty (BEPU) methodology. The GRS method is based on the combinatorial Wilks' formula set[2] compared to the original integral Wilks' formula[3]. A few years ago, the author described how the formula set of one-sided and two-sided Wilks' approaches could be analytically derived in terms of combinatorial numeric[4]. In the paper, multiple meanings of code runs were discussed with derivation of more general forms of equations and, specifically another formula for the two-sided approach for the 1st order was suggested against the present formula for the two-sided approach while confirming the validity of the present formula for the one-sided approach.

In the meantime, we have watched how the above approach evolves through one of the most active international-cooperation activity of the BEMUSE projects[5] under OECD/NEA, CSNI lead. Related to the application of the Wilks' formula, during the past years, it was one of very interesting findings to see that there still exists some openness depending on the expert judgments such as in the selection of the order of Wilks' formula or the number of random samples to deal with the meaning of

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the numerical outputs. For example, some organizations may use 100 code runs, some others use 124, etc., even though the 1st order 95th percentile 95 % confidence condition requires a less number of 59 code runs from the theoretical point of view. These higher numbers than the 59 are mostly ascribed to the adoption of a higher order approach than the 1st order. At this point, the authors would like to raise questions from two-folded aspects:

a) Which number should be more appropriate to meet the 95th / 95 % requirement?
b) Can we all agree with that the present requirement of 95th / 95 % is enough from the safety perspective?

Starting from the previous work of the main author, we have continued a further research; a) to better understand the meanings of the Wilks' formula set, b) to numerically validate the Wilks' related theoretics, and c) to determine a better application of the Wilks' formula set from the safety perspective.

It is noted that the focus of the present paper is rather made on discussing a better application of Wilks' approach by presenting applicable tables and supporting numerical simulation results, which are followed by summary and presentation of the Wilks one-sided and two-sided formulas, than presenting detailed derivations of formula set in order not to distract readers' attention.

2. Wilks' Combinatorial Formula Set and Statistically Meaningful Number of Simulations

For one-sided approach for the upper tolerance limit, the combinatorial form of Wilks' formula is given below:

$$Pr(CP(maximum) \geq \alpha; \text{the 1st order}) = 1 - \alpha^n \geq \beta, \quad (1))$$

where, n is the total number of code runs and α denotes the cumulative probability range, tolerance limits or percentile, which is defined as $\alpha = \int_{-\infty}^{x_U} f(x)dx$ and $f(x)$ is the PDF for continuous variable x and x_U is the upper tolerance limit for x. CP means cumulative probability, $Pr(CP(maximum) \geq \alpha)$ represents the confidence level of the maximum value to be located above α, and β is the judgment criterion of confidence level, for example, 95 %.

In a more general form, the Wilks' one-sided formula is as below for a general order of p:

$$Pr(CP(maximum) \geq \alpha;\ p\text{-}th\ order) = \sum_{k=0}^{n-p} {}_nC_k \alpha^k (1-\alpha)^{n-k} = 1 - \sum_{k=n-p+1}^{n} {}_nC_k \alpha^k (1-\alpha)^{n-k} \geq \beta. \quad (2))$$

In Equation (2), if p is 2 (2nd order one-sided), it means that two output values at least are to be located above the given percentile α with higher than the confidence level criterion of β. The resultant formula, when p equals 2, is then

$$1 - \alpha^n - n(1-\alpha)\alpha^{n-1} \geq \beta, \quad (3))$$

which has the exactly the same form as the 1st order two-sided formula.

It is noted that another formula was suggested in the foot-noted reference 4 against the present 1st order two-sided formula in Equation (3).

$$Pr\left\{(CP(x_m) \leq \frac{1-\alpha}{2}) \cap (CP(x_M) \geq \frac{1+\alpha}{2}); \text{the 1st order}\right\} = 1 + \alpha^n - 2\alpha^n \sum_{k=0}^{n} {}_nC_k \left(\frac{1-\alpha}{2\alpha}\right)^k \geq \beta \quad (4))$$

The 2nd order two-sided formula is newly presented in the below:

$$Pr\left\{(CP(x_m) \le \frac{1-\alpha}{2}) \cap (CP(x_M) \ge \frac{1+\alpha}{2}); the\ 2nd\ order\right\} =$$

$$2\sum_{k=5}^{n}\sum_{r=2}^{\left[\frac{k-1}{2}\right]} {}_{n}C_{k-r}\left(\frac{1-\alpha}{2}\right)^{r} {}_{n-k+r}C_{r}\left(\frac{1-\alpha}{2}\right)^{k-r}\alpha^{n-k} + \sum_{r=2}^{\left[\frac{n}{2}\right]} {}_{n}C_{r}\left(\frac{1-\alpha}{2}\right)^{r} {}_{n-r}C_{r}\left(\frac{1-\alpha}{2}\right)^{r}\alpha^{n-2r} \ge \beta \quad (5)$$

Equation (4) and Equation (5) are suitable for the two-sided approach when the lower limit and upper limit are symmetrically set around the center of the percentile axis. That is, for a given percentile α, the upper-left side of Equation (5) means the confidence level for the case where the 2nd largest maximum is located above $(1+\alpha)/2$-th percentile and the 2nd smallest minimum is located below $(1-\alpha)/2$-th percentile at the same time.

Various Tables for statistically meaningful number of simulations are derived from the above Equation (2) for different orders' one-sided approach, Equation (4) for the 1st order two-sided approach and Equation (5) for the 2nd order two-sided approach. Table 1 to Table 3 summarize the minimum numbers of code runs necessary for the 1st, 2nd and 3rd order statistics, respectively, when users require to perform the simulations for the one-sided approach. Table 4 and Table 5 summarize the numbers for the 1st and 2nd order statistics each for the two-sided approach. For example, the minimum number of required code runs for the condition of 95th percentile / 95 % confidence level are:

- 59 for 1st order one-sided,
- 93 for 2nd order one-sided,
- 124 for 3rd order one-sided,
- 146 for 1st order two-sided (against the present 93) and
- 221 for 2nd order two-sided.

Figure 1 shows a contour plot between the percentile and confidence levels for given number of code runs. Figure 2 shows the corresponding 3-dimensional surface plot. The relationship between the minimum required number of code runs and percentile / confidence levels are non-linear and much larger number of codes runs has to be added to increase the percentile level compared to the confidence level. Considering that the 95th percentile is the most widely accepted standard engineering practice not only in nuclear safety field but also in other fields, it seems more reasonable to take into account the importance of the confidence level than the percentile. The importance of the confidence level is more to be discussed in the following Section.

3. Numerical Validation and Discussion

Before jumping into discussion, we performed various numerical validation tests for the formula set presented in Section 2. Among them, Figure 1 shows one of the validation results for the newly derived the 2nd order two-sided formula in Equation (5) for the condition of 95th percentile and 95 % confidence level, which requires 221 code runs as shown in Table 5. In this case, the lower and upper percentile limits are 0.025 (25th percentile) and 0.975 (97.5th percentile), respectively. Then, numerical experiments were performed based on a few assumptions listed below:

- An unknown code output parameter x is assumed to follow the uniform random distribution (between 0.0 ~ 1.0) as a trial distribution. (Note that the Wilks' approach corresponds to distribution free approach.)
- A set of 221 code runs is assumed to constitute the trial code output distribution, and the corresponding 221 code output values are generated using a uniform pseudo-random generator.
- Then, a total of 100 sets of 221 code runs are simulated to investigate the statistical behavior, specifically for the 1st and 2nd largest maximum values and the 1st and 2nd smallest minimum values.

In Figure 3, nested two figures on the left column show the 1st maximum and minimum values and the other two figures on the right column show the 2nd maximum and minimum values from the 100 sets for each 221 code output simulations. From Figure 3, it can be identified that 3 sets out of 100 sets have the 2nd largest maximum values below the upper limit of 0.975 and 1 set out of 100 sets has the 2nd smallest minimum value above the lower limit of 0.025. Thus, the total confidence level is simply calculated as 96 % from (100 cases – (3+1) cases), which has a deviation of 1 % from the intended 95 %.

To fully validate the formula set in Section 2, the total number of test sets was increased from 100 to a million to get statistically more converged results. This test logic has been applied to all the major five cases which are listed at the end of Section 2 for the 95th percentile / 95 % confidence level. Table 6 lists the validation results which compare the numerically experimented confidence levels with the analytically derived confidence levels. The results demonstrate that the discrepancies are within 0.03 % between the numerical experiments and analytic solutions, thus it is considered that all the formulas as intended in this paper have been validated.

Now to answer for the first question in Section 1 of "Which number should be more appropriate to meet the 95th / 95 % requirement?", the conclusion is that the 59 code runs is enough for the 1st order one-sided approach while the 146 code runs is enough for the 1st order two-sided approach. It is noted that these values for the 1st order exactly fits the theoretically derived Wilks' approaches. Then, what is the use of a higher order approach? To answer this question, the meaning of a p-th order could be reminded that the p-th order means that at least p output values will be above the percentile α with an equal or higher confidence level than the preset criterion β. Considering the derivation of the Wilks' formula set, it is very natural to be tempted that we may be able to apply the p-th order statistics to explain the meaning of jointly related p output parameters from the same code runs. However, the author's opinion is that this approach is valid only when the p output parameters are totally independent and thus uncorrelated. And in reality, any major output parameters from a computer code which are inter-correlated from sharing the same input variables are not believed to be uncorrelated. Thus, when the correlation between any p output parameters cannot be properly evaluated and even when it was, the higher order approach might not be easily applicable from a practical point of view. So we would like to state just for a little while that a more number of trials than the 59 (for one-sided) or the 146 code runs (for two-sided) may not be very necessary.

For the next, let us think of the second question in Section 1 of "Can we all agree with that the present requirement of 95th / 95 % is enough from the safety perspective?". Without doubt, we know that international organizations have already agreed to use this criterion for the safety evaluation of nuclear power plants. Then, what about the practiced number of code runs of 100 or 124, as per higher order statistics, which is higher than the 59 for one-sided approach? For this, the author's viewpoint is that we may need to look into this kind of group behavior from the side of maybe possible confusion. For example, for the number of 124 code runs that is based on the 3rd order one-sided approach for 95 % confidence and the 95th percentile, it is not difficult to recognize that the number corresponds to a confidence level of 99.83 % for the same 95th percentile in terms of the 1st order one-sided approach. Therefore, if we cast the present behavior of using higher order statistics into the frame of the 1st order statistics, it is simply considered that we are relying on higher confidence level in reality. From this point of view, to authors it seems very reasonable that the experts in many organizations have been using a higher number of code runs than the 59 for their analyses since the number of 59 means that there exists 1 out-of 20 missing possibility from the intended 95 % confidence level. Can we simply say that it is very acceptable living with 5 % of maybe-dangerous conclusion? It is recognized that this question is in an area of open questions but it seems there should be a consensus on the number of code runs in the near future and also more considerations should be taken for the statistical meanings of all those numbers to ensure a higher level of safety standard for the nuclear power plants.

4. Concluding Remarks and Suggestions

Through the review of the BEMUSE project results, major documents for the tolerance limit evaluation approach and demonstration of the combinatorial Wilks' formula, our observation is concluded as follows:

i. The tolerance limit evaluation approach would be applicable to not only the safety analysis discipline but wider range of disciplines such as physics, materials in the near future. For a fit-for-purpose application, a more in-depth understanding of the tolerance limit approach might be necessary at the working group levels.

ii. The present practice of using a more than enough number of theoretically derived minimum numbers of code runs, for example 124 rather than 59, is expected to generate a higher confidence level than the target confidence level in terms of the 1st order statistics. This approach is concluded in two ways; a) it is reasonable in a sense that it ensures a higher confidence level than the 95 % in terms of the 1st order, however, b) it is unreasonable because it may not produce the intended results.

iii. From the safety perspective, the present practice of crediting the 95th percentile looks reasonable but the 95 % confidence level may or may not be high enough to ensure nuclear safety analysis results. The choice seems somewhat open yet as can be seen from different choices between organizations.

From the above conclusions, we recognize and suggest that there should be a consensus for proper applications of the tolerance limit evaluation approach between different organizations, disciplines where the approach can be referred to.

Table 1. Number of Required Code Runs for the First-order One-sided Approach

One sided 1st order		Percentile in Percentage				
		95.0	**96.0**	**97.0**	**98.0**	**99.0**
Confidence Level in Percentage	**95.0**	**59**	74	99	149	299
	96.0	63	79	106	160	321
	97.0	69	86	116	174	349
	98.0	77	96	129	194	390
	99.0	**90**	113	152	228	459
	99.5	104	130	174	263	528
	99.9	**135**	170	227	342	688

Table 2. Number of Required Code Runs for the Second-order One-sided Approach

One sided 2nd order		Percentile in Percentage				
		95.0	**96.0**	**97.0**	**98.0**	**99.0**
Confidence Level in Percentage	**95.0**	**93**	117	157	236	473
	96.0	99	124	166	249	500
	97.0	105	132	177	266	534
	98.0	115	144	193	290	581
	99.0	130	164	219	330	662
	99.5	146	183	245	369	740
	99.9	181	227	304	458	920

Table 3. Number of Required Code Runs for the Third-order One-sided Approach

One sided 3rd order		Percentile in Percentage				
		95.0	**96.0**	**97.0**	**98.0**	**99.0**
Confidence Level in Percentage	**95.0**	**124**	156	208	313	628
	96.0	130	163	218	328	658
	97.0	138	173	231	347	696
	98.0	148	186	248	374	749
	99.0	165	207	277	418	838
	99.5	182	229	306	461	924
	99.9	220	277	370	557	1119

Table 4. Number of Required Code Runs for the First-order Two-sided Approach

Two sided 1st order		Percentile in Percentage				
		95.0	**96.0**	**97.0**	**98.0**	**99.0**
Confidence Level in Percentage	**95.0**	**146**	183	244	366	734
	96.0	155	194	259	389	779
	97.0	166	208	278	418	837
	98.0	182	228	305	458	918
	99.0	**210**	263	351	527	1057
	99.5	237	297	397	597	1196
	99.9	**301**	377	503	757	1517

Table 5. Number of Required Code Runs for the Second-order Two-sided Approach

Two sided 2nd order		Percentile in Percentage				
		95.0	**96.0**	**97.0**	**98.0**	**99.0**
Confidence Level in Percentage	**95.0**	**221**	276	369	-[1)]	-
	96.0	231	289	386	-	-
	97.0	244	306	409	-	-
	98.0	263	329	440	-	-
	99.0	294	369	-	-	-
	99.5	325	407	-	-	-
	99.9	396	-	-	-	-

* Note: 1) The corresponding values could not be fully calculated yet from the limited utility code capability.

Table 6. Numerically Experimented and Theoretically Derived Confidence Levels

Percentile	Approach	N[1)]	Num.Exp.[2)]	Theoretical	Diff (%)
95th	1st order 1-sided	59	95.1622	95.1505	0.01
	2nd order 1-sided	93	95.0305	95.0024	0.03
	1st order 2-sided	146	95.1029	95.0934	0.01
	2nd order 2-sided	221	95.0894	95.1012	-0.01

* Note: 1) N: number of code runs for each set,

2) 1,000,000 tests were performed for each set of N code runs.

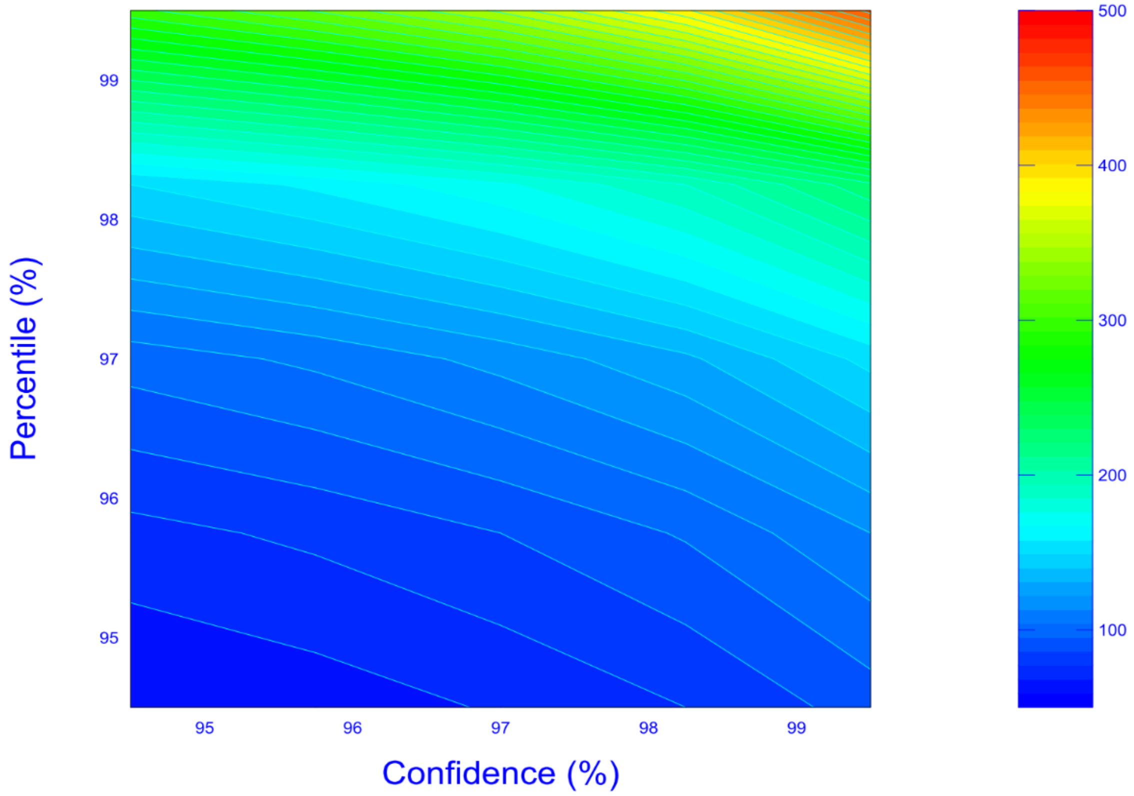


Figure 1. Contour Distribution between Confidence Level and Percentile for Given Number of Code Runs

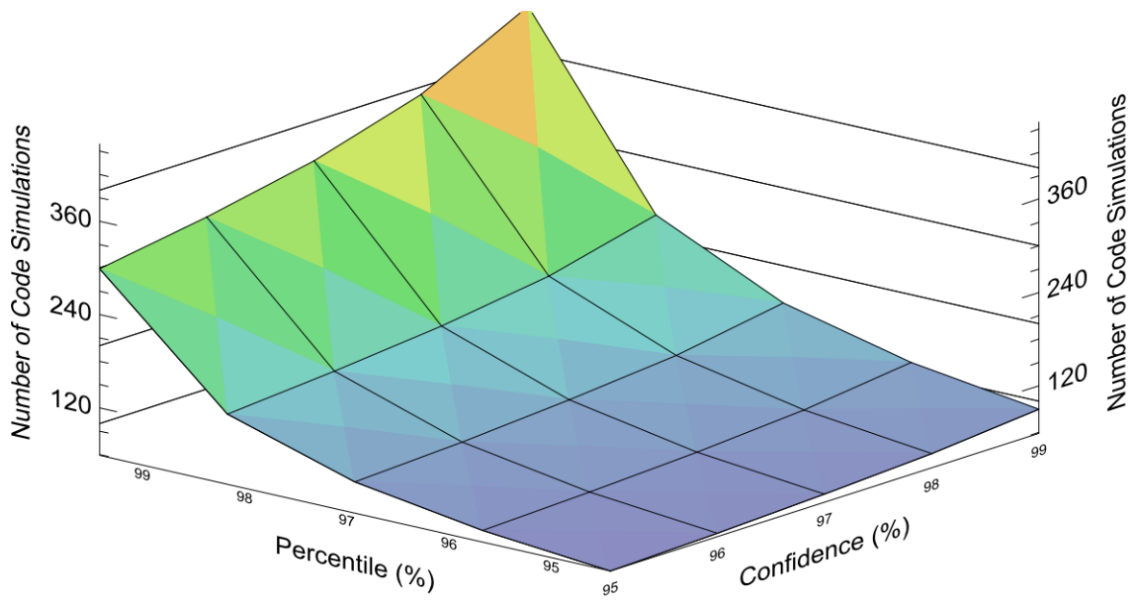
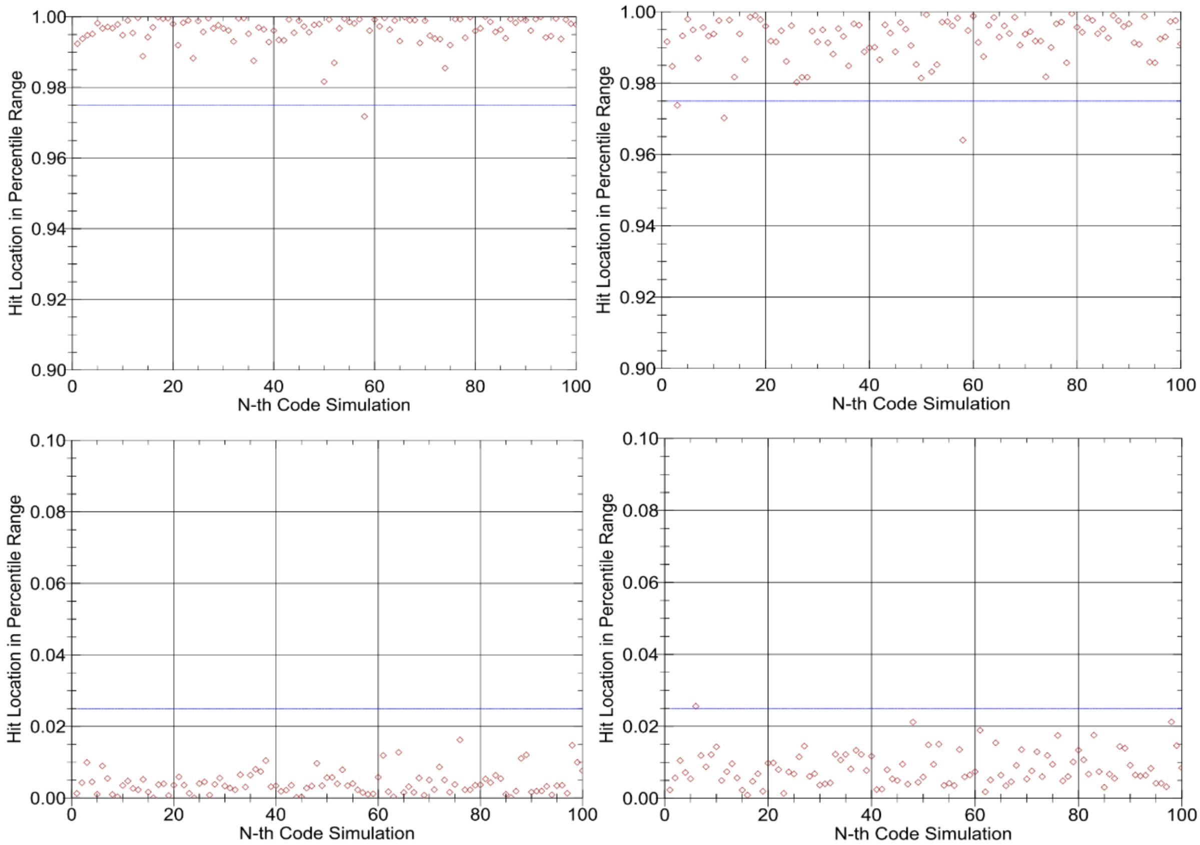


Figure 2. Trend of Number of Code Runs for Percentile and Confidence Frame

First Maximum and Minimums | Second Maximum and Minimums



* Note: The Percentile in Y axis in the above snapshots correspond to cumulative probability

Figure 3. Numerical Simulation of Max and Mini for 2nd Order Two-sided Statistics from 100 Test Cases of Each of 221 Code Run

An Integrated Approach for Characterization of Uncertainty in Complex Best Estimate Safety Assessment

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Abstract

This paper discusses an approach called Integrated Methodology for Thermal-Hydraulics Uncertainty Analysis (IMTHUA) to characterize and integrate a wide range of uncertainties associated with the best estimate models and complex system codes used for nuclear power plant safety analyses. Examples of applications include complex thermal hydraulic and fire analysis codes. In identifying and assessing uncertainties, the proposed methodology treats the complex code as a "white box", thus explicitly treating internal sub-model uncertainties in addition to the uncertainties related to the inputs to the code. The methodology accounts for uncertainties related to experimental data used to develop such sub-models, and efficiently propagates all uncertainties during best estimate calculations. Uncertainties are formally analyzed and probabilistically treated using a Bayesian inference framework. This comprehensive approach presents the results in a form usable in most other safety analyses such as the probabilistic safety assessment. The code output results are further updated through additional Bayesian inference using any available experimental data, for example from thermal hydraulic integral test facilities. The approach includes provisions to account for uncertainties associated with user-specified options, for example for choices among alternative sub-models, or among several different correlations. Complex time-dependent best-estimate calculations are computationally intense. The paper presents approaches to minimize computational intensity during the uncertainty propagation. Finally, the paper will report effectiveness and practicality of the methodology with two applications to a complex thermal-hydraulics system code as well as a complex fire simulation code. In case of multiple alternative models, several techniques, including dynamic model switching, user-controlled model selection, and model mixing, are discussed.

Keywords: Model Uncertainty, Code Structure, IMTHUA, Thermal-Hydraulics System Code

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Nomenclature

CSAU	Code Scaling, Applicability and Uncertainty Evaluation
ECCS	Emergency Core Cooling System
GRS	Gesellschat Fur Anlagen- und Reaktorsicherheit
HTC	Heat Transfer Coefficient
IMTHUA	Integrated Methodology for TH uncertainty analysis
ITF	Integrated Test Facility
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
MCMC	Markov Chain Monte Carlo
PCT	Peak Clad Temperature
PIRT	Phenomena Identification and ranking Process
PWR	Pressurized Water Reactor
SET	Separate Effect Test
TH	Thermal-Hydraulics
UMAE	Uncertainty Analysis Methodology based on Accuracy Extrapolation
USNRC	United States Nuclear Regulatory Commission

1. Introduction

The importance of more comprehensive treatment of uncertainties in complex computational codes has been recognized by the technical community and also in regulatory applications. An example is USNRC's ECCS licensing rules that require characterization of uncertainties in the use of best estimate computer codes [2]. An effective uncertainty analysis methodology in complex system models requires a comprehensive treatment of many uncertainty sources in its transient behavior. Code structure (model) is a crucial source of uncertainty in TH analysis results. The structure refers to such model features as the assembly of sub-models and correlations for simulating the various physical phenomena, system components for fluid and structural simulation, and cases where multiple sub-models may be available for code calculations. The model uncertainty is assessed in several of the existing TH code uncertainty analysis methodologies. For example, CSAU[3] uses correction multipliers for code assessment and applicability adjustments of the models based on the available models. UMAE[4,5] assesses the models qualitatively to account for code structure model uncertainty. The uncertainty approach proposed by GRS[3] considers the alternative models estimates as the major source of model uncertainty. For example, the models for critical heat flux were sampled based on three available models (total of three models) based on their applicability and credibility weights.

Uncertainty expressed probabilistically by using probability density functions is the most prevalent formal approach to quantitatively characterize unknowns. Using all available knowledge, including data, models, and expert opinion one may formally derive the corresponding uncertainties. In the case of TH code uncertainty analysis, this requires a number of qualitative and quantitative steps to consider the state of knowledge in the various contexts of interest[6].

The approach described in this paper, called IMTHUA, provides a formal way to assess the uncertainty of the best estimate TH system code calculations. Code structure in this research is limited to physical models with corresponding parameters implemented in the codes to simulate TH characteristics of accident scenarios. Numeric-related issues, including numerical resolution,

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[2] USNRC (1989), "Best-Estimate Calculations of Emergency Core Cooling System Performance", USNRC Regulatory Guidance 1.157.
[3] Technical Program Group (1989), "Quantifying Reactor Safety margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large Break Loss of Coolant Accident," Technical report NUREG/CR-5249 EG&G Idaho, Inc.
[4] CSNI (1998), "Uncertainty Analysis Study," NEA-CSNI/R(97)35/V.2-2.1 report
[5] D'Auria F. et al, (1998), "Overview of Uncertainty Issues and Methodologies," Proceeding ECD/CSNI Seminar on Best Estimate Methods in Thermal Hydraulic Analysis, Ankara, 437-458.

convergence methods, and styles, are not explicitly assessed as part of the code structure. However these issues are part of the IMTHUA methodology for uncertainty assessment, and have been addressed implicitly by code output updating step of the IMUTHA methodology for consideration of the missing/screened out sources of uncertainties. Figure 1[6,7] shows the overall steps involved in IMUTHA.

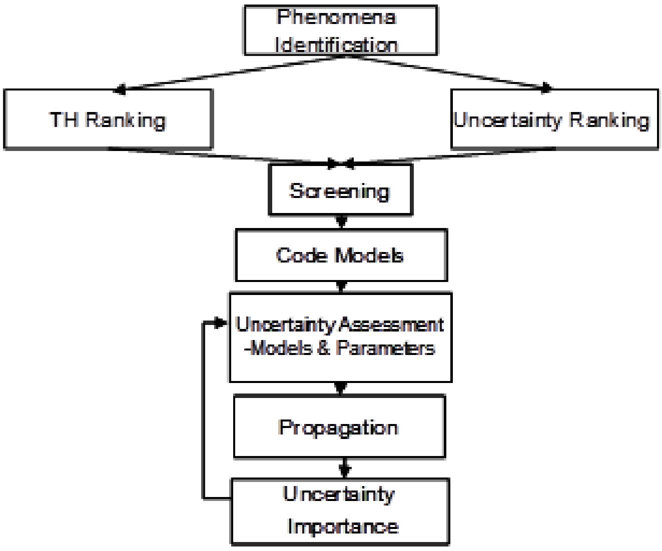

Figure 1: IMTHUA Methodology Flowchart Input-Based Phase Main Steps

2. Structure of TH codes and Complexities

A key characteristic of the TH systems codes is their complexity. This complexity has its roots in the composite structure of these systems consisting of many different elements (or sub-systems) whose state inevitably affects the state of the whole system. Its main implication is the dynamic (i.e. variable in time) and/or non-linear behavior. Systems with many interdependent elements (such as sophisticated modern process systems with multiple feedback loops) exhibit an unpredictable highly nonlinear behavior that can sometimes result in sudden output changes following a minor variation in the input (i.e., the so-called "butterfly effect"). TH codes are complicated because of their structures' inherent complexity, which is described below:

- Limited user control over code structure
- Lack of appropriate data and information about the basis of models, sub-models, and actual variables, such as HTC
- Large number of models and correlations (thousands) involved in the computations
- Dynamic code behavior in which only a small portion of the code models may be active during each time step, depending on the underlying simulation and system conditions
- Many horizontal and vertical regime phases in the code calculation, with fuzzy borders between them

[6] Pourgol-Mohamad, M., Modarres, M. and Mosleh A., (2009), Integrated Methodology for Thermal Hydraulics Uncertainty Analysis with Application, Volume 165/ No. 3/March 2009/ 333-359.
[7] [8] Pourgol-Mohamad M., Mosleh M., Modarres M., (2006), "Treatment of Uncertainties; Output Updating in Complex thermal-hydraulics (TH) Computational Codes", Proceedings of ICONE14 Conference, Miami, FL.

- Deficiency in precisely solving field equations for specific configurations due to coarse average nodes; for example, choked flow model is called in TH codes calculation when the results of momentum equation calculation is unsatisfactory. The code calls for a choked flow model for velocity calculation and replaces it with the previous calculation. For better resolution, TH codes are recently coupled with CFD codes for the sake of precise calculation where needed[8].

3. Code User Assumptions and Input Deck

The user input and user options for input preparation have great impact on code model uncertainties. User assumptions and decisions in preparing the input deck in how code models (structure) is utilized are critical with respect to the quantitative assessment of the code uncertainties. The user has a significant influence on the input deck, given the many options for the executable code preparations; the user also affects the computation in the various stages of code calculation, some of which are listed in Table 1, where user choices influences the overall calculation.

Table 1: TH Code Input Deck Preparation and User Options in Model Uncertainty[9]

User Domains	Impacts
System Nodalization	-Node Size -Component Selection -Node Numbers
Code Options	-Input parameters related to specific system characteristics -Input parameters needed for specific system components -Specification of initial and boundary conditions -Specification of state and transport property data -Selection of parameters determining time step size -Choice between engineering or alternative models, e.g., critical flow models -Efficiency of separators Parameter -Two-phase flow characteristics of main coolant pumps -Pressure loss coefficient for pipes, pipe connections, valves, etc.
Code Source Adjustments	-Multipliers (correlation coefficients) -Choice between engineering or alternative models, e.g., critical flow models in a specific time -Numerical scheme

4. TH System Code Structure

All uncertainty sources should ideally be explicitly considered in the analysis; however, this is impractical, due to a lack of knowledge and/or limited resources. These limitations are mainly due to access and scarcity of empirical data. Limitation in data sources is a significant barrier, especially with limited field data for studying nuclear facility dynamic (transient) behavior, and causes difficulties in quantifying uncertainties in the code structure. In cases where data were obtained from scaled-down experimental facilities, the model evaluations rely exclusively on the subjective interpretation of the information available at the time of the analysis. This leads to the conclusion that

[8] Aumiller D.L. at el, (2000), "A Coupled Relap5–3D/CFD Methodology with a Proof-of-Principle Calculation" 2000 International RELAP5 Users Seminar, Jackson Hole, Wyoming.

[9] Aksan, S.N., at el, (1994), "User Effects on the Transient System Code Calculations", NEA/CSNI/R(94)35

any attempt to address the issue of model uncertainty in a quantitative manner will rely mainly on expert judgment.

There are different approaches to modeling code structure uncertainties. The approach may view the structure as a “black box”, with no knowledge of the inner code structure, or as a “white box”, with explicit characterization of the internal code structure. A structural, or “white box” uncertainty assessment, allows the user to peek inside the code structure and focusing specifically on the internal sub-models and the state of data, information and knowledge that supports the inner structure of the code. The degree of model uncertainty varies among available methodologies, and is treated as a weight assignment for the alternative models and correlations in the GRS[10]. Code uncertainty assessment is also used in CSAU[3] and ASTRUM[11] for treating model uncertainties. IMTHUA considers the code structure shown in Figure 2 by treating the uncertainties associated with code’s sub-models and alternatives models, as well as the interaction between them[6].

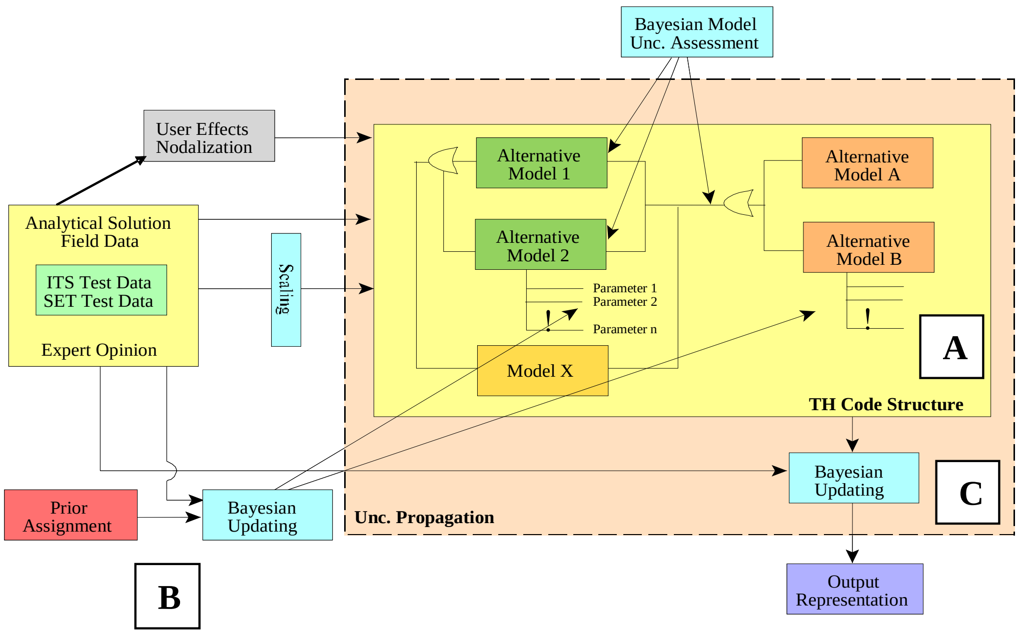

Figure 2: Code Structure Treated in IMTHUA Methodology for Uncertainty Quantification[13]

Treatment of the code structure uncertainty is shown in Step A of Figure 2. A key objective of the proposed method is the quantification of uncertainties due to model form (structure) as well as model parameters. In relation to model structure uncertainty, the methodology attempts to explicitly account for identifiable sources of structural uncertainty. This is applied both at the sub-model levels and also the entire TH code[6]. Input parameter uncertainty quantification is performed via the Maximum Entropy and/or and expert judgment methods, depending on the availability and type of information[6] (Step B in Figure 2). Hybrid of Input-Based and Output-Based Uncertainty Assessment (Step C in Figure 2) uncertainty analysis in IMTHUA is two-step uncertainty quantification. Table 2 lists the features of IMTHUA methodology in dealing with different aspects of TH system codes uncertainty.

[10] Glasaer H. et al., (1988), “NEA-CSNI Report; GRS Analyses for CSNI Uncertainty Methods Study (UMS),” Nuclear Energy Agency–Committee on the Safety of Nuclear Installations.
[11] Frepolli C., et al., (2004), “Realistic Large Break LOCA Analysis of AP1000 with ASTRUM,” Proc. 6th Int. Conf. Nuclear Thermal Hydraulics, Operations and Safety (NUTHOS-6), Nara, Japan.

IMTHUA was developed as a comprehensive framework for the assessment of uncertainties in TH system code results. Details of the methods and steps of IMTHUA as well as a discussion on some of its limitations are provided in reference 6. Key characteristics of IMTHUA are:

1. Modified PIRT[12]: A modified form of the PIRT process[13] is proposed. The two-step method identifies and ranks phenomena based on their (a) TH influence (using AHP[14]), and (b) uncertainty rating based on an expert judgment procedure.
2. White Box Treatment (Step A in Figure 2): A key objective of the proposed method is the quantification of uncertainties due to model form (structure) as well as model parameters. In relation to model structure uncertainty, the methodology attempts to explicitly account for identifiable sources of structural uncertainty. This is applied both at the sub-model levels and also the entire TH code.
3. Methods for input parameter uncertainty quantification via the Maximum Entropy and/or and expert judgment, conditioned on availability and type of information[13] (Step B in Figure 2). Bayesian methods were used to update such uncertainty distributions with new evidence when available. The information could be in the form of fully or partially relevant data, expert opinions, or qualitative assessments.
4. The two-step Hybrid of Input-Based and Output-Based Uncertainty Assessment (Step C in Figure 2). The first step quantifies code output uncertainties associated with identified structural and parametric uncertainties at input and sub-model levels. The second step updates this uncertainty distribution with any available integrated experimental data and validation information. This “output uncertainty correction” phase (the subject of this paper) is intended to at least partially account for code user choices (user effects), numerical approximations, and other unknown sources of uncertainties (model and parameter) not considered in the first phase.
5. Efficient uncertainty propagation through the use of modified Wilks’ tolerance limits sampling criteria to reduce the number of Monte Carlo iterations for the required accuracy.

Through assessing and propagating these uncertainties various output values are obtained, an example of which is shown in Figure 3 for LOFT Facility's fuel clad temperature. The code was executed 93 times. The uncertainty range was resulted by propagation of a set of parameter uncertainties and model uncertainties as discussed in detail in [6.] The fuel clad temperature uncertainty range was benchmarked with LOFT LB-1 experiment data.

[12] Pourgol-Mohamad M, Modarres M., Mosleh A. Modified Phenomena Identification and Ranking Table (PIRT) For Uncertainty Analysis, Proceedings of 14th International Conference on Nuclear Engineering, July 17-20, 2005, Miami, Florida, USA.

[13] Pourgol-Mohamad M (2007a), “Integrated Methodology on Thermal Hydraulics Uncertainty Analysis (IMTHUA),” Ph.D Thesis, University of Maryland, College Park.

[14] Saaty T.L., “Models, methods, concepts & applications of the analytic hierarchy process,” Kluwer Academic Publishers, Boston 2001.

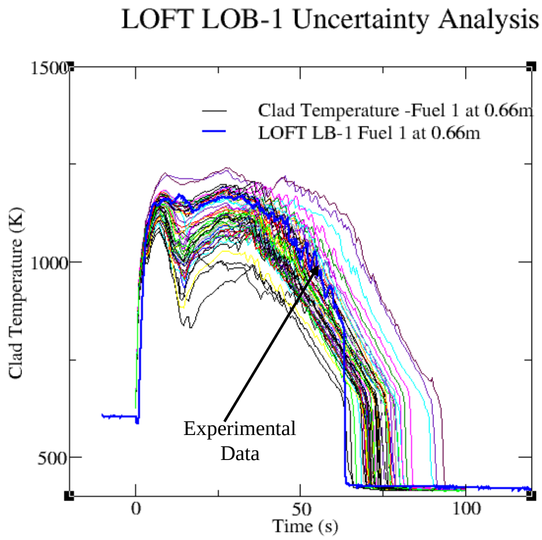

Figure 3: Input-Based Uncertainty Propagation Results vs. Experimental Data

Table 2: Features of IMTHUA Methodology

Location of Uncertainty	Property for which Uncertainty was Developed	Methodology
Input	**Parameters**	**Use of Data in Probabilistic Methods**
	Models	**Expert Judgment, Corrective Factors/Biases**
	Input Parameter Restriction?	**No**
	Dependency on No. of Code Calculations?	**No**
	Dependency Consideration	**No**
	Input Uncertainty Assignment	**Maximum Entropy/Bayesian**
Propagation	**Parameters**	**Order Statistics-Based Wilks Tolerance Limit**
	Models	**Order Statistics-Based Wilks Tolerance Limit**
Output	**Parameters**	**NA**
	Models	**Use of Integrated Data/Bayesian Approach**
	Consideration of Missed/Screened Out Sources of Uncertainty	**Yes**

5.0 Concluding Remarks

Structural models have important contributions to the quantified results of TH codes. This paper discusses strategies in structural model uncertainty assessment, along with other sources of uncertainty. The main thrust of the methodology is to efficiently utilize all available types of data to identify important sources of uncertainty, and to assess the magnitude of their impact on the uncertainty of the TH code output values. Single sub-model uncertainties, as well as alternative models, were treated in code structure. Depending on the conditions and on the availability of information and data, different solutions are proposed for uncertainty assessment of the models. A Bayesian solution was proposed for single and multiple models' structure uncertainty assessment along with mixing, switching, maximization/minimization, and user effect consideration for alternative models. Future work could cover the following areas:

- A method for systematic separation of aleatory and epistemic uncertainties in the final results.
- The ability to account for temporal uncertainties, since in some cases uncertainty of physical phenomena has magnitude as well as temporal dimensions.
- Representation of uncertainty in temporal dimensions, i.e., interpreting and representing the output uncertainty results correctly for time-variant output parameters.

Acknowledgments

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Information Synthesis in Uncertainty Studies : Application to the Analysis of the BEMUSE Results

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1. Introduction

To demonstrate that the nuclear power plants are designed to respond safely at numerous postulated accidents computer codes are used. The models of these computer codes are an approximation of the real physical behaviour occurring during an accident. Moreover the data used to run these codes are also known with a limited accuracy. Therefore the code predictions are not exact but uncertain.
To deal with these uncertainties, "best estimate" codes with "best estimate" input data are used to obtain a best estimate calculation and it is necessary to derive the uncertainty associated to their estimations. For this reason, regulatory authorities demand in particular to technical safety organization such as the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN) to provide results taking into account all the uncertainty sources to assess safety quantities are below critical values. Uncertainty analysis can be seen as a problem of information treatment and a special effort on four methodological key issues has to be done.
The first one is related to information modelling. In safety studies, one can distinguish two kinds of uncertainty. The first type, called aleatory uncertainty, is due to the natural (such as spatial) variability of an observed phenomenon and cannot be reduced by the arrival of new information. The second type, called epistemic uncertainty, can arise from imprecision (a variable has a fixed value which is badly known due to systematical error in measurement or lack of data, knowledge or experiment). Contrary to the previous one, this uncertainty can be reduced by increasing the state of knowledge (use of devices giving more precise measurements, expert providing a more informative opinion,…).. *Performing a relevant information modelling therefore requires to work with a mathematical formalism flexible enough to faithfully treat both types of uncertainties.*
The second one deals with information propagation through a computer code. It requires to run the codes several times and it is usually achieved thanks to a coupling to a statistical software. The complexity of the propagation is strongly connected to the mathematical framework used for the information modelling. The more general (in order to faithfully represent the real state of knowledge on uncertainties) the mathematical framework, the more time consuming the propagation should be. Therefore, the key point is here *to construct a numerical treatment for uncertainty propagation which reduces the computational cost and can be applied to complex models used in practice*.
In nuclear safety studies, different uncertainty analyses using different codes and implying different experts are generally performed. Deriving benefits from these analyses appears to be a problem of information synthesis which is the third key issue. Indeed each uncertainty study can be viewed as an information source on quantities of interest. It is then useful *to define formal methods to combine all these information sources in order to improve the reliability of the results and to detect possible conflicts (if any) between the sources*.
The efficiency of an uncertainty analysis requires a reliable quantification of the information associated to uncertainty sources. This quantification is addressed in the fourth key issue. It consists in exploiting the information related to available experiments and to the comparison code/experiment to infer the uncertainty attached to the code input parameters. Therefore, the crucial points stand in *the choice of an experimental database sufficiently representative and exhaustive of the considered phenomenon and in the construction of an efficient treatment (numerical procedure, experts' judgement) to perform this inference.*

The two first points have been deeply studied in the frame of the OECD BEMUSE Program. In particular, it came out that statistical approaches, based on Monte-Carlo techniques, are now

sufficiently robust for the evaluation of uncertainty on a LB-LOCA transient. Concerning the mixing between aleatory and epistemic uncertainties, we refer to [1] and [2] for a full description of a new methodology to combine these two types of uncertainties and its application to nuclear study as well. In this paper, we focus on the third issue and present some recent developments proposed by IRSN to derive formal tools in order to improve the reliability of an analysis involving different information sources. It is applied to exhibit some important conclusions from the two BEMUSE benchmarks. For sake of completeness, we recall that the last key issue is the main topic of the new OECD PREMIUM Program where IRSN is involved.
This work is organized as follows. Section 2 is devoted to a quick overview of the BEMUSE Program and to the problem of information synthesis that we have to cope in order to analyse the results provided by each participant. Two formal tools proposed by IRSN to evaluate and combine information are then introduced in Section 3. They are finally applied to derive some important conclusions from the two BEMUSE benchmarks.

2. BEMUSE Program and information synthesis

A wide spectrum of uncertainty methods applied to Best Estimate codes exists and is used in research laboratories but their practicability and/or validity is not sufficiently established to support general use of codes and acceptance by industry and safety authorities. In this context, the OECD BEMUSE Program represented a step for a reliable application of high-quality best-estimate and uncertainty evaluation methods. The activity of the Program consisted in two main steps devoted to applications of uncertainty methodologies to Large Break LOCA scenarios. The first step is related to best-estimate and uncertainty/sensitivity analyses of the LOFT L2-5 test (Phases 2 and 3). LOFT is the only integral test facility with a nuclear core where safety experiments have been performed. The second step concerns best-estimate and uncertainty/sensitivity analyses of a nuclear power plant (Phases 4 and 5). They were performed on the nuclear Zion reactor that was shut down in 1998 following 25 years in service.

2.1 Uncertainty analysis

Most of the BEMUSE participants, except UNIPI ([3]), followed the same statistical uncertainty analysis methodology. It consists in choosing a probability distribution function (pdf) to represent each uncertain parameter. If dependencies between uncertain parameters are known, they are quantified by correlation coefficients. Uncertainty propagation through the computer code is performed thanks to Monte-Carlo simulations ([4]). In Monte-Carlo simulation, the computer code is run repeatedly, each time using different values for each of the uncertain parameters. These values are drawn from the probability distributions and dependencies chosen in the previous step. In this way, one value for each uncertain parameter is sampled simultaneously in each repetition of the simulation. The results of a Monte-Carlo simulation lead to a sample of the same size for each output quantity. Using the law of large number, the output sample is used to get any typical statistics of the code response such as mean or variance and to determine the cumulative distribution function (CDF). The CDF allows one to derive the percentiles of the distribution (if X is a random variable and F_X its CDF, the α-percentile, $\alpha \in [0;1]$, is the deterministic value X_α such that $F_X(X_\alpha)=\text{Proba}(X \le X_\alpha)=\alpha$). These percentiles then lead to quantitative insight on the likelihood of the code response to exceed a critical value but are also used to exhibit the so-called uncertainty margins.

2.2 Uncertainty results

An uncertainty analysis was performed by each participant for the two benchmarks LOFT and Zion. Six output parameters were considered by all participants. They were of two kinds: scalar output parameters (First Peak Cladding Temperature, Second Peak Cladding Temperature and Time of accumulator injection, Time of complete quenching of fuel rods) and time trend output parameters (Maximum cladding temperature and Upper plenum pressure). Figure 1 displays the uncertainty margins obtained by each participant for the four scalar outputs in the Zion case ([5]). They are summarized by a lower and upper bound. The best-estimate calculation is provided as well.

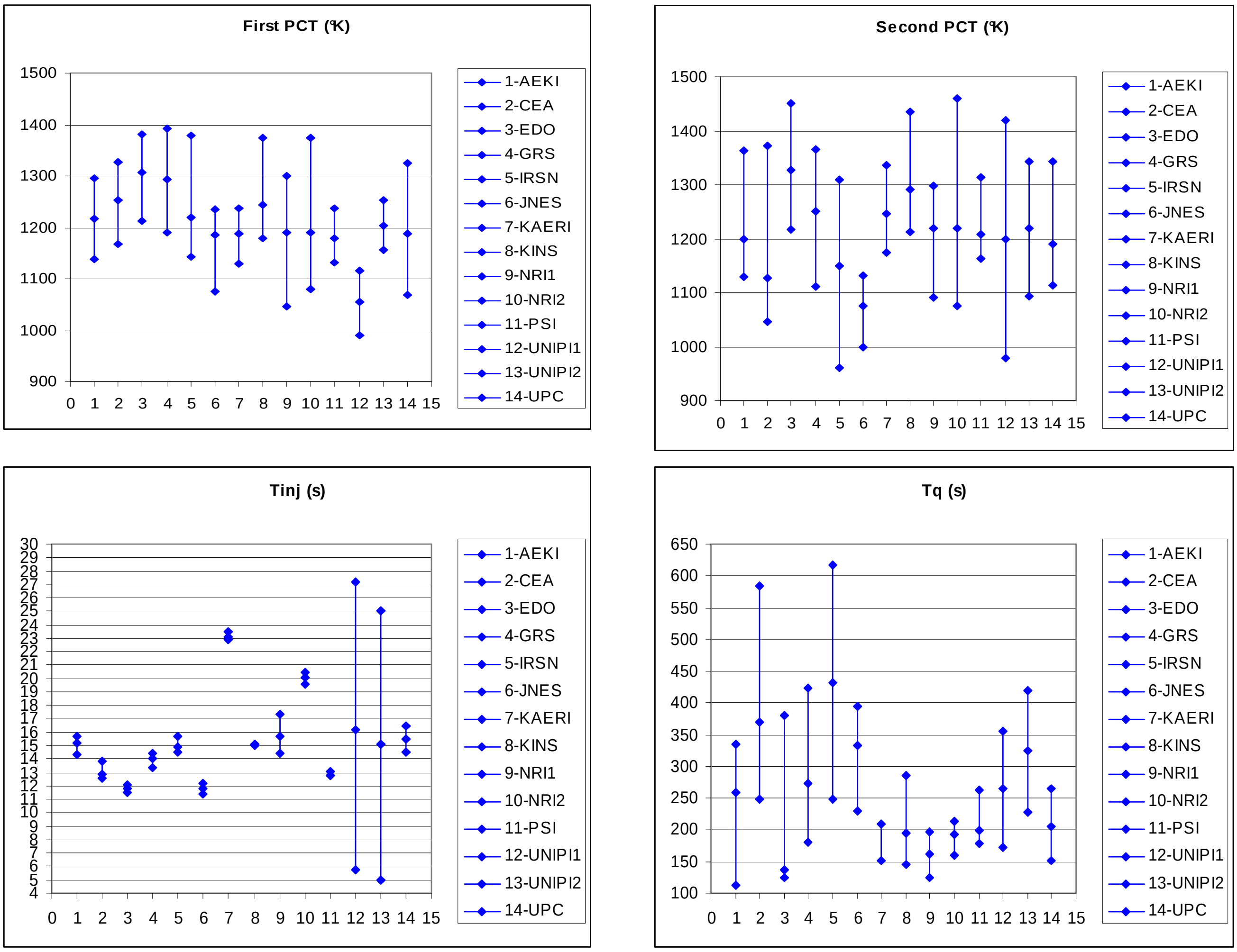

Figure 1: Uncertainty margin and reference calculation given by all participants for the Zion benchmark.

A straightforward analysis of the results depicted by Figure 1 points out a large discrepancy between reference calculations (250 K between EDO and UNIPI1 for PCT1). Moreover, some differences between the width of uncertainty margins are noticeable (for PCT1: 294 K for NRI2 and 96 K for UNIPI2). As a result, there is no overlapping when taking into account the information given by all participants. However, there might be some groups of participants (the users of the same code for example) that provided some results that are in agreement. This information is difficult to extract from Figure 1 since there are 4 output variables and 14 participants. Therefore, for the remaining of the paper, we propose to construct and apply formal tools in order to provide a more complete and reliable synthesis when multiple information sources are involved.

3. Formal tools to synthesize uncertainty margins coming from multiple sources of information

3.1 Construction

Most of the formal approaches proposing to handle information given by multiple sources consist in three main steps:

- Modelling of information provided by each source: it consists in choosing a mathematical framework to represent the available information such that the lower and upper bounds of an uncertainty margin as well as some particular values within this interval (reference calculation).

- Evaluation of the quality of the information: it requires to define and compute numerical criteria in order to take into account the precision of the information and its coherence with observed reference values (if any).
- Information fusion: it implies the definition and the application of fusion operators to build a summary of all information provided by the sources.

We describe in what follows how to perform these different steps in the frame of the probability and possibility theories. Full details and ampler discussions can be found in [6], [7] and [8].
In the following, we will consider that N sources $s_{1},...,s_{N}$ provide information about V different variables $X_{1},...,X_{V}$

3.1.1 Information modelling

Probabilistic approach

For each variable of interest, information provided by sources is modelled by a finite set of $B+1$ percentiles with the $k\%$ percentile denoted $q_{k\%}$. If the same $B+1$ percentiles are provided for each variable $X_{1},...,X_{V}$, then the information on each variable can be modelled by an histogram or probability density $p = (p_1 , . . . , p_B)$ made of B interpercentiles. In the following, we will denote $q_l = q_{0\%}$ and $q_u = q_{100\%}$ the entire range of variation of one variable.
The figure below represents the cumulative distribution function (CDF) corresponding to a temperature for which available information is $q_{0\%}$ = 500 K, $q_{5\%}$ = 600 K,$q_{50\%}$ = 800 K,$q_{95\%}$ = 900 K, $q_{100\%}$ = 1000 K. The corresponding probability density p = (0.05, 0.45, 0.45, 0.05) is pictured in dashed lines.

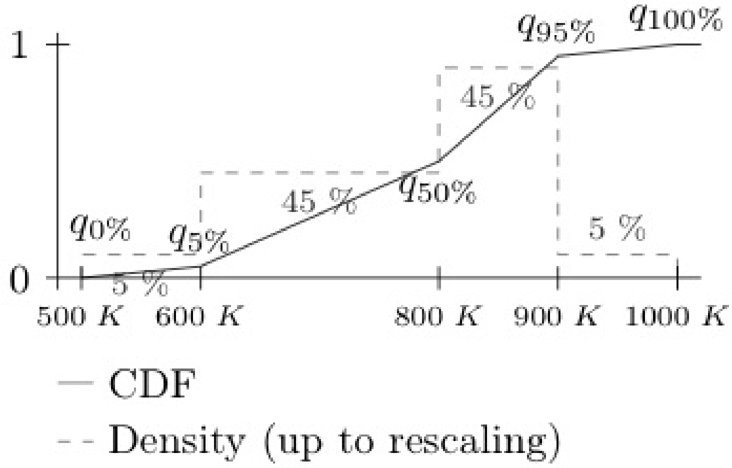

Figure 2: Example of information modelling within the probabilistic framework.

Possibilistic approach

A possibility distribution is formally a mapping $\pi : \Re \rightarrow [0, 1]$. From this distribution, we can define two (conjugate) measures describing the likelihood of events:

- the possibility measure: $\Pi(A) = Sup_{x\in A}\ \pi(x)$
- the necessity measure: $N (A) = 1 - \Pi(A^c)$

with A^c the complement of A. To every value $\alpha \in [0, 1]$ we can associate an α-cut A_α which is the set $A_\alpha = \{x|\pi(x) \geq \alpha\}$. When interpreted as a family of probability distributions ([9]), a possibility distribution can be seen as a collection of nested intervals with lower confidence levels, that is $P (A_\alpha) \geq N (A_\alpha)$ $(N (A_\alpha) = 1 - \alpha)$, with P an imprecisely known probability.

The figure below represents a possibility distribution corresponding to a temperature where information consists of four intervals [750 K, 850 K], [650 K, 900 K], [600 K, 950 K], [500 K, 1000 K] which have respective confidance levels of 10%, 50%, 90% and 100%.

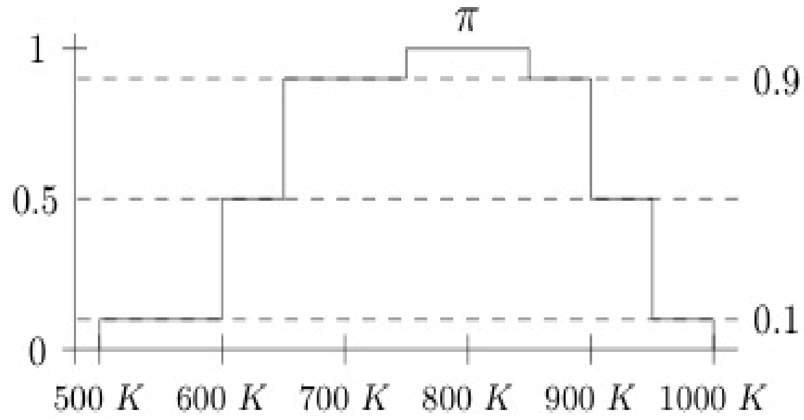

Figure 3: Example of information modelling within the possibilistic framework.

3.1.2 Information evaluation

The information provided by the sources is evaluated by two different criteria, measuring different qualities:

- Informativeness: it measures the precision of the information. The more precise a source is, the more useful it is
- Calibration: it measures the coherence between information provided by the source and a reference (such as experimental) value.

<u>Probabilistic approach</u>

Informativeness: let $p = (p_1 , . . . , p_B)$ be the probability density modelling the source information for a variable X, and $u = (u_1 , . . . , u_B)$ be the uniform density for the same variable on the same range of variation (u = (0.20, 0.40, 0.20, 0.20) for the example of Figure 2). The informativeness $I(p,u)$ for this source on variable X is simply the relative entropy

$$I(p,u) = \sum_{i=1}^{B} p_i \log\left(\frac{p_i}{u_i}\right)$$

and if the source gives information on V variables, the global informativeness is simply the mean of informativeness scores over all these variables.

Calibration: given a set of V variables $X_1,...,X_V$ for which the source has provided information and for which we have reference values, calibration scores assume that, for a perfectly calibrated source, p_1% of the true values will fall in interpercentile 1, p_2% in interpercentile 2, ... Now, if in reality r_1V values fall in interpercentile 1, r_2V in interprecentile 2, we can build an empirical distribution $p = (p_1 , . . . , p_B)$, and compute the divergence of p from r. (i.e., the surprise of learning r when p is thought to be the right answer)

$$I(r,p) = \sum_{i=1}^{B} r_i \log\left(\frac{r_i}{p_i}\right)$$

which is minimal (=0) if and only if *r=p*. As it is known that *2*N*I(r,p)* tends to a chi-square distribution with *B-1* degree of freedom (*N* being the number of observed values), the final value of the calibration for a given source *s* is

$$Cal_p(s) = 1 - \chi^2_{B-1}\left(2 * N * I(r,p)\right)$$

The final score is then obtained as the product of both calibration and informativeness (the product is used as a "good" source has to be informative and well calibrated)

Possibilistic approach

Informativeness: let π_s be the distribution modelling source *s* information for one variable and π_{ign} the distribution corresponding to interval $[q_l,q_u]$ ($\pi_{ign}(x)=1$ if $x \in [q_l,q_u]$, zero otherwise). Then, the informativeness of source *s* is simply

$$I(\pi, s) = \frac{|\pi_{ign}| - |\pi_s|}{|\pi_{ign}|}$$

with $|\pi_s|$ the cardinality of π_s which reads $|\pi_s| = \int_{q_l}^{q_u} \pi(x)dx$. This is simply the area under the distribution. If a source provides information for multiple variables, the final informativeness is simply the average of all informativeness over all variables.

Calibration: assume the (true, reference) value x* of a variable is available. Then, the calibration of source *s* is simply the value $\pi_s(x^*)$, that is how much the value x* is judged plausible by the information given by the source. If information is given for multiple variables, the final calibration score is again the average of all calibration scores. Note that, in the case of possibilistic approach, calibration is computed individually for each variable (whereas it was done globally in the probabilistic approach).

3.1.3 Information fusion

As for information fusion, it is common to distinguish three main kinds of behaviors:

- conjunctive: equivalent to take the intersection between the information provided by the sources, conjunctive fusion assumes the reliability of all sources, and allows to quantify the conflict among them. It produces precise but potentially unreliable results in case of strong conflict,
- disjunctive: equivalent to take the union between the information provided by the sources, disjunctive fusion makes the conservative assumption that at least one source is reliable. It produces in general imprecise but reliable results,
- arithmetic (weighted) mean: assumes independence between sources and produces a result between disjunction and conjunction.

The probability approach as tool of information modelling is mathematically well-founded and its practical interest has been confirmed by many applications over the years. The arithmetic weighted is the usual approach to synthesize probability distributions ([6]). Also, there are no real counterparts to set intersections and unions in the probability field, therefore conjunction and disjunction are difficult to formally define. Moreover, it is not always meaningful to relate the uncertainty attached to information by a probability especially in case of epistemic uncertainty. Therefore, we propose to perform information fusion in the framework of possibility theory ([9]).

Let $\{\pi_i\}_{i=1,\dots,N}$ be a set of possibility distributions (for example, each distribution corresponds to the choice of an expert to represent the uncertainty attached to a given uncertain parameter), one associates to each of the three different previous behaviors the following fusion operator:

$$\text{Disjunction: } \pi_{\cup}(x) = \max_{i=1,\dots,N} (\pi_i(x)),$$
$$\text{Conjunction: } \pi_{\cap}(x) = \min_{i=1,\dots,N} (\pi_i(x)),$$
$$\text{Arithmetic mean: } \pi_{mean}(x) = \sum_{i=1}^{N} (\pi_i(x)).$$

Probabilistic (arithmetic mean) and possibilistic fusions are illustrated in Figure 4 when two sources provide information.

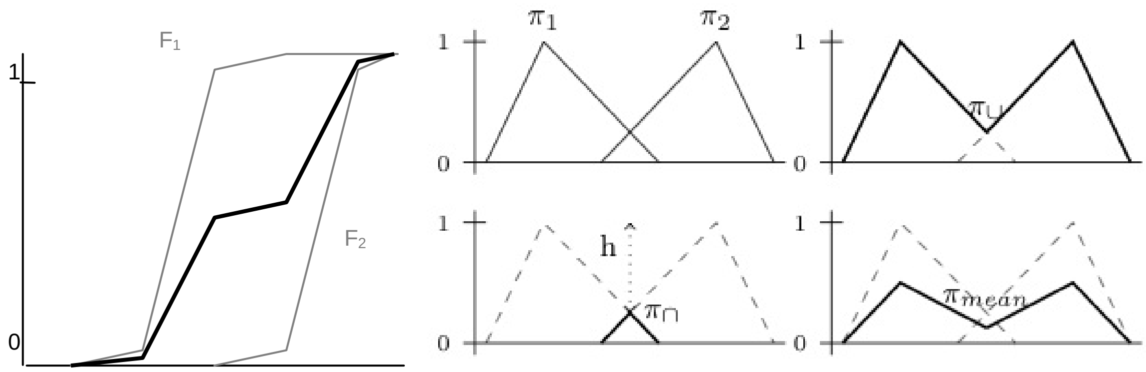

Figure 4: Examples of fusion operators. Left, probabilistic framework (arithmetic mean, CDF), right, possibilistic framework (disjunction, conjunction and arithmetic mean, possibility distributions)

Note that the possibility-based approach offers a greater flexibility than the probabilistic one since it allows a clearer distinction between disjunction and conjunction, making easier the detection of possible conflicts between the sources (the conflict is quantified on Figure 4 by the distance h). This is emphasized by the application to the BEMUSE benchmark which is described in the following section.

3.2 Synthesis of the results provided by the BEMUSE participants on the Zion benchmark

We have limited this study to the following four scalar outputs: First Peak Cladding Temperature (PCT1), Second Peak Cladding Temperature (PCT2), Accumulator Injection Time (Tinj) and Complete Core Quenching Time (Tq).
It should be kept in mind that this study is preliminary and that its main purposes are to show that such methods can be useful in the analysis of the information and to make some first conclusions about the benchmark. A full study would require a longer and deeper work, but interesting conclusions were already drawn from this preliminary study. All the results have been computed by the means of the SUNSET software developed by IRSN.

First step:

The first step was to model the information provided by each source. Thus, for each variable, the entire variation domain $[q_l,q_u]$ was chosen to be the entire variation between the minimal and maximal values given by all participants (e.g., for PCT1, the minimum/maximum value among all participants was 991K/1393K and therefore $[q_l,q_u]$=[991K,1393K]). Then for each participant i and for each safety quantity q, a triplet (q^i_1, q^i_2, q^i_3) of values was defined from its uncertainty range and its reference value. At the end, the information relative to a quantity q and provided by the participant i was modelled by a quintuplet $(q_l, q^i_1, q^i_2, q^i_3, q_u)$. We then choose two different mathematical modellings to represent this information.

Probabilistic modelling: the quintuplet (q_l, q^i_1, q^i_2, q^i_3, q_u) of values was respectively assigned to the percentiles: 0%, 5%, 50%, 95% and 100%. For example, for PCT1, if a participant has provided an uncertainty range of [1142K,1379K] with a reference value of 1218K, we have $q_{0\%}$ = 991K, $q_{5\%}$ = 1142K, $q_{50\%}$ = 1218K, $q_{95\%}$ = 1379K, $q_{100\%}$ = 1393K.
Possibilistic modelling: in the same way, we used the quintuplet (q_l, q^i_1, q^i_2, q^i_3, q_u) to define a possibility distribution: 0%, 5%, 100%, 5% and 0%.

The second step is source evaluation (informativeness and calibration). It allows us to evaluate the quality of the information delivered by each source. It is possible provided we possess, for each variable, an experimental value, which is not the case for the Zion benchmark. However, this evaluation has been performed in a previous study in the frame of BEMUSE phase 3 (LOFT L2-5 experiment, [10]).

Second step: source evaluation on the LOFT L2-5 benchmark:

We summarize in what follows the main results of this evaluation keeping in mind that it is a primary study in order to illustrate how new tools coming from uncertainty theories can help in the analysis of information. More details can be found in [8]. Table 1 provides the results obtained for the four considered variables. The global score has been computed by product between the informativeness and the calibration. Indicated are the ranks for each participant and each criterion.

Table 1. Evaluation of the results provided by each participant on the LOFT L2-5 benchmark (in the "participants" columns, the used code is given)

Participants	Prob. Approach			Poss. Approach		
	Inf.	Cal.	Global	Inf.	Cal.	Global
CEA (CATHARE)	8	5	6	8	6	7
GRS (ATHLET)	4	1	1	3	7	6
IRSN (CATHARE)	5	2	2	6	1	1
KAERI (MARS)	9	5	7	9	8	8
KINS (RELAP5)	3	5	5	7	3	3
NRI-1 (RELAP5)	7	2	3	5	5	4
NRI-2 (ATHLET)	6	8	8	4	2	2
PSI (TRACE)	1	10	10	1	10	10
UNIPI-1 (RELAP5)	10	2	4	10	4	5
UPC (RELAP5)	2	9	9	2	9	9

Several conclusions can be drawn from this table:

- User effect on results: from Table 1, it can be seen that users of a same code (KINS, NRI1, UPC, UNIPI-1 for RELAP5, CEA and IRSN for CATHARE, GRS and NRI2 for ATHLET) can receive very different scores. This indicates that the user expertise, as well as how uncertainties are modeled, play important roles in the quality of the final result.
- Uncertainty underestimation: both PSI and UPC have received very good informativeness scores with bad calibration scores. This indicates and confirms (from previous reports) that they probably both tend to underestimate the uncertainties in the input parameters.
- Uncertainty overestimation: in both approaches, UNIPI-1 has the lowest rank as for informativeness, while having a good calibration score. This indicates that, compared to other participants, its method has in this case a tendency to overestimate final uncertainties, while guaranteeing a good accuracy.

The four "most reliable" sources (i.e. the four institutes with the highest global score) identified from Table 1 are recalled in Table 2.

Table 2. The four "most reliable" sources on the LOFT benchmark identified by both methods.

Rank	Prob. approach	Poss. approach
1	GRS	IRSN
2	IRSN	NRI-2
3	NRI-1	KINS
4	UNIPI-1	NRI-1

<u>Third step:</u>

The different fusion operators defined previously are applied in the sequel to synthesize the information provided by each participant. Concerning the quenching time output, we did not integrate KAERI results since they do not obtain complete core quench in the upper bound case of the uncertainty band.
We first integrate the previous ranking (Table 2) in the analysis of the Zion results to check if the information provided by the "most reliable" sources of the LOFT benchmark is still in agreement for this new benchmark keeping in mind that the number of participants is not the same in Phase 3 (11 participants) and 5 (14 participants).
Figure 5 displays the resulting CDF after a probabilistic arithmetic mean fusion.

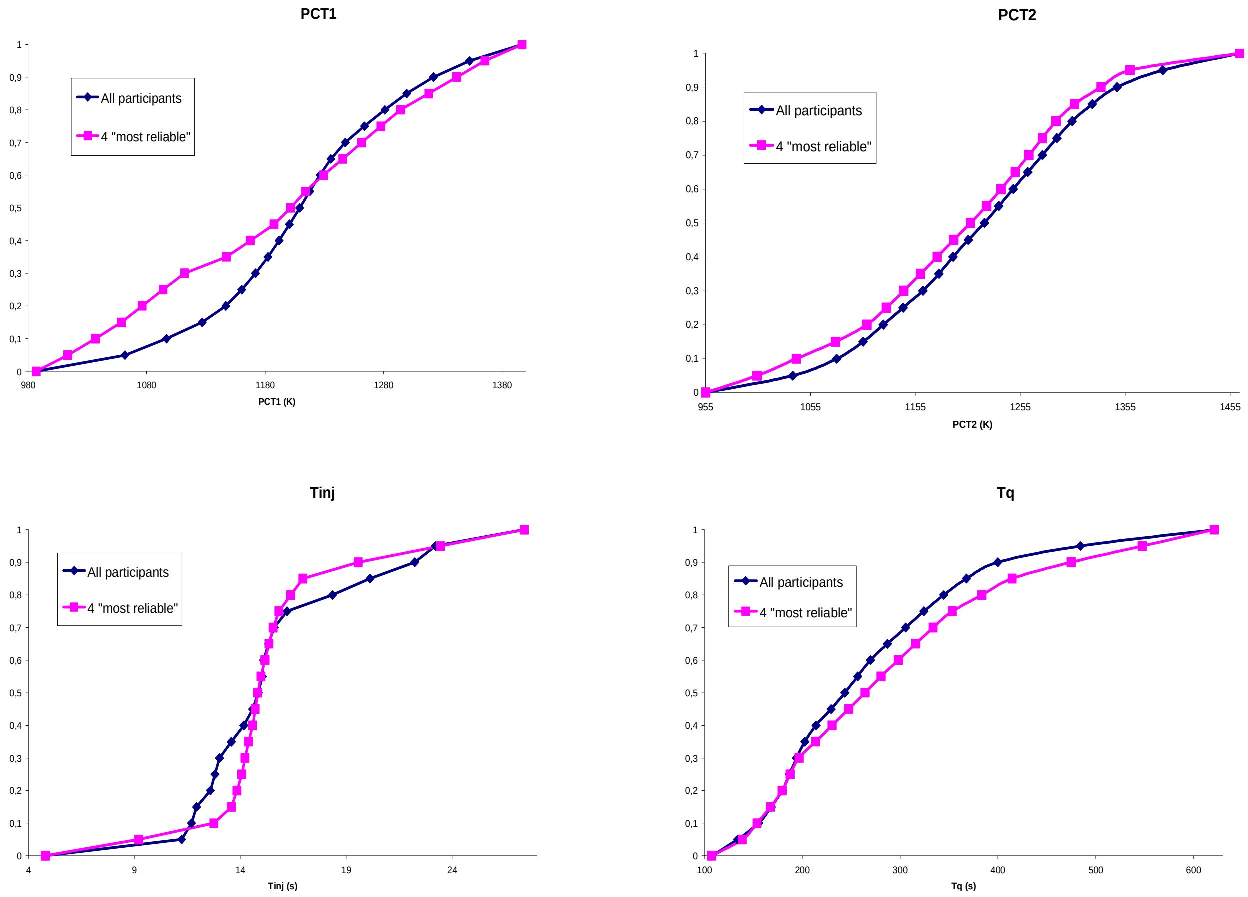

Figure 5: Fusion of the ZION results following the probabilistic approach (arithmetic mean, cumulative distribution function), in pink, provided by the four "most reliable" sources on the LOFT benchmark, in blue, all participants.

It appears that for the four outputs of interest, the probability distributions associated to the four "most reliable" sources of the LOFT benchmark is close to the distribution aggregated from all participants. It is due to the arithmetic mean that tends to average the resulting distribution. Note that, in the case of

Tinj, the two CDFs lead to narrow uncertainty margins. This could be expected since this output only depends on the depressurization parameter which has a narrow uncertainty margin at the beginning.

In order to exhibit a potential conflict between participants, it is easier to work in a possibilistic framework with a conjunctive fusion operator (Figure 6).

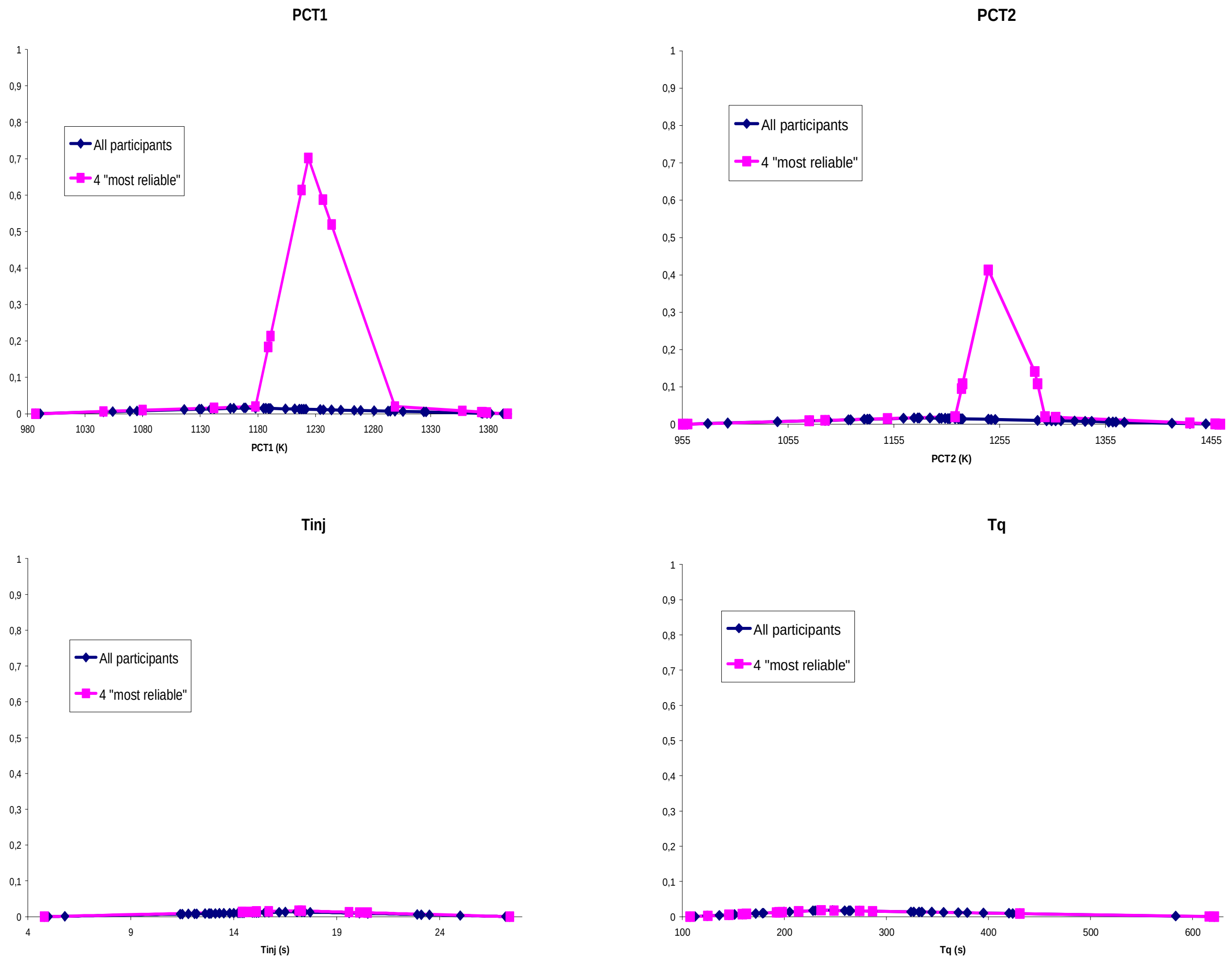

Figure 6: Fusion of the ZION results following the possibilistic approach (conjunction, possibillity distribution), in pink, provided by the four "most reliable" sources on the LOFT benchmark, in blue, all participants.

Several relevant results come out from this figure:

- The results provided by all the participants are highly conflicting for every output variable.
- Considering the four "most reliable" participants identified from the LOFT benchmark increases the coherence of the results for variables related to temperature (first and second peak cladding temperatures).
- Participants are more conflicting as for time variables. In this case, the only reliable synthesis is, according to us, the one based on the union of information provided by each participant (i.e. imprecise but reliable fusion) (Figure 7).

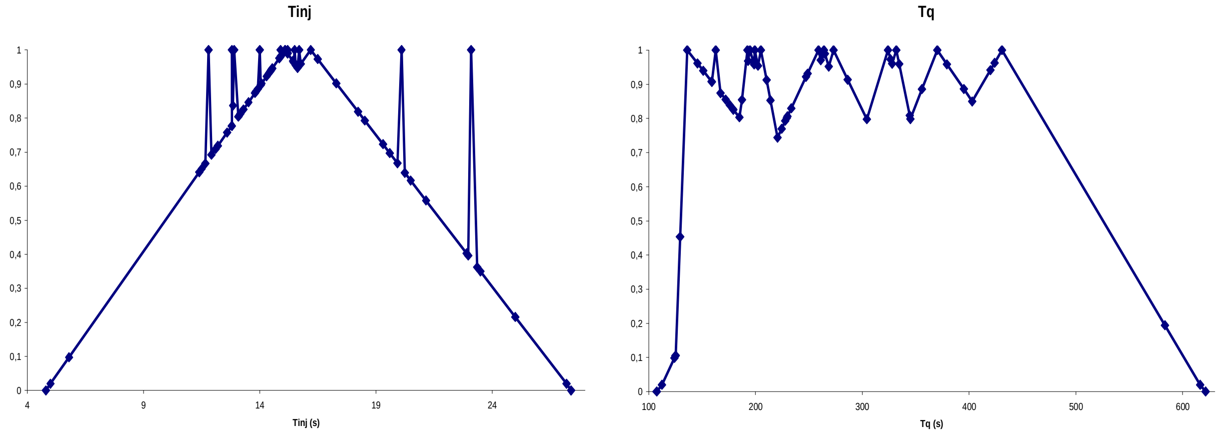

Figure 7: Fusion of the ZION results provided by all participants following the possibilistic approach (disjunction, possibillity distribution).

Note that for the accumulator injection time, the uncertainty margins are very narrow indicating that the uncertainties which have been taken into account don't impact this output variable.

Moreover, Figure 7 left also shows that half of the participants provided close reference calculations (AEKI, IRSN, KINS, NRI1, UNIPI1, UNIPI2 and UPC). These participants are expected to be in agreement in their analysis and their results can be combined to increase the coherence of the synthesis as for Tinj (see Figure 8).

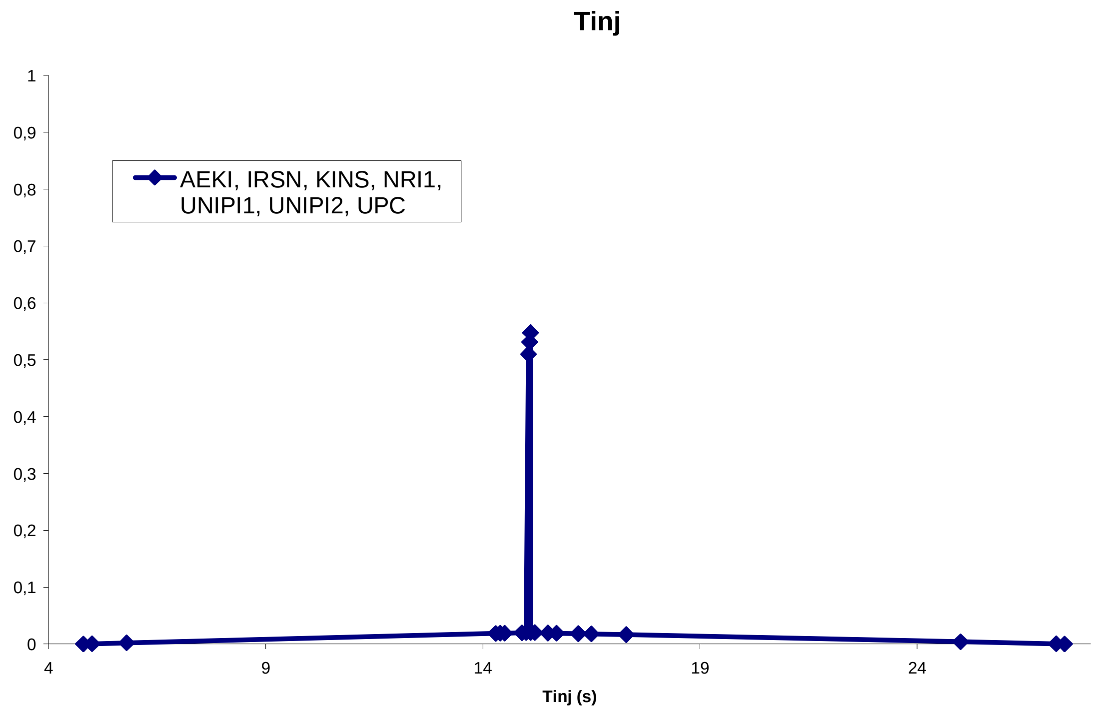

Figure 8: Fusion of the ZION results provided by the 7 participants with close reference calculation, following the possibilistic approach (conjunction, possibillity distribution).

- The four "most reliable" participants identified from the LOFT benchmark are usually less coherent for the Zion one. Moving from a small scale test facility to a real nuclear reactor is not straightforward in term of result prediction.

It is also interesting to study the influence of the used code on the derivation of uncertainty margins. This is done in the rest of this section (Figure 9). This comparison does not take into account the number of users that can differ from one code to another as well as the difference in modelling (e.g. 1D/3D modelling of the reactor vessel) that can be used in the various versions of a same code.

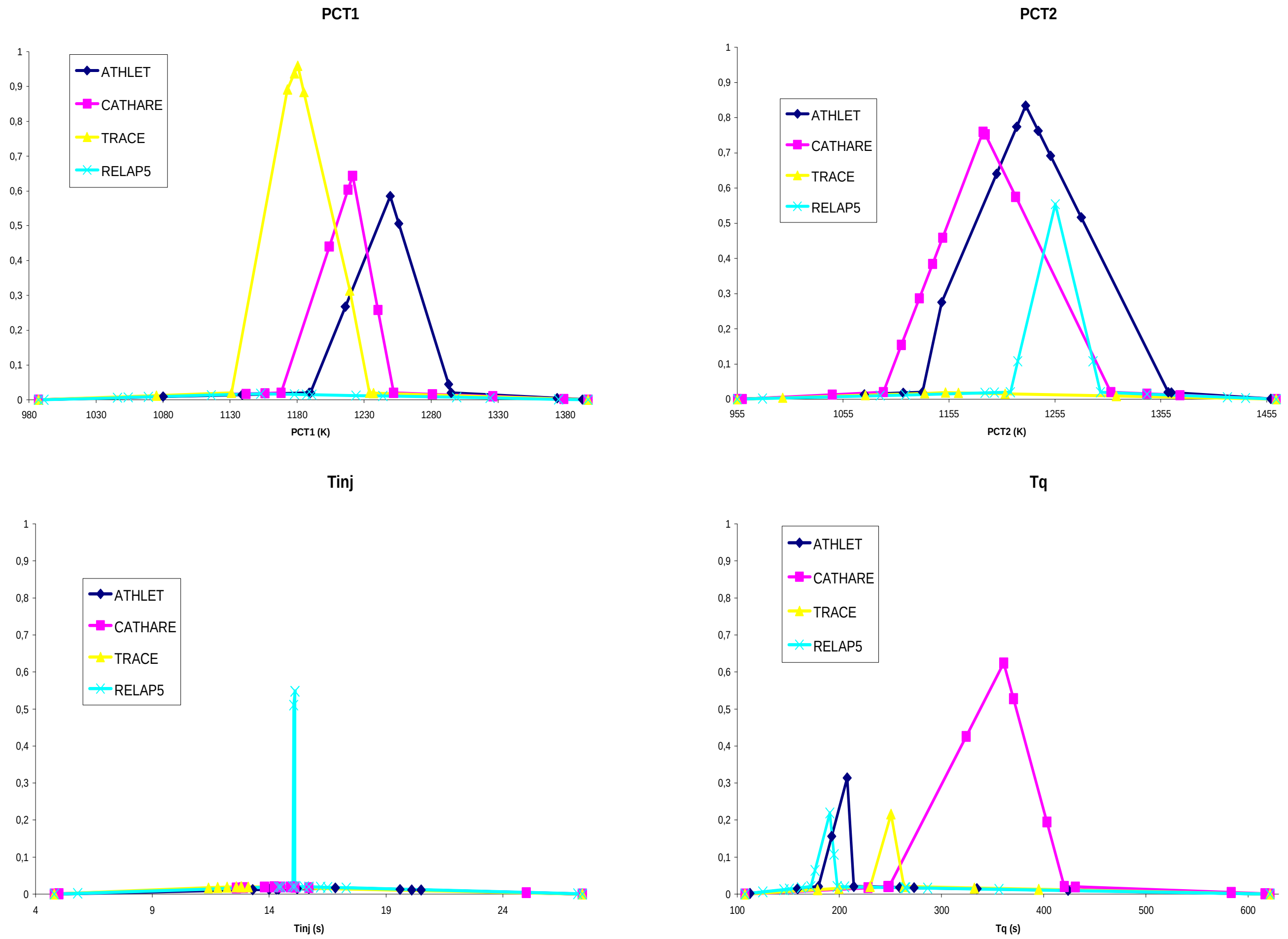

Figure 9: Fusion of the ZION results with respect to the used code and following possibilistic framework (conjunction, possibility distribution).

It appears that:

- Code effect is not negligible in the computation of reference calculation and in the derivation of uncertainty margins: For PCT1, TRACE users tend to provide reference calculations lower than the other code users (a difference of 40 K, resp. 70 K, compared to CATHARE, resp. ATHLET). Moreover, CATHARE users get the largest uncertainty band width for most of the variables of interest which leads to a large support for the resulting distribution of Figure 9.
- For a large majority of codes, users are coherent when estimating uncertainty related to the first and second peak cladding temperatures. Concerning time variables, only one code among four exhibits a low degree of conflict (RELAP5 for Tinj and CATHARE for Tq). This can be explained by the dispersion of reference calculations among users of a same code (for example as for Tinj, 14.9-12.9=2s for CATHARE IRSN and CEA users) and a narrow uncertainty band (1.2s). This arises again the question of reliability in the prediction of accumulator injection time and complete core quenching time.

4. Conclusion

Information synthesis is a key issue in nuclear safety studies integrating uncertainties since different uncertainty analyses using different codes and implying different experts are generally performed. In order to improve the reliability of the results, we described in this paper some recent developments performed by IRSN related to the construction of formal tools to synthesize uncertainty margins coming from multiple sources of information. They are based on probability or possibility theory. They allow one to quantify the quality (exactness and precision) of the provided information and to combine all the information sources in order to exhibit some common features or detect possible conflicts between them.

These tools were then applied in the frame of the BEMUSE Program for the Zion benchmark. It appeared that the information given by all participants is highly conflicting for the four outputs of interest (first and second peak cladding temperature, injection and quenching time). Considering sub-groups of participants has increased the coherence of the results related to temperature which is not the case for time variables. Each sub-group has been identified according to the quality of the given information or with respect to the used computer code. In this last case, one could emphasize the effect of the code, which is not negligible.

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OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations
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Supporting Qualified Database for Uncertainty Evaluation

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Abstract

Uncertainty evaluation constitutes a key feature of BEPU (Best Estimate Plus Uncertainty) process. The uncertainty can be the result of a Monte Carlo type analysis involving input uncertainty parameters or the outcome of a process involving the use of experimental data and connected code calculations. Those uncertainty methods are discussed in several papers and guidelines (IAEA-SRS-52, OECD/NEA BEMUSE reports).

The present paper aims at discussing the role and the depth of the analysis required for merging from one side suitable experimental data and on the other side qualified code calculation results. This aspect is mostly connected with the second approach for uncertainty mentioned above, but it can be used also in the framework of the first approach.

Namely, the paper discusses the features and structure of the database that includes the following kinds of documents:

1. The" RDS-facility" (Reference Data Set for the selected facility): this includes the description of the facility, the geometrical characterization of any component of the facility, the instrumentations, the data acquisition system, the evaluation of pressure losses, the physical properties of the material and the characterization of pumps, valves and heat losses;
2. The "RDS-test" (Reference Data Set for the selected test of the facility): this includes the description of the main phenomena investigated during the test, the configuration of the facility for the selected test (possible new evaluation of pressure and heat losses if needed) and the specific boundary and initial conditions;
3. The "QP" (Qualification Report) of the code calculation results: this includes the description of the nodalization developed following a set of homogeneous techniques, the achievement of the steady state conditions and the qualitative and quantitative analysis of the transient with the characterization of the Relevant Thermal-Hydraulics Aspects (RTA);
4. The EH (Engineering Handbook) of the input nodalization: this includes the rationale adopted for each part of the nodalization, the user choices, and the systematic derivation and justification of any value present in the code input respect to the values as indicated in the RDS-facility and in the RDS-test.

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1. Introduction

The present paper discusses the role and the depth of the analysis required for merging from one side suitable experimental data and on the other side qualified code calculation results. The availability of an experimental qualified database is of outmost importance for the validation and qualification of code calculation. Such database can be used to demonstrate that the code results are reliable[1], it can constitute the basis for an independent code assessment[2] and finally the basis for an uncertainty evaluation methodology[3, 4]. As discussed in several papers and guidelines an uncertainty methodology must rely on the availability of a qualified code and qualified procedures. The development of a documentation including the Reference Data Set (RDS) of the facility and of the tests, the Qualification Report (QR) of the code calculations and the Engineering Handbook (EH) constitutes an approach, envisaged by IAEA[5, 6], to set up a qualified experimental database.

In order to frame and to outline the role of a qualified database for performing a best estimate and uncertainty analysis, section 2 summarizes first the approach for performing the uncertainty analysis, whereas section 3 gives more details about the UMAE[1] (Uncertainty Methodology Based on Accuracy Extrapolation) process for qualifying code calculations and about the CIAU[3, 4] (Code with the capability of Internal Assessment of Uncertainty) method that needs a qualified set of experimental and code calculation results as input for performing a qualified uncertainty analysis. Section 4 presents the features the database shall have to be considered qualified and finally section 5 provides short conclusions.

2. Approaches for performing Uncertainty Analysis

The features of two independent approaches for estimating uncertainties are reviewed below. The propagation of code input errors (Fig. 1): this can be evaluated as being the most adopted procedure nowadays, endorsed by industry and regulators. It adopts the statistical combination of values from selected input uncertainty parameters (even though, in principle an unlimited number of input parameters can be used) to calculate the propagation of the errors throughout the code. To this approach belongs the so-called “GRS method”[7] and the majority of methods adopted by the nuclear industry. Although the entire set of the actual number of input parameters for a typical NPP (Nuclear Power Plant) input deck, ranging up to about 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 suitable PDF (Probability Density Function) are then assigned for each of the uncertain input parameter actually considered in the analysis. The number of computations needed for obtaining the desired confidence in the results can be determined theoretically by the Wilks formula[8]. Subsequently, the identified computations (ca. 100) are performed using the code under investigation to propagate the uncertainties inside the code, from inputs to outputs (results). The logical steps of the approach are depicted in Fig. 1.

The main drawbacks of such methods are connected with: (i) the need of engineering judgment for limiting (in any case) the number of the input uncertain parameters; (ii) the need of engineering judgment for fixing the range of variation and the PDF for each input uncertain parameter; (iii) 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 and (iv) the process of selecting the (about) 100 code runs is not convergent.

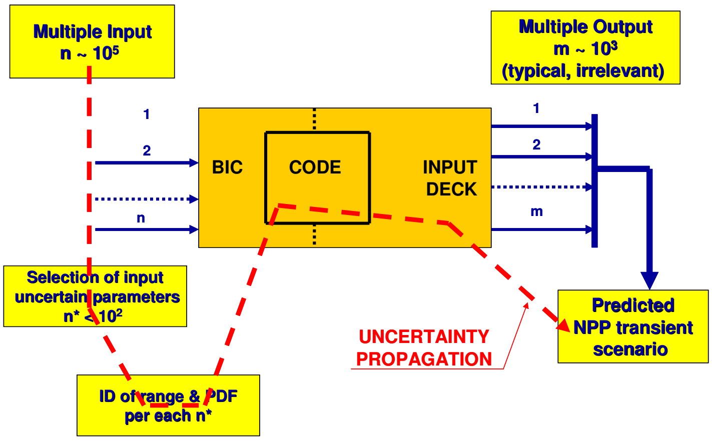

Figure 1. Uncertainty methods based upon propagation of input uncertainties (GRS method).

The second approach (Fig. 2), reviewed as the propagation of code output errors, is representatively illustrated by the UMAE-CIAU (Uncertainty Method based upon Accuracy Extrapolation[1] 'embedded' into the Code with capability of Internal Assessment of Uncertainty[2,3]). Note that this class of methods includes only a few applications from industry. The use of this method depends on the availability of 'relevant' experimental data, where here the word 'relevant' is connected with the specific NPP transient scenario under investigation for uncertainty evaluation. Assuming such availability of relevant data, which are typically Integral Test Facility (ITF) data, and assuming the code correctly simulates the experiments, it follows that the differences between code computations and the selected experimental data are due to errors. If these errors comply with a number of acceptability conditions[1], then the resulting (error) database is processed and the 'extrapolation' of the error takes place. Relevant conditions for the extrapolation are:

- Building up the NPP nodalization with the same criteria as was adopted for the ITF nodalizations;
- Performing a similarity analysis and demonstrating that NPP calculated data are "consistent" with the data measured in a qualified ITF experiment.

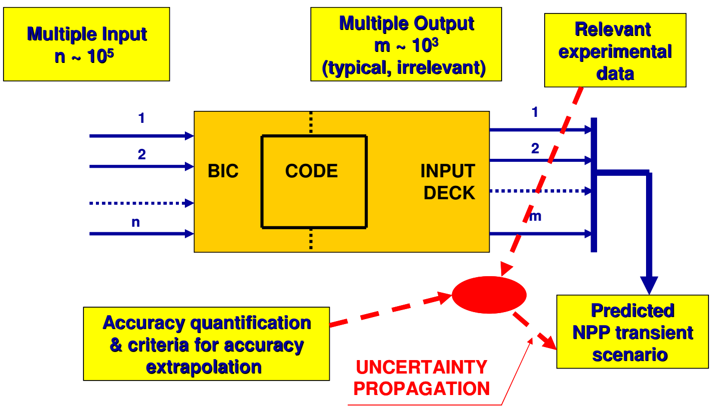

Figure 2. Uncertainty methods based upon propagation of output uncertainties (CIAU method).

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

The development and availability of a supporting qualified database for performing the uncertainty evaluation is mostly connected with the second approach (propagation of output uncertainties), but it is strictly recommended also in the framework of the first approach (propagation of input uncertainties) for demonstrating the quality level of the analysis.

3. UMAE-CIAU Method

3.1 The UMAE Methodology: the Engine of the CIAU Method

The UMAE[1], whose flow diagram is given in Fig. 3, is the prototype method for the description of "the propagation of code output errors" approach. The method focuses not on the evaluation of individual parameter uncertainties but on the propagation of errors from a suitable database calculating the final uncertainty by extrapolating the accuracy from relevant integral experiments to full scale NPP. Considering ITF of reference water cooled reactor, and qualified computer codes based on advanced models, the method relies on code capability, qualified by application to facilities of increasing scale. Direct data extrapolation from small scale experiments to reactor scale is difficult due to the imperfect scaling criteria adopted in the design of each scaled down facility. So, only the accuracy (i.e. the difference between measured and calculated quantities) is extrapolated.

Experimental and calculated data in differently scaled facilities are used to demonstrate that physical phenomena and code predictive capabilities of important phenomena do not change when increasing the dimensions of the facilities (see right loop FG in Fig. 6). Other basic assumptions are that phenomena and transient scenarios in larger scale facilities are close enough to plant conditions. The influence of user and nodalizations upon the output uncertainty is minimized in the methodology. However, user and nodalization inadequacies affect the comparison between measured and calculated trends; the error due to this is considered in the extrapolation process and gives a contribution to the overall uncertainty.

The method utilizes a database from similar tests and counterpart tests performed in ITF, that are representative of plant conditions. The quantification of code accuracy (step 'f' in Fig. 3) is carried out by using a procedure based on the Fast Fourier Transform Based Method (FFTBM[9]) characterizing the discrepancies between code calculations and experimental data in the frequency domain, and defining figures of merit for the accuracy of each calculation. Different requirements have to be fulfilled in order to extrapolate the accuracy. Calculations of both ITF experiments and NPP transients are used to attain uncertainty from accuracy. Nodalizations are set up and qualified against experimental data by an iterative procedure, requiring that a reasonable level of accuracy is satisfied. Similar criteria are adopted in developing plant nodalization and in performing plant transient calculations (see left loop FG in Fig. 6). The demonstration of the similarity of the phenomena exhibited in test facilities and in plant calculations, accounting for scaling laws considerations (step 'k' in Fig. 6), leads to the Analytical Simulation Model, i.e. a qualified nodalization of the NPP.

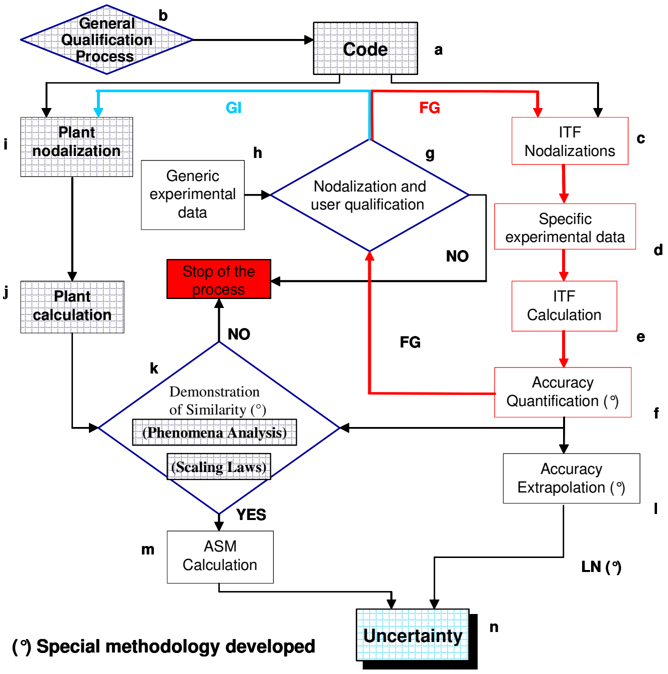

Figure 3. UMAE flow diagram (also adopted within the process of application of CIAU).

The qualified database, constituted by the RDS, QR and EH, is at the basis for a consistent application of the UMAE methodology and in particular for the steps characterizing the red loop FG that are pre-requisites for deriving the uncertainty by the CIAU. In fact, the availability of a RDS of the facility is mandatory for the step 'c' related with the development of the nodalization and the RDS of the selected test represents the specific experimental data at step 'd'. The QR demonstrates the fulfillment of the acceptability criteria at steps 'f' and 'g' of the code calculation for the selected test and thus the acceptance and the associated quality level of the input nodalization. The EH documents all the values of the qualified input (i.e. all steps in loop FG) in a systematic way.

3.2 The CIAU Method

The basic idea of the CIAU[2, 3] can be summarized in two parts:

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

A key feature of CIAU is the full reference to the experimental data (see loop FG in Fig. 3). Accuracy from the comparison between experimental and calculated data is extrapolated to obtain uncertainty. A solution to the issues constituted by the "scaling" and "the qualification" of the computational tools is embedded into the method[10, 11] 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.

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

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

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

An idea of the architecture of the CIAU methodology can be derived from Fig. 4. Two processes can be distinguished: the "Error Filling Process" (similar to path FG in Figure 3) 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) are picked up from the NPP statuses.

The qualified database constituted by the RDS, QR and EH is the supporting tool to the "Error Filling Process".

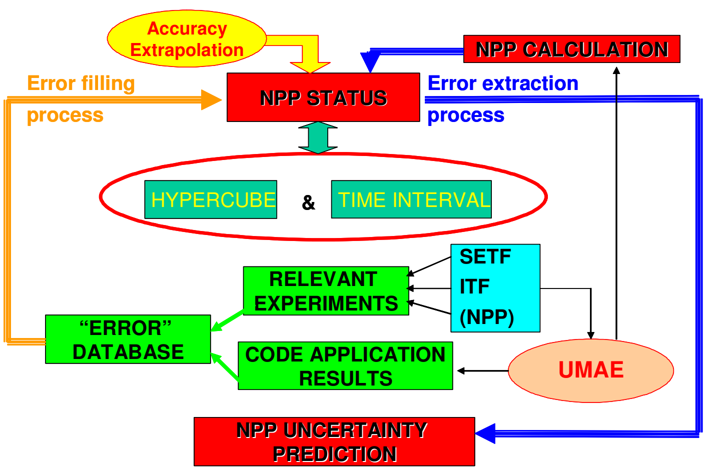

Figure 4. CIAU Method: "Error Filling Process" and "Error Extraction Process".

4. The Qualified Database for Uncertainty Evaluation

The experimental database for the nuclear technology is mainly constitutes by the experiments available through the OECD/CSNI Integral Test Facility (ITF)[12] and Separate Test Facility Effect (STF)[13, 14] matrix. Those database collect over thirty years of experiments: separate effects test for individual phenomena, integral test for large break LOCA, small break LOCA, transients, beyond design basis accidents and accident management in PWRs, BWRs and WWERs type of reactor. The enormous amount of information has been used for the code assessment in the framework of V&V activities. The availability of the experimental database constitutes also the pre-requisite for the creation of a qualified 'error' database of system thermal-hydraulic responses to be used for the uncertainty evaluation in the method based on "extrapolation of output errors".

The information contained in the experimental reports together with the code input nodalization are the sources to be elaborated in a systematic way by a qualified database made up of the following documents:
1) The Reference Data Set for the selected facility, RDS-facility;
2) The Reference Data Set for the selected test of the facility, RDS-test;
3) The Qualification Report, QR;
4) The Engineering Handbook, EH.

Figure 5. Figure 5 shows the link between the RDS, the Input deck, the QR and the EH. The black lines indicate the time sequence of the activities, the blue lines constitutes the feedback for review and the red lines are the necessary input to developed the input deck and the EH. The whole process is based on continuous review and exchange of information between the analysts involved in the activities. An independent review of each report is guarantee by the fact that the developer of the EH is different by the input deck developer, and the input deck developer is different from who work on the RDS.

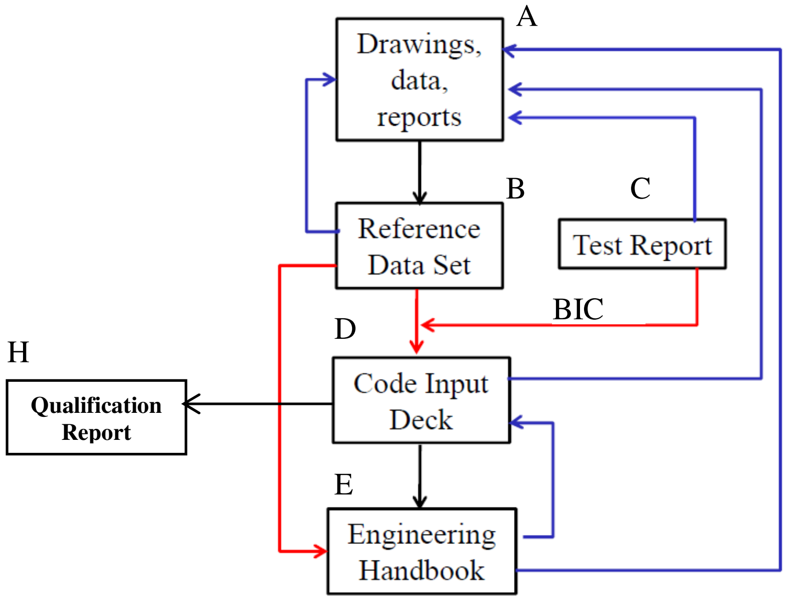

Figure 5. Flow chart of the RDS, Input Deck, QR and EH interconnections.

Block *A* in Fig 5 is related to the collection of relevant drawings and reports of the selected facility. This documentation constitutes the basis for writing down the RDS (block *B*). The writing of the RDS is also the first step of the review process, each documents is checked against is consistency with other source of information till the creation of a final documentation set for the particular facility.

The subsequent block of the chart (block *C*) is related to the creation of a RDS for the selected test that has to be analyzed. The RDS of the test will contain the definition of test conditions, the set points and the boundary conditions. The RDS of the facility and of the test are at the basis of the code input development. The development of an input (block *D*) must follow a preconfigured set of nodalization strategies described in a dedicated document whose goal (not described here) is to collect the nodalization strategies, user choices and model selections to be used for the development of any other input. A review process of the RDS took place in this phase: the input developer uses the RDS to extract the necessary information for the input preparation together with the availability of the original documentation already collected. Potential errors and misinterpretations maybe identified and corrected in the RDS. The writing down of the RDS-test also constitutes a review process of the RDS-facility, in deed the RDS-facility is continuously used to check the consistency of the two reports.

One of the reasons for the need of the RDS is connected with the duration of the experimental campaign that usually takes place in each facility (more or less from five to ten years). During those years, different modifications can be made to the facility configuration in order to improve the fidelity of the facility to the reference plant, to reduce the heat losses, to install a more detailed instrumentation apparatus, etc... Such information and modifications are obviously not part of the original documentation and in general could be only partially reflected in separated reports and documents. Thus, the goal of the RDS is to analyze the enormous amount of available documentation and to solve possible contradictions coming out from different reports in order to produce a consistent and homogenous set of data of the facility.

Once the code input file has been produced, there is the need to qualify the achieved code results following appropriate procedures as discussed in the UMAE methodology. The QR (block *H*) collects the results of the qualification procedures of the code input.

The engineering handbook (block *E*) constitutes the final step for the set up of a qualified database useful for the uncertainty methodology. The IAEA report[6] states that a *"document contains a full description of how the database has been converted into an input data deck for a specific computer code"* should be available. This is the goal of the EH: it not only describes the nodalization of the facility based on the input file and the calculation notes made available by the input developer but it also provides the engineering justifications of the user choices and the explanation of possible discrepancies with the general nodalization strategies.

At this step, a final review process of the three set of documents is also performed: any entry in the input deck is checked between the calculation notes and the RDS of the facility, errors or inconsistencies found in the input are tracked and reported and appropriate countermeasures are taken. For the criterion of the independence of the review process, it is of outmost importance that the engineer in charge of the EH is different from the input deck developer (the last one is involved in the preparation of the EH only for the description of the "nodalization rationale" and for the "user choices").

4.1 The RDS-facility and the RDS-test

The first step for the database creation is constituted by the collection of the relevant experimental information in a document called Reference Data Set handbook. The relevant design data of the facility and of the test are organized in order to be ready to be used for the development of the code input data file. The data and the organization of the data are not dependent on the code selected for the analysis.

To perform a proper data collection, the IAEA suggest to:

- Check the quality of input data,
- Resolve the contradiction coming out from the data,
- Explain information on geometry, thermal and hydraulic properties,
- Perform an independent review,
- Carry-out a quality control of the database by means of relevant quality assurance procedures,
- Developed a database in a code independent form.

Two kinds of RDS are necessary to the creation of an input deck: the RDS of the facility (RDS-facility) and the RDS of each test (RDS-test) that has to be included in the database. Figure 6 shows the relationship between those documents.

The RDS related with the design of a facility may consists of the following sections[15]:

- Layout of the facility,
- Collection of geometrical data (length, volumes, areas, elevations) for each subsystem and component of the facility,
- Collection of specific data for complex component (pumps, valves, heaters, etc…),
- Identification of geometrical discontinuities and evaluation of pressure loss coefficients (normal operation),
- Material properties,
- Measurement system,
- Nominal heat losses.

The RDS of a particular test in a facility may consists of the following sections[16]:

- Description of the test and phenomena occurring in the experiment,
- Characterization of the Boundary and Initial Conditions (BIC),
- Characterization of trips and logic signals occurring during the transient,
- Measurement data,
- Specific heat losses,
- Evaluation of possible additional pressure loss coefficients,
- Thermal hydraulic system behaviour description.

4.2 The Input Deck

A preliminary consideration to set up a nodalization should first addresses the kind of facility and the related type of problem to be investigated. It should be noted that in the present context the word "facility" is here intended both as a real plant and as a test rig. The level of detail of both full or reduced scale facilities should be comparable even though, in case of an experimental rig, some auxiliary or ancillary

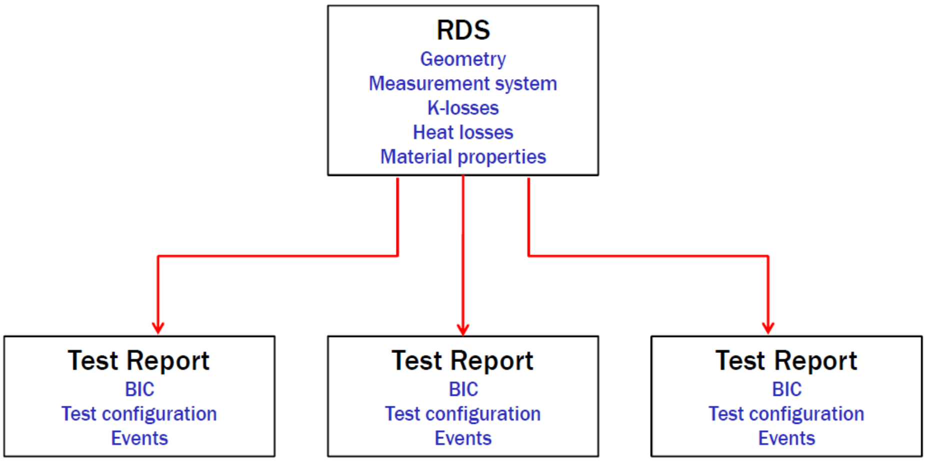

Figure 6. The RDS-facility and the RDS-Tests.

systems are simplified or even not reproduced, rather in case of a real plant "side" systems consumes code resources.

The set up of a nodalization passes through different steps:

- Nodalization preparation: main choices of the model characteristics and preliminary code resources distribution;
- Nodalization schematization: build up of the discretization of the various parts of the considered facility;
- Input writing: translation into the code language of the schematization.

The input data file is developed starting from the RDS-facility and RDS-test. Together with the data file, calculation notes to document modelling decisions have to be elaborated by the code input developer. Input data file and calculation notes pass through a peer review process (see Figure 5 and Figure 8) before the final version of the input is issued.

The process of nodalization preparation includes the analysis of the facility lay-out, its main geometrical characteristics and its working modalities. In addition the analyst shall have in mind which phenomena are expected to occur in the considered facility (due to geometrical specificities and/or constraints) and in the specific (or set of) transient. At this regard, the OECD code validation matrixes[12, 13, 14], constitute an irreplaceable tool in which most of the ITF and SETF operated in the world are considered.

An assessment on code resources distribution shall be made by the analyst prior to set up the related model, to ensure that sufficient level of details (e.g. number of active channels, number of equivalent U-tubes, etc.) is put in relevant part of the plant without neglecting or making too simple the rest of the plant. Definition of sufficient level of detail shall take into account the objectives of the analysis. As an example the detailed simulation of a RPV could be enough in case of very short term core analysis, but it is completely insufficient in case of global plant reproduction.

Outcome of this initial stage of the code input development is a draft of the nodalization sketch with the indication of the main features of the nodalization (e.g. number of active channels, number of equivalent U-tubes, etc.).

The nodalization development is the logical subsequent step of the nodalization preparation during which the basic idea of the model are fully developed up to design a complete nodalization. Feedbacks on the previous stage could come out during the development of the nodalization, especially when the code user starts to have a more complete picture of the model. Iterative paths could be requested if basic changes influence relevant or wide zones of the facility. The availability of the report containing the guidelines for the development of the nodalization and the main strategies to be adopted (block F in Figure 8) is a fundamental supporting tool for the code input developers.

The writing of the input deck consists in the translation of the developed nodalization into the code language. In principal it needs 'just' the knowledge of the code's syntax and the use of the related manual. Attention shall be put in order to avoid typing error (very difficult to be detected in case of plausible value from the code internal check point of view). In addition a rational in the deck structure shall be followed: readability of the input deck may suggest to clearly identifying the main sections which include trips; hydrodynamic data; heat structures data; 0DNK cards; control variables; additional requested variable (if any).

4.3 The Qualification Report

The Qualification Report (QR) is necessary to demonstrate that the code results are qualitative and quantitative acceptable with respect to fixed acceptance criteria[3]. In particular for the method based on extrapolation of output error, the availability of QR is a pre-requisite to demonstrate the qualification level of the 'error' database.

Without going into the details of the qualification procedures[3] (illustrated in Figure 7), it is however important to outline the minimum amount of information shall be contained in a qualification report:

- The demonstration of the geometrical fidelity (block *e* in Figure 7) of the model respect to the facility. A non exhaustive list of the characteristic to be verified is given here: primary and secondary volume, active and non-active structure heat exchange area, volume versus height curves, component relative elevation, flow area of specific components, etc…;
- The qualification at steady-state level (block *d* in Figure 7), i.e. the demonstration of the capability of the model to reproduce the steady-state qualified condition of the test;
- The qualification at transient level (block *h* in Figure 7). 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.

When the acceptability criteria at blocks 'f' and 'j' are met the input deck is considered qualified.

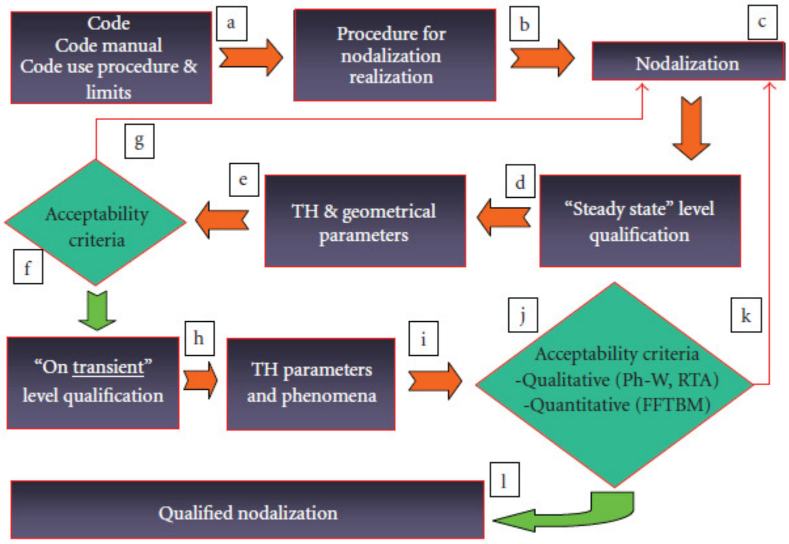

Figure 7. Flow chart of the nodalization qualification procedure.

4.4 The Engineering Handbook

The creation of a plant model for computer codes is an interactive procedure that includes the selection of the nodalization scheme, the preparation of the code input deck and the associated documentation of the activities.

The engineering handbook constitutes the technical rationale for the input model. It summarizes for each component of the model's input file the documentation used and provides engineering justification of the adopted assumptions. Each input's entry is fully described and documented and the calculation notes of the input developer are included in the EH allowing an easier understand of the input values. It finally makes a cross link between the RDS (both for the facility and for the test), the code and the input data files.

As the EH documents in detail how the RDS have been converted into an input deck for the particular computer code, one can say that the final goal of the EH is to preserve and make easier the transfer of knowledge about the input (i.e. the user choices and the technical rationale behind the development of the input) from one group to an other one.

A typical EH shall contain:
(a) The methods and the assumptions used to convert the RDS information into the code input data;
(b) All calculations made to convert the technical plant data to the necessary format for the input deck (i.e. the traceability of the information);
(c) The nodalization schemes of the components as well as for the complete system being modeled;
(d) The adequate description and explanation of all adopted modeling assumptions.

The EH is set up once the input deck has been qualified and frozen, any changes to the input deck, prior peer review approval, must be documented in the EH.

In the process described in the present paper, the writing down of the EH constitutes the final step of the review activities for the whole process (from the blue prints to the input deck and the EH). The availability of the calculation notes (block *G* in Figure 8), prepared by the input developer, allows to check possible inconsistencies in the input deck respect to the RDS data and any revealed discrepancy is carefully reviewed and solved through the proper corrective action that has to be documented.

The structure proposed for the EH is similar to the organization of a typical (RELAP5) code input file in order to result as much as possible user friendly. A non-exhaustive list for the content of the EH is give here below:

- Introduction;
- Nodalization description:
    - Primary side, with subsections related to relevant primary circuit zones, i.e. hot leg, cold leg, etc... Each zone has a description of the rationale, user choices, models, geometry, junctions and heat structures entries;
    - Secondary side, with subsections related to relevant secondary circuit zones, i.e. steam generator downcomer, steam generator riser, separator, etc... Each zone has a description of the rationale, user choices, models, geometry, junction and heat structures entries;
    - Primary and Secondary Side Boundary Conditions;
    - Primary and Secondary Side heat Losses;
- General table:
    - Core power;
    - Heat Losses;
- Material Property;
- Logical and Control System:
    - Control variables;
    - Artificial controllers;
    - Logical and variables trips.

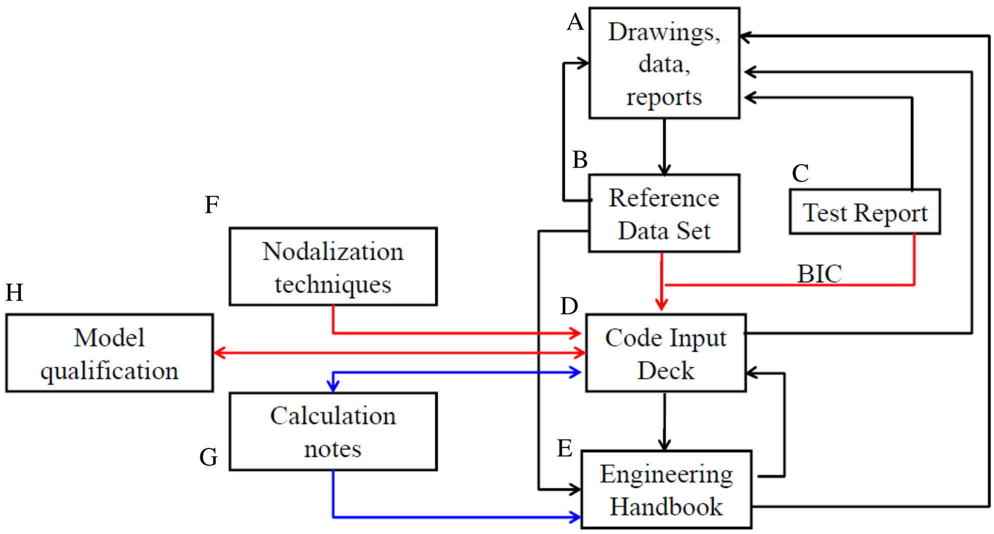

Figure 8. Flow Chart of the Input Development, Review and Qualification Procedures.

5. Conclusions

The paper presents the features and structure of the database that includes a series of documents whose goal is to demonstrate the qualification level of the achieved code results. This aspect is mostly connected with a Best Estimates Plus Uncertainty analysis based on “propagation of code output errors” approach. Such qualified database is a fundamental pillar to justify the qualification level of the error database, but its availability should be also a mandatory requirement for any analysis based on best estimate approach. By-product of this activity is the concrete possibility to make traceable any input’s value and any user choice respectively derived and taken from the blue prints to the final achieved code results.

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A PROCEDURE FOR CHARACTERIZING THE RANGE OF INPUT UNCERTAINTY PARAMETERS BY THE USE OF THE FFTBM

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Abstract

In the last years various methodologies were proposed to evaluate the uncertainty of Best Estimate (BE) code predictions. The most used method at the industrial level is based upon the selection of input uncertain parameters, on assigning related ranges of variations and Probability Distribution Functions (PDFs) and on performing a suitable number of code runs to get the combined effect of the variations on the results.

A procedure to characterize the variation ranges of the input uncertain parameters is proposed in the paper in place of the usual approach based (mostly) on engineering judgment. The procedure is based on the use of the Fast Fourier Transform Based Method (FFTBM), already part of the Uncertainty Method based on the Accuracy Extrapolation (UMAE) method and extensively used in several international frameworks.

The FFTBM has been originally developed to answer questions like "How long improvements should be added to the system thermal-hydraulic code model? How much simplifications can be introduced and how to conduct an objective comparison?". The method, easy to understand, convenient to use and user independent, clearly indicates when simulation needs to be improved.

The procedure developed for characterizing the range of input uncertainty parameters involves the following main aspects:

a) One single input parameter shall not be 'responsible' for the entire error |exp-calc|, unless exceptional situations to be evaluated case by case;
b) Initial guess for Max and Min for variation ranges to be based on the usual (adopted) expertise;
c) More than one experiment can be used per each NPP and each scenario. Highly influential parameters are expected to be the same. The bounding ranges should be considered for the NPP uncertainty analysis;
d) A data base of suitable uncertainty input parameters can be created per each NPP and each transient scenario.

1. Introduction

Several approaches have been proposed to quantify the accuracy of a given code calculation[1, 2, 3, 4]. Even though these methods were able to give some information about the accuracy, they were not considered satisfactory because they involved some empiricism and were lacking of a precise mathematical meaning. Besides, engineering subjective judgment at various levels is deeply inside in proposed methods.

Generally, the starting point of each method is an error function, by means of which the accuracy is evaluated. Some requirements were fixed which an objective error function should satisfy:
- at any time of the transient this function should remember the previous history;
- engineering judgment should be avoided or reduced;
- the mathematical formulation should be simple;
- the function should be non-dimensional;
- it should be independent upon the transient duration;
- compensating errors should be taken into account (or pointed out);
- its values should be normalized.

The Fast Fourier Transform Based Method (FFTBM) has been developed taking into account the above requirements and to answer questions like “How long improvements should be added to the system thermal-hydraulic code model? How much simplifications can be introduced and how to conduct an objective comparison?”. The method, easy to understand, convenient to use and user independent, clearly indicates when simulation needs to be improved.

The objective of the paper is to propose a procedure to characterize the ranges of variation of the input uncertain parameters in place of the usual approach based (mostly) on engineering judgment for the uncertainty methods based on the propagation of input error. The proposed procedure is based on an alternative way to use the FFTBM respect to its common application inside the UMAE approach.

In order to give a full picture about the framework where the FFTBM is used, in section 2 a short description about the approaches to perform uncertainty analysis is given. More details about the methods that need the FFTBM and the existing interconnections between these tools is provided in section 3, whereas section 4 contains the description of the FFTBM. Section 5 presents the use of the FFTBM as a tool for characterizing the ranges of variation of the input uncertain parameters.

2. Approaches for performing Uncertainty Analysis

The features of two independent approaches for estimating uncertainties are reviewed below. The propagation of code input errors (Fig. 1): this can be evaluated as being the most adopted procedure nowadays, endorsed by industry and regulators. It adopts the statistical combination of values from selected input uncertainty parameters (even though, in principle an unlimited number of input parameters can be used) to calculate the propagation of the errors throughout the code. To this approach belongs the so-called “GRS method”[5] and the majority of methods adopted by the nuclear industry. Although the entire set of the actual number of input parameters for a typical NPP (Nuclear Power Plant) input deck, ranging up to about 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 suitable PDF (Probability Density Function) are then assigned for each of the uncertain input parameter actually considered in the analysis. The number of computations needed for obtaining the desired confidence in the results can be determined theoretically by the Wilks formula[6]. Subsequently, the identified computations (ca. 100) are performed using the code under investigation to propagate the uncertainties inside the code, from inputs to outputs (results). The logical steps of the approach are depicted in Fig. 1.

The main drawbacks of such methods are connected with: (i) the need of engineering judgment for limiting (in any case) the number of the input uncertain parameters; (ii) the need of engineering judgment for fixing the range of variation and the PDF for each input uncertain parameter; (iii) 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 and (iv) the process of selecting the (about) 100 code runs is not convergent.

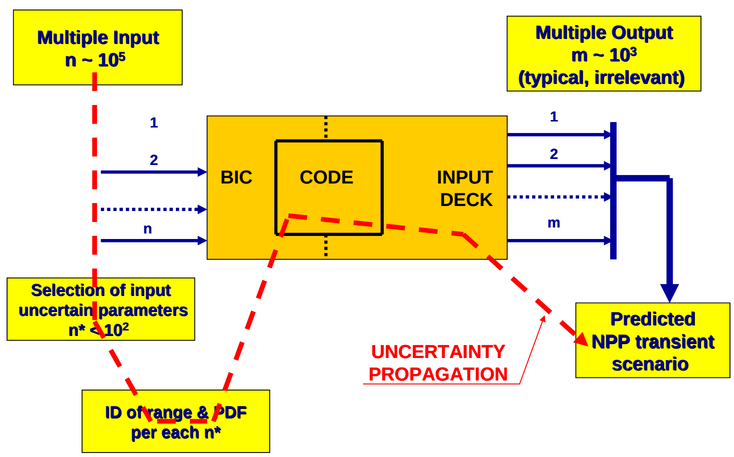

Figure 1. Uncertainty methods based upon propagation of input uncertainties (GRS method).

The second approach (Fig. 2), reviewed as the propagation of code output errors, is representatively illustrated by the UMAE-CIAU (Uncertainty Method based upon Accuracy Extrapolation[7] 'embedded' into the Code with capability of Internal Assessment of Uncertainty[8,9]). Note that this class of methods includes only a few applications from industry. The use of this method depends on the availability of 'relevant' experimental data, where here the word 'relevant' is connected with the specific NPP transient scenario under investigation for uncertainty evaluation. Assuming such availability of relevant data, which are typically Integral Test Facility (ITF) data, and assuming the code correctly simulates the experiments, it follows that the differences between code computations and the selected experimental data are due to errors. If these errors comply with a number of acceptability conditions[7], then the resulting (error) database is processed and the 'extrapolation' of the error takes place. Relevant conditions for the extrapolation are:

- Building up the NPP nodalization with the same criteria as was adopted for the ITF nodalizations;
- Performing a similarity analysis and demonstrating that NPP calculated data are "consistent" with the data measured in a qualified ITF experiment.

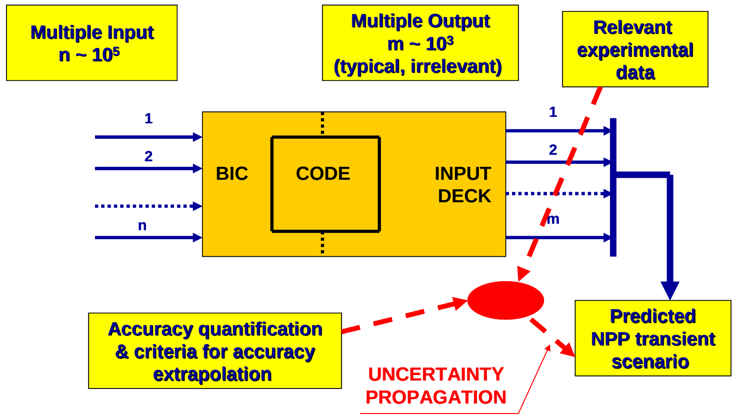

Figure 2. Uncertainty methods based upon propagation of output uncertainties (CIAU method).

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

The FFTBM procedure here proposed to characterize the ranges of variation of the input uncertain parameters is mostly connected with the second approach (i.e. the propagation of output uncertainties).

3. UMAE-CIAU Method

3.1 The UMAE Methodology: the Engine of the CIAU Method

The UMAE[7], whose flow diagram is given in Fig. 3, is the prototype method for the description of “the propagation of code output errors” approach. The method focuses not on the evaluation of individual parameter uncertainties but on the propagation of errors from a suitable database calculating the final uncertainty by extrapolating the accuracy from relevant integral experiments to full scale NPP. Considering ITF of reference water cooled reactor, and qualified computer codes based on advanced models, the method relies on code capability, qualified by application to facilities of increasing scale. Direct data extrapolation from small scale experiments to reactor scale is difficult due to the imperfect scaling criteria adopted in the design of each scaled down facility. So, only the accuracy (i.e. the difference between measured and calculated quantities) is extrapolated.

Experimental and calculated data in differently scaled facilities are used to demonstrate that physical phenomena and code predictive capabilities of important phenomena do not change when increasing the dimensions of the facilities (see right loop FG in Fig. 6). Other basic assumptions are that phenomena and transient scenarios in larger scale facilities are close enough to plant conditions. The influence of user and nodalizations upon the output uncertainty is minimized in the methodology. However, user and nodalization inadequacies affect the comparison between measured and calculated trends; the error due to this is considered in the extrapolation process and gives a contribution to the overall uncertainty.

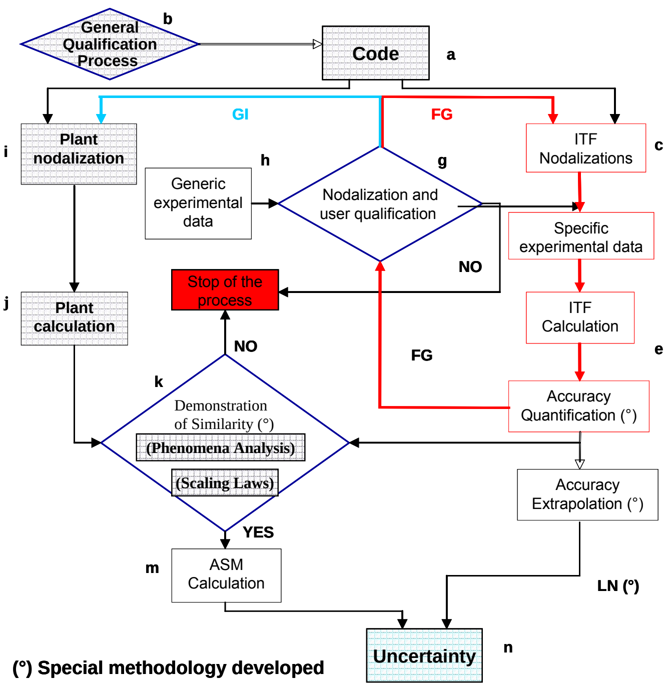

Figure 3. UMAE flow diagram (also adopted within the process of application of CIAU).

The method utilizes a database from similar tests and counterpart tests performed in ITF, that are representative of plant conditions. The quantification of code accuracy (step ‘f’ in Fig. 3) is carried out by using a procedure based on the Fast Fourier Transform Based Method (FFTBM[10]) characterizing the discrepancies between code calculations and experimental data in the frequency domain, and defining figures of merit for the accuracy of each calculation. Different requirements have to be fulfilled in order to extrapolate the accuracy. Calculations of both ITF experiments and NPP transients are used to attain uncertainty from accuracy. Nodalizations are set up and qualified against experimental data by an iterative procedure, requiring that a reasonable level of accuracy is satisfied. Similar criteria are adopted in developing plant nodalization and in performing plant transient calculations (see left loop FG in Fig. 6). The demonstration of the similarity of the phenomena exhibited in test facilities and in plant calculations, accounting for scaling laws considerations (step ‘k’ in Fig. 6), leads to the Analytical Simulation Model (ASM), i.e. a qualified nodalization of the NPP.

3.2 The CIAU Method

The basic idea of the CIAU[8, 9] can be summarized in two parts:

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

A key feature of CIAU is the full reference to the experimental data (see loop FG in Fig. 3). Accuracy from the comparison between experimental and calculated data is extrapolated to obtain uncertainty. A solution to the issues constituted by the "scaling" and "the qualification" of the computational tools is embedded into the method[11, 12] 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.

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

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

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

An idea of the architecture of the CIAU methodology can be derived from Fig. 4. Two processes can be distinguished: the "Error Filling Process" (similar to path FG in Figure 3) 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) are picked up from the NPP statuses.

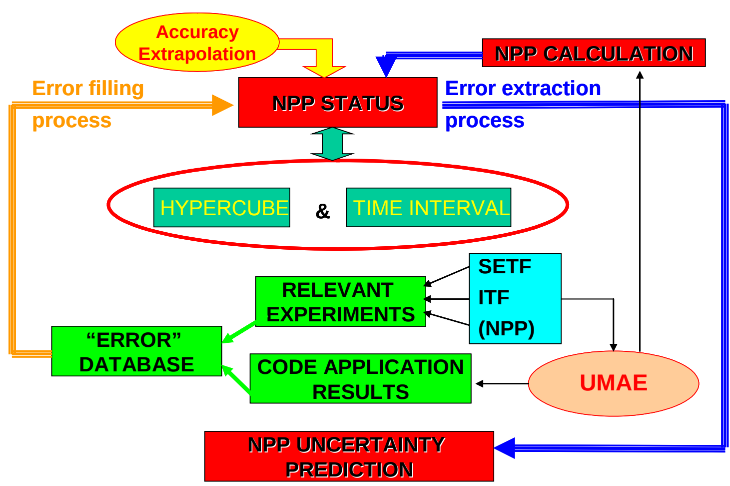

Figure 4. CIAU Method: "Error Filling Process" and "Error Extraction Process".

4. The FFTBM Tool

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

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

The information contained in this time dependent function, continuously varying, should be condensed to give a limited number of values which could be taken as indexes for quantifying accuracy. This is allowed because the complete set of instantaneous values of DF(t) is not necessary to draw an overall judgment about accuracy.

Integral approaches satisfy this requirement, since they produce a single value on the basis of the instantaneous trend of a given function of time. On the other hand, searching for functions expressing all the information through a single value, some interesting details could be lost. Therefore, it would be preferable to define methodologies leading to more than one value in order to characterize the code calculation accuracy.

Information that comes from the time trend of a certain parameter, be it a physical or a derivate one, may be not sufficient for a deep comprehension of a concerned phenomenon; in such a case, it may be useful to study the same phenomenon from other points of view, free of its time dependence. In this context, the complete behaviour of a system in periodic regime conditions (periodic conditions due to instability phenomena are explicitly excluded), can be shown by the harmonic response function that describes it in the frequency domain.

Furthermore, the harmonic analysis of a phenomenon can point out the presence of perturbations otherwise hidden in the time domain.

4.1 The FFT algorithm

It is well known that the Fourier transform is essentially a powerful problem solving technique. Its importance is based on the fundamental property that one can analyze any relationship from a completely different viewpoint, with no lack of information with respect to the original one. The Fourier transform can translate a given time function g(t), in a corresponding complex function defined, in the frequency domain, by the relationship:

$$\tilde{g}(f) = \int_{-\infty}^{+\infty} g(t) \cdot e^{-j \cdot 2\pi \cdot f \cdot t} dt \tag{2}$$

Afterwards, it is assumed that the experimental and calculated trends, to which the Fourier transform is applied, verify the analytical conditions required by its application theory; i.e., it is assumed that they are continuous (or generally continuous)(1) in the considered time intervals with their first derivatives, and

(1) : i.e. discontinuous only in a finite number of points. The existence of the Fourier Transform is guaranteed if g(t) is summable according to Lebesgue on the real axis.

absolutely integrable in the interval[13] (-∞, +∞) [(2)]. This last requirement can be easily satisfied in our case, since the addressed functions assume values different from zero only in the interval (0, T). Therefore:

$$\tilde{g}(f) = \int_0^T g(t) \cdot e^{-j \cdot 2\pi \cdot f \cdot t} dt \tag{3}$$

The Fourier integral is not suitable for machine computation, because an infinity of samples of g(t) is required. Thus, it is necessary to truncate the sampled function g(t) so that only a finite number of points are considered, or in other words, the discrete Fourier transform is evaluated. Truncation introduces a modification of the original Fourier transform (the Fourier transform of the truncated g(t) has a rippling); this effect can be reduced choosing the length of the truncation function as long as possible.

When using functions sampled in digital form, the Fast Fourier Transform (FFT) can be used. The FFT is an algorithm that can compute more rapidly the discrete Fourier transform. To apply the FFT algorithm, functions must be identified in digital form by a number of values which is a power of 2. Thus, if the number of points defining the function in the time domain is:

$$N = 2^{m+1} \tag{4}$$

the algorithm gives the transformed function defined in the frequency domain by 2^m+1 values corresponding to the frequencies f_n, in which T is the time duration of the sampled signal:

$$f_n = \frac{n}{T} \qquad (n = 0, 1, \ldots, 2^m) \tag{5}$$

Taking into account the fact that the adopted subroutine packages evaluate the FFT normalized to the time duration T, from Eqs. (3) and (5) it can be seen that $|\tilde{g}(0)|$ represents the mean value of the function g(t) in the interval (0,T), while $|\tilde{g}(f_i)|$ represents the amplitude of the *i*-th term of the Fourier polynomial expansion for the function g(t). Generally, the Fourier transform is a complex quantity described by the following relationship:

$$\tilde{g}(f) = \mathrm{Re}(f) + j \cdot \mathrm{Im}(f) = |\tilde{g}(f)| \cdot e^{j\theta(f)} \tag{6}$$

where:

- Re(f) is the real component of the Fourier transform
- Im(f) is the imaginary component of the Fourier transform
- $|\tilde{g}(f)|$ is the amplitude or Fourier spectrum of g(t)
- θ(f) is the phase angle or phase spectrum of Fourier transform.

It is well known that:

[(2)] : i.e. $\int_{-\infty}^{+\infty} |g(t)| dt < \infty$

$$\left|\tilde{g}(f)\right| = \sqrt{(\mathrm{Re}(f))^2 + (\mathrm{Im}(f))^2} \tag{7}$$

$$\theta(\mathrm{f}) = \mathrm{tg}^{-1}\,\frac{\mathrm{Im}(\mathrm{f})}{\mathrm{Re}(\mathrm{f})}$$

4.2 The FFTBM algorithm

The method developed to quantify the accuracy of code calculations is based on the amplitude of the FFT of the experimental signal and of the difference between this one and the calculated trend. In particular, with reference to the error function ΔF(t), defined by the Eq. (1), the method defines two values characterizing each calculation:

1. The ***Dimensionless Average Amplitude, AA:***

$$\mathrm{AA} = \frac{\sum_{n=0}^{2^m}\left|\widetilde{\Delta \mathrm{F}}(\mathrm{f}_n)\right|}{\sum_{n=0}^{2^m}\left|\tilde{\mathrm{F}}_{\mathrm{exp}}(\mathrm{f}_n)\right|} \tag{8}$$

2. The ***Weighted Frequency, WF:***

$$\mathrm{WF} = \frac{\sum_{n=0}^{2^m}\left|\widetilde{\Delta \mathrm{F}}(\mathrm{f}_n)\right|\cdot \mathrm{f}_n}{\sum_{n=0}^{2^m}\left|\widetilde{\Delta \mathrm{F}}(\mathrm{f}_n)\right|} \tag{9}$$

The AA factor can be considered a sort of "*average fractional error*" of the addressed calculation, whereas the weighted frequency WF gives an idea of the frequencies related with the inaccuracy[(3)].

The accuracy of a code calculation can be evaluated through these values, by representing the discrepancies of the addressed calculation with respect to the experimental data with a point in the WF-AA plane. The most significant information is given by AA, which represents the relative magnitude of these discrepancies; WF supplies a different information allowing to better identify the character of accuracy. In fact, depending on the transient and on the parameter considered, low frequency errors can be more important than high frequency ones, or vice versa.

[(3)] : In fact, it really represents the centre of gravity of the amplitude spectrum of Fourier transform of the difference function ΔF(t); therefore, it evaluates the kind of error of the addressed calculation (which can be more or less relevant depending on the characteristics of the analyzed transient)

Trying to give an overall picture of the accuracy of a given calculation, it is required to combine the information obtained for the single parameters into average indexes of performance. This is obtained by defining the following quantities:

$$(\mathrm{AA})_{\mathrm{tot}} = \sum_{\mathrm{i}=1}^{\mathrm{N_{var}}} (\mathrm{AA})_{\mathrm{i}} (\mathrm{w_f})_{\mathrm{i}} \tag{10}$$

$$(\mathrm{WF})_{\mathrm{tot}} = \sum_{\mathrm{i}=1}^{\mathrm{N_{var}}} (\mathrm{WF})_{\mathrm{i}} (\mathrm{w_f})_{\mathrm{i}} \tag{11}$$

with: $\sum_{i=1}^{N_{\mathrm{var}}} (w_f)_i = 1$, where:

- $\mathrm{N_{var}}$ is the number of parameters selected (to which the method has been applied);
- $(\mathrm{w_f})_\mathrm{i}$ are weighting factors introduced for each parameter, to take into account their importance from the viewpoint of safety analyses.

The need of $(\mathrm{w_f})_\mathrm{i}$ definition derives from the fact that the addressed parameters are characterized among other things by different importance and reliability of measurement. Thus, each $(\mathrm{w_f})_\mathrm{i}$ takes into account of:

- "*experimental accuracy"*: experimental measures of thermalhydraulic parameters are characterized by a more or less sensible uncertainty due to:
  - intrinsic characteristics of the instrumentation;
  - assumptions formulated in getting the measurement;
  - un-avoidable discrepancies existing between experimental measures and the code calculated ones (mean values evaluated in cross-sections, volume centers, or across junctions, etc.);
- "*safety relevance*": particular importance is given to the accuracy quantification of calculations concerned with those parameters (e.g. clad temperature, from which PCT values are derived) which are relevant for safety and design.

Last, a further contribution is included in the weighting factors definition; this is a component aiming at accounting for the physical correlations governing most of the thermalhydraulic quantities. Taking as reference parameter the primary pressure (its measurement can be considered highly reliable), a normalization of the AA values calculated for other parameters with respect to the AA value calculated for the primary side pressure is carried out. Doing thusly, the weighting factor for the generic *j*-th parameter, is defined as:

$$(\mathrm{w_f})_j = \frac{(\mathrm{w_{exp}})_\mathrm{j} \cdot (\mathrm{w_{saf}})_\mathrm{j} \cdot (\mathrm{w_{norm}})_\mathrm{j}}{\sum_{\mathrm{j}=1}^{\mathrm{N_{var}}} (\mathrm{w_{exp}})_\mathrm{j} \cdot (\mathrm{w_{saf}})_\mathrm{j} \cdot (\mathrm{w_{norm}})_\mathrm{j}} \tag{15}$$

and:

$$\sum_{j=1}^{N_{var}} (w_f)_j = 1 \qquad (16)$$

where:

- N_{var} is the number of parameters to which the method is applied;
- $(w_{exp})_j$ is the contribution related to the experimental accuracy;
- $(w_{saf})_j$ is the contribution expressing the safety relevance of the addressed parameter;
- $(w_{norm})_j$ is component of normalization with reference to the average amplitude evaluated for the primary side pressure.

This introduces a degree of engineering judgment that has been fixed by a proper and unique definition of the weighting factors[14] (see Table 1)). The most suitable factor for the definition of an acceptability criterion, therefore, for using the method, is the average amplitude *AA*. With reference to the accuracy of a given calculation, it has been defined the following acceptability criterion:

$$(AA)_{tot} < K \qquad (17)$$

where K is an acceptability factor valid for the whole transient. The lower the $(AA)_{tot}$ value is, the better the accuracy of the analyzed calculation (i.e., the code prediction capability and acceptability is higher). On the other hand, $(AA)_{tot}$ should not exceed unity in any part of the transient (AA = 1 means a calculation affected by a 100% error). Because of this requirement, the accuracy evaluation should be performed at different steps during the transient.

With reference to the experience gathered from previous application of this methodology, K = 0.4 has been chosen as the reference threshold value identifying good accuracy of a code calculation. In addition, in the case of upper plenum pressure, the acceptable threshold is given by K = 0.1.

Table 1. Selected weighting factor components for typical thermalhydraulic parameters.

Parameter	ID	w_{exp}	w_{saf}	w_{norm}
Primary pressure	PP	1.0	1.0	1.0
Secondary pressure	SP	1.0	0.6	1.1
Pressure drops	PD	0.7	0.7	0.5
Mass inventories	MS	0.8	0.9	0.9
Flow rates	FR	0.5	0.8	0.5
Fluid temperatures	FT	0.8	0.8	2.4
Clad temperatures	CT	0.9	1.0	1.2
Collapsed levels	LV	0.8	0.9	0.6
Core power	PW	0.8	0.8	0.5

5. Characterization of the Ranges of Variation of the Input Uncertain Parameters by FFTBM

5.1 The FFTBM procedure

A procedure to characterize the boundaries of the input uncertain parameters shall give emphasis to the connection between the 'objective sources' of uncertainty[9] and the list of input uncertain parameters.

It is worth noting that 'objective sources' of uncertainty and 'suitable lists' of input uncertain parameters should be considered for uncertainty method design and application, respectively. Moreover, both sets of parameters (i.e. 'objective sources' of uncertainty and 'suitable lists') are part of recognized international documents. Namely, reference is made for the sources of uncertainty to an OECD/CSNI (Organization for Economic Cooperation and Development / Committee on the Safety of Nuclear Installations) document issued in 1998 and to a more recent IAEA (International Atomic Energy Agency) document, issued in 2008. The lists of input uncertain parameters are derived from the application of various uncertainty methods in the UMS[5]and BEMUSE projects[15] completed or nearly completed under the umbrella of OECD/CSNI.

The proposed procedure is based on the use of the FFTBM plus series of considerations as the one here below:

a) One single input parameter shall not be 'responsible' for the entire error |exp-calc|, unless exceptional situations to be evaluated case by case;
b) Initial guess for Max and Min for variation ranges have to be based on the usual (adopted) expertise;
c) More than one experiment can be used per each NPP and each scenario. Highly influential parameters are expected to be the same. The bounding ranges should be considered for the NPP uncertainty analysis;
d) A data base of suitable uncertainty input parameters can be created per each NPP and each transient scenario.

Once the facility and the experiment has been chosen based on the selected NPP transient, the main steps of the procedure proposed in Figure 5 are:

- To run the Reference Case (RC),
- To select the Responses (R),
- To derive the AA_R^{REF} for each selected response by FFTBM,
- To select a set of Input Uncertainty Parameters (IP),
- To run sensitivity cases for which identified input parameter,
- To apply FFTBM to the sensitivity cases to obtain $AA_R^{*,IP}$,
- To apply the criteria for identifying the range [Min, Max],
- To discard not relevant Input Uncertainty Parameters.

About the criteria to be applied for the identification of the range of the input parameters, seven different criteria are under investigation to test the procedure. Each criterion includes a set of statements to be satisfied. The seven proposed criteria are listed hereafter:

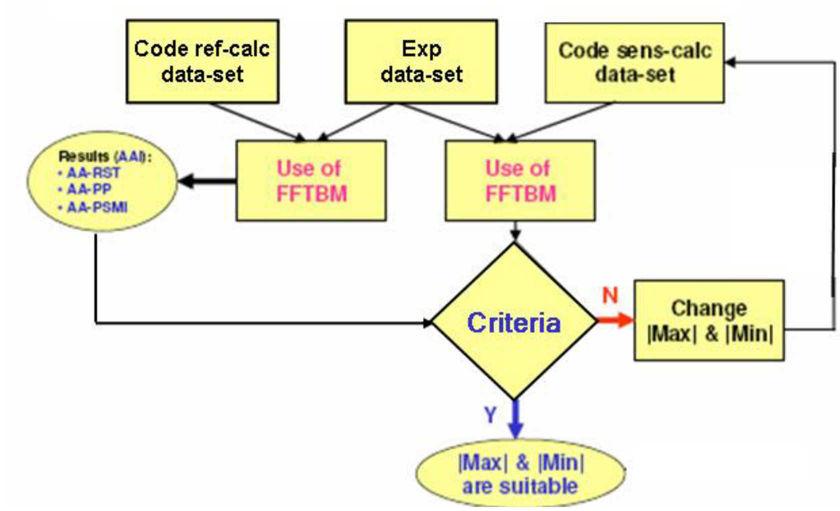

Figure 5. FFTBM procedure for identifying the range of variation of input uncertain parameters.

CR1: ***CR1.a:*** $AA_p^{*,IP} < 0.1$ p: pressure

CR1.b: $MAX(AA_{R_i}^{*,IP} / AA_{R_i}^{REF}) - 1 < 0$ i=1,…,N N responses

CR2: ***CR2.a:*** $AA_p^{*,IP} < 0.1$

CR2.b: $AA_G^{*,IP} / AA_G^{REF} - 1 < T1$ $AA_G^{*,IP} = \sqrt{\sum_{i=1}^{N} \left(AA_{R_i}^{*,IP}\right)^2}$ T1: Acceptability threshold #1

CR2.c: $MAX(AA_{R_i}^{*,IP} / AA_{R_i}^{REF}) - 1 < T2$ T2: Acceptability threshold #2

CR3: ***CR3.a:*** $AA_p^{*,IP} < 0.1$

CR3.b: $AA_G^{*,IP} / AA_G^{REF} - 1 < T1$ $AA_G^{*,IP} = \sqrt[N]{\prod_{i=1}^{N} AA_{R_i}^{*,IP}}$

CR3.c: $MAX(AA_{R_i}^{*,IP} / AA_{R_i}^{REF}) - 1 < T2$

CR4: ***CR4.a:*** $AA_p^{*,IP} < 0.1$

CR4.b: $AA_G^{*,IP} / \sqrt{N} - 1 < T1$ $\qquad AA_G^{*,IP} = \sqrt{\sum_{i=1}^{N} \left(AA_{R_i}^{*,IP} \big/ AA_{R_i}^{REF} \right)^2}$

CR5: ***CR5.a:*** $AA_p^{*,IP} < 0.1$

CR5.b: $AA_G^{*,IP} - 1 < T1$ $\qquad AA_G^{*,IP} = \sqrt{\dfrac{\sum_{i=1}^{N} \left(AA_{R_i}^{*,IP} \right)^2}{\sum_{i=1}^{N} \left(AA_{R_i}^{REF} \right)^2}}$

CR6: ***CR6.a:*** $AA_p^{*,IP} < 0.1$

CR6.b: $AA_G^{*,IP} / AA_G^{REF} - 1 < T1$ $\qquad AA_G^{*,IP} = \dfrac{1}{N} \cdot \sum_{i=1}^{N} AA_R^{*,IP}$

CR7: ***CR7.a:*** $AA_p^{*,IP} < 0.1$

CR7.b: $AA_G^{*,IP} - 1 < T1$ $\qquad AA_G^{*,IP} = \sqrt{\dfrac{\sum_{i=1}^{N} \left(AA_{R_i}^{*,IP} \cdot w_{f_i} \right)^2}{\sum_{i=1}^{N} \left(AA_{R_i}^{REF} \cdot w_{f_i} \right)^2}}$

w_{f_i} FFTBM weighting factors (see Eq. 15 and Table 1)

5.2 Sample applications of FFTBM procedure

The FFTBM procedure has been applied to three tests so far:

1. Marviken Test CFT04,
2. Edwards pipe,
3. LOBI tests A1-83 (analysis currently on going).

In this section the summary of the results achieved for the “Edwards pipe” application is provided. The sketch of the nodalization and of the boundary conditions of the experiment are given in Figure 6. The responses of interest are the pressure and the void fraction measured at about middle length of the pipe. The input parameters that have been initially investigated were about 10. Through the use of the FFTBM, the input parameters that determine a little variation of AA values in correspondence of large variations of those parameters were discharged (e.g. the pipe roughness and the thermal non-equilibrium constant in the Henry-Fauske critical flow model). Finally the analysis was reduced to investigate the following input parameters:

1. The form loss coefficient (Kloss),
2. The initial fluid temperature,

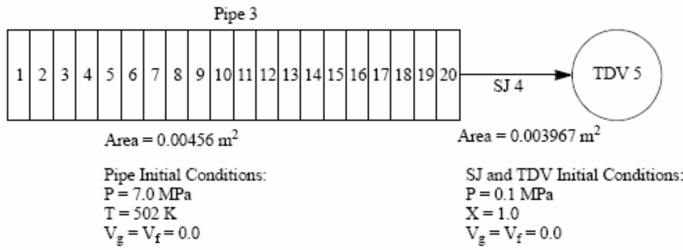

Figure 6. Nodalization and Boundary condition of the Edward pipe.

3. The break area,
4. The "Henry‐Fauske" choked flow model, discharge coefficient.

Figure 7 shows the results of the application of the FFTBM procedure in relation with the statement 'a' of each criteria (i.e. $AA_p^{*,IP} < 0.1$) for the four selected input parameters. The attention shall be focused on the AA value for the pressure (AA_P) and its limit equal to 0.1. The corresponding ranges of variation $[Min(IP), Max(IP)]_a$ for each of the four input parameters satisfying the statement 'a' are then identified.

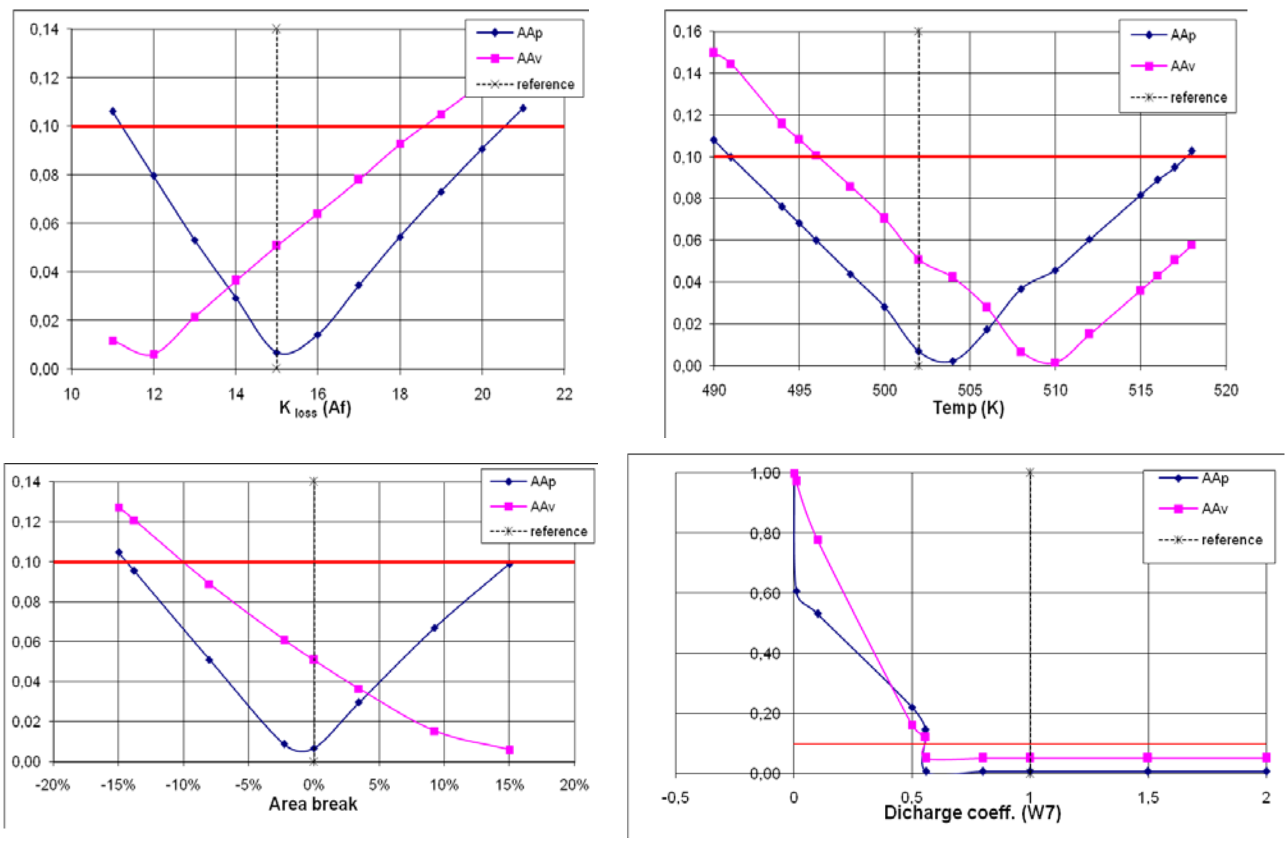

Figure 7. Values of AA_P and ranges of variation of input parameters.

In order to consider the overall accuracy of a calculation, one of the seven criteria proposed in section 5.1 can be applied. The results are shown in Figure 8 together with different values for the acceptability thresholds T. It shall be noted that the y-axis represents the values of the left side of the inequalities and that the statement criteria CR1.b is the same of the statements CR2.c and CR3.c. The corresponding ranges of variation $[Min(IP), Max(IP)]_{b,c}$ for each of the four input parameters satisfying the statements 'b' and 'c' are then identified. The final range of variation [Min(IP), Max(IP)] for each input parameter is then calculated by the intersection of the ranges of variation derived for the statements 'a', 'b' and 'c' for each criterion.

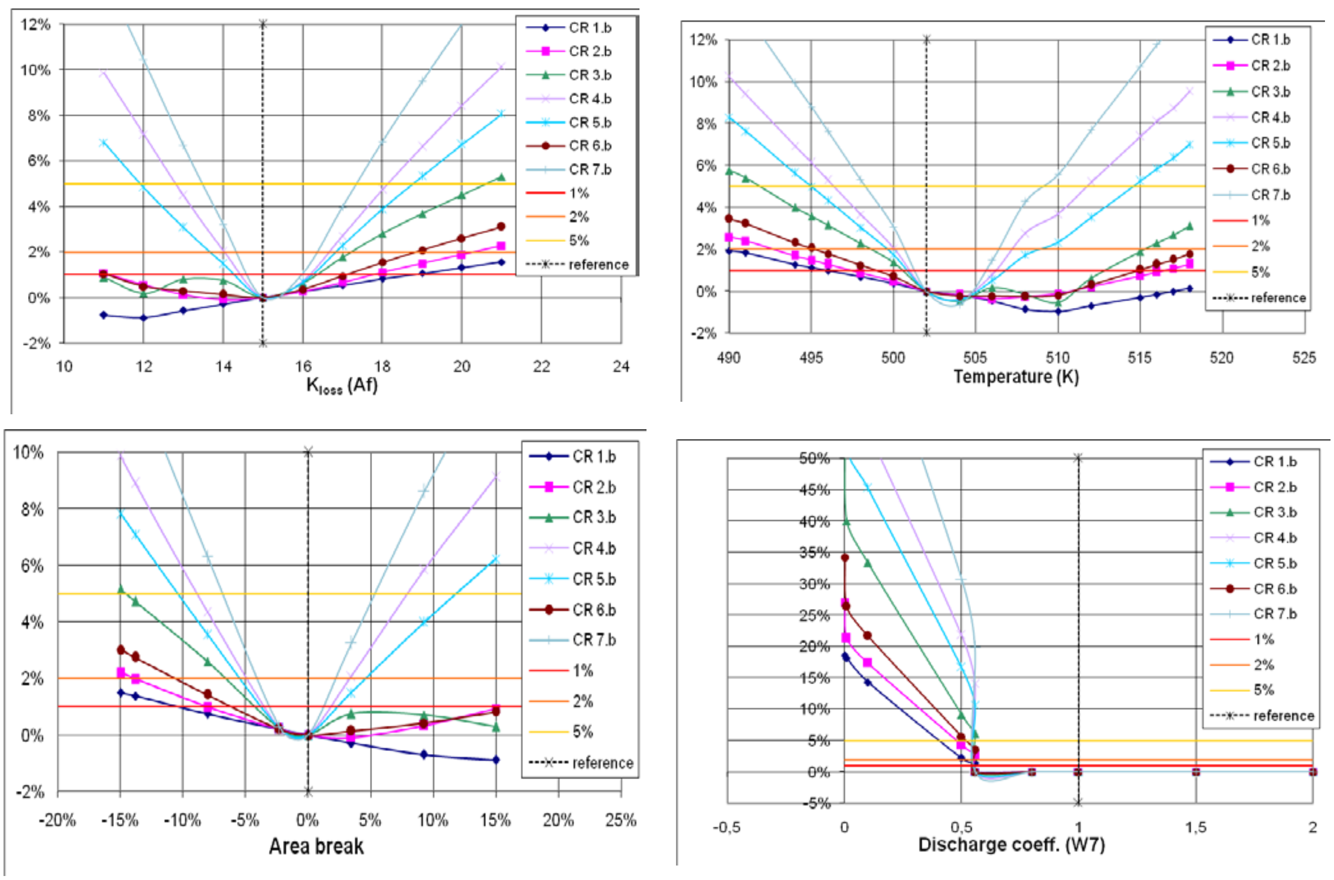

Figure 8. Results of the FFTBM procedure and ranges of variation of input parameters depending on different criteria.

6. Conclusions

The FFTBM allows a quantitative judgment for a given calculation. Each set of two curves constituted by a calculated and a measured time trend can be processed by FFTBM. The transformation from time to the frequency domain avoids the dependence of the error from the transient duration. Weight factors are attributed to each time trend to make possible the summing up of the error and the achievement of a unique threshold for accepting a calculation. The quantitative accuracy evaluation by FFTBM must be carried out following the demonstration that the calculation is qualitatively acceptable.

The objective of the paper is to propose a FFTBM-based procedure to characterize the ranges of variation of the input uncertain parameters in place of the usual approach based (mostly) on engineering judgment for the uncertainty methods based on the propagation of input error. The application to a simplified problem like Edwards pipe has been discussed.

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OECD/CSNI Workshop, Barcelona (Spain), Nov.16-18, 2011

Invited

The Findings from the OECD/NEA/CSNI UMS (Uncertainty Method Study)

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Abstract

Within licensing procedures there is the incentive to replace the conservative requirements for code application by a "best estimate" concept supplemented by an uncertainty analysis to account for predictive uncertainties of code results. Methods have been developed to quantify these uncertainties. The Uncertainty Methods Study (UMS) Group, following a mandate from CSNI (Committee on the Safety of Nuclear Installations) of OECD/NEA (Organization for Economic Cooperation and Development / Nuclear Energy Agency), has compared five methods for calculating the uncertainty in the predictions of advanced "best estimate" thermal-hydraulic codes.

Most of the methods identify and combine input uncertainties. The major differences between the predictions of the methods came from the choice of uncertain parameters and the quantification of the input uncertainties, i.e. the wideness of the uncertainty ranges. Therefore, suitable experimental and analytical information has to be selected to specify these uncertainty ranges or distributions.

After the closure of the Uncertainty Method Study (UMS) and after the report was issued comparison calculations of experiment LSTF-SB-CL-18 were performed by University of Pisa using different versions of the RELAP 5 code. It turned out that the version used by two of the participants calculated a 170 K higher peak clad temperature compared with other versions using the same input deck. This may contribute to the differences of the upper limit of the uncertainty ranges. A 'bifurcation' analysis was also performed by the same research group also providing another way of interpreting the high temperature peak calculated by two of the participants.

1. Introduction

Some computer codes that model reactor accidents contain deliberate pessimisms and unphysical assumptions. It is then argued that the overall predictions are worse than reality (for example, calculated fuel temperatures are higher than reality). These conservative evaluation models, for example, were provided to satisfy US licensing requirements for such pessimistic calculations up to 1988, ref. [1] (see below for a historical outline of the subject).

The other approach is to design a code to model all the relevant processes in a physically realistic way with the intention that the predictions are not biased either in a pessimistic or optimistic direction (”best estimate”). The main motivations for the development of advanced best estimate thermal hydraulic codes were:

- To describe the physical behavior realistically without individual conservative assumptions. This should calculate the mean of the data rather than providing a bound to the data.
- To obtain more economical operation of reactors by relaxing technical specifications and core operating limits set by conservative evaluation model calculations. This might include prolonging fuel cycles, up-rating power and justifying the continued operation of some reactors against modern safety standards.
- To develop effective accident management measures based on realistic analysis.

These realistic computer codes can approximate the physical behavior with more or less accuracy. The inaccuracies are stated during the usual code validation process. Agreement of calculated results with experimental data is often obtained by choosing specific code input options or by changing parameters in the model equations. These selected parameters usually have to be changed again for different experiments in the same facility or a similar experiment in a different facility in order to obtain agreement with data. A single “best estimate” calculation with an advanced realistic code would give results of one code run of unknown accuracy. To make such results useful, for example if they are to be compared with limits of acceptance, the uncertainty in the predictions then has to be calculated separately. Uncertainty analysis methods therefore had to be developed to estimate safety to justify reactor operation. A particular stimulus was the U.S. Nuclear Regulatory Commission (USNRC) revision of its regulations in 1989 to permit the use of realistic models with quantification of uncertainty in licensing submissions for Emergency Core Cooling Systems, ref. [2]. This kind of approach to the licensing is identified as BEPU (Best Estimate Plus Uncertainty) and requires the application of qualified Uncertainty methods.

In addition uncertainty analysis can be used to assist research prioritization. It can help to identify models that need improving and areas where more data are needed. It can make code development and validation more cost-effective.

The Uncertainty Methods Study (UMS) compares different methods to estimate the uncertainty in predictions of advanced best estimate thermal hydraulic codes by applying the methods to a particular experiment. This paper summarizes the comparison reported in ref. [3] and includes a discussion from more recent evaluations of the results. An outline of the milestones for the application of BEPU is given in advance.

2. A Historic Outline for Accident Analyses

Accidents and related scenarios in nuclear power plants were considered to demonstrate the safety of NPP in the ‘50s when computers did not exist. Experiments, pioneering thermal-hydraulics models and engineering evaluations (this can be expressed as engineering judgment, consideration of lack of knowledge, introducing conservative safety margins, etc.) were the basis of the reactor safety analyses accepted by regulators at the time.

More systematic thermal-hydraulic studies and experiments were conducted in the ‘60s, noticeably concerning individual ‘physical’ phenomena like Two-Phase Critical Flow, Critical Heat Flux, Depressurization/Blow-down, etc. New findings from those researches were considered in reactor safety and licensing documents.

Massive use of computers for nuclear reactor safety started in the ‘70s. The accident analysis could also benefit of primitive numerical codes and of results of lately called integral-system experiments. The nuclear regulatory point-of-view was well established with the publication of the ‘Interim Acceptance Criteria for ECCS’, by the USNRC (US AEC at the time), ref. [4]. This triggered a wide variety of researches aimed at the evaluation of safety margins focusing on the estimation of the maximum temperature on the surface of fuel rods following Large Break Loss of Coolant Accident (LB-LOCA). The Appendix K to the paragraph 10-CFR-50.46 of the Code of Federal Regulation, followed (i.e. the 'Interim Criteria') in 1974, ref. [1], basically confirming the structure of the document by USNRC, 1971 (10 CFR 50, including the Appendixes, is a continuously updated document). 'Conservatism' is the key-word which characterizes the application of Appendix K in licensing analyses. During the same decade, the WASH-1400 or the "Rasmussen Report" was issued addressing the relevance of PSA also producing significant results from the execution of probabilistic analyses, see ref. [5].

Robust, user-friendly versions of lately called system-thermal-hydraulic codes (or computational tools) were available in the '80s. The importance of V & V (Verification and Validation), soon became clear (see e.g. ref. [6]) as a mandatory process to be completed before application of those computational tools to safety and licensing. In this context, the bases were set for addressing the scaling issue, e.g. see ref. [7], for a summary status on the subject. International activities were conducted at CSNI (Committee on the Safety of Nuclear Installations of OECD/NEA, Organization for Economic Cooperation and Development / Nuclear Energy Agency) to propose viable ways for V & V, e.g. refs. [8], [9] and [10], also involving the evaluation of the user effect, ref. [11], and the recognition of the role of the input deck (or nodalization) and qualification, e.g. ref. [12]. Appendix K to 10 CFR 50.46 continued to be used during the decade for licensing purposes.

The need for uncertainty methods suitable for predicting unavoidable errors to be added to the results of calculations performed by system thermal-hydraulic codes became clear at the beginning of '90s (or even at the end of '80s). Working approaches were proposed, e.g. by USNRC, ref. [13] 1989, by GRS, ref. [14] and by University of Pisa, ref. [15]. Research and Development (R & D) activities started to have robust (e.g. with results independent of the user) and qualified uncertainty methods, e.g. ref. [3] (the UMS report concerned in the present paper). The US NRC issued the Regulatory Guide (RG) 1.157, ref. [16]: the application of system thermal-hydraulic codes was envisaged, even though recommending the use of selected conservative models. Those models should be applied in phenomenological areas where the knowledge was not considered satisfactory. Requirements in the RG 1.157 gave guidelines how to perform uncertainty analysis when using best estimate codes in licensing. 10 CFR 50.46 opened the possibility to use best estimate codes for LOCA analyses in licensing, however, in that case uncertainty analysis is required. The acronym BEPU for "best estimate plus uncertainty" started to circulate.

Applications of BEPU approaches in licensing processes definitely started in the '00s. The following key events (not an exhaustive list, not in the order of importance) give an idea of the technology development in the area:

a) AREVA proposed a BEPU methodology to analyze the LBLOCA for the licensing of Angra-2 NPP in Brazil, ref. [17] (the method is noticeably different from what is described by Martin and O'Dell, in ref. [18]). The submission was analyzed by the Regulatory Authority of Brazil which also requested the application of different uncertainty methods by assessors independent from AREVA, e.g. see refs. [19] and [20].

b) USNRC issued the RG 1.203, ref. [21], and opening for the possibility of adopting the BEPU approach in licensing.

c) CSNI launched and completed the six-year project BEMUSE, e.g. ref. [22]. The general aim of BEMUSE was to demonstrate the maturity of uncertainty methods and approaches with main concern to LBLOCA applications. The objective was basically achieved, but differences in the results by participants mainly in predicting reflood time caused the need for a careful interpretation of results. The difficulty in harmonizing, from the side of applicants of uncertainty methods, the choice of input uncertainty parameters and the related ranges of variations was an outcome from the project.

d) Two key documents were issued by IAEA, i.e. refs. [23] and [24]. The former (Safety Report Series, SRS 52) deals with the description of workable uncertainty approaches and methods. The latter (Specific Safety Guide, SSG-2 on deterministic safety analysis) proposes the BEPU approach in licensing as consistent with the technological state of art in the area of accident analysis.

e) Best Estimate (BE) conferences (BE-2000 and BE-2004) were held under the auspices of the American Nuclear Society (ANS) in 2000 and 2004, i.e. refs. [25] and [26]. The series of conferences was continued by V & V Workshops held in Idaho Falls and in Myrtle Beach, i.e. refs. [27] and [28], respectively. The importance given by the international community to BEPU methods was evident from those conferences.

f) A variety of BEPU applications all over the world during the concerned decade, mostly within the license renewal framework, are summarized in ref. [29].

Definitely, the first decade of the current millennium is characterized by the breakthrough for the use of BEPU methods in licensing of NPP. The '10s decade started with the submission of Chapter 15 of the FSAR of the Atucha-II NPP to the Regulatory Authority in Argentina, by the NA-SA utility. In this case, the entire FSAR is based on the BEPU and the approach itself was submitted in advance (and endorsed by the Regulatory Body), e.g. ref. [30]. In this case the adopted uncertainty method was the CIAU (Code with capability of Internal Assessment of Uncertainty), ref. [31]. The approval process from the side of the Regulatory Authority at the time of the present paper is still on-going.

3. Objectives of UMS

The CSNI Task Group on Thermal Hydraulic System Behaviour held a Workshop to discuss the different uncertainty methods in 1994. The Workshop recommended that there should be an international comparison exercise between the available uncertainty analysis methods with the objectives, ref. [32]:

1. To gain insights into differences between features of the methods by:
   - comparing the different methods, step by step, when applied to the same problem;
   - comparing the uncertainties predicted for specified output quantities of interest;
   - comparing the uncertainties predicted with measured value;
   - and so allowing conclusions to be drawn about the suitability of methods.
2. To inform those who will take decisions on conducting uncertainty analyses, for example in the light of licensing requirements.

The CSNI approved the Uncertainty Methods Study (UMS) at its meeting in December 1994, with these objectives. The UK was given the task of leading the study. The study was performed from May 1995 through June 1997.

4. Uncertainty Methods applied

The methods compared in the UMS are summarized in Table 1. The methods (and those discussed at the 1994 Workshop) may be divided into three groups according to their basic principles, refs. [3], [33] and [34]:

- The University of Pisa method, the Uncertainty Method based on Accuracy Extrapolation (UMAE), extrapolates the accuracy of predictions from a set of integral experiments to the reactor case or experiment being assessed (e.g. see also ref. [15]).

The other methods rely on identifying uncertain models and data and quantifying and combining the uncertainties in them. They fall into two kinds:

- The AEA Technology method that characterises the uncertainties by "reasonable uncertainty ranges" and attempts to combine these ranges with a bounding analysis.
- Methods which assign probability distributions to uncertainty ranges for uncertain input parameters and sample the resulting probability density at random in the space defined by the uncertainty ranges. In the UMS, the GRS, IPSN and ENUSA methods are of this kind; so also is the Code Scaling Applicability and Uncertainty Evaluation (CSAU) Method as demonstrated in ref. [35]. The probability used here is due to imprecise knowledge and is not probability due to stochastic or random variability.

Table 1. Summary of Methods Compared in the UMS Study.

Participant	Code Version Used	Method Name and Type
AEA Technology, UK	RELAP5/MOD3.2	AEAT Method. Phenomena uncertainties selected, quantified by ranges and combined.
University of Pisa, Italy	RELAP5/MOD2 cycle 36.04, IBM version CATHARE 2 version 1.3U rev 5	Uncertainty Method based on Accuracy Extrapolation (UMAE). Accuracy in calculating similar integral tests is extrapolated to plant.
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany	ATHLET Mod 1.1 Cycle A	GRS Method. Phenomena uncertainties quantified by ranges and probability distributions (PDs) and combined.
Institut de Protection et de Sûreté Nucléaire (IPSN), France	CATHARE 2 version 1.3U rev 5	IPSN Method. Phenomena uncertainties quantified by ranges and PDs and combined.
Empresa Nacional del Uranio, SA (ENUSA), Spain	RELAP5/MOD 3.2	ENUSA Method. Phenomena uncertainties quantified by ranges and PDs and combined.

5. Experiment used for UMS

The participants and the Task Group of Thermal Hydraulic System Behavior agreed that the International Standard Problem (ISP) 26 experiment, LSTF SB-CL-18, should be used. LSTF SB-CL-18 is a 5% cold leg small break LOCA experiment conducted in the ROSA-IV Large Scale Test Facility (LSTF). The LSTF is located at the Tokai Research Establishment of JAERI and is a 1/48 volumetrically scaled, full height, full pressure simulator of a Westinghouse type 3423 MWth pressurized water reactor. The experiment simulates a loss of off-site power, no high pressure injection (HPIS), the accumulator system initiates coolant injection into the cold legs at a pressure of

4.51 MPa, the low pressure injection system initiates at 1.29 MPa. Thus, the experiment considers a beyond design basis accident. Because of the ISP the experiment was already well documented. Much helpful information has been provided by Y. Kukita of the Japan Atomic Energy Research Institute (JAERI).

Although ISP 26 was an open ISP participants in the UMS have not used experimental measurements from the test, for example the break flow or the secondary pressure. In the same way other experiments performed in the LSTF were not used. This means not allowing the related measurements to influence the choice of input uncertainty ranges and probability distributions of the input uncertainties. Submissions included a written statement of the justification used for input uncertainty ranges and distributions used and were reviewed at workshops. All other assumptions made were listed and reviewed.

Two dry-out situations at different timing (see below) characterize the ISP 26 experiment: this makes challenging the application of uncertainty methods.

6. Comparison of the uncertainty methods

A detailed description of the methods is given in refs. [3], [33] and [34], see also ref. [36]. An excerpt from comparison tables provided in those documents is given in Table 2. In this way a step by step comparison is performed and the main similarities and differences between the methods are characterized.

Table 2. Comparison between the uncertainty methods adopted in UMS: an excerpt from comprehensive tables (refs. [3], [33], [34] and [36]).

Feature	AEA Technology	University of Pisa	Statistic methods - GRS, IPSN, ENUSA
Characterization of uncertainties	Reasonable uncertainty ranges	Accuracy: difference between prediction and measurement	Ranges and probability distributions
Selection of important uncertainties	Yes (9 and 7 in two demonstration studies)	No	GRS, IPSN: No (52 and 25 here) ENUSA: PIRT (25)
Combination and propagation of uncertainties	Analyst explores uncertainty space and decides when to stop	Measured accuracy can be extrapolated to plant if: - criteria on integral test data satisfied - data, nodalization, users are qualified - relevant thermal hydraulic aspects are in data base - measured/calculated ratio scattered about 1.0;	Uncertainty space sampled at random according to combined probability distribution
Number of code runs	22 and 50 in two studies	One for each test in the data base and one for the plant	For one sided 95%/95% tolerance/confidence limit: 59; for two-sided interval: 93
Use of specific data for scaling	During code validation	Yes	During code validation
Use of response surface to approximate result	No	No	No
Use of biases on results	No	Possibly	No

The results from the comparison among the methods can be summarized as follows:

a) Three of the methods (i.e. GRS, IPSN and ENUSA) are based upon the approach 'propagation of input uncertainties'; one method (i.e. UMAE) is based upon the approach 'extrapolation of output error' (this classification was introduced about one decade after the end of the UMS, i.e. see ref. [23]). All these methods make use of statistics at different levels.

b) One method (i.e. AEAT), based on the approach 'propagation of input uncertainties', is fully based upon the control by analysts at each step of the adopted procedure and makes no use of statistics. It is an unfortunate situation that this method has no (openly available) application other than the UMS.

c) Engineering judgment is needed by all methods; however, in the case of UMAE engineering judgment is used in the development phase and in the case of other methods this is needed for each application. The expert judgment in setting up an input deck can be checked by applying acceptance criteria by use of the Fast Fourier Transform in the UMAE method.

d) Experimental data are used for validation of computer codes selected for uncertainty evaluation. However, in the case of UMAE experimental data are strictly needed to derive uncertainty. A similar statement can be applied in relation to the scaling issue: indirectly, all methods require the consideration of scaling, but in the case of UMAE, the availability of scaled data from experiments is mandatory to derive uncertainty. However, the gap between integral test facilities to reactor size is significant.

e) Among the methods which use statistics, see item a), the GRS method takes benefit of a theoretical derivation (i.e. the Wilks formula, ref. [37]) to identify the number of needed calculations (or code runs). Vice-versa expert judgment at various extents is needed by the AEAT method for each application, except in the case of UMAE where only one calculation is needed (actually, more code runs may be needed in the cases when 'biases' are concerned, ref. [15]), and to optimize the nodalization.

f) All methods which are based on the approach 'propagation of input uncertainties' have to identify proper input uncertainty parameters and related ranges of variation (see also the conclusions and the PREMIUM - Post BEMUSE REflood Models Input Uncertainty Methods - project). For those methods, the propagation of uncertainty occurs by code calculations varying the identified input uncertainties. In the case of UMAE, either an unqualified code or nodalization is causing large uncertainty bands. However, the extrapolation of accuracy to get the uncertainty requires a specific qualification process.

7. Comparison of results from the application of the methods

Results from the UMS comparison project are comprehensively described in ref. [3] and summarized in ref. [38]. In the following, a synthesis of the discussion in ref. [38] is presented.

The participants were asked to make uncertainty statements, based on their calculation results, for:

1 Functions of time:

- Pressurizer pressure.
- Primary circuit mass inventory.
- Rod surface temperature for one assigned position

2 Point quantities:

- First peak clad temperature.
- Second peak clad temperature.
- Time of overall peak clad temperature (the higher of the two).
- Minimum core pressure difference at an assigned position.

As a result of the short time-scale of the UMS project and of limited funding, some of the uncertainty ranges adopted for methods following the ‘propagation of input uncertainty’ approach may have been less well defined and, as a consequence, wider than would have been the case, for example, if the calculations were for a plant safety analysis. This had consequences to the AEAT and the ENUSA uncertainty predictions.

7.1 Results for the time functions

The uncertainty ranges calculated for the point quantities and for the functions of time, i.e. pressurizer pressure, primary system mass inventory and hot rod surface temperature for the assigned location are given in graphical form in refs. [3] and [38]. Here, results are discussed only in relation to the rod surface temperature, Fig. 1.

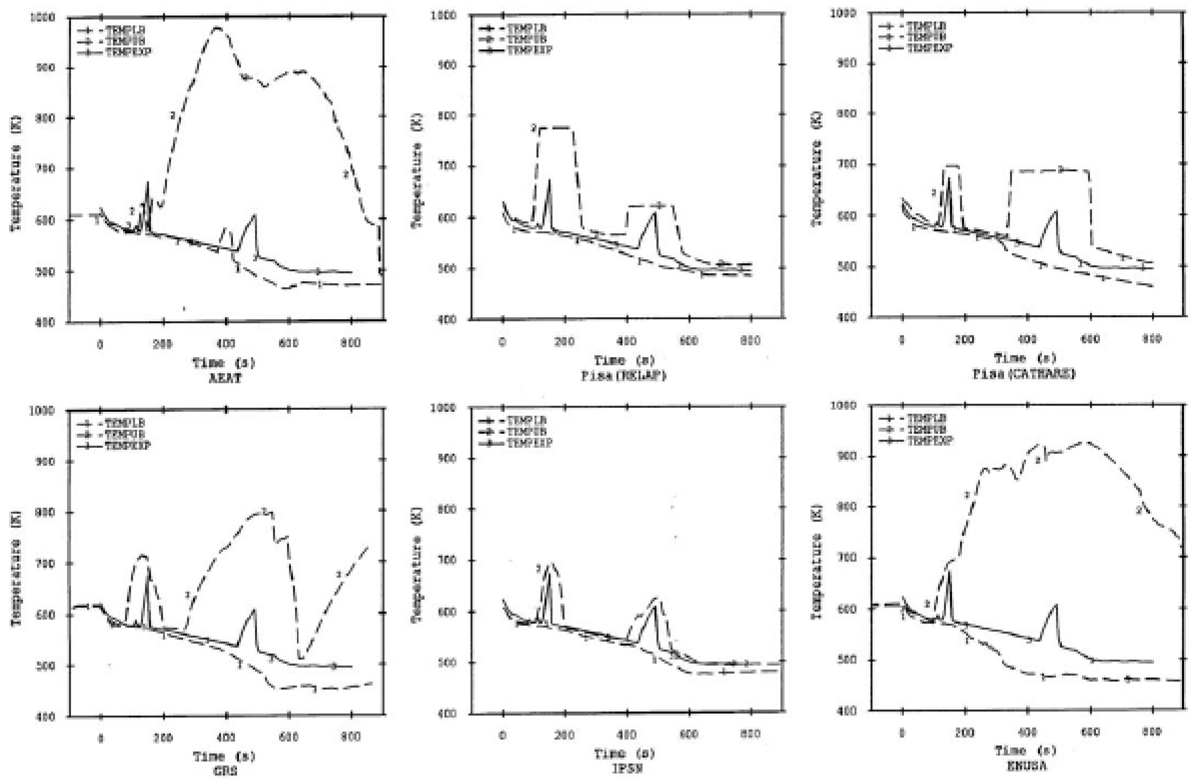

Fig. 1 – UMS results related to rod surface temperature.

7.1.1 Overall evaluation

The AEAT and ENUSA predictions are remarkably similar. The same code (RELAP5), the same base input deck and mostly the same input uncertainties were used. However, they applied different methods, i.e. bounding (AEAT) versus probabilistic treatment (ENUSA) of the uncertainties, and the numbers of adopted input uncertainty parameters are different (7 by AEAT and 25 by ENUSA).

The two applications of the Pisa (UMAE) method using different computer codes (RELAP and CATHARE) are broadly similar. The clad temperature ranges for the second dry-out are similar. Those ranges for the first dry-out calculated by CATHARE gave a narrower range because the base case calculation did not predict the dry-out (ref. [38]).

Of the probabilistic methods, IPSN gave the narrowest ranges. This is attributed to the choice of parameters made and to the narrow ranges adopted for the input uncertainty parameters. The characterization of input uncertainty parameters, of the number of those parameters and of the related ranges of variation could not be optimized owing to the time constraints of the UMS project. An actuation of the CCFL option in the CATHARE code (at the upper core plate and at the inlet plena of the steam generators) was needed to predict the first dry-out.

The large uncertainty ranges for clad temperature calculated by AEAT, ENUSA and GRS prompted discussion among the participants about the reasons for them.

In the case of AEAT and ENUSA this may be due to unrealistically large uncertainty ranges for CCFL parameters, agreed upon in order to complete the study on time (as already mentioned). Another contribution may come from the used RELAP5 version MOD 3.2, as discussed in chapter 8.

The measured maximum clad temperature during the second heat-up is well below that of the first one. In the GRS calculations, many of the calculated time histories exhibit the contrary, namely a maximum during the second heat-up. The uncertainty of the maximum temperature during the second heat-up is about double of that during the first. In some code runs a partial dry-out occurs due to fluid stagnation in the rod bundle which causes an earlier second heat-up, and as a consequence, a larger increase of the clad temperature.

The large uncertainty is striking, and the sensitivity measures indicate which of the uncertainties are predominantly important. These sensitivities are presented in Volume 2 of ref. [3]. Main contributions to uncertainty come from the critical discharge model and the drift in the heater rod bundle. Other contributions that are worth mentioning come from the bypass cross section upper down-comer - upper plenum. As a consequence, improved state of knowledge about the respective parameters would reduce the striking uncertainty of the computed rod surface temperature in the range of the second core heat-up most effectively.

7.1.2 Adding details to the evaluation

Problems were identified in predicting the first dry-out by AEAT, by the UMAE application with CATHARE and by IPSN: this is due to the physical phenomenon of loop seal filling and clearing. Proposed solutions when performing the uncertainty analyses had minor impact on the overall results.

The occurrence of late quench times predicted for the upper uncertainty band, up to about 600 s (see Fig. 1) was questioned since the issuing of the report ref. [3]. AEAT analyzed cases where quench times at the concerned rod surface location was calculated after 570 s. This occurred when:

- there was a third dry-out, or
- when loop seal B (broken loop) does not clear (or clears late) which is associated with low break flow or modeling the off-take at the top of the cold leg, or
- when loop seal A (intact loop) did not clear combined with CCFL parameters maximize the hold-up of water in the Steam Generator (SG) and SG inlet plenum, or
- in cases where the CCFL parameters that maximize the hold-up of water were used in the SG and SG inlet plenum combined with relative low break flow.

Namely, in 17 out of the 50 cases reported by AEAT a third dry-out was detected. The third dry-out leads to relatively small temperature transients which are localized at the top of the core. This is not correlated with high peak clad temperatures.

The third dry-out conditions were also calculated by ENUSA and GRS. GRS found a third dry-out in 2 out of 99 calculations. They were associated with high bypass flow cross section and high condensation. High condensation keeps the pressure low so the accumulators inject continuously. However the accumulator water is discharged through the break. Although the input uncertainties leading to these cases differ from those found by AEAT, they also do not calculate clearance of the broken loop seal. Some other GRS calculations without loop seal clearance do not predict a third dry-out, however.

8. 'Post-UMS' evaluations

The information provided in chapters 3 to 7 above considers the original summary of UMS in ref. [3], and the key evaluations from refs. [33], [34], [36] and [38]. In the following, results from computational activities with regard to UMS, and performed after the end of the UMS project, are outlined.

8.1 Use of different code versions at UNIPI

University of Pisa performed comparison calculations of experiment LSTF-SB-CL-18 using different versions of the RELAP 5 code as reported in Fig. 2, i.e. MOD 2 (old code in Fig. 2), MOD 3.2 (intermediate code in Fig. 2) and MOD 3.2.2 (new code in Fig. 2), ref. [39]. The MOD 2 code version was used by the University of Pisa and MOD 3.2 by AEA Technology as well as by ENUSA in the UMS. 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. This may contribute to the relative high upper limit of the uncertainty ranges calculated by AEAT and ENUSA. That explanation is also in agreement with the AEAT peak clad temperature of 787 K at 300 s for their reference calculation using nominal values for the input parameters, without calculating the first heat-up. The measured second peak is 610 K at 500 s.

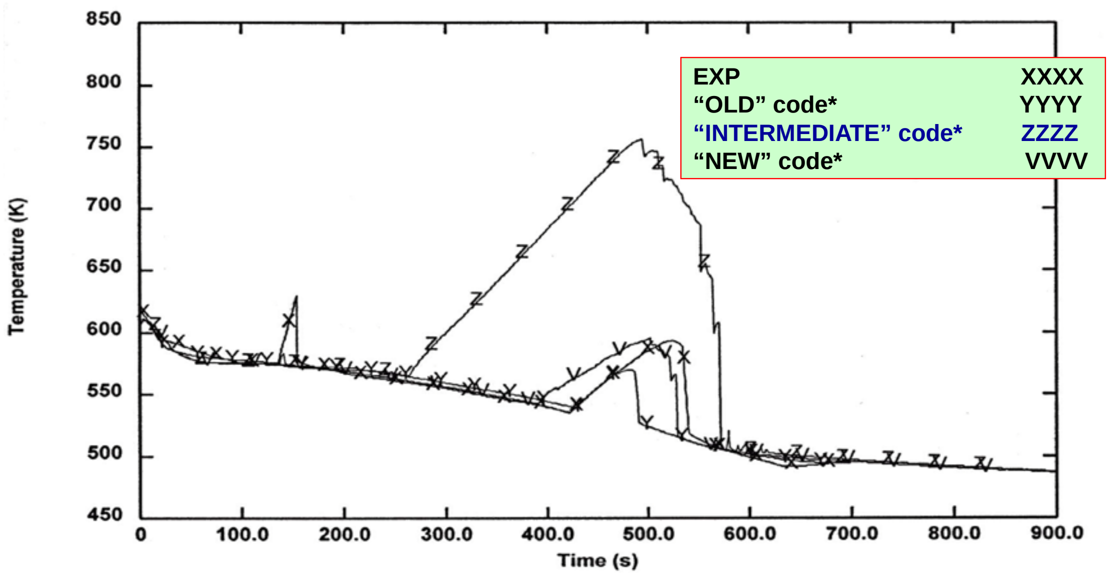

Fig. 2 – UNIPI UMS reference calculation performed with different versions of the Relap5 code.

8.2 New GRS calculation

A new application of the GRS method to the UMS problem was performed by GRS using revised probability distributions for the most important uncertain parameters. These were identified by sensitivity measures in the original UMS analysis: contraction coefficient of critical discharge flow and drift in the heater rod bundle. A newer ATHLET version Mod 1.2, cycle A was used instead of Mod 1.1, Cycle A. The result was a shorter duration of the second heat-up and lower peak clad temperatures during the second heat-up, as shown in Fig. 3. More details can be found in ref. [40].

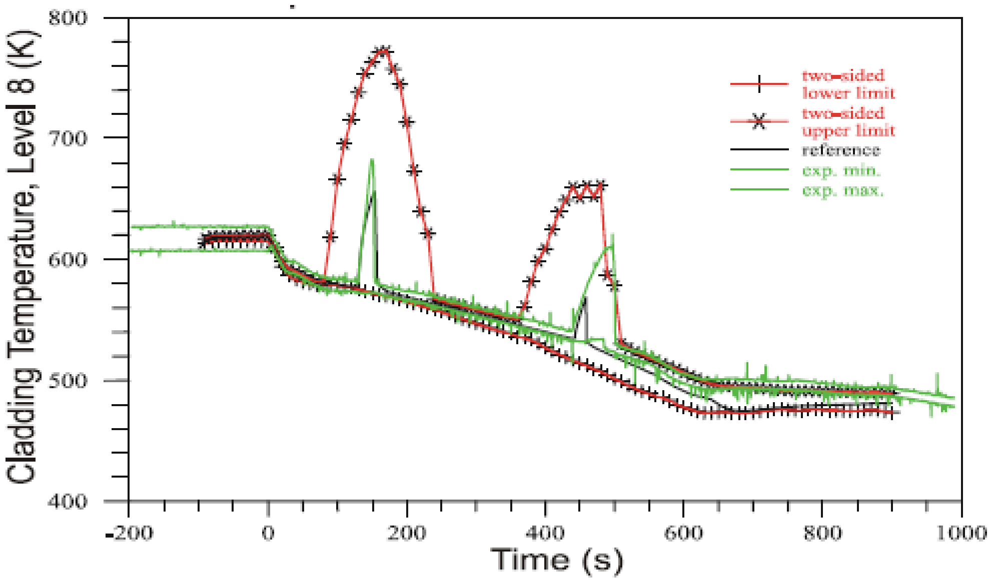

Fig. 3 – GRS UMS revised calculated uncertainty range compared with measured minimum and maximum values of rod clad temperature.

8.3 Bifurcation analysis performed at UNIPI

Thermal-hydraulic transient scenarios in NPP can occur when bifurcations bring the transient evolution far from the best-estimate deterministic prediction, thus invalidating the original uncertainty evaluation. Therefore a bifurcation analysis may be necessary. The related results should be intended to be 'outside' the 95% confidence level which characterizes the results from the 'normal' uncertainty analysis.

An activity was completed at University of Pisa as discussed in ref. [41]. The selected reference transient for the analysis is the same experiment (and related calculation) as for UMS.

Starting points for the bifurcation analysis are:

- The identification of type '1' and type '2' bifurcations.
- The knowledge of the uncertainty characterizing the parameters which affect the bifurcation.

Namely, type '1' and type '2' bifurcations are associated with the actuation of any system during the concerned transient (e.g. scram, valve opening-closure, pump-trip, accumulator intervention, etc.) and with the occurrence of cliff-edge type thermal-hydraulic phenomena (primarily CHF occurrence), respectively. Furthermore, the knowledge of the uncertainty characterizing the parameters which affect the bifurcation is the result of the uncertainty study.

For instance, in the case of accumulator intervention, let us consider the diagram pressure versus time where upper and lower uncertainty lines are reported. Then, the accumulator start may come at any time identified by the intercepts defined by the horizontal line drawn at the nominal accumulator pressure and the upper and the lower pressure boundary curves. Similarly, the actuation of a steam relief valve, e.g. in the secondary side of SG, having reached the opening pressure, may occur during a time interval depending upon the width of the uncertainty bands for SG pressure.

Once 'n' possible bifurcation points have been identified on the boundaries of the uncertainty bands, 'n' new calculations are performed starting from the initial conditions: by properly fixing input parameters the 'n' bifurcation points are reached. Branches are calculated starting from each of the 'n' points, thus creating a bifurcation tree. More details can be found in ref. [41].

In the case of the UMS test scenario, the envelope of the bifurcation calculations can be seen in Fig. 4: the bifurcation envelope 'resembles' the AEAT and the ENUSA upper uncertainty boundaries in Fig. 1, thus supporting the conclusion that those boundaries are 'conservative'.

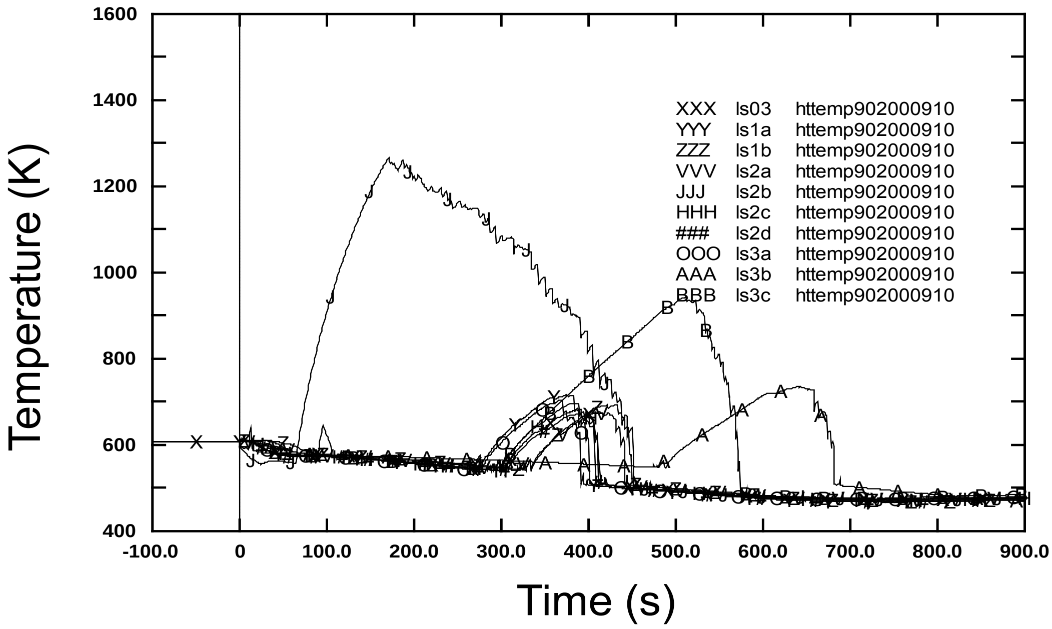

Fig. 4 – UNIPI bifurcation analysis performed in relation to UMS reference calculation.

8.4 Application of CIAU

The CIAU procedure, see chapter 2 and refs. [23] and [31], has been applied to the UMS task producing the results in Fig. 5. Predicted uncertainty bands appear similar to those predicted by the UMAE in Fig. 1 and by the GRS method in Fig. 3.

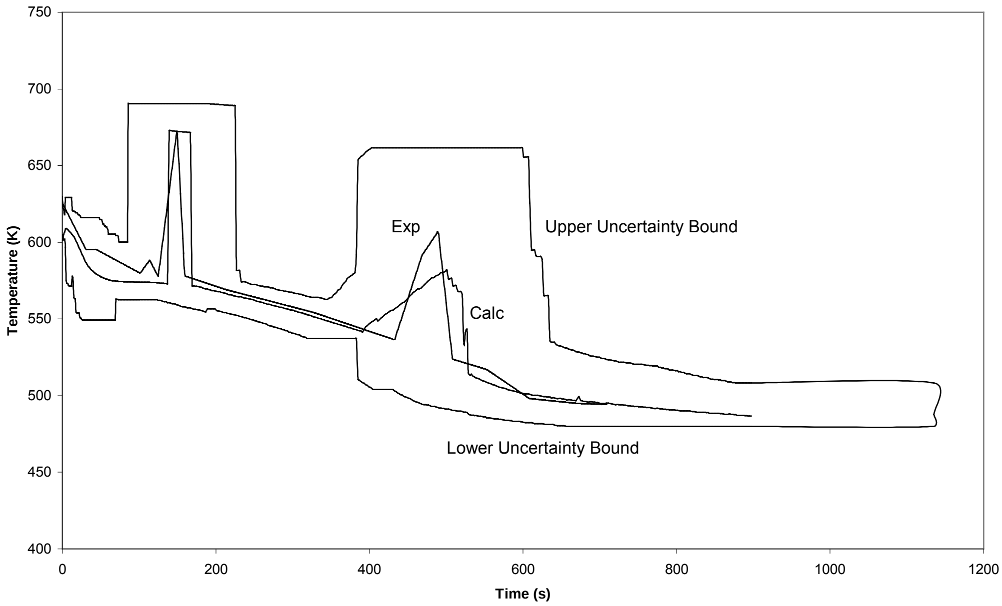

Fig. 5 – UNIPI CIAU application to UMS.

9. Conclusion

Five uncertainty methods for advanced best estimate thermal hydraulic codes have been compared. Most of the methods identify and combine input uncertainties. Three of these, the

GRS, IPSN and ENUSA methods, use probability distributions and one, the AEAT method, performs a bounding analysis. One of the methods, the University of Pisa method, is based on accuracy extrapolation from integral experiments. To use this method, stringent criteria on the code and experimental data base must be met.

Firstly, the basis and the characteristics of the methods were compared within UMS, secondly, the uncertainty predictions obtained with these methods applied to the LSTF small break LOCA test SB-CL-18 were compared.

Three (out of 5) methods, GRS, ENUSA and IPSN follow the approach 'propagation of code input uncertainty'. The selection of input uncertainty parameters and related range of variations constitutes a challenge for those methods. One method, AEAT, is based on a number of steps controlled by the analyst: the challenges in this case are the availability of suitable expertise and the reproducibility of results. One method, UMAE proposed by University of Pisa, follows the approach later on identified as 'extrapolation of code output error'. The method is based upon processing of the error in calculating experimental transient scenarios which are relevant to the specific application. The challenges are the availability of suitable experimental data and the demonstration that those data are actually 'relevant' to the specific application.

The presented uncertainty bands for the concerned transient predicted by the different methods and users leads to the following conclusions:

a) Differences in calculating the wideness of the time-dependent uncertainty bands by the various methods are significant and may cause misleading conclusion by a reader (of the UMS report) not familiar with the (UMS) activity.

b) Large band wideness is calculated by AEAT and ENUSA which may raise concerns related to the capability of codes and their applicability to the prediction of NPP transients.

c) In contrast to the above large bands, a very small band wideness is calculated by IPSN: those bands are small compared with the spread of results observed in the conduct of ISP 26 where several qualified participants calculated the same transient applying state-of-the-art qualified codes.

d) The set of results calculated by UMAE (2 applications within UMS), by GRS (post-UMS calculation) and by CIAU (post-UMS calculation) show similar results and are consistent with the current capabilities of codes. These might be considered as reference results from the UMS.

Explanations have been provided and can be found or can be evaluated from the various reports (see the list of references) in relation to items a) to c). These may be synthesized as follows:

- Resources (man-power) invested in the UMS analysis is not consistent with the needed resources.
- Too much judgement applied for determining the input uncertain parameters and the related ranges of variation without sufficient use of validation experience. Mainly, too much conservatism was introduced to the ranges of uncertain input parameters (except in the IPSN application).
- Use of a 'non-frozen' or not sufficiently verified version of one code.

As a follow-up of UMS, the OECD/CSNI BEMUSE project was promoted and has recently been completed. The inadequacy related to input uncertainty parameters, 2nd bull item above, was also identified in BEMUSE. Then, the OECD/CSNI PREMIUM project has just been launched addressing the issues of selection of input uncertain parameters and determining (possibly by an analytical approach) the related ranges of variation.

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Main Results of the OECD BEMUSE Progamme

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Abstract

The BEMUSE (Best Estimate Methods Uncertainty and Sensitivity Evaluation) Programme promoted by the Working Group on Analysis and Management of Accidents (WGAMA) and endorsed by the Committee on the Safety of Nuclear Installations (CSNI) represents an important step towards reliable application of high-quality best-estimate and uncertainty and sensitivity evaluation methods. The methods used in this activity are considered to be mature for application, including licensing processes. Skill, experience and knowledge of the users about the applied suitable computer code as well as the used uncertainty method are important for the quality of the results.

1. Objectives of the programme

The CSNI BEMUSE programme is focused on the application of uncertainty methodologies to Large Break Loss of Coolant Accident (LB-LOCA) scenarios in Pressurized Water Reactors (PWR).

The objectives of the programme are:
- To evaluate the practicability, quality and reliability of Best-Estimate (BE) methods including uncertainty and sensitivity evaluation in applications relevant to nuclear reactor safety.
- To develop common understanding from the use of those methods.
- To promote and facilitate their use by the regulatory bodies and the industry.

Operational objectives include an assessment of the applicability of best estimate and uncertainty and sensitivity methods to integral tests and their use in reactor applications. The justification for such an activity is that some uncertainty methods applied to BE codes exist and are used in research organisations, by vendors, technical safety organisations and regulatory authorities. Over the last years, the increased use of BE codes and uncertainty and sensitivity evaluation for Design Basis Accident (DBA), by itself, shows the safety significance of the proposed activity. Uncertainty methods are used worldwide in licensing of loss of coolant accidents for power uprates of existing plants, for new reactors and new reactor developments. End users for the results are expected to be industry, safety authorities and technical safety organisations.

2. Main steps

The programme was divided into two main steps, each one consisting of three phases. The first step is to perform an uncertainty and sensitivity analysis related to the LOFT L2-5 test, and the second step is to perform the same analysis for a Nuclear Power Plant (NPP) LB-LOCA) The programme started in January 2004 and was finished in September 2010.

- Phase 1: Presentation “a priori” of the uncertainty evaluation methodology to be used by the participants; lead organization: IRSN, France.
- Phase 2: Re-analysis of the International Standard Problem ISP-13 exercise, post-test analysis of the LOFT L2-5 large cold leg break test calculation; lead organization: University of Pisa, Italy [1].
- Phase 3: Uncertainty evaluation of the L2-5 test calculations, first conclusions on the methods and suggestions for improvement; lead organization: CEA, France [2].
- Phase 4: Best-estimate analysis of a NPP-LBLOCA; lead organization: UPC Barcelona, Spain [3].
- Phase 5: Sensitivity analysis and uncertainty evaluation for the NPP LBLOCA, with or without methodology improvements resulting from phase 3; lead organization: UPC Barcelona, Spain [4].
- Phase 6: Status report on the area, classification of the methods, conclusions and recommendations; lead organization: GRS, Germany [5].

The participants of the different phases of the programme and the used computer codes are given in Table 1.

Table 1 Participants and used codes

No.	Organisation	Country	Code	Participation in Phases
1	AEKI	Hungary	ATHLET2.0A	1, 2, 4, 5
2	CEA	France	CATHARE2V2.5_1	1, 2, 3, 4, 5
3	EDO “Gidropress“	Russia	TECH-M-97	2, 4, 5
4	GRS	Germany	ATHLET1.2C/ 2.1B	1, 2, 3, 4, 5
5	IRSN	France	CATHARE2V2.5_1	1, 2, 3, 4, 5
6	JNES	Japan	TRACE ver4.05	1, 2, 3, 4, 5
7	KAERI	South Korea	MARS 2.3/ 3.1	2, 3, 4, 5
8	KINS	South Korea	RELAP5 mod3.3	1, 2, 3, 4, 5

9	NRI-1	Czech Republic	RELAP5 mod3.3	2, 3, 4, 5
10	NRI-2	Czech Republic	ATHLET2.0A/ 2.1A	1, 2, 3, 5
11	PSI	Switzerland	TRACE v4.05 5rc3	1, 2, 3, 4, 5
12	UNIPI-1	Italy	RELAP5 mod3.2	1, 2, 3, 4, 5
13	UNIPI-2	Italy	CATHARE2V2.5_1	4, 5
14	UPC	Spain	RELAP5 mod3.3	1, 2, 3, 4, 5

3. Used methods

Two classes of uncertainty methods were applied. One propagates "input uncertainties" and the other one extrapolates "output uncertainties".

The main characteristics of the statistical methods based upon the propagation of input uncertainties is to assign probability distributions for these input uncertainties, and sample out of these distributions values for each code calculation to be performed. The number of code calculations is independent of the number of input uncertainties, but is only dependent on the defined probability content (percentile) and confidence level. The number of calculations is given by Wilks' formula [6]. By performing code calculations using variations of the values of the uncertain input parameters, and consequently calculating results dependent on these variations, the uncertainties are propagated in the calculations up to the results. Uncertainties are due to imprecise knowledge and the approximations of the computer codes simulating thermal-hydraulic physical behaviour.

The methods based upon extrapolation of output uncertainties need available relevant experimental data, and extrapolate the differences between code calculations and experimental data at different reactor scales [7]. The main difference of this method compared with statistical methods is that there is no need to select a reasonable number of uncertain input parameters and to provide uncertainty ranges (or distribution functions) for each of these variables. The determination of uncertainty is only on the level of calculation results due to the extrapolation of deviations between measured data and calculation results.

The two principles have advantages and drawbacks. The first method propagating input uncertainties is associated with order statistics. The method needs to select a reasonable number of variables and associated range of variations and possibly distribution functions for each one. Selection of parameters and their distribution must be justified. Uncertainty propagation occurs through calculations of the code under investigation. The "extrapolation on the outputs" method is based on fundamental statistics to derive uncertainties, and needs to have "relevant experimental data" available. In addition, the sources of error cannot be derived as result of application of the method. The method seeks to avoid engineering judgement as much as possible.

In BEMUSE, the majority of participants used the statistical approach, associated with Wilks' formula. Only University of Pisa used its method extrapolating output uncertainties. This method is called the CIAU method, Code with (the capability of) Internal Assessment of Uncertainty. The reason why this method is not used by other participants is the high effort needed to get the data base for deviations between experiment and calculation results in CIAU. That time and resource consuming process has been performed only by University Pisa for the codes CATHARE and RELAP5 for the time being. The data base is available only there.

4. Selected results

4.1 Application to LOFT L2-5 experiment

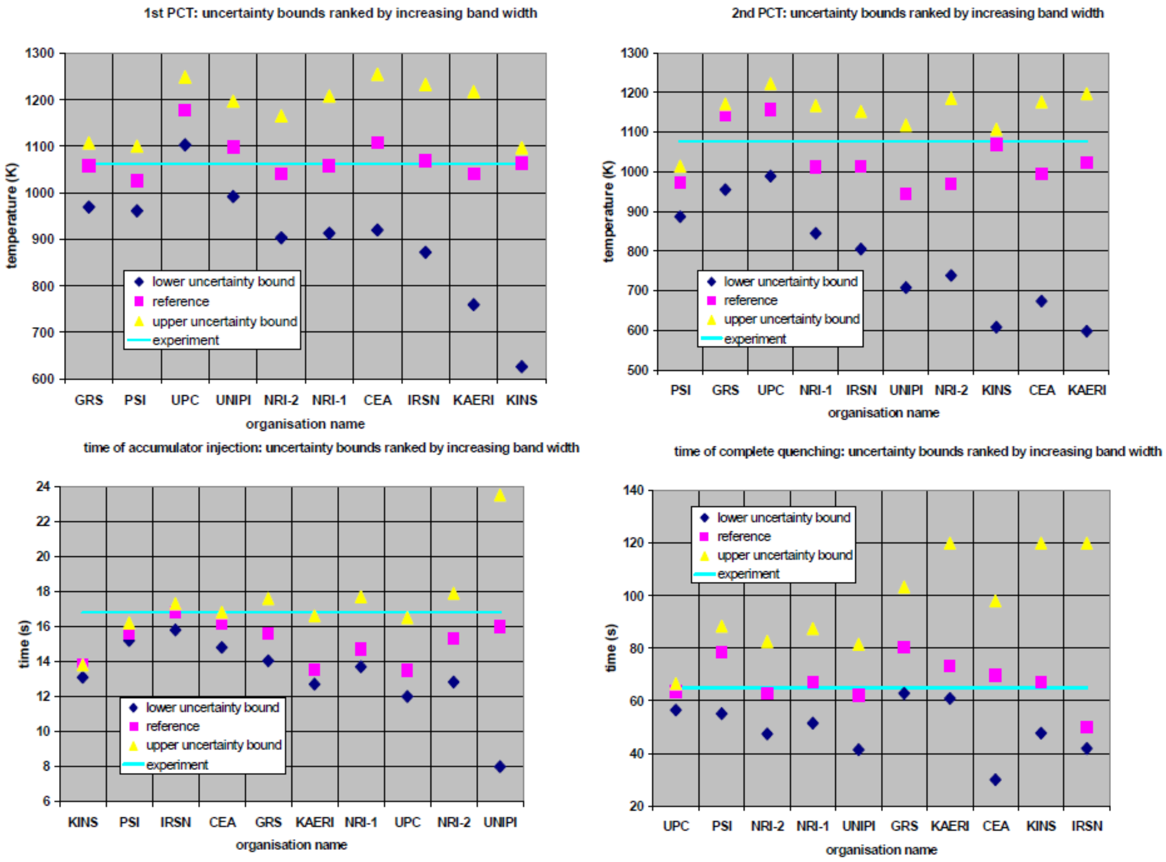
Based on procedures developed at University of Pisa, a systematic qualitative and quantitative accuracy evaluation of the code results have been applied to the calculations performed by the participants for LOFT test L2-5 in BEMUSE phase 2. The test simulated a 2 x 100% cold leg break. LOFT was an experimental facility with nuclear core. A Fast Fourier Transform Based Method (FFTBM) was performed to quantify the deviations between code predictions and measured experimental data [1]. The proposed criteria for qualitative and quantitative evaluation at different steps in the process of code assessment were carefully pursued by participants during the development of the nodalisation, the evaluation of the steady state results and of the measured and calculated time trends. All participants fulfilled the criteria with regard to agreement of geometry data and calculated steady state values.

The results of uncertainty bands for the four single-valued output parameters first peak cladding temperature (PCT), second peak cladding temperature, time of accumulator injection and time of complete quenching for the calculations of the LOFT L2-5 test are presented in Figure 1 [2]. It was agreed to submit the 5/95 and 95/95 estimations of the one-sided tolerance limits, that is, to determine both tolerance limits with a 95% confidence level each. They are ranked by increasing band width. It was up to the participants to select their uncertain input parameters.

The following observations can be made:

- First PCT: The spread of the uncertainty bands is within 138-471 K. The difference among the upper 95%/ 95% uncertainty bounds, which is important to compare with the regulatory acceptance criterion, is up to 150 K and all but one participant cover the experimental value. One participant (UPC) does not envelop the experimental PCT, due to a too high lower bound. Two reasons can explain this result: Among all the participants, on the one hand, UPC has the highest reference value; on the other hand, its band width is among the narrowest ones. KINS attribute their low lower uncertainty bound to a too high value of maximum gap conductance of the fuel rod.
- Second PCT: In this case, one participant (PSI) does not envelop the experimental PCT, due to a too low upper bound. The reasons are roughly similar to those given for the first PCT: PSI, as several participants, calculates a too low reference value, but has also the specificity to consider an extremely narrow upper uncertainty band. The spread of the uncertainty bands is within 127-599 K. The difference among the upper 95%/ 95% uncertainty bounds, which is important to compare with the regulatory acceptance criterion, is up to 200 K.
- Time of accumulator injection: Four participants among ten calculate too low upper bounds (KINS, PSI, KAERI and UPC), whereas CEA finds an upper bound just equal to the experimental value. These results are in relationship with the prediction of the cold leg pressure reaching the accumulator pressure 4.29 MPa. The band widths vary within 0.7-5.1 s for all the participants except for UNIPI which finds a much larger band, equal to 15.5 s. This is mainly due to the consideration of time error for the pressure transient calculated by UNIPI.

Figure 1 Uncertainty analysis results of LOFT L2-5 test calculations for four single-valued output parameters compared with experimental data



Time of complete quenching: All the uncertainty bands envelop the experimental value, even if the upper bound is close to the experimental value for one participant. The width of the uncertainty range varies from 10 s to more than 78 s. If the core is not yet quenched at the end of the calculation as it is the case for two participants (KAERI, KINS), or if there are several code failures before the complete quenching (IRSN), the upper bound is plotted at 120 s in Figure 1.

First suggestions for improvement of the methods have not been proposed as result of the exercise; however, recommendations for proper application of the statistical method were given, see under "Conclusions".

4.2 Application to Zion nuclear power plant

The scope of phase 4 was the simulation of a LBLOCA in a Nuclear Power Plant using experience gained in phase 2. Reference calculation results were the basis for uncertainty evaluation, to be performed in the next phase. The objectives of the activity are 1) to simulate a LBLOCA reproducing the phenomena associated to the scenario, and 2) to have a common, well-known basis for the future comparison of uncertainty evaluation results among different methodologies and codes [3].

The activity for the Zion Nuclear Power Plant was similar to the previous phase 2 for the LOFT experiment. The UPC team together with UNIPI provided the database for the plant, including RELAP5 and TRACE input decks. Geometrical data, material properties, pump information, steady state values, initial and boundary conditions, as well as sequence of events were provided. The nodalisation comprised generally more hydraulic nodes and axial nodes in the core compared with the LOFT applications.

It is important to point out that, as the plant was in permanent shutdown condition from 1998, no detailed information could be made available if needed during the development of the project. In order to work on this problem along with plant parameters, the main features of the LBLOCA scenario were specified in order to ensure common initial and boundary conditions.

Figure 2 Calculated time trends for the Zion NPP

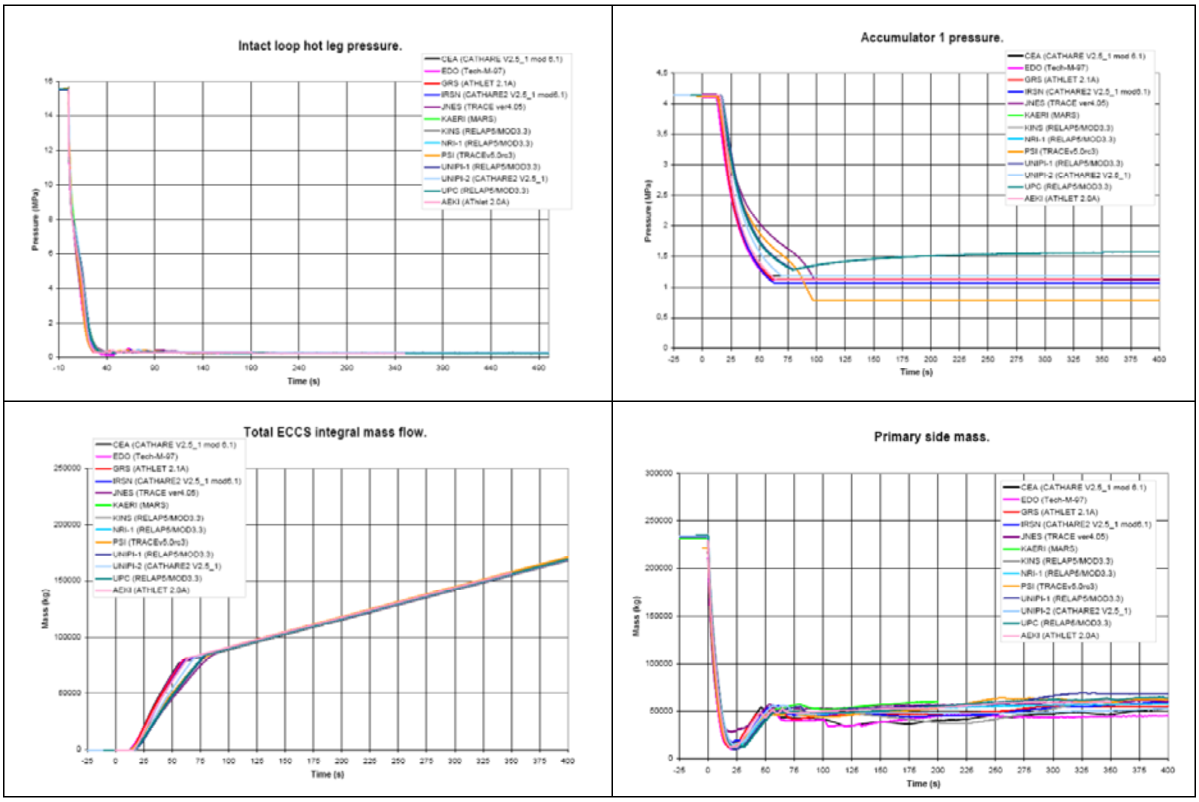
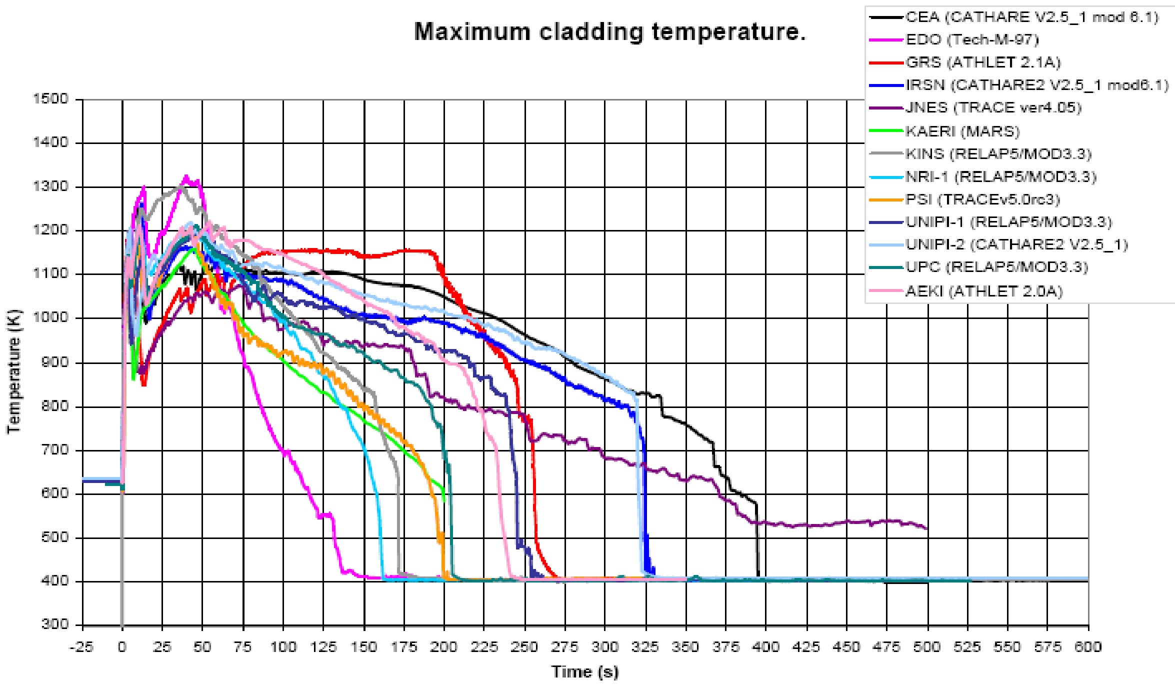


Figure 3 Calculated maximum cladding temperature versus time for the Zion NPP



Results of reference calculations

Most of the events related to the scenario (see Figure 2) are strongly dependent on primary pressure time trend. Despite of the dispersion shown in some of the figures, some events are predicted in a consistent way by participants among these:

- Subcooled blowdown ended
- Cladding temperature initially deviated from saturation (DNB in core)
- Pressurizer emptied
- Accumulator injection initiated
- LPIS injection initiated

Events related to the partial top-down rewet (see Figure 3) need some explanation. After analyzing the corresponding figures, despite of a non-negligible dispersion, the shape of the curves shows some consistency. All participants predict a first PCT, a temperature decrease (at the initiation of the partial rewet) and a further temperature increase (at the end of the partial rewet). These events are not so clearly shown when participants are asked to define a time quantity related to each event but there is a general agreement on the shape of the curves. Clearly the time trend analysis (instead of the simple comparison of the time of occurrence of the events) is the best way to show the discrepancies and similarities among results.

The calculated maximum cladding temperatures versus time are shown in Figure 3. The highest difference in calculated maximum peak cladding temperatures (PCT) between the participants is 167 K (EDO “Gidropress”: 1326, KAERI: 1159 K), what is lower than the difference of BEMUSE phase 2 calculations of the LOFT test.

Results of uncertainty analysis

Phase 5 dealt with a power plant [4], like phase 4. There was no available documentation concerning the uncertainties of the state of the plant, initial and boundary conditions, fuel properties, etc. To solve this situation, it was agreed to provide common information about geometry, core power distribution and modelling. In addition, a list of common input parameters with its uncertainty was prepared. This was done due to the results of phase 3, calculating the LOFT experiment, showing quite a significant dispersion of the uncertainty ranges by the different participants. This list of common uncertain input parameters with their distribution type and range was prepared by the CEA, GRS and UPC teams for the nuclear power plant. These parameters were strongly recommended to be used in the uncertainty analysis when a statistical approach was followed. Not all participants used all proposed parameters. Some considered only those without any model uncertainty. The list is shown in Table 2.

Table 2 Common input parameters associated with a specific uncertainty, range of variation and type of probability density function.

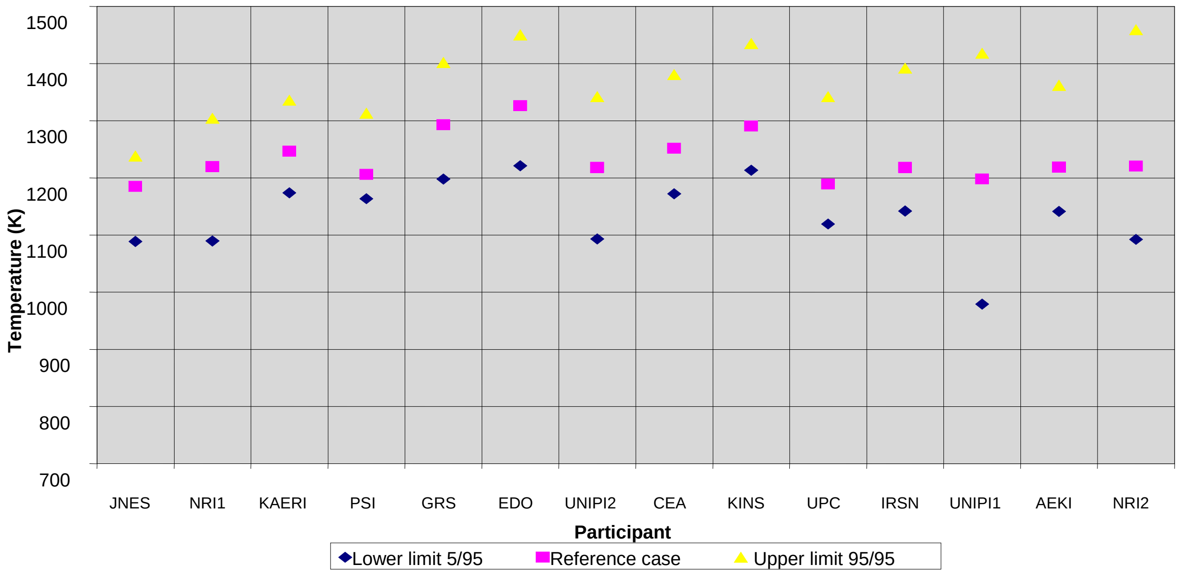
Phenomenon	Parameter	Imposed range of variation	Type of pdf	Comments
Flow rate at the break	Containment pressure	[0.85, 1.15]	Uniform	Multiplier.
Fuel thermal behaviour	Initial core power	[0.98; 1.02]	Normal	Multiplier affecting both nominal power and the power after scram.
	Peaking factor (power of the hot rod)	[0.95; 1.05]	Normal	Multiplier.
	Hot gap size (whole core except hot rod)	[0.8; 1.2]	Normal	Multiplier. Includes uncertainty on gap and cladding conductivities.

Phenomenon	Parameter	Imposed range of variation	Type of pdf	Comments
	Hot gap size (hot rod)	[0.8; 1.2]	Normal	Multiplier. Includes uncertainty on gap and cladding conductivities.
	Power after scram	[0.92; 1.08]	Normal	Multiplier
	UO2 conductivity	[0.9, 1.1] (T_{fuel} <2000 K) [0.8,1.2] (T_{fuel} >2000 K)	Normal	Multiplier. Uncertainty depends on temperature.
	UO2 specific heat	[0.98, 1.02] (T_{fuel} <1800 K) [0.87,1.13] (T_{fuel} >1800 K)	Normal	Multiplier. Uncertainty depends on temperature.
Pump behaviour	Rotation speed after break for intact loops	[0.98; 1..02]	Normal	Multiplier.
	Rotation speed after break for broken loop	[0.9; 1.1]	Normal	Multiplier.
Data related to injections	Initial accumulator pressure	[-0.2; +0.2] MPa	Normal	
	Friction form loss in the accumulator line	[0.5; 2.0]	Log-normal	Multiplier.
	Accumulators initial liquid temperature	[-10; +10] °C	Normal	
	Flow characteristic of LPIS	[0.95 ; 1.05]	Normal	Multiplier.
Pressurizer	Initial level	[-10; +10] cm	Normal	
	Initial pressure	[-0.1; +0.1] MPa	Normal	
	Friction form loss in the surge line	[0.5; 2]	Log-normal	Multiplier.
Initial conditions: primary system	Initial intact loop mass flow rate	[0.96; 1.04]	Normal	Multiplier. This parameter can be changed through the pump speed or through pressure losses in the system.

Phenomenon	Parameter	Imposed range of variation	Type of pdf	Comments
	Initial intact loop cold leg temperature	[-2; +2] K	Normal	This parameter can be changed through the secondary pressure, heat transfer coefficient or area in the U-tubes.
	Initial upper-head mean temperature	[T_{cold} ; T_{cold} + 10 K]	Uniform	This parameter refers to the "mean temperature" of the volumes of the upper plenum.

The main results of the calculated uncertainty bands can be seen for the single valued code results maximum peak cladding temperature in Figure 4. This temperature is defined as the maximum fuel cladding temperature value, independently of the axial or radial location in the active core during the whole transient. It is the main parameter to be compared with its regulatory acceptance limit in LOCA licensing analyses. For comparison purposes it was agreed to submit the 5/95 and 95/95 estimations of the one-sided tolerance limits, that is, to determine both tolerance limits with a 95% confidence level each.

Figure 4 Calculated uncertainty bands of the maximum PCT of Zion NPP LB-LOCA

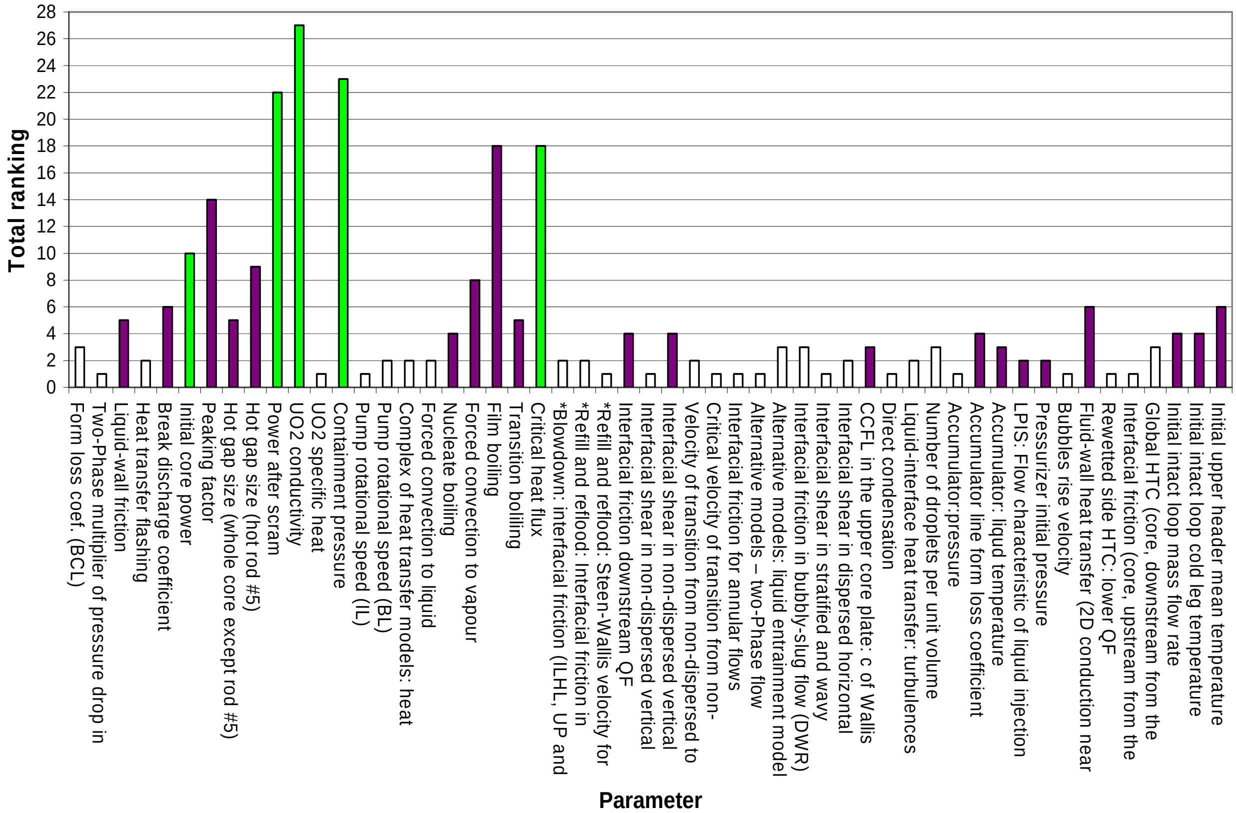


Comparing results for the maximum PCT, there is an overlap region of, roughly, 15K (between 1221K and 1238K). This region is very small. When not including participants with extreme values of the uncertainty bands, it is possible to obtain a better overlap region. EDO, JNES and PSI considered only the proposed common input parameters from Table 2 without model uncertainties.

Results of sensitivity analysis

Sensitivity analysis is here a statistical procedure to determine the influence of uncertain input parameters on the uncertainty of the output parameter (result of code calculations). Each participant using the statistical approach provided a table of the most relevant parameters for four single valued output parameters and for two time trends (maximum cladding temperature and upper-plenum pressure), based on their influence measures. To synthesize and to compare the results of these influences, they are grouped in two main “macro” responses. The macro response for core cladding temperature comprise first, second and maximum peak cladding temperature, maximum cladding temperature as function of time before quenching and time of complete core quenching. The summary of the total ranking by participants is shown in Figure 5. Such information is useful for further uncertainty analysis of a LB-LOCA. High ranked parameters are fuel pellet heat conductivity, containment pressure, power after scram, critical heat flux and film boiling heat transfer.

Figure 5 Total ranking of the influence of input uncertainties on cladding temperature per uncertain input parameter for Zion NPP LB-LOCA



5. Conclusions and recommendations

The methods used in this activity are considered to be mature for application, including licensing processes. Differences are observed in the application of the methods, consequently results of uncertainty analysis of the same task lead to different results. These differences raise concerns about the validity of the results obtained when applying uncertainty methods to system analysis codes. The differences may stem from the application of different codes and uncertainty methods. In addition, differences between applications of statistical methods may mainly be due to different input uncertainties, their ranges and distributions. Differences between CIAU applications may stem from different data bases used for the analysis. However, as it was shown by all BEMUSE phases from 2 through 5, significant differences were observed between the base or reference calculation results.

When a conservative safety analysis method is used, it is claimed that all uncertainties are bounded by conservative assumptions. Differences in calculation results of conservative codes would also be seen, due to the user effect such as different nodalisation and code options, like for best estimate codes used in the BEMUSE programme. Difference of code calculation results have been observed for a long time, and have been experienced in all International Standard Problems where different participants calculated the same experiment or a reactor event. The main reason is that the user of a computer code has a big influence on how a code is used. The objective of an uncertainty analysis is to quantify the uncertainties of a code result. An uncertainty analysis may not compensate for code deficiencies. Necessary pre-condition is that the code is suitable to calculate the scenario under investigation.

Consequently, before performing uncertainty analysis, one should concentrate first of all on the reference calculation. Its quality is decisive for the quality of the uncertainty analysis. More lessons were learnt from the BEMUSE results. These are:

- The number of code runs, which may be increased to 150 to 200 instead of the 59 code runs needed when using Wilks' formula at the first order for the estimation of a one-sided 95/95 limit tolerance. More precise results are obtained, what is especially advisable if the upper tolerance limit approaches regulatory acceptance criteria, e.g. 1200°C PCT.
- For a proper use of Wilks' formula, the sampling of the input parameters should be of type Simple Random Sampling (SRS). Other types of parameter selection procedures like "Latin-Hypercube-Sampling" or "Importance-Sampling" may therefore not be appropriate for tolerance limits.
- Another important point is that all the code runs should be successful. At a pinch, if a number of code runs fail, the number of code runs should be increased so that applying Wilks' formula is still possible. That is the case supposing that the failed code runs correspond to the highest values of the output, e.g. PCT.

In addition to the above recommendations, the most outstanding outcome of the BEMUSE programme is that a user effect can also be seen in applications of uncertainty methods, like in the BEMUSE programme. In uncertainty analysis, the emphasis is on the quantification of a lack of precise knowledge by defining appropriate uncertainty ranges of input parameters, which could not be achieved in all cases in BEMUSE. For example, some participants specified too narrow uncertainty ranges for important input uncertainties based on expert judgement, and not on sufficient code validation experience. Therefore, skill, experience and knowledge of the users about the applied suitable computer code as well as the used uncertainty method are important for the quality of the results.

Using a statistical method, it is very important to include influential parameters and provide distributions of uncertain input parameters, mainly their ranges. These assumptions must be well justified. An important basis to determine code model uncertainties is the experience from code validation. This is mainly provided by experts performing the validation. Appropriate experimental data are needed. More effort, specific procedures and judgement should be focused on the determination of input uncertainties.

This last point is an issue for recommendation for further work. Especially, the method used to select and quantify computer code model uncertainties and to compare their effects on the uncertainty of the results could be studied in a future common international investigation using different computer codes. That may be performed based on experiments. Approaches can be tested to derive these uncertainties by comparing calculation results and experimental data. Other areas are selection of nodalisation and code options. This issue on improving the reference calculations among participants is fundamental in order to obtain more common bands of uncertainties of the results. Discussions are underway to initiate an international activity in this area.

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Discussion of OECD LWR Uncertainty Analysis in Modelling Benchmark

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Abstract

The demand for best estimate calculations in nuclear reactor design and safety evaluations has increased in recent years. Uncertainty quantification has been highlighted as part of the best estimate calculations. The modelling aspects of uncertainty and sensitivity analysis are to be further developed and validated on scientific grounds in support of their performance and application to multi-physics reactor simulations. The Organization for Economic Co-operation and Development (OECD) / Nuclear Energy Agency (NEA) Nuclear Science Committee (NSC) has endorsed the creation of an Expert Group on Uncertainty Analysis in Modelling (EGUAM). Within the framework of activities of EGUAM/NSC the OECD/NEA initiated the Benchmark for Uncertainty Analysis in Modelling for Design, Operation, and Safety Analysis of Light Water Reactor (OECD LWR UAM benchmark). The general objective of the benchmark is to propagate the predictive uncertainties of code results through complex coupled multi-physics and multi-scale simulations. The benchmark is divided into three phases with Phase I highlighting the uncertainty propagation in stand-alone neutronics calculations, while Phase II and III are focused on uncertainty analysis of reactor core and system respectively.

This paper discusses the progress made in Phase I calculations, the Specifications for Phase II and the incoming challenges in defining Phase 3 exercises. The challenges of applying uncertainty quantification to complex code systems, in particular the time-dependent coupled physics models are the large computational burden and the utilization of non-linear models (expected due to the physics coupling).

1. Introduction

The demand for best estimate calculations in nuclear reactor design and safety evaluations has increased in recent years. Uncertainty quantification has been highlighted as part of the best estimate calculations. The modeling aspects of uncertainty and sensitivity analysis are to be further developed and validated on scientific grounds in support of their performance and application to multi-physics reactor simulations. The Organization for Economic Co-operation and Development (OECD) / Nuclear Energy Agency (NEA) Nuclear Science Committee (NSC) has endorsed the creation of an Expert Group on Uncertainty Analysis in Modeling (EGUAM). This Expert Group reports to the Working Party on Scientific issues of Reactor Systems (WPRS). Since it addresses multi-scale/multi-physics aspects of uncertainty analysis, it works in close co-ordination with the benchmark groups on coupled neutronics-thermal-hydraulics simulations and on coupled core-plant problems. The Expert Group also coordinates its activities with the Group on Analysis and Management of Accidents (GAMA) of the Committee on Safety of Nuclear Installations (CSNI).

Within the framework of activities of EGUAM/NSC the OECD/NEA initiated the Benchmark for Uncertainty Analysis in Modeling for Design, Operation, and Safety Analysis of Light Water Reactor (OECD LWR UAM benchmark). The general objective of the benchmark is to propagate the uncertainty through complex coupled multi-physics and multi-scale simulations. The benchmark is divided into three phases with Phase I highlighting the uncertainty propagation in stand-alone neutronics calculations, while Phase II and III are focused on uncertainty analysis of reactor core and system respectively.

The objective of the performed work has been to define, conduct, and summarize an OECD benchmark for uncertainty analysis in best-estimate coupled code calculations for design, operation, and safety analysis of LWRs. Reference systems and scenarios for coupled code analysis are defined to study the uncertainty effects for all stages of the system calculations. Measured data from plant operation are available for the chosen scenarios.

The utilized technical approach has been to establish a benchmark for uncertainty analysis in best-estimate modeling and coupled multi-physics and multi-scale LWR analysis, using as bases a series of well defined problems with complete sets of input specifications and reference experimental data. The objective is to determine the uncertainty in LWR system calculations at all stages of a coupled reactor physics/thermal hydraulics calculation. The full chain of uncertainty propagation from basic data, engineering uncertainties, across different scales (multi-scale), and physics phenomena (multi-physics) are tested on a number of benchmark exercises for which experimental data are available and for which the power plant details have been released. The principal idea is: a) to subdivide the complex system/scenario into several steps or Exercises, each of which can contribute to the total uncertainty of the final coupled system calculation; b) to identify input, output, and assumptions for each step; c) to calculate the resulting uncertainty in each step; d) to propagate the uncertainties in an integral system simulation for which high quality plant experimental data exists for the total assessment of the overall computer code uncertainty.

The general frame of the OECD LWR UAM benchmark consists of three phases with three exercises for each phase:

Phase I (Neutronics Phase)

- Exercise 1 (I-1): “Cell Physics” focused on the derivation of the multi-group microscopic cross-section libraries.
- Exercise 2 (I-2): “Lattice Physics” focused on the derivation of the few-group macroscopic cross-section libraries.
- Exercise 3 (I-3): “Core Physics” focused on the core steady state stand-alone neutronics calculations.

Phase II (Core Phase)

- Exercise II-1: "Fuel Physics" focused on fuel thermal properties relevant for transient performance.
- Exercise II-2: "Time-Dependent Neutronics" focused on stand-alone depletion and kinetics performance.
- Exercise II-3: "Bundle Thermal-Hydraulics" focused on thermal-hydraulic fuel bundle performance.

Phase III (System Phase)

- Exercise III-1: "Core Multi-physics" focused on coupled neutronics/thermal-hydraulics core performance.
- Exercise III-2: "System Thermal-Hydraulics" focused on thermal-hydraulics system performance.
- Exercise III-3: "Coupled Core/System" focused on coupled neutron kinetics thermal-hydraulic core/thermal-hydraulic system performance.
- Exercise III-4: Comparison of Best Estimate calculations with uncertainty quantification vs. conservative calculations.

For each exercise Input (I), Output (O), and propagated Uncertainty (U) parameters are identified. One of the important tasks is identifying the sources of Input (I) uncertainties for each Exercise - which input uncertainties are propagated from the previous exercise and which one are the new ones. Other important parameters to be defined are the Output (O) uncertainties and propagated Uncertainty parameters (U) for each Exercise, which is directly related to the objective of each Exercise. The Output (O) uncertainties are for specified output parameters for each Exercise, used to test (evaluate) the utilized uncertainty method. The propagated Uncertainty parameters (U) are output parameters, which are selected to be propagated further through the follow-up Exercises in order to calculate the overall resulting uncertainty

The OECD LWR UAM benchmark is organized following the modular approach, which allows for a maximum flexibility and benefit to the participants:

a. Participants can participate in the 3 Phases and in all exercises propagating the uncertainties through all stages of coupled reactor physics/thermal hydraulics calculation scheme
b. Alternatively participants can participate in selected exercises (the benchmark team will provide them with the input data following the established format of previous OECD/NEA benchmarks) and just follow the activities in the other exercises
c. There are several types of operating LWRs to be followed in this benchmark representative of a Boiling Water Reactor (BWR), a Pressurized Water Reactor (PWR) and a Water-Water Energy Reactor (VVER). Participants can model one or more reactor types depending on their interests
d. For each Exercise two types of test problems are designed – numerical test problems (provided with reference solutions) for these types of LWR reactors and experimental test problems obtained from publicly available databases.

The selected LWR types are based on previous benchmark experiences and available data, in order to address all industrial issues and participants interests:

a. Representative of operating PWR based on Three Mile Island 1 (TMI-1) Nuclear Power Plant (NPP);
b. Representative of operating BWR based on Peach Bottom-2 (PB-2) NPP;
c. Representative of operating VVER-1000 based on Kozloduy-6 and Kalinin-3 (V1000) NPPs;
d. Representative of Generation III PWRs with UOX (Uranium OXide) and MOX (Mixed Oxide) cores.

The intention is to follow the established calculation scheme for LWR design and safety analysis in the nuclear power generation industry and regulation

2. Discussion of Phase I

Phase I is focused on understanding uncertainties in prediction of key reactor core parameters associated with LWR stand-alone neutronics core simulation. Such uncertainties occur due to input data uncertainties, modelling errors, and numerical approximations. The chosen approach in Phase I is to select/propagate for each exercise the most important contributors which can be treated in a practical manner. The cross-section uncertainty information is considered as the most important source of input uncertainty for Phase I. The cross-section related uncertainties are propagated through the three exercises of Phase I.

The Specification for Phase I, which includes the definitions of Exercise 1 of Phase I (I-1), of Exercise 2 of Phase I (I-2), and of Exercise 3 of Phase I (I-3) was finalized [1, 2].

Exercise I-1 propagates the uncertainties in evaluated Nuclear Data Libraries - NDL - (microscopic point-wise cross sections) into multi-group microscopic cross-sections used as an input by lattice physics codes and associated multi-group covariance matrices. The participants can use any of the major NDLs such as ENDF, JEFF, and JENDL. The development of nuclear data covariance files is in progress in major NDLs. For the purposes of the OECD LWR UAM benchmark the availability of relative covariance data is important for: all relevant nuclides (actinides, fission products, absorbers and burnable poisons, structural materials and etc.) present in the reactor core and reflector regions of LWRs; covering the entire energy range of interest (from 0 to 10 MeV); and all relevant reaction cross-section types. The SCALE-6.0 [3] covariance library is the recommended source of cross-section uncertainties in near term to be used and propagated in the OECD LWR UAM benchmark. In the longer term, when the complete covariance data becomes available together with evaluated nuclear data files, these more consistent data can be used without difficulty since the OECD LWR UAM benchmark establishes a framework for propagating cross-section uncertainties in LWR design and safety calculations.

The SCALE-6.0 covariance library and updated versions of processing tools ANGELO and LAMBDA have been made available for the UAM benchmark work. The SCALE-6.0 covariance library replaces the earlier incomplete SCALE-5.1 covariance library, and it has the following important features:

a. Assumes that the same relative uncertainties can be applied to different evaluated data sets;

b. Approximate uncertainties span the full energy range 0-20 MeV;

c. Approximate uncertainties are included for all the reaction cross-sections for all materials present in LWR cores;

d. Includes uncertainties in the fission spectra, which is very important in multi-group reactor calculations;

e. The 44-group covariance library is “generic” (problem-independent) and thus the participants have to address problem-specific resonance self-shielding effects for performing sensitivity and uncertainty analysis;

f. The SCALE-6.0 44-group covariance library uses "high-fidelity" covariance data from ENDF/B-VII.0, preleased ENDF/B-VII.1, and ENDF/B-VI.8, as well as JENDL 3.3. These data include all energy, reaction, and nuclide correlations specified in the evaluations. For materials with no ENDF/B or JENDL covariance evaluations, approximate data were taken from the "Low-Fidelity" covariance project by BNL, LANL, and ORNL. These data are available from the National Nuclear Data Center website. The approximate covariances have correlations between energy groups, but not reactions or materials.

In order to perform a comparative analysis of the multi-group cross-section uncertainty data obtained after processing within Exercise I-1, test problems are devised or utilized from the previously defined benchmarks and from the available experimental data. The first set of problems consists of two-dimensional fuel pin-cell test problems representative of BWR Peach Bottom (PB)-2, PWR TMI-1, and VVER-1000 Kozloduy-6. These problems are analyzed at Hot Zero Power (HZP) conditions and Hot Full Power (HFP) conditions. To enhance the differences between the three numerical test cell cases (PWR, BWR and VVER) for the Exercise I-1, the HFP case of the BWR is changed to 40% void fraction instead of 0%. Hence the PWR and BWR cases are for square pitch but with different spectrum, while the VVER case is for triangular pitch. In addition to the numerical test cases, experimental test cases 2.13 and 2.19 are utilized based on Kritz data. Figure 1 shows the temperature of different test cases used within the framework of Exercise I-1.

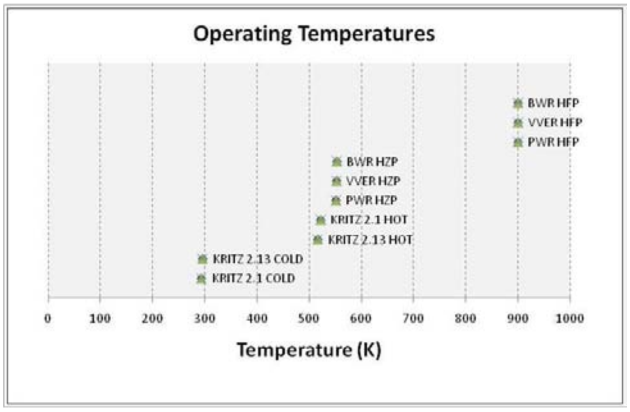

Figure 1. The operating temperature of the reactors and experiments

Representatives of Generation III PWR pin cell test cases have been added also to the Exercise I-1 in the Version 2.0 of the Specification on Phase I [2], including UOX, UOX with Gd_2O_3, and MOX pin cells.

In addition to defining different phases and exercises the benchmark team works on providing reference solutions for the numerical test problems for each exercise as well as conducting comparative analyses for each exercise using numerical and experimental test cases. Such study was conducted for Exercise I-1 utilizing the TSUNAMI-1D sequence of SCALE-6.0 [4]. The uncertainty in k_{eff} seemed to increase with decreasing temperature – see Figure 2. The largest contributor to the uncertainty is due to the ^{238}U (n,γ) reaction. The majority of neutron captures in ^{238}U occurs at intermediate energy and thus as the temperature increases, the neutron spectrum shifts into the epithermal range.

The fuel cells, representative of Generation III PWR reactor designs, were analyzed for the purpose of comparing effect of the material composition on the uncertainty calculations. Three types of unit cells were analyzed at HFP: these include MOX, UOX, and UOX with Gd_2O_3. The multiplication factors and their uncertainties are presented in Table 2.

For each unit cell, the calculated k_{eff} increases with increasing enrichment of fissile material while the uncertainty in k_{eff} decreases. With increasing fissile material, the amount of ^{238}U is reduced. This is later found to be the most important nuclide contributor to the uncertainty in k_{eff} in the unit cell. The uncertainties of the MOX fuel cells are nearly twice than that of UOX fuel. The presence of ^{239}Pu plays an important role in the increase in uncertainty.

Recently an extension, named Pin Cell Burnup Case, was added to the Exercise I-1 with objective to address the uncertainties in the depletion calculation due to the basic nuclear data as well as the impact of processing the nuclear and covariance data [5]. The purpose of this calculation case is to evaluate criticality values, reactions and collapsed cross-sections, and nuclide concentrations computed as well as their uncertainties for depletion in a simple pin-cell model. The proposed fuel test is a typical fuel rod from the TMI-1 PWR 15x15 assembly design.

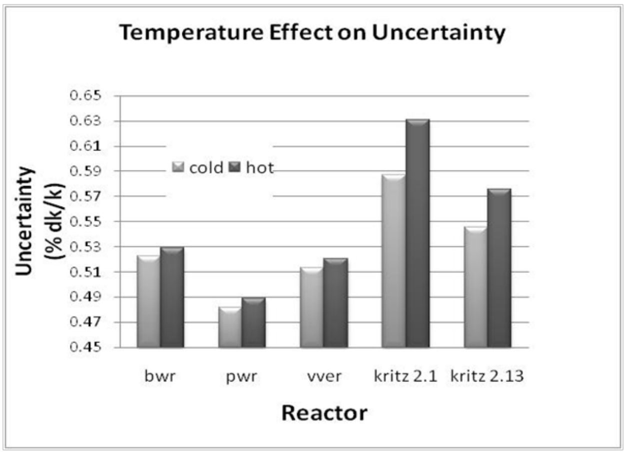

Figure 2. Changes in uncertainty in k_{eff} due to temperature

Table 2. Uncertainty in k_{eff} in PWR Generation III unit cells

Fuel	Compositions	K_{eff}	Uncertainty in K_{eff} (% Δ-k/k)	Largest Nuclide Reaction Cross-section Contributor to Uncertainty
MOx	9.8% ^{239}Pu	1.095097	0.9398	^{238}U (n,n')
	6.5% ^{239}Pu	1.057435	0.9651	^{238}U (n,n')
	3.7% ^{239}Pu	1.014537	0.9877	^{238}U (n,n')
UOx	4.2% ^{235}U	1.245378	0.5115	^{238}U (n,γ)
	3.2% ^{235}U	1.176433	0.5367	^{238}U (n,γ)
	2.1% ^{235}U	1.051125	0.5910	^{238}U (n,γ)
$UOxGd_2O_3$	2.2% ^{235}U	0.216013	1.7667	^{238}U (n,n')
	1.9% ^{235}U	0.199026	1.9334	^{238}U (n,n')

In the current established calculation scheme for LWR design and safety analysis, the multi-group microscopic cross-section libraries are an input to lattice physics calculations to generate few-group assembly-homogenized macroscopic cross-section libraries. The multi-group cross-section uncertainties (multi-group cross-section covariance matrix) are propagated to few-group cross-section uncertainties (few-group cross-section covariance matrix) within the framework of Exercise I-2. In Exercise I-2 manufacturing and geometry (technological) uncertainties are added to account for them in lattice physics calculations – see Table 3.

SCALE 6.1 with Generalized Perturbation Theory (GPT) implemented is now available for the participants [6]. Few-group covariance matrix can be obtained using GPT through the NEWT module of SCALE-6.1. The first set of test problems for Exercise I-2 consists of test problems representative of BWR PB-2, PWR TMI-1, and VVER-1000 Kozloduy-6 defined on assembly spatial scale – see Figure 3. These problems are analyzed at HZP conditions and HFP conditions to account for spectrum changes. For BWR case also different void fraction conditions are considered. PWR Generation III UOX and MOX assembly cases have been added also recently - see Figure 4. Continuous energy Monte Carlo calculations provide reference solutions. The second set of problems is based on publically available experimental data.

Table 3. Manufacturing uncertainties for TMI-1

Parameter	Distribution	Mean	Sigma
Fuel density	Normal	10.283 g/cm^3	0.05666666 g/cm^3
Fuel pellet diameter	Normal	0.9391 cm	0.000433333 cm
Gap thickness	Normal	0.0186 cm	0.0008 cm
Clad thickness	Normal	0.0673 cm	0.0008333333 cm
U235 enrichment	Normal	4.85 %	0.07466 %

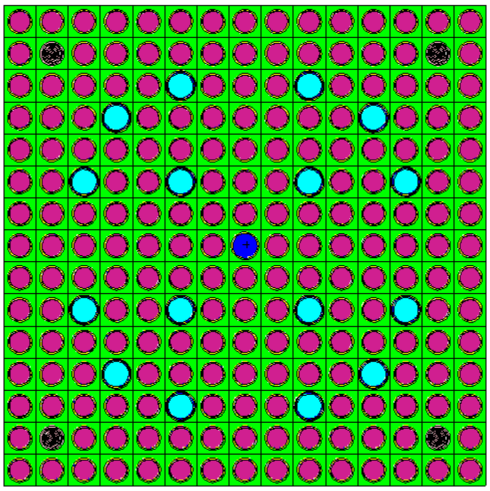

Figure 3. TMI MCNP5 Assembly Model

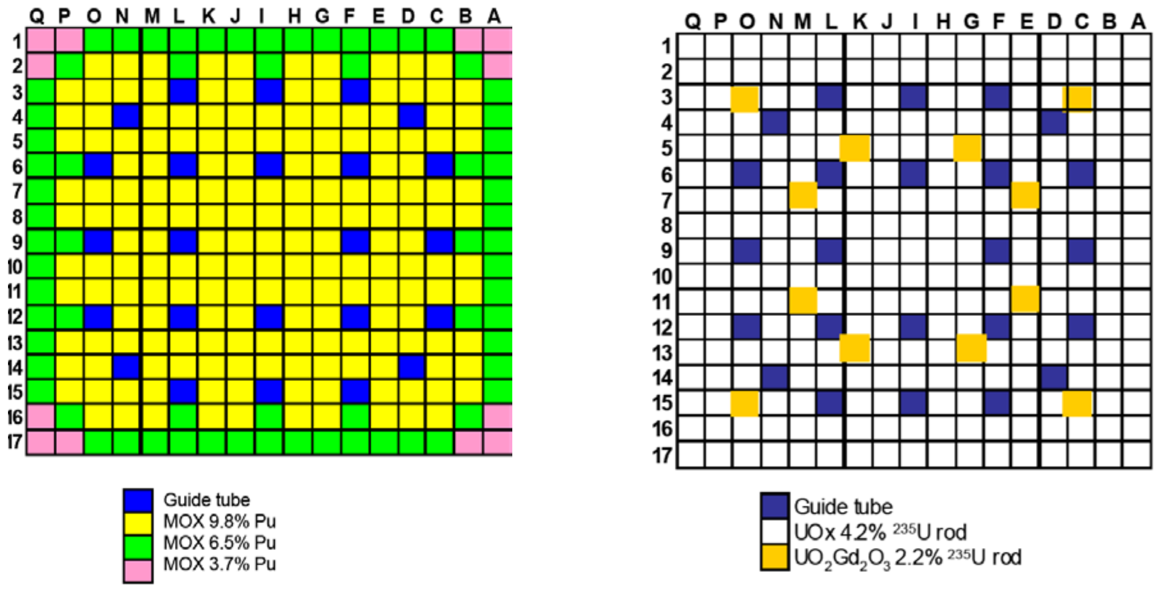

Figure 4. Specification of PWR Generation III assemblies

In the current LWR calculation scheme the lattice averaged (homogenized) few-group cross-sections are an input to core calculations The few-group cross-section uncertainties (few-group covariance matrix) are obtained by participants as output uncertainties within the framework of Exercise I-2. In Exercise I-3 the few-group cross-section uncertainties are input uncertainties and are propagated to uncertainties in evaluated stand-alone neutronics core parameters. The propagation of the input uncertainties through core calculations to determine uncertainties in output core parameters within the framework of Exercise I-3 requires utilization of a core simulator code, and as the benchmark experience shows statistical sampling techniques. The primary differences between the different uncertainty analysis methods are the moment at which the perturbations are applied and the origin of the covariance matrix. One approach is to sample the multi-groups covariance matrix while in the second approach the few-group covariance matrix is then sampled.

Test problems on two different levels are defined to be used for Exercise I-3:

a. HZP test cases based on the realistic LWR designs – TMI-1, PB-2 and Kozloduy-6 (for which the continuous energy Monte Carlo method is used for reference calculations) – see Figure 5. Two

Generation II PWR core test cases are considered – a UOX core and mixed UOX/MOC core – see Figure 6.

b. Documented experimental benchmark plant cold critical data and critical lattice data

In summary, the Phase I of the OECD LWR UAM benchmark is focused on stand-alone neutronics core calculations and associated prediction uncertainties. Concerning the relevance of the experimental data used for Phase I, the desire has been to utilize experiments, which were conducted as close as possible to actual reactor conditions. Phase I does not analyze uncertainties related to cycle and depletion calculations. No feedback modelling is assumed. It addresses the propagation of uncertainties associated with few-group cross-section generation, but it does not address cross-section modelling (it will be addressed in the following Phases).

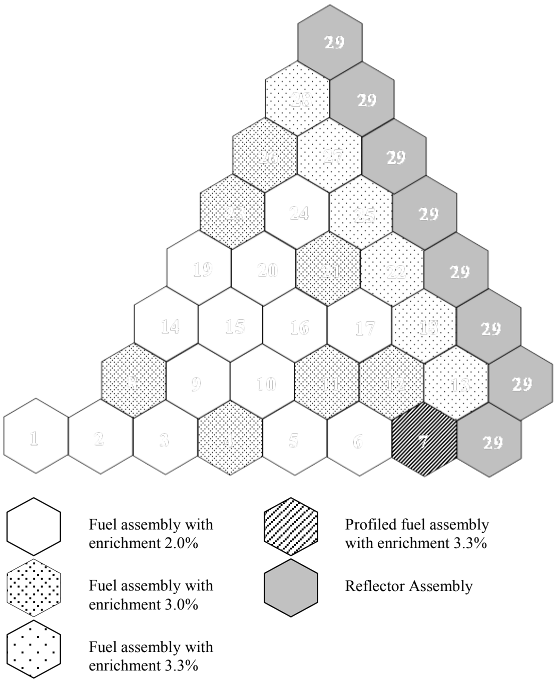

Figure 5. VVER-1000 Core

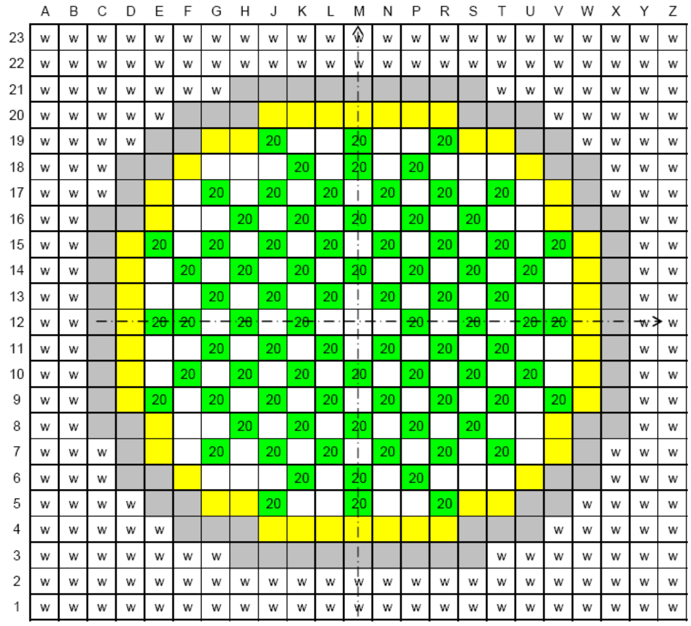

Figure 6. Generation III PWR MOX core

In order to investigate the uncertainties of kinetics parameters the SNEAK (fast core problem) was added as an optional test case to the test problems for Exercise I-3 since it has a unique set of experimental data for β_{eff} uncertainties and can be used as an example on how to calculate uncertainty in β_{eff}. The two high-quality reactor physics benchmark experiments, SNEAK-7A and 7B (Karlsruhe Fast Critical Facility) are part of the International Reactor Physics Benchmark Experiments (IRPhE) database [7]. It was demonstrated that the energy field responsible for β_{eff} uncertainty is the same for fast and thermal reactors i.e. the SNEAK cases are relevant to the any kinds of kinetic parameters calculations validation.

Table 4. Measured kinetics parameters and associated uncertainties

Parameters	SNEAK 7A	SNEAK 7B
Λ, μs	0.180	0.159
Λ, uncertainty	4.15%	3.21%
β	0.00395	0.00429
β, direct uncertainty	2.0 %	2.5%
β, uncertainty	2.4%	N/A

Table 5. Participation in Phase I

Organization	Country	Exercise I-1	Exercise I-2	Exercise I-3
PSU/UPC	USA/Spain	X	X	X
MkMaster/UPisa	Canada/Italy	X	X	
ORNL	USA	X	X	X
GRS	Germany	X	X	X
PSI	Switzerland	X	X	
VTT	Finland	X	X	
UPM	Spain	X	X	
KFKI	Hungary	X	X	
CEA	France	X	X	X
UM	USA	X	X	X
TUM	Germany	X		
NECSA	South Africa	X	X	
JNES	Japan	X	X	
LPSC/CNRS-PM	Canada/France	X		

3. Discussion of Phase II

The obtained output uncertainties from Phase I of the OECD LWR UAM benchmark are utilized as input uncertainties in the remaining two phases – Phase II (Core Phase) and Phase III (System Phase). Phase II addresses time-dependent neutronics, core thermal-hydraulics and fuel performance, without any coupling between the three physics phenomena. In other words the Phase II takes into account other physics involved in reactor simulation, i.e. Thermal-Hydraulics and Fuel Physics and introduces time-dependence. For Phase II - Core Phase it is suggested to start on all exercises simultaneously i.e. each participant can start working on any exercise of Phase II – there is no a recommended sequential order as in Phase I. The Version 1.0 of the specification for Phase II has been prepared [8].

The Exercise II-1 is focused on determining uncertainties in fuel temperature, which is important for Doppler feedback modelling. This exercise is divided in two parts:

a. Steady State - Exercise II-1a;

b. Transient - Exercise II-1b.

The Exercise II-2 addresses uncertainties in time-dependent reactivity insertion, total power evolution and power peaking factors, exposure and isotopic content. It is also subdivided in two parts:

a. Depletion – Exercise II-2a;

b. Neutron Kinetics – Exercise II-2b.

The Exercise II-3 deals with uncertainties in moderator temperature, density and void fraction, which are related to the moderator feedback. It consists of two sub-exercises:

a. Steady State – Exercise II-3a;

b. Transient – Exercise II-3b.

The test cases for Exercise II-1 are assembled from the International Fuel Performance Experiments Data Base (IFPE) at OECD/NEA and from the IAEA Co-ordinated Research Programme on Fuel Modelling at Extended Burnup - FUMEX. Various uncertainty sources are accounted for in Exercise II-1 (Manufacturing, Modelling and Boundary conditions). The specified manufacturing and geometry (technological) uncertainties for Exercise II-1 are consistent with those specified for Exercise I-2. It is important to take into account the uncertainty in gap conductance.

Additional sub-exercise is added to Exercise II-2 – Exercise II-2a – Standalone neutronics depletion to study the propagation of uncertainties in burn-up (depletion) calculation on fuel assembly level.

The test cases for Exercise II-3 are numerical (following TMI-1, PB-2, VVER-1000 fuel bundles) and experimental test cases (using the databases of OECD/NRC BFBT and PSBT benchmarks). The experimental test cases are designed to test uncertainty and sensitivity methods on similar designs (as close as possible to the designs to be followed) by comparisons with measured data with uncertainties. For Exercise II-3 the data bases of the of OECD/NRC BFBT and PSBT benchmarks are utilized for PWR and BWR test cases [9, 10]. For example for PWR bundle test cases the uncertain Input (I) parameters include boundary conditions, power shapes, geometry, and modelling parameters:

1. Modeling parameters used in the codes as:
   - Friction factors (single-phase factor, two-phase multiplier, heating corrector);
   - Single-phase and two-phase heat transfer coefficients;
   - Boiling/condensation and interfacial mass transfer factors;
   - Turbulence and mixing coefficients;
   - Void drift model parameters;
   - Spacer loss coefficient; etc.
2. Geometry effects as:
   - Diameter of the heated rod;
   - Thickness of the cladding;
   - Flow area of the sub-channel;
   - Wetted perimeter of the sub-channel, etc.
3. Boundary condition effects as:
   - Mass flow rate;
   - Inlet fluid temperature (or inlet sub-cooling);
   - System pressure;
   - Power.

The measurement uncertainties of void fraction for the PWR bundles are given in Table 7. Uncertainties for void fraction are under-estimated since these uncertainties cover only the measurement technique but do not include the bias due to the geometry.

The Output (O) parameters are pressure drop, CHF/DNB, moderator density, temperature and void distribution while the Propagated uncertainty parameters (U) are moderator density, temperature and void distribution. The utilized Assumptions (A) are – stand-alone thermal-hydraulic steady state and transient modeling.

Table 7. Estimated Accuracy for Void Fraction Measurements

Quantity	Accuracy
Void fraction measurement	
CT measurement	
Gamma-ray beam width	1 mm
Subchannel averaged (steady-state)	3% void
Spatial resolution of one pixel	0.5 mm
Chordal measurement	
Gamma-ray beam width (center)	3 mm
Gamma-ray beam width (side)	2 mm
Subchannel averaged (steady-state)	4% void
Subchannel averaged (transient)	5% void

4. Discussion of Phase III

Phase III will include system thermal-hydraulics and coupling between fuel, neutronics and thermal-hydraulics for steady-state, depletion and transient analysis. Figure 7 shows the envisioned interactions between input and output parameters in Exercise III-1.

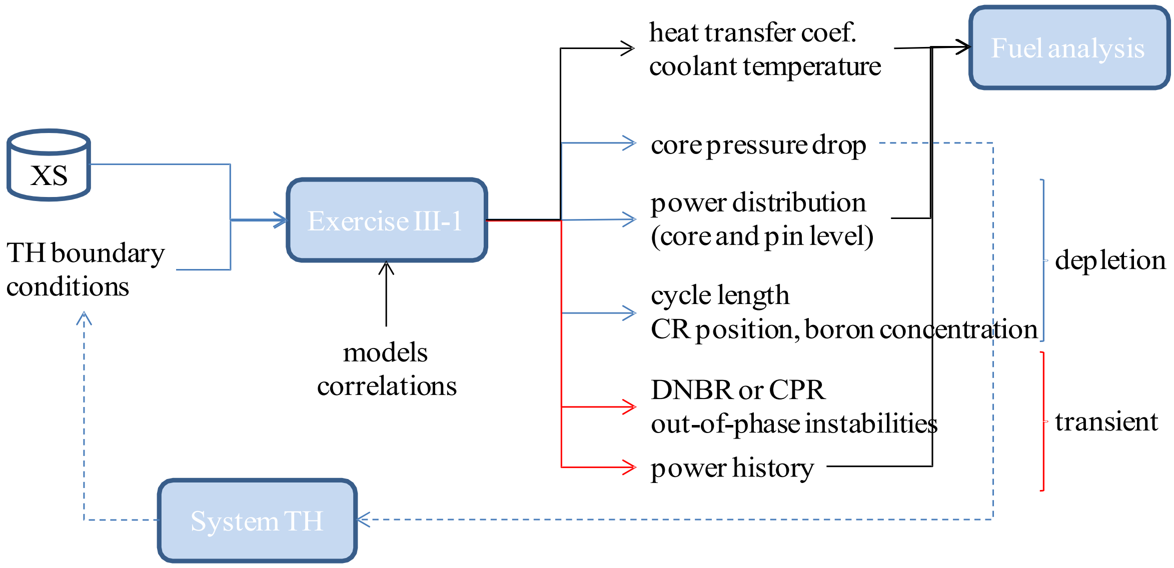

Figure 7. Main input and output parameters in Exercise III-1

At the last OECD LWR UAM Benchmark workshop it was decided to move the time of conducting Exercise III-2 "System Thermal-Hydraulics" (which is system simulation without coupling with 3D coupled core thermal-hydraulics and neutronics) in parallel to Phase II activities as well as to add one more exercise III-4 – Comparison of Best Estimate calculations with uncertainty quantification vs. conservative calculations [11]. Benchmark participants have also agreed to rank transients (to be analyzed in Phase III) according to their complexity (models and uncertainties) and to their relevance for the industry and regulation in order to select the final targets of Phase III.

Interactions and cooperation have been established between the LWR UAM benchmark team, with help of the OECD/NEA secretariat, and the following groups:

a. OECD/NEA NSC Expert Group on Fuel Performance;

b. OECD/NEA NSC Expert Group on Uncertainty Analysis for Criticality Safety Assessment;

c. OECD/NEA NSC Expert Group on Burn-up Credit Criticality Safety;

d. OECD/NEA CSNI BEMUSE Project;

e. IAEA CRP on Uncertainty Analysis in HTGR Modelling.

5. Conclusions

The planning for the OECD LWR UAM benchmark activities envisions completing the benchmark activities on Phase I in the 2011-2012 time framework and beginning in 2011 in parallel the activities on Phase II.

The expected results of the OECD LWR UAM project are:

a. Systematic consideration of uncertainty and sensitivity methods in all steps. This approach will generate a new level of accuracy and will improve transparency of complex dependencies;

b. Systematic identification of uncertainty sources;

c. All results will be represented by reference results and variances and suitable tolerance limits;

d. The dominant parameters will be identified for all physical processes;

e. Support of the quantification of safety margins;

f. The experiences of validation will be explicitly and quantitatively documented;

g. Recommendations and guidelines for the application of the new methodologies will be established;

h. Experience on sensitivity and uncertainty analysis. Several methods will be used and compared:
   - Deterministic methods;
   - Statistical sampling techniques;
   - Hybrid methodologies.

i. New developments such as Adapted Global Sensitivity Analysis and Non-linear ESM approach to Hessian Matrix Construction.

The benefits of participation for different organizations in this activity are seen as:

a. To develop, propose and/or validate advanced uncertainty and sensitivity methodology;

b. Have access to different techniques in sensitivity / uncertainty analysis;

c. Compare and exchange of know-how, resolve difficulties with the world experts;

d. Improve understanding of model validity and their limitation;

e. Provide evidence to model simplification;

f. Have access to high quality integral experiments from experimental facilities and operating power reactors;

g. Acquire competence in quantifying confidence bounds for physics and safety parameters in best estimate methods required for licensing.

It is expected that the application of coupled codes for reactor design and safety analyses will be continuously growing. In fact, they are the only means to perform best-estimate calculations for accident conditions with a tight coupling of neutronics and thermal-hydraulics effects. The current tendencies in coupled code developments are towards systematic integration of uncertainty and sensitivity analysis with simulations for safety analysis. The OECD LWR UAM benchmark activity is designed to address current regulation needs and issues related to practical implementation of risk informed regulation. Establishing such internationally accepted LWR UAM benchmark framework offers the possibility to accelerate the licensing process when using coupled best estimate methods.

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PREMIUM – Benchmark on the quantification of the uncertainty of the physical models in the system thermal-hydraulic codes

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ABSTRACT

PREMIUM (Post BEMUSE Reflood Models Input Uncertainty Methods) is an activity launched with the aim to push forward the methods of quantification of physical models uncertainties in thermal-hydraulic codes. It is endorsed by OECD/NEA/CSNI/WGAMA.

The benchmark PREMIUM is addressed to all who applies uncertainty evaluation methods based on input uncertainties quantification and propagation. The benchmark is based on a selected case of uncertainty analysis application to the simulation of quench front propagation in an experimental test facility. Application to an experiment enables evaluation and confirmation of the quantified probability distribution functions on the basis of experimental data. The scope of the benchmark comprises a review of the existing methods, selection of potentially important uncertain input parameters, preliminary quantification of the ranges and distributions of the identified parameters, evaluation of the probability density function using experimental results of tests performed on FEBA test facility and confirmation/validation of the performed quantification on the basis of blind calculation of Reflood 2-D PERICLES experiment.

Introduction

The identification of the input uncertainties and probabilistic quantification of their uncertainty are essential for the uncertainty and sensitivity analyses. The results of BEMUSE project have shown that the input uncertainties quantification is of great weight for BEPU (Best Estimate Plus Uncertainty) analyses [1]. Whereas, the quantification of uncertainties regarding to the analysed facility and initial and boundary conditions of the transient is mainly the question of information that usually can be obtained; the quantification of physical models uncertainty is an extensive process involving experimental data, numerical methods, expert judgement. Since, the model uncertainties are frequently the dominant factors in uncertainty analyses a special attention has to be paid to the methodologies applied for their quantification.

The basis for physical model uncertainties quantification is evaluation of separate effect experiments. Other information sources are experience from code validation, survey of expert state of knowledge, published data about model uncertainties and if necessary application of theoretical limitation. The preferable way of model uncertainties quantification is the comparison of code predictions with experimental data from separate effect tests. However, there are some phenomena where no single effect tests are available. An example of such a phenomenon is reflooding of the reactor after core dry out. On the basis of selected reflooding tests a benchmark has been defined with the aim to push forward the quality of physical model uncertainties quantification. The main objective of the benchmark is to compare existing approaches, estimate their quantification capabilities and to find out proper methods to quantify uncertainties of models describing phenomena for which no single effect tests exist. The additional benefit of the benchmark is the evaluation of uncertainties quantification performed by participants for reflooding modelling in their codes. Reflooding is one of the key issues of LOCA simulation and even participants using engineering judgement only would profit from the validation of their probability distribution functions.

Since, the PREMIUM is started only a short time before and no results are available, the presentation will concentrate on the scope and methodologies that are going to be applied in the frame of the benchmark.

Quantification of physical model uncertainties

Quantification of the model uncertainties can be performed in different ways. It is generally accepted that for those phenomena for which separate effect tests exist the quantification can be performed comparing results of code models with experimental data for selected phenomena. For each phenomenon for which uncertainties are to be quantified, a stationary separate effect experiment should be found where the phenomenon could be clearly identified and uncertainties of physical models addressing these phenomena can be quantified directly. Such an experiment will allow quantifying input uncertainties of physical models describing specific phenomenon directly comparing calculation results and measurements.

For phenomena for which no single effect tests exist but there are available so called “intermediate” experiments, the situation is more complicated. In fact there is no method which can be clearly preferred. Possibilities are, e.g. statistical methods like CIRCE (CEA) [2], [3], KIT method [4], [5] or newly proposed FFTBM approach (Uni Pisa) [6], “trial-and-error” method by performing calculations of selected experiments related to the investigated phenomenon, definition of “biases”, evaluation of information and experience from model development and may be others.

It seems to be generally accepted that a “good practice” of model uncertainty quantification is to apply preferably separate effect tests and only if unavoidable engineering judgement.

Methodology of uncertainties quantification on the basis of separate effect tests can be regarded already as “standard”. Therefore, the main goal of the benchmark is to compare existing approaches, if possible define new approaches, identify necessary improvements for each approach and finally give recommendations in particular for quantification of uncertainties of models describing phenomena for which no single effect tests exist.

Search of adequate experiments for quantification of models for which no single effect tests are available

First of all an experiment had to be found which involves phenomena that could be clearly identified and physical models responsible for these phenomena could be quantified on the basis of other experiments. It has been looked for an experiment where some phenomena could be quantified on the basis of separate effect tests but for some other phenomena such tests do not exist and more sophisticated methods have to be applied. It is for example the case of reflooding experiments. For code models calculating enhancement of heat transfer very close to the quench front and the relative velocity downstream from the quench front separate effect test do not exist. The influence of these models can only be seen at their effect on quench front propagation which is measured. Sophisticated methods are needed to determine these input uncertainties.

For performing a successful check of model uncertainty quantification, it is necessary that other potentially important uncertainties, like uncertainties of spatial modelling can be eliminated. This is the case for relatively simple test facilities, where 1-D approximation is suitable and no particular problems should arise by discretization and development of an input data set.

In this context reflooding experiments appears to be suitable for the benchmark application:

- Some reflooding tests are available.
- Geometry of the test section is quite simple and average experienced user should not have any problem with its correct simulation.
- In reflooding only a few physical phenomena are involved, it means it should be possible to identify the reason of differences comparing the results of different calculations (participants) with experimental data.

For the confirmation/validation phase in the proposed benchmark PERICLES [7] experiment has been chosen. FEBA/SEFLEX [8] experiment was proposed as support for input uncertainties quantification. Both tests cover similar field of application. PERICLES test facility consists of test section with larger number of fuel rod simulators. Additionally, at PERICLES effect of radial power distribution is investigated. The selected test sequence follows typical way of uncertainty analysis. Input uncertainties are quantified on the basis of test facilities and applied for reactor geometry of the much larger scale. At PERICLES test facility parallel to tests with radial power distribution, tests with uniform radial power distribution were performed, where the 2D effect should be minimized. Therefore, it should be possible to separate this effect by analysis of benchmark results.

Outline of the benchmark

Phase 1 – Introduction and methodology review

Phase 1 is dedicated to the benchmark introduction. The final specification of the benchmark and the time schedule should be fixed. The detailed specification of FEBA test facility and reflooding experiments performed at this test facility will be supplied to the participants. The participants who have at their disposal methodology for quantification of uncertainties of the physical models should present them. The other participants will get with it an overview of the available methods and obtain possibility to choose one of them for application. They can also apply other ways of input uncertainties quantification like making use of modelling accuracy estimation from model development and validation or apply expert judgement and then evaluate initial uncertainties in the step 3 on the basis of reflood experiments. Additionally, alternative experiments to the FEBA/SEFLEX are to be presented, when some participants prefer to use them for quantification of reflooding uncertainties in their codes.

Phase 2 – Identification of important input uncertainties and their initial quantification

The reflooding is a frequently considered phenomenon in the nuclear reactor safety analyses. It can be expected, that majority of participants, if not all of them, performed already BEPU analyses of an accident where reflooding was one of the most important phenomena.

Phase 2 is in fact the first step of the main part of the benchmark. In this step the participants should identify and perform preliminary quantification of the potentially important uncertain input parameters regarding FEBA experiment or any other equivalent reflooding experiment. This process composes of following steps:

- Identification of influential phenomena
- Identification of the associated physical models and parameters depending on the used code
- Preliminary quantification of the identified input uncertainties

The preliminary quantification of the model uncertainties is expected to be done by all participants independently in a usual manner, as it would be done by a typical application. The preferred way of the model uncertainties quantification is comparison with separate effect tests, if such tests are available.

Phase 3 – Evaluation of the physical model uncertainties

This phase is particularly addressed to the models related to the phenomena for which separate effect tests do not exist, as it is the case for heat transfer enhancement at the quenching front.

The participants, who have in their disposal a tool for quantification of input uncertainties on the basis of intermediate experiments, should apply it using measured data from the FEBA/SEFLEX or another experiment. Other participants should control their preliminary defined uncertainty ranges (defined in the phase 2) by comparison of results of calculations with varied uncertain input parameters with measured data. This should enable to improve uncertainty ranges estimated in previous step, e. g.: by engineering judgement, or confirm their validity for the reflooding experiment.

For the evaluation step the tests from the FEBA/SEFLEX program have been chosen. However, if any participants prefer, they can use their own reflood experiment, provided that it is sufficiently validated and they accept to make their experimental results available for other participants.

The test runs of the FEBA experiment seems to be sufficient for performing the quantification (or evaluation) of the selected input uncertainties. If any participants found that more experimental data are needed or they would like to extend the evaluation on fuel rod simulators of different construction (with gap and different cladding material) the measured data of SEFLEX experiment can be used in addition.

FEBA/SEFLEX Program

The purpose of the FEBA/SEFLEX program was to obtain an insight into the most important heat transfer mechanisms during reflood phase of LOCA and to broaden the data base for the development and assessment of improved thermal-hydraulics models. The FEBA/SEFLEX program has been performed at KfK Karlsruhe, Germany. The FEBA and SEFLEX experiment were published inclusive experimental results. A detailed description of the FEBA experiment series I and II can be found in KfK reports: KfK 3657 [8] and 3658 [9]. Description of SEFLEX experiment with unblocked arrays in KfK reports: KfK 4024 [10] and KfK 4025 [11].

The test facility was designed for the reflooding tests with possibility of maintaining constant flooding rates and constant back pressure.

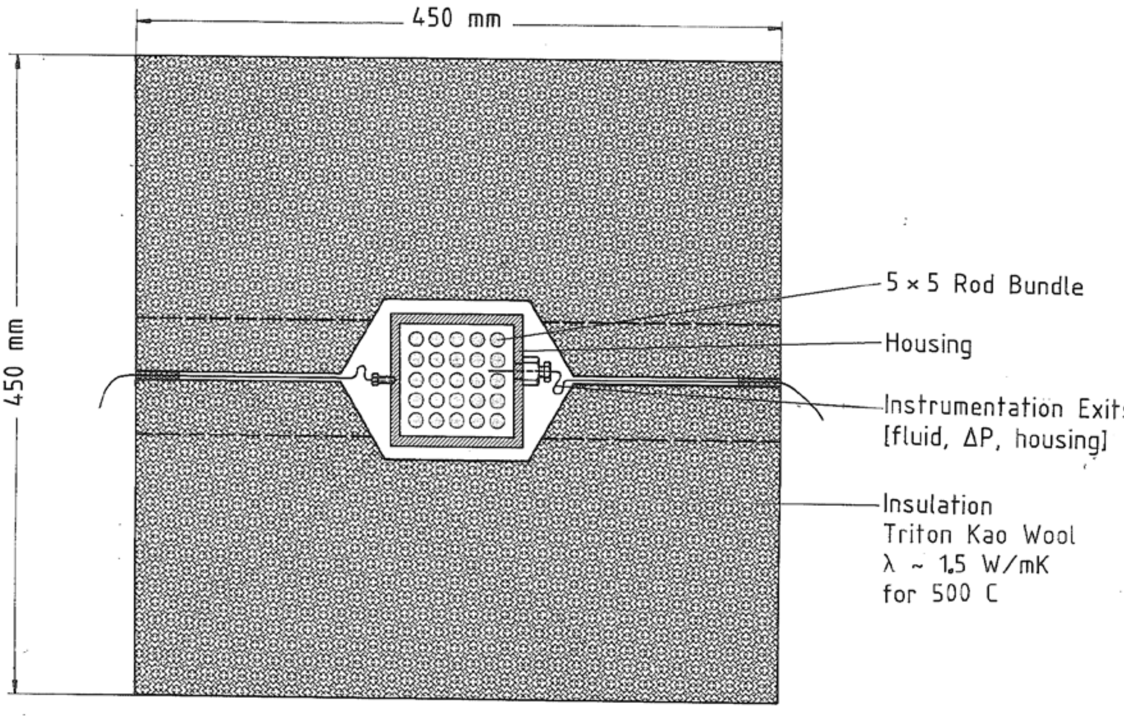

Figure 1. FEBA test section – cross-section view.

FEBA Experiment

The FEBA experiment (Flooding Experiments with Blocked Array) consisted of tests performed on a full-length 5 x 5 rod bundle of PWR fuel rod dimensions utilizing electrically heated rods with a cosine power profile approximated by 7 steps of different power density in axial direction. The cross-section of the FEBA test section is shown in the Figure 1 and the cross-section of the heater road in the Figure 2.

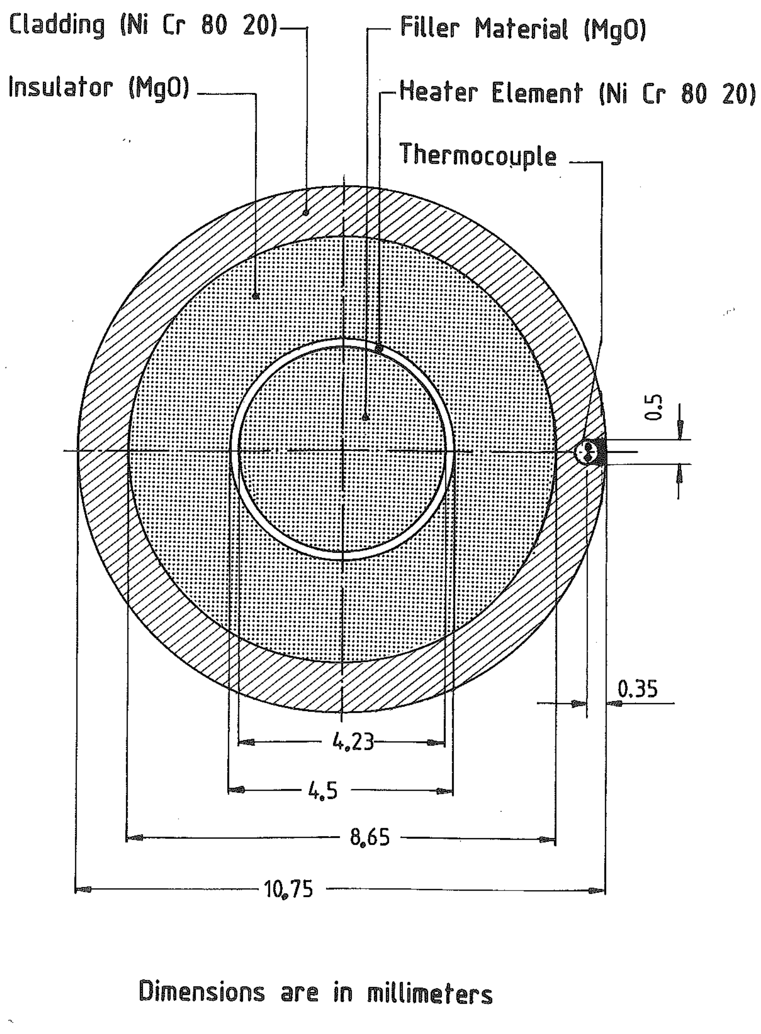

Figure 2. Cross section of the FEBA heater rod

Eight test series were performed under idealized reflood conditions using forced feed and system pressure as fixed boundary conditions, without taking into account the effects of reactor cooling system. The tests series were conducted to study the effect of grid spacers and of coplanar blockages with different blockage rates.

The series I and II (test runs are listed in the Tables 1 and 2) were performed for unblocked rod arrays and they are suitable as a basis for quantification of uncertainties of flooding simulation in PERICLES test facility.

Table 1

Series I: Base line tests with undisturbed bundle geometry with 7 grid spacers

Test No.	Inlet velocity (cold), cm/s	System pressure, Bar	Feed water temperature, °C		Bundle power, kW	
			0-30 s	End	0 s	Transient
223	3.8	2.2	44	36	200	120% ANS*
216	3.8	4.1	48	37	200	120% ANS
220	3.8	6.2	49	37	200	120% ANS
218	5.8	2.1	42	37	200	120% ANS
214	5.8	4.1	45	37	200	120% ANS
222	5.8	6.2	43	36	200	120% ANS

* – 120% of decay heat according to ANS curve, is measured value in all test runs

Table 2

Series II: Investigation of the effect of a grid spacer, with 6 grid spacers (without grid spacer at the bundle midpoint)

Test No.	Inlet velocity (cold), cm/s	System pressure, Bar	Feed water temperature, °C		Bundle power, kW	
			0-30 s	End	0 s	Transient
234	3.8	2.0	46	37	200	120% ANS
229	3.8	4.1	53	38	200	120% ANS
231	3.8	6.2	54	40	200	120% ANS
233	5.8	2.0	47	37	200	120% ANS
228	5.7	4.1	50	37	200	120% ANS
230	5.8	6.2	48	37	200	120% ANS

SEFLEX Experiment

The aim of the SEFLEX (fuel rod Simulator Effects in FLooding EXperiments) [7] experiment was investigation of the influence of the design and the physical properties of different fuel rod simulators on heat transfer and quench front progression in unblocked und blocked rod bundles during the reflood phase of LOCA in a PWR reactor. In the frame of SEFLEX experiment 4 test series were performed. Test series I and II were performed without blockage.

Series I - Rods with helium filled gaps between Zircaloy claddings and alumina pellets and 7 grid spacers

Series II - Rods with argon filled gaps between Zircaloy claddings and alumina pellets and 7 grid spacers

Comparison of SEFLEX tests Series I and II boundary conditions with FEBA tests is shown in the Table 3.

Table 3

Characteristic of SEFLEX tests series I and II

Experiment /Test series	Test No.	Cladding material	Gap gas	Inlet velocity, cm/s	System pressure, bar	Feed water temperature, °C
SEFLEX/I	05	Zircaloy	Helium	3.8	2.1	40
SEFLEX/I	03	Zircaloy	Helium	3.8	4.1	40
SEFLEX/I	06	Zircaloy	Helium	5.8	2.1	40
SEFLEX/I	04	Zircaloy	Helium	5.8	4.1	40
SEFLEX/II	07	Zircaloy	Argon	3.8	2.1	40
FEBA		Stainless Steal	Gapless	3.8 – 5.8	2.1 – 6.2	40

Phase 4 – Confirmation/Verification

The input uncertainties are code related. Therefore, the quantified probability distribution functions can vary considerably and in the frame of the phase 4 they are to be compared and verified by comparison with experimental data of the flooding experiment PERICLES. It should be done performing uncertainty analysis of selected test runs from PERICLES 2D reflood experiment. The participants have to use the input uncertainties obtained in the previous two steps. The verification of the input uncertainties will be performed by comparison of calculated uncertainty ranges of selected output parameters like cladding temperature or quench front propagation with corresponding measured values. In addition the differential pressure drop along the test section can be compared as an important boundary condition for the heat transfer and quenching process. The correspondence of measured values with determined uncertainty limits can be estimated using the method proposed by IRSN [12]. If necessary the sensitivity analyses can be included for clarification of reasons of potential differences between results obtained with different codes.

The comparison will confirm or not the input uncertainty ranges quantified in the previous steps of the benchmark. The results should enable to elaborate some conclusions concerning the applicability and quality of methodologies used for input uncertainties quantification. It is expected that on this basis some recommendations for quantification of model uncertainties can be formulated.

For the confirmation/validation step five test runs from the 2-D PERICLES reflood experiment have been chosen. The selected tests runs have not been published up to now. This enables performing of blind analyses as it was desired in the benchmark specification.

The reflood 2-D PERICLES experiment

PERICLES [4] has been carried out to study reflood, with a special focus on 2D effects in a PWR core where the rod power is not identical from one assembly to the other. The experiment consists of three different assemblies. These assemblies are contained in a vertical housing with a rectangular section, each assembly containing 7*17 = 119 full length heater rods. Thus, the total number of heater rods is 357. The fuel rod simulators are without gap, with stainless steel cladding, boron nitride as insulator and Nichrome V as heating material.

The rods are heated by two independent electrical power sources, giving the possibility to heat more the central assembly B (the ‘hot’ assembly) than the two lateral ones A and C (the ‘cold’ assemblies). The length of the rods is equal to the length of the channel (3656 mm) and their diameter is equal to 9.5 mm. The axial power distribution has a cosine profile approximated by 11 steps of different power density in the axial direction.

In the REFLOODING tests of the PERICLES experiment, the three assemblies are initially empty of water. During a first stage, the rods are electrically heated in order to bring the cladding temperatures to some initial values. When these values are reached, water is injected with a constant rate at the bottom of the test section in order to reflood the three assemblies, the rod power being kept constant. During the reflooding stage, a quench front goes up in each assembly. The reflooding stage is finished when in all three assemblies the quench fronts reach the top of the assemblies (the velocities of the three quench fronts may be slightly different because of the different rod power between the hot and the cold assemblies). The arrangement of fuel assemblies in PERICLES test section is shown in Figure 3 and the cross-sectional view of one fuel assembly in the Figure 4.

Roughly 40 tests have been conducted. For the validation step, it was proposed to consider only 5 out of them, the features of which are given in the table 4 below. In this table, N_{nom}(HA) and N_{nom}(CA) are the nominal heat fluxes in the Hot and Cold Assemblies respectively, Fxy is their ratio, GO is the inlet mass velocity entering the bottom of each assembly during the reflooding stage, T_{wi} is the initial cladding temperature in the middle of each assembly (at the beginning of the reflooding stage), DT is the sub-cooling of water entering in the assemblies and P is the pressure of the test section.

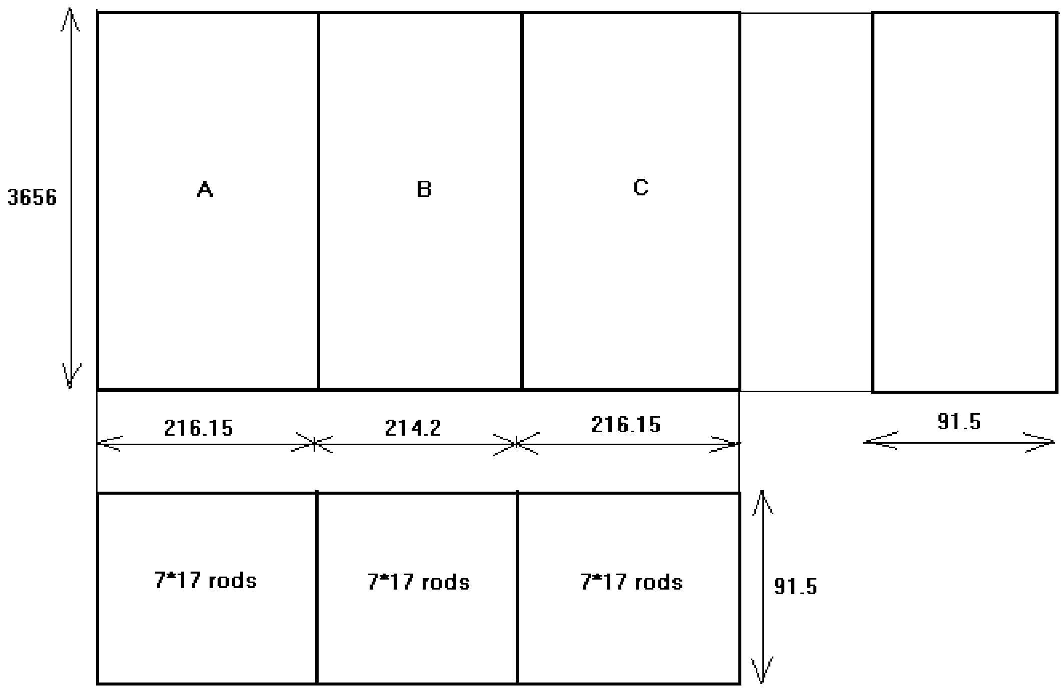

Figure 3. Arrangement of fuel assemblies in 2-D PERICLES experiment

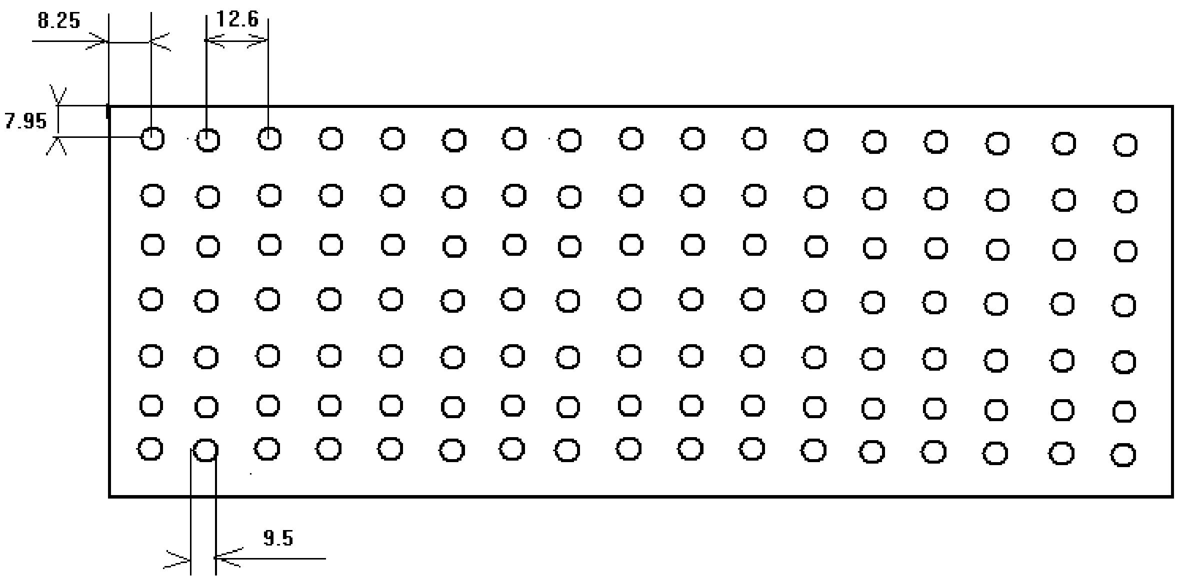

Figure 4. PERICLES assembly.

Table 4

Selected 2D PERICLES test runs

Test No	N_{nom} (HA) W/cm^2	N_{nom} (CA) W/cm^2	Fxy	GO (HA) g/cm^{2}s	GO (CA) g/cm^{2}s	T_{wi} (HA) °C	T_{wi} (CA) °C	DT °C	P (bar)
RE0062	2.93	2.93	1	3.6	3.6	600	600	60	3
RE0064	4.2	2.93	1.435	3.6	3.6	600	475	60	3
RE0069	2.93	2.93	1	3.6	3.6	475	475	60	3
RE0079	4.2	2.93	1.435	3.6	3.6	600	475	90	3
RE0080	4.2	2.93	1.435	5	5	600	475	60	3

Test RE0064 is the reference test, each of the other tests are obtained by changing one parameter. The test RE0069 allows investigating the effect of the radial peaking factor Fxy. In tests RE0062 and RE0069 the effect of the imposed initial cladding temperature T_{wi} at the beginning of the reflooding stage (for a radial peaking factor Fxy equal to 1) is investigated. The test RE0080 shows the effect of the inlet mass velocity GO. The test RE0079 shows the effect of the value of the sub-cooling of the water entering the test section.

Another important point is the large number of measured cladding temperatures in the test section. Per assembly, there are roughly 30 fuel rods with thermocouples at 6 elevations, including 3 elevations far from the spacer grids. As a consequence, we can easily have at our disposal 50 to 60 independent temperature measurements. Considering the progression of the quench fronts in the hot assembly and in the cold assemblies provides also valuable data.

Of course, it is advisable to use a 2-D modelling of the experiment. Nevertheless, 3 parallel channels can also be used, if cross-flows are taken into account.

Conclusions - Objectives and benefits of the benchmark

The main objective of the benchmark is to compare existing approaches, estimate their quantification capabilities and to give recommendation for quantification of model uncertainties, which describe phenomena for which no single effect tests exist. The benefit of the benchmark is additionally evaluation of uncertainties quantification performed by each participant for reflooding modelling in their codes. Reflooding is one of the key issues of LOCA simulation and even participants using engineering judgement would profit from the validation of their probability distribution functions.

Even if they will not apply an advanced methodology they can at least improve their probability distribution functions.

It is expected that on the basis of benchmark results some recommendations for quantification of model uncertainties can be formulated. If desired, an attempt of a "good praxis" guide elaboration for model uncertainties quantification can follow the finalisation of the benchmark.

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Use and application of "best estimate plus uncertainty" methods. A regulatory view

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Abstract

Regulatory environment is characterized by its prevention to change. Any move has to be solidly founded. Application in licensing of Best Estimate Plus Uncertainty (BEPU) methods is not an exception. Typically a fully deterministic approach as described in IAEA SSG-2 is used for design bases accident analyses in current Nuclear Power Plants. In recent years the use of BEPU methodologies is gaining favor in the nuclear technology community as a way to optimize design and operation while preserving full compliance with applicable regulation.

This paper has its focus on the regulatory relevance of the use of BEPU in licensing practices. A regulatory analysis describing the rationale of the evolution from classic deterministic methods to BEPU as well as a selected set of topics around the implications of the use of BEPU methods is shown.

To finalize some conclusions and thoughts of possible further developments of these methods are drawn.

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1. Introduction.

Regulatory environment is characterized by its prevention to change. The reasons of this prevention are manifold. It is in the benefit of the licensing process to have a stable and coherent regulation such that the expectations can be reasonably predicted both in resources and results; that is, a well defined regulatory framework is needed. Secondly and perhaps more relevant it is the fact that moving from a regulation that has proved to adequately fulfill the mission to protect people, as is the case of classic deterministic assessment, to a new one like the Best Estimate Plus Uncertainty (BEPU), calls for further justification.

Utilities' economical reasons arising from the overconservatism of classic Deterministic Safety Assessment (DSA) may justify the move to BEPU methods, but in no way prevail upon safety. Only if it is proved that, with the current knowledge, the level of safety achieved by the application of BEPU is acceptable, while filling the gaps in the current deterministic methods, a regulatory change may be justified.

IAEA SSG-2 [1] describes in sufficient detail the different analytical approaches to deal with deterministic safety assessments of nuclear power plants. Typically a fully deterministic approach, Options 1 or 2 as described in the aforementioned guide, is used for design bases accident analyses in current Nuclear Power Plants (NPP). Nevertheless, in recent years the use of BEPU methodologies is gaining favor in the nuclear technology community as a way to optimize design and operation, while preserving full compliance with applicable regulation.

In the following sections, a regulatory analysis around the use of BEPU in licensing practices describing the evolution from classic deterministic methods to BEPU and its consistency with fundamental safety principles, as well as a selected set of topics around the implications of the use of BEPU methods is shown.

2. Regulatory analysis.

Nuclear safety regulation is not an exception to other industrial activities heavily regulated because of their potential impact in terms of health and safety of the people i.e.: chemical, transport, pharmaceutical, food, etc. All these industries have in common the existence of basic safety principles developed in legally binding regulations to protect the affected patrimony (life, environment, property) as a result of the unwanted effects resulting from accidents, use or any other means.

The set of basic safety principles can be derived from applicable regulation in different countries with origin on international treaties or constitutional mandates. Focusing in nuclear power industry, IAEA "Fundamental Safety Principles" SF-1[2] compiles and develops these safety principles for the case of civil nuclear activities. Obviously IAEA agreements like the Nuclear Safety Convention (NSC) [3] and Vienna Convention on Civil Liability for Nuclear Damage (VCCLND) [4] make explicit provision to the fulfilment of those principles i.e.: NSC art.1 states its objective of protecting individuals, society and environment by means of prevention and mitigation, and VCCLND establishes as objective to repair by means of indemnification and reinstatement.

Although not explicitly present in these Conventions, it is also of relevance to BEPU methods to mention the controversial precautionary principle i.e.: as established in Art.174 of the Treaty establishing the European Community and developed in COM(2000)1 [5] which elaborates the manner in which the Commission intends to apply the precautionary principle when faced with taking decisions relating to the containment of risk. A precautionary action is defined as: *"a decision exercised where scientific information is insufficient, inconclusive, or uncertain and where there are indications that the possible effects may be potentially dangerous and inconsistent with the chosen level of protection"*. So it is important to note how the application of such principle calls for the acquisition of scientific knowledge in order to determine with sufficient certainty the risk in question, as well as observe how once the certainty level is quantified and risk reduction is feasible to a societal acceptable level a decision can be adopted.

All these principles: precaution, prevention, mitigation, indemnification and reinstatement, materialize in each NPP licensing basis, whereas the design basis conforms, in broad terms, the technical targets to fulfil the licensing basis. Two complementary and systematic methods i.e.: deterministic and probabilistic, are the relevant tools to assess in an integrated way the plant safety. NPP have been originally licensed based on the deterministic design of the safety related Structures, System and Components (SSC). On the other hand Probabilistic Safety Assessment (PSA) methodologies have given a new perspective to safety analysis when hypothesis of the deterministic design break down and so go beyond the basis for the design of those SCC (i.e.: concurrent independent failures, escalation, etc).

The challenge faced by the designers of the current fleet of NPP was immense, and it is fair to recognise the good job they did as they had to deal with a new technology plagued of new questions and need for answers. Basically a theory of protection had to be built from scratch and concepts like defence in depth, redundancy, diversity, etc. were introduced in the design [6]. Similarly the corresponding accident analysis with the associated methodology of classification, acceptance criteria and analytical tools was developed. For the case of USA regulation, this process resulted in the issue of the well known 10CFR 50 app.A (1971) [7], 10CFR 50.46 app.K (1974) [8] (interesting because of its prescriptive nature), R.G. 1.70 (1972)[9] and technical standards like ANSI/ANS 18.2 (1973)[10]. For the case of the European Union the Nuclear Regulators Working Group contributed since its creation in 1972 to develop a set of safety principles later compiled in COM(81)519 [39].

It is clear that engineering capabilities at that time (early 70's) were heavily conditioned by the limited scientific and technology knowledge, limited data and computing capacity. As a result a common engineering practice was adopted making use of the so called worst case scenario as well as the use of large safety margins together with the adoption of defence in depth principles. All of that resulted in the deterministic methodology.

Consistently with that environment, IAEA Glossary [40], defines deterministic analysis as:

"-Analysis using, for key parameters, single numerical values (taken to have a probability of 1), leading to a single value of the result.

-In nuclear safety, for example this implies focusing on accident types, releases and consequences, without considering the probabilities of different event sequences.

-Typically used with either "best estimate" or "conservative" values, based on expert judgement and knowledge of the phenomena being modelled"

Without going into a deeper critical analysis of the definition, it is apparent that no room for BEPU methods based on the propagation of uncertainties is considered, while otherwise it is consistent with Options 1 and 2 as defined in [1].

As nuclear technology matured and especially after the Three Mile Island accident, extensive experimental work was made in order to gain data and so to reduce the scientific uncertainty associated to complex thermal-hydraulic phenomena. International programs like LOFT [11] and a multitude of International Standard Program (ISP) under the auspice of the Organisation for Economic Co-operation and Development (OECD), largely contributed to the development of a data base suitable for code validation and verification [12], all that resulting in the development of Thermal Hydraulic (TH) realistic system codes representing the best available knowledge. This new generation of codes pretend to be a realistic image of the physics involved in real plant transients or accidents, but as a result of the imperfect knowledge of nature, as well as from the mathematics formalism reduction (averaging process, discretization, numerics, etc.) they produce uncertain results. The improved knowledge of TH phenomena supported by a qualified experimental data base [13] allowed for the relaxation of prescriptive regulation like [8], giving room to Best Estimate (BE) methods accompanied by an uncertainty analysis (see 10.CFR 50.46 (a)(1)(i)). This was recognised in the so called Code Scalability Application and Uncertainty (CSAU) method [14] where a structured approach for the licenciability of this sort of methods compatible with such rule was described and supported in R.G.1.157 [15] and more recently R.G.1.203 [36].

As a recognition of the relevant role that BEPU methods were to take in nuclear safety analysis, international organisations like OECD and the International Atomic Energy Agency (IAEA) adopted an strategy to advance in the qualification of the different methods (e.g. Uncertainty Methods Study (UMS) [16], Best Estimate Methods Uncertainty and Sensitivity Evaluation (BEMUSE) [17], Uncertainty Analysis in Modelling (UAM) [18] OECD benchmarks), and to assess the integration of BEPU methods in design and operation of Nuclear Power Plants (NPP) as collected in the following set of IAEA references [19, 1, 37, 38]). These efforts continue and the present workshop is a good example of it.

3. Selected topics

In this section a number of different topics are developed. All of them relate to the implications, mainly regulatory, that derive from the use of BEPU methods with regard the conservative deterministic methods traditionally used. A more detailed description can be found in [20, 21, 22].

3.1 Deterministic safety analysis and probabilistic safety margin

From a regulatory perspective it is important to note how BEPU methods have an impact on the widely used concept of safety margin, and in particular the move from a traditional concept of safety margin mainly based on the discrepancy between a maximum design load and the capacity at which a safety function is lost. The use of BEPU methods derives into a definition of the safety margin based on likelihood measures, typically probabilities. The use of the Traditional Safety Margin (TSM) has the benefit to allow for both significant simplifications on design methods and an easy way to build up conservatism, typically verified by extensive sensitivity analyses. On the contrary, because of the nature of the metrics it does not allow for a collective assessment of safety margins, and an improper use may give a misleading view of “having enough margin”.

The use of a probabilistic definition of safety margin is not entirely new in the deterministic safety analysis. Acceptance criteria are established, in the regulation of different countries, with regard to the failure probability of individual fuel rods and failure probability of a given number of fuel rods against anticipated operational occurrences.

The definition of “probabilistic safety margin” was proposed in early works related to the combination of best-estimate analysis and Level 1 probabilistic safety assessment [44].

Following [21], the probabilistic safety margin (PSM) associated to a safety output V during a transient or accident A is defined as:

$$PSM\left(V \mid A\right) \equiv PR\left(V \in R_V \mid A\right) \qquad (1)$$

, where R_V is the ‘‘acceptance region’’ established by the regulator for output V. The complementary of R_V can be termed the “forbidden region”. These regions are bounded by the safety limits. (1) is a probability conditioned to the occurrence of A.

The difference to 1 of the PSM is the probability of “exceeding the limits”:

$$P_{EX}\left(V \mid A\right) \equiv 1 - PSM\left(V \mid A\right) = PR\left(V \in R_V^* \mid A\right) \qquad (2)$$

, where R^*_V is the forbidden region for V.
Defining the safety margin as a probability has several advantages, because it is nondimensional, ranges in the interval [0, 1], and can be applied to both escalar and multidimensional safety outputs. Furthermore, probabilistic margins can be combined according to the laws of probability.

As an example, let us suppose a safety barrier B having several failure modes, the k-th failure mode being typified by a scalar safety output V_k with an upper safety limit L_k, k = 1, . . ., F. A safety margin can be assigned to the barrier, conditioned to the accident A:

$$PSM(B \mid A) \equiv PR\left\{ \bigcap_{k=1}^{F} \left(V^k < L^k\right) \mid A \right\} \qquad (3)$$

, which is the probability of no failure conditioned to A. It is an example of PSM for a multidimensional safety output.

Similarly and for a given initiating event (IE) with derived sequences $A_1,\ldots,A_s$, the definition (1) can be extended to the IE:

$$PSM(V \mid IE) \equiv PR(V \in R_V \mid IE) \qquad (4)$$

, expressible, according to the law of total probability, as:

$$PSM(V \mid IE) = \sum_{j=1}^{S} p_j \; PSM(V \mid A_j) \qquad (5)$$

That is, the margin for the initiator is a weighted average of the margins for sequences, the weights being the conditional probabilities:

$$p_j \equiv PR(A_j \mid IE)$$

An example of this may be a LOCA analysis with several combinations of single failures, giving rise to different sequences to analyze.

Consider now that all the initiating events which induce transients of V could be grouped into M categories IE_i, i = 1, . . .M . The frequency of V exceeding the limits is:

$$\nu\left(V \in R_V^*\right) = \sum_{i=1}^{M} \nu_i \left[1 - PSM(V \mid IE_i)\right] \qquad (6)$$

, where υ_i is the frequency of IE_i . In (6) the frequencies of initiators combine with the exceedance probabilities:

$$1 - PSM(V \mid IE_i) = \sum_{j=1}^{S_j} p_{ij} \left[1 - PSM(V \mid A_{ij})\right] \qquad (7)$$

A_{ij} being the j-th sequence evolving from the i-th initiator, and p_{ij} its conditional probability:

$$p_{ij} \equiv PR(A_{ij} \mid IE_i)$$

In (6) the uncertainty about V, represented by the PSM, merges with the aleatory uncertainty represented by the frequencies υ_i. We conclude that probabilistic safety margins can combine with initiator frequencies and produce exceedance frequencies, which constitute the plant risk.

Let us analyse how PSM and consequently BEPU methods fit within a deterministic safety analysis methodology. With this aim, we suppose that V is a scalar safety output upon which an upper safety limit L has been imposed. The PSM for V given an initiating event IE is:

$$PSM(V \mid IE) \equiv PR(V < L \mid IE) \qquad (8)$$

, and can be expressed in terms of the probability distributions of both V and L:

$$PSM(V \mid IE) = \int_{-\infty}^{+\infty} f_L(s) F_V(s \mid IE) ds \qquad (9)$$

, where f_L is the probability density function (pdf) of L and F_V is the cumulative distribution function (cdf) of V conditioned to the occurrence of A. (9) is a convolution integral.

Let us now suppose that V_b is a large value of V obtained from a limiting scenario evolving from IE and calculated with a conservative methodology. Typically, V_b would correspond to a Design Basis Transient (DBT). This means that the probability of V being less than V_b conditioned to IE is very close to 1. Notice that such probability is formally a PSM of V with respect to V_b, and therefore has the sense of an analytical safety margin. As proved in [22], from (9) derives the following inequality:

$$PSM(V \mid IE) > PSM(V \mid DBT) \cdot PR(V \leq V_b \mid IE) \qquad (10)$$

, with

$$PSM\left(V \mid DBT\right) \equiv PR\left(V_b < L \mid IE\right)$$

, which is the PSM for the design basis transient.

(10) illustrates that a high margin for V in the DBT assures a high margin in the enveloped IE

Considering, as in (6), M categories of initiating events, and using (10), we obtain an upper bound for the exceedance frequency of L

$$\nu\left(V > L\right) < \sum_{k=1}^{M} \nu_k PR\left(V \leq V_{bk}\right) PR\left(V_{bk} \geq L\right) + \sum_{k=1}^{M} \nu_k PR\left(V > V_{bk}\right) \qquad (11)$$

,where a DBT has been defined for each category. The first addend in the right hand side of (11) represents the contribution of the transients enveloped by the DBTs (those within the design basis) and the second one is the residual contribution stemming from the not enveloped fraction (beyond design basis sequences, BDBS). Notice that, in order to obtain a general result, the limit L has been considered as uncertain

The right hand side of (11) gives the exceedance frequency of L supposing that the BDBS result in a limit violation (i.e. progress beyond design basis assumptions).

If the DBT is adequately chosen, the residual term can be neglected against the main one, and the maintenance of the safety margin is assured through the enveloping character and the safety margin of the DBTs. This is the basis of the deterministic design approach. In the realistic methodologies of DSA, the value V_b of the output V in the enveloping transient is an uncertain rather than a constant value. But the cornerstone of the method still remains: the high safety margin is assured through the definition of an enveloping transient with a safety margin high enough.

Note that (11) reduces to the classical fully deterministic approach when $PR(V_{bk} \geq L) = 0$ and $PR(V \leq V_{bk}) = 1$ (a strong statement that calls for proof) and as a result the frequency of exceedance for a given IE is that of the BDBS (e.g. several independent single failures). Also note that a traditional safety margin, defined as L minus V_b may be misleading as it carries no information on the probability to exceed the safety limit, so it needs to be supplemented by a validation against a BEPU method. This in fact has been a driving force for the development of BEPU methods, and justifies a regulatory move as said before. Proof like this has been incorporated into regulation. An example are the “requirements for LOCA safety analyses” of RSK [23]

As a result of this analysis it is demonstrated that the use of BEPU methods, termed Option 3 in [1], is fully compatible with a deterministic methodology. It is possible to move from option 3 to option 2 (BE code plus conservative single valued hypothesis) by being more conservative in the assignation of probability distributions to uncertain inputs, thus producing a “Better Estimate” approach. As a limit case, fixed conservative values (i.e. degenerate probability distributions) can be assigned to all the uncertain inputs, and therefore a single calculation will carry all the information (option 1).

3.2 Probabilistic acceptance criteria.

As for any other hazardous industry, NPP risk analysis is based on a societal acceptable risk in terms of damage to public and individuals, and the risk is measured in terms of frequency of exceedance of the allowed damage, which is built as the cumulative frequency of individual events. Current NPP deterministic design is characterized by a strategy of events classification [10, 24, 25, 26, 27], mostly using the frequency of the initiating event as the criterion to establish the category. An exception is [25].

Given the frequency of the IE, it is allocated to a corresponding category and by means of the accident analysis verified the fulfilment of the applicable acceptance criteria in terms of barriers damage and dose limits i.e. consequence analysis.

Under a fully deterministic approach the probability of an accident progressing beyond the deterministic design basis of the affected SSC is negligible (partially verified by means of PSA level 1 analysis). For BEPU analyses, generally a non-zero probability exists to surpass the safety limit and as a consequence an escalation in the damage to happen. In order to estipulate a limit to the PSM it is basic to <u>constrain the allowable damage for the residual cases</u> that under a BEPU method will violate the limit. In other words, the probability to exceed the safety limit should be commensurate to the consequences in terms of damage arising from those cases.

An acceptance criterion for the PSM of the enveloping transient should read:

$$PR\{PSM(V \mid DBT) > M_0\} \geq 1-\alpha \qquad (12)$$

, that is, the margin must be higher than M_0, a value close to 1, and a high statistical confidence (α is a low value).

The requirement of high statistical confidence is due to the finite size of the random samples used in the calculation of the PSM. The values of both M_0 and α are imposed by the regulatory authority. Typically M0 and 1-α have been both set to 0.95

Recalling the definition (1) of PSM, the criterion (12) may be expressed in a fully probabilistic fashion:

$$PR\{PR\{V \in R_V\} > M_0\} \geq 1-\alpha \qquad (13)$$

, so that the presence of V in the acceptance region is required with high levels of probability and statistical confidence.

An alternative formulation of (13) is

$$PR\{PR(V \in R_V^*) < 1-M_0\} \geq 1-\alpha \qquad (14)$$

, now in terms of the probability of “exceeding limits” (i.e. entering the forbidden region)

Assessing the PSM by a pure Monte Carlo procedure can be envisaged as a computational binomial experiment. Each Monte Carlo run has two possible outcomes: “success” if V falls inside the acceptance region, and “fail” otherwise. This is captured by a random variable S taking one the value 1 for success and 0 for fail. S is a Bernoulli random variable, with parameter PSM(V|DBT)

Thus, the PSM is the expected value of S:

$$PSM\left(V \mid DBT\right) = E\left[S\right] \qquad (15)$$

The acceptance criterion may be multidimensional, i.e. a condition on a multidimensional safety output, or a set of conditions on scalar safety outputs. Very well-known examples are the criteria about the LOCA-ECCS analyses, imposed on Peak Clad Temperature (PCT), Core Wide Oxidation (CWO) and Local Clad Oxidation (LCO). In this case, pure Monte Carlo runs are also binomial experiments producing an index S.

3.3 PSM calculation methods: uncertainty methodology.

When uncertainty is taken into account, DSA acceptance criteria adopt the probabilistic form stated in (12)-(14). There is no regulatory prescription about the methodology to use in the check of such criteria.

As stated in [20] there are two basic procedures for checking the acceptance criteria with the form (12-(14), from a random sample of the output V:

1) Construction of a tolerance region for V, with coverage/confidence levels of at least $M_0/(1-\alpha)$, and check that such region is inside the acceptance region RV.
2) Construction of a lower confidence limit, with level $1-\alpha$, and check that it is higher than M_0

BEPU methods, almost unanimously, focus on the procedure 1). Pioneering works as [13, 14, 28, 29] followed multistep methodologies to qualify and perform a quantification of reactor safety margin, all of them having in common the use of Monte Carlo sampling of a response surface metamodel. Most of current models perform the uncertainty analysis by <u>propagation of the uncertainty</u> embedded in probabilistic density functions of input and model variables through the evaluation model i.e.: a qualified code and plant model, to produce an uncertain output. The method described in [30] make use of the <u>accuracy quantification</u> of the output by means of an elaborated treatment of applicable experimental data bases.

The aforementioned methodologies are mostly nonparametric (i.e. distribution-free) in the sense that do not assume any parametric form for the probability distribution of V. All methodologies based on Wilks’ method, pioneered by [31] have a nonparametric approach.

In some methodologies, parametric methods are used, assuming a parametric distribution for the outputs (mainly the normal one). Such assumptions require demonstration, mainly by using goodness-of-fit tests.

Now we will refer to the check of acceptance criteria by the procedure 2), based on the construction of a confidence limit of the PSM from a simple random sample (SRS) of V. This is the simplest sampling procedure, and will be assumed here because it is almost unanimously used in the current BEPU methodologies. The SRS is supposed to be obtained by repeatedly running the code for a SRS of inputs. To simplify, we will assume a fixed acceptance region, so that the acceptance limits are not uncertain.

A lower confidence limit with level 1-α for the PSM is a statistic P_L such that:

$$PR\{P_L \leq PSM(V \mid DBT)\} = 1-\alpha \tag{16}$$

That is, (0, P_L) is a one-sided (1-α)-confidence interval

If the obtained P_L is higher than M_0 , the criterion (12) is fulfilled.

Methods to set up confidence limits on a probability (as PSM) can be parametric or nonparametric, and a survey of them is found in [42]. Nonparametric (i.e. distribution-free) methods can be classified as frequentist (or classical) and Bayesian.

Frequentist methods only make use of the information provided by R and N, where R is the number of values in the SRS of V falling inside the acceptance region. So, R is a binomial variable for N runs and a “success probability” equal to the PSM. We mention the Clopper-Pearson interval (the so-called “exact method”), which gives as lower confidence limit

$$P_L = {}_{\alpha}beta(R, N-R+1) \tag{17}$$

, that is, the α-quantile of the beta distribution with parameters R and N-R+1.

Bayesian methods can accommodate “a priori” information about the parameter to be estimated, in the form of a prior probability distribution. The application of Bayes’ theorem combines the information of the random sample with that of the prior, producing a posterior probability distribution for the parameter. The simplest case is when the prior is a beta distribution, beta(γ,δ), because it produces a posterior which is also beta, beta(γ+R , δ+N-R). Then, it is immediate to find a lower credible limit as

$$P_L = {}_{\alpha}beta(\gamma + R, \delta + N - R) \tag{18}$$

Both (17) and (18) tend to the single value R/N when N$\rightarrow\infty$. But for small or moderate N both methods can give quite different values, depending on the chosen prior. Of course, the main problem in the Bayesian method is precisely the election of the prior. The information conveyed by a beta(γ,δ) prior could be described as “γ successes in γ+δ runs”. The prior information about a PSM could be obtained from previous or analogous analyses, in NPPs or integral facilities, and experts’ elicitation.

The Bayesian formulation can even represent the complete lack of “a priori” information”, by using the so-called noninformative priors, for instance the uniform distribution, beta(1,1), or the Jeffreys prior, beta(1/2 ,1/2)

Parametric methods can also be used in the confidence limit estimation, when parametric distributions are assumed for V and L. Goodness-of-fit tests must be applied to the data, before confidently assigning the distributions.

When the expected value of the PSM is quite large (i.e. very close to 1), it is clear that the acceptance limits are located in the tails of the V distribution. In such cases, there are sampling procedures to check (12)-(14) more efficiently than SRS, mainly stratified sampling and importance sampling, which focus on the tail regions of V close to the acceptance limits.

3.4 BEPU methodologies and validation.

BEPU methodologies must undergo a verification and validation process before being accepted for the licensing task. In the validation process, real data, from experimental facilities and plants, are compared with predictions of the methodology.

The relevance of Integral Effect Tests (IET) in validation must be stressed. They were formerly planned as a means to qualify the predictive capacities and accuracy of codes [32], but they also play a fundamental role in the verification of uncertainty methodologies itself.

It may be argued that model uncertainty results against IET data will be affected by scale distortion, particularly when some models have been qualified against 1:1 power to volume mockups. On the other hand, it is not evident [33] that the magnitude of scale distortions could suffice to discard the outcome of the BEPU method validation against IET.

IET have as peculiarity the controlled environment in which the experiments are performed. This means that uncertainties about initial and boundary conditions are reduced with regard to those from a NPP, whereas the model uncertainties remain.

Once verified and valicated, BEPU methodologies may also be used to validate other methodologies, aimed to obtain bounding results. The use of a BEPU methodology for validating a bounding methodology is quite similar to that in licensing; in both cases there are acceptance criteria, focused on the non exceedance of a safety limit or a conservatively calculated value.

On the other hand, the validation of realistic model is a different exercise from both the licensing and the conservative validation. Now the acceptance criteria are focused on the closeness of the calculated values of the important outputs to their real values, taking into account both prediction and measurement uncertainties. These important outputs can be different physical magnitudes, or magnitudes in different spatial locations and/or in different time instants.

There are many techniques to make the comparison between predicted and real values. When we have a Monte Carlo random sample of the outputs, a possibility is to formulate the validation exercise as a statistical hypothesis test. As a very simple example, let us suppose a experiment where a scalar output V is measured, giving a value E with a negligible uncertainty. The validation process could be based on testing the following hypothesis:

H_0: “E is compatible with the distribution of V“

For testing H_0, an acceptance interval may be set up as follows. Given a SRS of size N for V, the probability that a new observation of V is bracketed by two OS is [43].

$$PR\{V_{r:N} < V < V_{r:N}\} = \frac{s-r}{N+1} \tag{19}$$

i.e. it only depends on the two orders and N. If the probability (25) equates to 1- α, [$V_{r:N}$, $V_{s:N}$] is an acceptance interval for testing H_0 with significance level α. If we decide to use the sample extremes (s=N , r=1), the minimum size needed for the sample is obtained by equating (25) to 1-α and solving for N:

$$N_{\min} = \left\lceil \frac{2-\alpha}{\alpha} \right\rceil \tag{20}$$

For α=0.05, (20) gives a minimum size of 39. So, if a random sample of size 39 is obtained for V, and the measured value E is between the extremes of the sample, we accept the hypothesis H_0.

This is a simple example, using order statistics. Several other approaches exist.

The validation is performed for the class of accident scenarios dealt with by the methodology.

3.5 Characterisation of models uncertainty

At the core of most BEPU methods it is the characterisation of the uncertainty of models relevant to the dominant phenomena of the scenarios being simulated. This uncertainty is epistemic in nature, due to lack of knowledge and imperfection of models. The calculation of model uncertainty from experimental data has been recognised [17] as an element of improvement of the methodologies. Some international exercises are being set up about this topic [34].

From the regulatory standpoint this concern has also been subject to discussion. The use of probability theory to model epistemic uncertainty has been called into question, especially when the information is scarce. Even when probability distributions are used, the combination of epistemic and aleatory uncertainties is questioned. Anyway, probability distributions should be assigned with a certain degree of conservatism to be acceptable in the BEPU analysis.

As an example, when probability distributions are assigned to model parameters from a sample of experimental values, the statistical uncertainty (stemming from the finite size of the sample) should be taken into account. A possibility is to obtain the empirical cumulative distribution function (ecdf) supplemented with confidence bands (see fig 1), either one-sided or two-sided. Furthermore, such bands can be either pointwise or global.

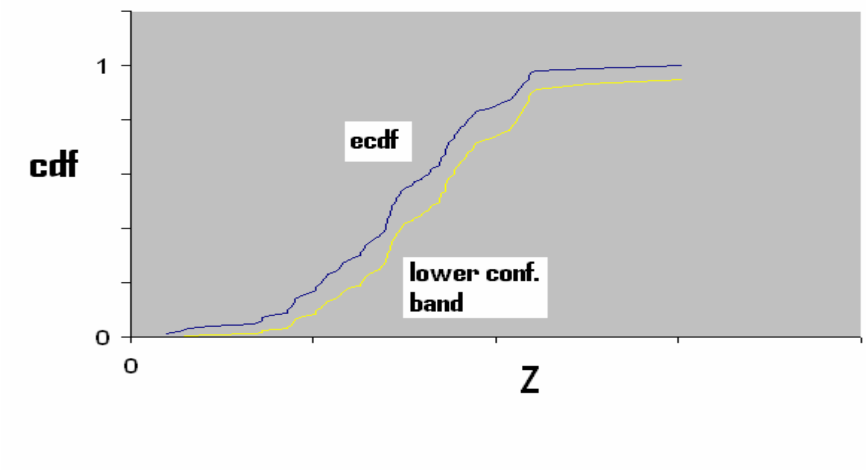

Fig.1 Uncertainty on the empirical cumulative distribution

3.6 Impact on Technical Specifications

Technical Specifications (TS) and methodologies of accident analyses are closely connected. It is by means of the accident analysis that the operational space and systems operability requirements are verified against the fulfilment of acceptance criteria. In that sense those process variables that are initial conditions of the design bases events are incorporated to their extreme values into the limiting conditions for operation (LCO), thus defining the safe boundaries for operation. This is clearly shown in the CNS [3], art.19 ii) *"operational limits and conditions derived from the safety analysis, tests and operational experience are defined and revised as necessary for identifying safe boundaries for operation."*

Current interpretation is compatible with options 1 & 2 from ref [1], where LCO values of variables or parameters part of a Design Basis Transient (DBT) are used, that is, DBT envelops extreme values of the operation space.

BEPU methods often include uncertainty on the operational parameters dealt with in the Technical Specifications as LCO so that probability distributions are assigned to such parameters. Typically the LCOs are located in the tails of the real density functions (pdfs) of the parameters. Because of that, BEPU methodologies based on pure Monte Carlo propagation (i.e. simple random sampling of the inputs) with a small or medium number of runs will not provide a real exploration of the regions close to the LCOs. In that sense, BEPU results do not strictly bound the allowed operation space, e.g. operation with all variables, or a number of them, sufficiently close to their LCO is not demonstrated to be a safe initial condition for the corresponding initiating event.

The crucial point is that safety analyses must prove not only that the *real* operation is safe, but also that the *allowed* operation of the plant is safe. BEPU methodologies are not fit for obtaining TS limits; rather, their compatibility with them must be proved. A strict procedure to do that should fix the TS parameters on their limit values upon performing the BEPU analysis. But looser procedures could be envisaged [41], by assigning to the operational parameters probability distributions which are not based on the real operation, but on the allowed operation of the plant, so that they assign a non negligible, or even significant value to the probability of violating the operating limits (see fig. 2). For BEPU methodologies based on pure Monte Carlo, the goal is a significant exploration of the regions close to the TS limits. In [41] several criteria are proposed to accomplish such goal.

These modified probability distributions assigned to the operational parameters must be object of surveillance, to check that they really bound the measured values in the plant

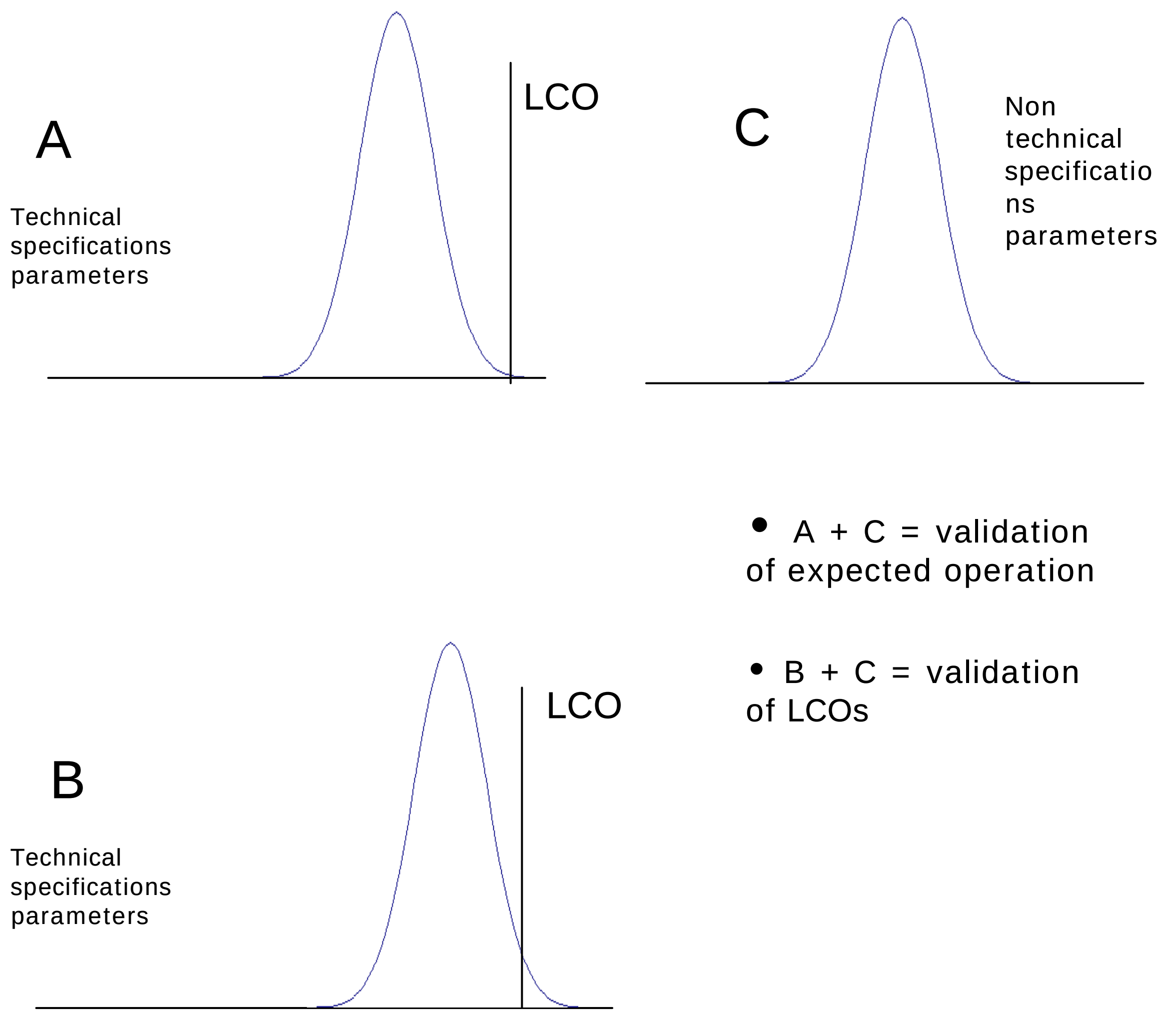

Fig 2 Treatment of uncertain variables and Technical Specifications.

3.7 Code accuracy and user qualification.

Prior to the development of an uncertainty method, code eligibility should be performed and accuracy determined through a verification and validation process.

Code accuracy should be evaluated for the selected application scenarios for a “nominal case” avoiding unjustified “tuning”. Modeled physics (including special models), spatial and time discretization as well as numerical solutions must rely on sound principles and be consistent with the code functional requirements through a validation process. Code accuracy can only be claimed if confirmed through an exhaustive contrast against experimental, operational data and even analytical solutions.

As full scale validation is rarely possible, claimed code accuracy may be subject to questioning because of scale distortion between real plant and experimental facilities. It is important to confirm that code accuracy is preserved through scale variations or in other case identify the reason, e.g. special models, empirical correlations, etc and quantify the impact.

Code user effect has long been recognized as an important if not limiting element to consider in any nuclear safety application of thermal hydraulic system codes [35]. In order to limit such effect, appropriate training of the individuals, technology competence and resources are basic. For the case of licensing calculations, strict design procedures should be required.

Conclusions

1. Nuclear safety assessment based on classical Deterministic Safety Assessment (DSA) has proved to be a robust methodology in support of current Nuclear Power Plants Design

2. Evolution towards Best Estimate Plus Uncertainty (BEPU) methods was a lengthy process result of a coherent effort from the nuclear safety technology community and fully consistent with regulation.

3. The move from classic DSA to BEPU methods implied the adoption of probabilistic acceptance criteria, and the definition of probabilistic safety margins (PSM)

4. The probabilistic definition of safety margin allows the use of combination rules derived from probability theory.

5. BEPU methods rigorously allow for the confirmation of the conservative nature of classic deterministic methods.

6. A variety of uncertainty treatment methods possibilities exist which are in principle compatible with regulation.

7. Verification of BEPU methods against Integral Effect Tests (IET) is needed. Development of specific methods to deal with that would be desirable.

8. Special care should be exercised when developing probability distributions for model parameters. OECD PREMIUM benchmarks will hopefully test different methods to deal with the issue.

9. Implementation of BEPU methods with licensing purposes should appropriately consider its impact on Technical Specifications of the nuclear power plant.

10. Code accuracy should be demonstrated to be scale independent for licensing scenarios where the BEPU method is intended to be applied.

11. Strict rules with regard to code user qualification and design procedures should be part of the methodology assessment process.

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Uncertainty Methods Framework Development for the TRACE Thermal-Hydraulics Code by the U.S.NRC

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1. Introduction

The Code of Federal Regulations, Title 10, Part 50.46 requires that the Emergency Core Cooling System (ECCS) performance be evaluated for a number of postulated Loss-Of-Coolant-Accidents (LOCAs). The rule allows two methods for calculation of the acceptance criteria ; using a realistic model in the so-called "Best Estimate" approach, or the more prescriptive following Appendix K to Part 50. Because of the conservatism of Appendix K, recent Evaluation Model submittals to the NRC used the realistic approach. With this approach, the Evaluation Model must demonstrate that the Peak Cladding Temperature (PCT), the Maximum Local Oxidation (MLO) and Core-Wide Oxidation (CWO) remain below their regulatory limits with a "high probability." Guidance for Best Estimate calculations following 50.46(a)(1) was provided by Regulatory Guide 1.157. This Guide identified a 95% probability level as being acceptable for comparisons of best-estimate predictions to the applicable regulatory limits, but was vague with respect to acceptable methods in which to determine the code uncertainty. Nor, did it specify if a confidence level should be determined. As a result, vendors have developed Evaluation Models utilizing several different methods to combine uncertainty parameters and determine the PCT and other variables to a high probability.

In order to quantify the accuracy of TRACE calculations for a wide variety of applications and to audit Best Estimate calculations made by industry, the NRC is developing its own independent methodology to determine the peak cladding temperature and other parameters of regulatory interest to a high probability. Because several methods are in use, and each vendor's methodology ranges different parameters, the NRC method must be flexible and sufficiently general. Not only must the method apply to LOCA analysis for conventional light-water reactors, it must also be extendable to new reactor designs and type of analyses where the acceptance criteria are less well defined. For example, in some new designs the regulatory figure of merit (FOM) has been the reactor vessel level rather than PCT or MLO. Thus, the methods developed by the NRC must be capable of being used for a wide range of plant designs and potential FOMs.

Development and selection of an uncertainty methodology for the NRC was based on a review of available methods and a set of criteria considered important to agency goals and objectives. The criteria included regulatory acceptance of the statistical method, feasibility of implementation, extensibility of the methodology, ease of use and effort to perform analyses, and the effort to implement the methodology. Based on these criteria, an overall approach consistent with Code, Scaling, Applicability and Uncertainty (CSAU) but using order statistics is being adopted. To implement the methodology in an efficient manner, the Symbolic Nuclear Analysis Package (SNAP) has been modified to perform a Monte Carlo sampling of code input parameters, submit multiple cases for execution, and statistically evaluate the results. SNAP performs the statistical analysis using DAKOTA (Design and Analysis Toolkit for Optimization and Terascale Applications), which is a code developed by the SANDIA national laboratories specifically for sensitivity and uncertainty analyses. This paper outlines the NRC approach to uncertainty, describes the tools and activities in progress, and presents the rationale for several decisions.

2. Regulatory Needs and Usage

The TRACE nuclear systems analysis computer code is being developed by the NRC to be used to perform analyses for loss-of-coolant accident (LOCA) and other transients for a wide range of existing and new reactor plants. Licensees and vendors are making increasing use of various uncertainty methods in combination with realistic licensing calculations submitted to the NRC. To assist in the review and approval of these submittals, the U.S. Nuclear Regulatory Commission (NRC) is developing capabilities for a generalized uncertainty method. Use of an uncertainty methodology has the advantage of yielding a quantitative margin to safety limits, along with a measure of the confidence that can be placed on the results. Sensitivity studies making use of uncertainty methods are also expected to provide useful insight on code development priorities. Models and correlations that can be identified as having dominant effects on analysis results due to their uncertainty become candidates for improvement and additional research.

Various methods have been developed to assess uncertainty in thermal-hydraulic systems codes. In general, the focus has been on the uncertainty in computer code predictions to obtain the plant response during a loss-of-coolant-accident (LOCA). The output parameter of interest has usually been the peak clad temperature (PCT). However, as new plant designs and new analysis applications are received by the NRC it is becoming apparent that other output parameters are of significant regulatory interest. In some new plant designs with passive safety systems, the figure-of-merit (FOM) has been the liquid level in the reactor vessel. Demonstrating, with certainty, that the coolant level remains above the top of the core assures that the more typical regulatory criteria (peak cladding temperature, local cladding oxidation, and core-wide oxidation) are satisfied. In other applications, DNBR margin is an important FOM. Thus, as uncertainty method must be adaptable as the regulatory FOM change from application to application.

To accommodate these expected needs, the thermal-hydraulic code TRACE and the user interface SNAP are being developed so that code uncertainty can be determined. Flexibility is being built-in to the NRC tools, so that the staff can examine the effect of individual uncertainty contributors and also estimate the uncertainty in several different output parameters. This paper describes the initial NRC efforts and overall development effort.

3. Uncertainty Methods

Selection and development of an uncertainty process for TRACE started with a review and evaluation of available methods. Of particular interest are the practical aspects of uncertainty methods, and how they are applied in an analysis. Overall code uncertainty was the subject of the Code Scaling, Applicability, and Uncertainty Evaluation Methodology (CSAU) approach, which was developed to provide a general and systematic methodology to quantify thermal hydraulic code uncertainty. Although the method was developed in the 1980s, much of the CSAU approach is used today, particularly with the Phenomena Identification and Ranking Table (PIRT) as a means of defining the overall problem. Details of the CSAU approach are presented in NUREG/CR-5249[1].

One of the important contributions that came from CSAU was the use of a Phenomena Identification and Ranking Table (PIRT) to guide development of an uncertainty methodology. PIRTs are developed by a group of technical specialists who have expertise in the reactor system and are

[1] Boyack, B., et al., Quantifying Reactor Safety Margins, NUREG/CR-5249, December 1989

knowledgeable in the thermal hydraulic behavior expected during the class of transients being considered. As described in NUREG/CR-5249, the ranking technique is designed to direct the examination of uncertainty to those processes that have the most significant impact on the parameter being analyzed. Technical specialists involved with the PIRT development examine the transients in question to identify the underlying thermal hydraulic processes and phenomena. Then, the processes and phenomena are ranked with respect to the expected effect on the transient.

There are numerous approaches to analyze uncertainty. Most have been developed within the field of statistics and several of these approaches have been considered and applied in the field of nuclear safety. Nuclear systems codes present a challenge, however, since these codes can require large computational times for a single transient. If one had the ability of making an "infinite" number of computer runs, one could develop and analyze the exact distribution of the output variable to determine uncertainty. Since that ability does not exist in a practical sense when using nuclear systems codes, the issue becomes the number of runs needed to quantify uncertainty. Thus, a key aspect of selection of an uncertainty method is the number of code runs that needed to quantify the uncertainty of a code output parameter as defined by the probability of exceeding a given value to a certain confidence limit.

In the CSAU study, a response surface was used to determine the uncertainty in the peak clad temperature. Response surface methodology is a powerful tool for the development of empirical models of process variables for which little is known. It is widely used for optimization of chemical or biological industrial processes. Sampling of response surfaces was among the earliest methods for determining the uncertainty in computer code results. A response surface is basically a polynomial fit to the data. The number of experiments required to create the response surface depends on the complexity of the polynomial and the number of inputs selected. Unfortunately, a LOCA and most other transients related to nuclear safety is complex and can involve processes that are difficult to simulate. This creates difficulty in selecting parameters for the polynomial and adequate coverage of all processes that influence the result.

It is noteworthy that the CSAU calculation matrix was based on a 7 factor, mixed (3 to 5) level experimental design that utilized a total of 184 TRAC runs (NUREG/CR-5249, p 82). The CSAU report also notes that a complete three-level design would require more than 2000 TRAC calculations. With modern computer "clusters", making 180+ TRACE runs is feasible, although for some applications this would still require a great deal of time. A key aspect of uncertainty methods is to evaluate other approaches used in uncertainty and make the number of TRACE runs required manageable. Hence, a method that requires 180+ calculations is at least suspect, especially so in considering other issues related to response surface methodology.

More recently, non-parametric methods have been used to determine the uncertainty in code output variables. Non-parametric methods were first developed in the 1940's and have been used in a variety of fields, including manufacturing. The methods published by S.S. Wilks [2,3] in the early 1940s have been used as the basis for the non-parametric statistical approaches.

Of particular interest is the method by Guba, et al.[4] who discuss a statistical approach in the context of best-estimate analysis, that can be formulated as:

$$X_{BE}(\text{licensing}) = X_{\text{base case}} + \text{Plant uncertainty} + \text{Code uncertainty}$$

[2] Wilks, S. S., Determination of Sample Sizes for Setting Tolerance Limits, Annals of Mathematical Statistics, Vol. 12, No. 1, pp. 91-96, 1941.

[3] Wilks, S. S., Statistical Prediction with Special Reference to the Problem of Tolerance Limits, Annals of Mathematical Statistics, Vol. 13, No. 4, pp. 400-409, 1942

[4] Guba, A., Makai, M., and Pal, L., Statistical Aspects of Best Estimate Method – I, Reliability Engineering and System Safety Volume 80, pg 217-232, 2003

where the purpose of the uncertainty evaluation is to provide assurance that X_{BE} which has a probability of 95% or more will not exceed the acceptance criteria for a given accident. Guba et al. focuses on the determination of code uncertainty in his paper. Guba's basic premise is that the observed randomness of the code output variable is due to randomness of the code input variables. He does discuss the possibility of nonlinearities due to chaotic behavior in the computer model, however he assumes that there is no chaotic behavior in the computer code. Given N independent calculations of the random code output and a continuous probability density function with an unknown distribution, using Wilkes, Guba proposes that the upper one-sided tolerance interval is

$$\beta = 1 - \gamma^{N}$$

where β is the confidence level, γ is the probability and N is the number of computer runs. For a 95% probability and 95% confidence level, the value of N becomes 59. This equation is applicable to the case where there is a single code output parameter of interest. Thus, a code methodology with this formulation may reduce the number of calculations (compared to the response surface approach in CSAU) and also provide a confidence level.

To meet the requirements of 10CFR50.46, the criteria for peak clad temperature (< 2200 °F), total oxidation of the cladding (0.17 of the total cladding thickness before oxidation, and the total hydrogen generation (0.01 of the total amount of the fuel cladding were to oxidize and produce hydrogen) must be met. As a result, there are three code output parameters of interest. According to Guba, the sample size must be increased to account for the additional parameters. The equation used to determine the number of computer runs needed for three parameters, assuming a one-sided tolerance interval is

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

where p is the number of parameters. For the case where p = 3, and for a 95% probability and 95% confidence level, the value of N becomes 124. While still a large number of calculations, this is feasible provided the code is robust and has few failures due to numerical difficulties.

To determine an appropriate uncertainty method for the Next Generation Nuclear Plant (NGNP), these and other potential approaches to the treatment of code uncertainty were examined and evaluated. It was concluded that that an order statistics based methodology be used in the NGNP EM uncertainty analysis based on considerations of:

- Regulatory Acceptance
- Feasibility of implementation
- Extensibility of methodology
- Ease of use and effort to perform analyses
- Effort to implement methodology

Regulatory acceptance was considered to the acceptability of the method in past and present licensing applications. Ordered statistics is an approach that is being used by some applicants and international regulatory bodies[5,6,7]. Another key factor in the suitability of the proposed methodology was its

[5] C. Frepoli, "An Overview of the Westinghouse Realistic Large Break LOCA Evaluation Model", Science and Technology of Nuclear Installations, Vol. 2008.

extensibility – the ability of the methodology to be adapted to changes in the structure, codes, and inputs. Since in any of the available methods a large number of calculations is anticipated, the "ease of use" was likewise an important consideration. While the focus of the uncertainty method evaluation was its application to the NGNP, the findings were considered general and can be applied to other systems such as light-water reactors (LWR). Thus, an ordered statistics based approach was selected for TRACE applications. Because 10 CFR50.46 requires that three parameters be within specified limits (peak cladding temperature, maximum local oxidation, and total core-wide oxidation) the method proposed by Guba et al. will be used. The framework for code uncertainty using TRACE will allow a user to specify the number of desired output variables, along with the desired probability and confidence level and the expression by Guba will determine the necessary number of calculations. This will allow users to calculate multiple output parameters and also examine the sensitivity of the results to one's definition of a "high probability."

4. Uncertainty Contributors

The effort to implement an uncertainty methodology depends on several factors; the need to create new tools and/or the necessary modification of existing tools, and the resources required to accomplish these tasks. The effort is seen to involve the type and number of uncertainty contributors. The approach in CSAU was, in effect, to use the PIRT to reduce the number of input parameters that require uncertainty quantification to a manageable set. While the PIRT process is enlightening, PIRTs are subjective as they are based on the judgment and experience of the participating group of engineers. In addition, PIRTs are specific to a particular plant design and transient. Thus, in GRS method[8], the guidelines suggest that the use of judgment to determine which parameters are important is not required and all inputs with uncertainty should be included, "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.". Guidelines similar to the GRS approach put more effort on eliminating uncertain input parameters based on a PIRT, such that the number of code runs and number of input parameters requiring uncertainty information is limited[9] .

In the NRC uncertainty methodology currently under development, the CSAU approach of using a PIRT is being followed. However, as the NRC's objective is to develop a tool by which an applicant's method can be evaluated, the uncertainty parameter selection is being expanded beyond what might be contained in an individual PIRT. Rather, for each transient applicable PIRTs were reviewed, a comprehensive PIRT containing processes for a particular transient and plant type was developed. Large break LOCA (LBLOCA), small break LOCA (SBLOCA), and Main Steamline Break (MSLB) transients were considered. Plant types included Westinghouse 2-loop PWRs with upper plenum injection, Westinghouse 3 and 4 loop PWRs, Combustion Engineering PWRs, Babcock & Wilcox PWRs, General Electric BWRs, AP1000, EPR, APWR, ESBWR, and the ABWR. For major plant type (PWR or BWR) and transient (LBLOCA, SBLOCA, MSLB) a comprehensive PIRT was established.

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[6] R. P. Martin and L. D. O'Dell, "AREVA's realistic large break LOCA analysis methodology", Nuc. Eng. And Design, Vol. 235, 2005.

[7] L. Blake, G. Gavrus, J. Vecchiarelli, and J. Stoklosa, "Best Estimate Plus Uncertainty Analysis of LBLOCA for a Pickering B CANDU Reactor", ASME 2010 3rd Joint USEuropean Fluids Engineering Summer Meeting, August 2010.

[8] H. G. Glaeser, "Uncertainty Evaluation of Thermal-Hydraulic Code Results", ANS International Meeting on Best Estimate Methods in Nuclear Installations Safety Analysis, Washington, November 2000.

[9] "Report on the Uncertainty Methods Study", NEA/CSNI/R(97)35, June 1998.

These comprehensive PIRTs identified two overall categories of uncertainty parameters:

- TRACE Code Model Parameters
- Input Model Parameters

That is, the PIRTs were used to identify which uncertainty terms could be ranged by varying input to TRACE, and which terms could be varied only with a modification of the TRACE executable. Input model parameters also included parameters such as break discharge coefficient, containment back-pressure and power shape, each of which could be considered as boundary conditions to the analysis. This helped to define the overall framework and approach. Parameters that should be varied as part of the input can be addressed using SNAP, which is the NRC's tool for input deck preparation. Model parameter changes must be made in TRACE, but with new input parameters to provide instructions to the code for a particular run. Preparations are now being made to modify TRACE so that models and correlations can be ranged. Specifics on how this is to be accomplished, is described in the next section on SNAP and the framework for making uncertainty calculations.

5. SNAP Interface and Implementation

The framework to be employed in uncertainty analyses is comprised of three principle items. These items specify and define the means through which the analysis is performed and the uncertainty in the selected FOM calculated. In broad terms the required items are:

- Specification of an uncertainty quantification methodology
- A means of performing any calculations required for the uncertainty and sensitivity analyses
- A tool or set of tools for executing the analysis, including the uncertainty analysis and linked code executions

The framework for uncertainty analyses is based on SNAP and DAKOTA. SNAP is used to coordinate the analysis and facilitate communication between each of the component of an evaluation model. DAKOTA is used to perform the quantification of uncertainty and the sensitivity analyses.

SNAP[10] (Symbolic Nuclear Analysis Package) is a framework created to simplify analysis for supported engineering codes. SNAP provides utilities for graphically setting up models/input as well as facilities for running and monitoring supported codes. SNAP has been designed to allow users to create custom plug-ins for the support of additional codes. SNAP also contains functions to allow for the coupling of codes (using Job Streams) under SNAP[11]. It is possible to include codes that do not have SNAP plug-ins (Black box applications); however, full advantage of the features in SNAP cannot be utilized in these applications. SNAP also supports parametric inputs and has support for the DAKOTA code. The uncertainty quantification (UQ) features implemented in SNAP makes heavy use of the DAKOTA code.

[10] Applied Programming Technology, Inc., "Symbolic Nuclear Analysis Package (SNAP), User's Manual", April 2007.

[11] Applied Programming Technology, Inc., "Symbolic Nuclear Analysis Package (SNAP), Job Stream Design", DRAFT, July, 2010.

DAKOTA[12] (Design and Analysis Toolkit for Optimization and Terascale Applications) is a code developed by the SANDIA National Laboratory that is designed to perform sensitivity and uncertainty analyses. DAKOTA provides a means of performing detailed sensitivity and uncertainty analyses for general simulation codes. DAKOTA acts as an interface that permits the imposition of uncertainty on code inputs, the execution of a simulation code, and post processing to perform detailed sensitivity and uncertainty calculations. The structure of DAKOTA allows for the execution of simulation codes through a direct library interface, system calls, or a process fork.

DAKOTA is capable of generating samples (either random or using a Latin hypercube design) from a variety of distributions. These distributions include uniform, normal, triangular, gamma, beta, gumbel, and others. In addition, discrete parameters can be drawn from binomial, hypergeometric, Poisson, and others. A correlation matrix can be supplied to DAKOTA which specifies the relationship between the uncertain parameters.

DAKOTA is capable of performing both local and global sensitivity results on the FOM. DAKOTA is also capable of generating a variety of surrogate models relating the input parameters to the FOM. These include, least squares polynomial regression and Gaussian surfaces.

Sensitivity methods include:

- Morris One-At–Time (OAT) (used as a data screening technique in DAKOTA)
- Variance based approaches (similar to the Sobol method)
- Correlation (both Pearson and Spearmann)
- Local gradients

The process flow showing the communications between SNAP and DAKOTA is shown in Figure 1.

The SNAP graphical-user-interface simplifies the required user inputs for DAKOTA as well as controlling and gathering the output from the DAKOTA processes. A soon to be available feature of the SNAP UQ implementation allows for multiple interactions with DAKOTA after the Monte-Carlo sampled runs have been completed, thus permitting SNAP to use DAKOTA's more advanced uncertainty analysis features such as reliability methods; FORM (First Order Reliability Method) and SORM (Second Order Reliability Method), nested (or second order) methods, as well as others.

6. Using SNAP to Perform an Uncertainty Quantification (UQ) Analysis

The process that an analyst must go through when using the SNAP uncertainty quantification (UQ) plug-in is straightforward. In fact, certain aspects of the process may be implemented as SNAP based "templates" to allow faster processing of common types of analysis.

Under the current design of the SNAP UQ plug-in a UQ analysis procedure would follow the following steps, also shown conceptually in Figure 2.

1) Select appropriate plant model

2) Identify model input parameters relevant to the transient. SNAP engineering templates could be useful here to recommend to the analyst what parameters are important for particular transients.

[12] B. M. Adams, et. al., "DAKOTA. A Multilevel Parallel Object-Oriented Framework for Design Optimization, Parameter Estimation, Uncertainty Quantification, and Sensitivity Analysis: Version 5.0 Developers Manual", Sandia , SAND2010-2185, May 2010.

3) Use SNAP to "Tag" input parameters to designate the range and distributions to be used in the UQ process. Again, SNAP engineering templates can be used to suggest useful ranges and values.

4) Specify the parameters of the UQ, such as identifying the outputs of interest for this UQ and specifying the desired probability and confidence limits.

5) Start the SNAP UQ job-stream. The SNAP UQ job-stream then:

- Generates the input to DAKOTA,
- DAKOTA then generates the random variates
- SNAP integrates the random variates into the model
- SNAP then spawns a TRACE run corresponding to each set of variates
- SNAP collects the results and generates another DAKOTA input deck
- DAKOTA performs the requested analysis
- SNAP collects the DAKOTA results and formats then into a report as specified by the analyst

A visualization of the SNAP UQ job-stream is shown in Figure 3.

7. Status of the SNAP UQ plug-in Capabilities

The following features and capabilities are currently present in the SNAP UQ plug-in:

1. Automatic calculation of the number of samples required for an order statistics uncertainty analysis
2. Implementation of all DAKOTA Distribution types
3. Report the Results of Uncertainty Analysis
4. Implement Uncertainty Analysis in Engineering Templates (this provides the capability for UQ on a group of codes)

The following features are being actively developed and should be available in SNAP by mid-2012[13]:

1. Inclusion of Sensitivity Analysis Options
2. Inclusion of Variance Based and Morris Sensitivity Analysis Capability
3. Inclusion of Local Sensitivity Analysis Capability
4. Implement Looping Capability in Engineering Templates

8. Summary

The framework for an uncertainty methodology to be used with TRACE has been formulated and is under active development. The framework makes use of SNAP to rapidly perform the numerous individual calculations that must be performed. DAKOTA is used to perform the statistics. While the PIRT process originally developed as part of CSAU is used to identify uncertainty parameters, the methodology is not restricted, nor dependent on the parameters that may have been selected. A user will be able to add (to reduce) the number of parameters, if that is needed as part of the investigation. Recommended sets for each plant and transient will be developed as default, however, the user will have the ability to examine assumptions on the bias and uncertainty of a selected parameter. Both plant input parameters and code model parameters can be ranged. Flexibility is being built in to the framework to allow evaluation of most plant types of interest to the NRC.

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[13] Applied Programming Technology, Inc., "Symbolic Nuclear Analysis Package (SNAP), Uncertainty Analysis Plug-in Design", DRAFT, August 2010.

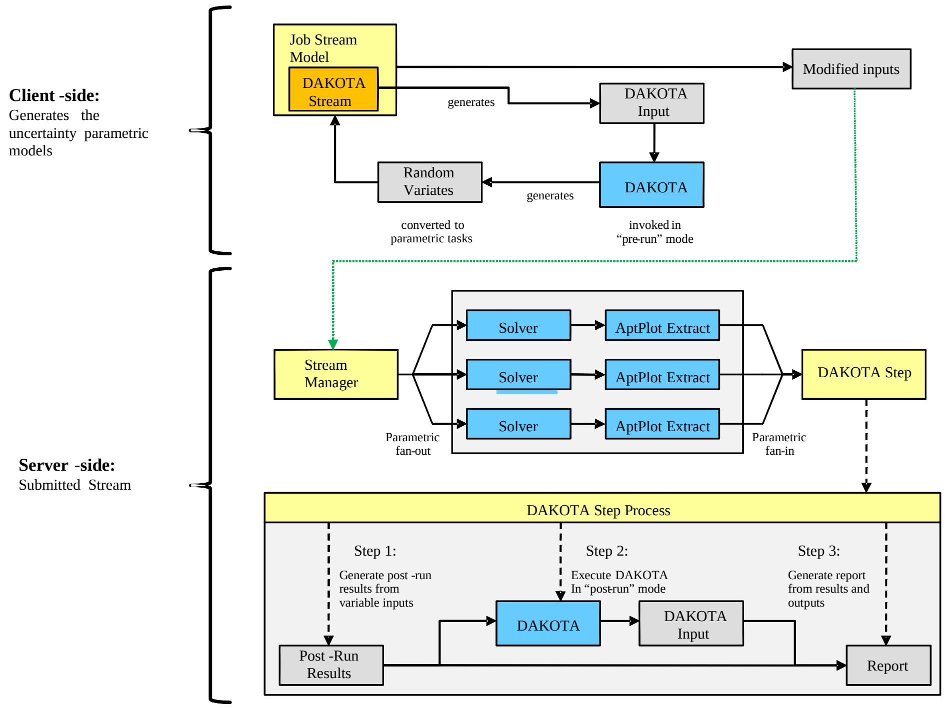

Figure 1. Process flow showing the communications between SNAP and DAKOTA.

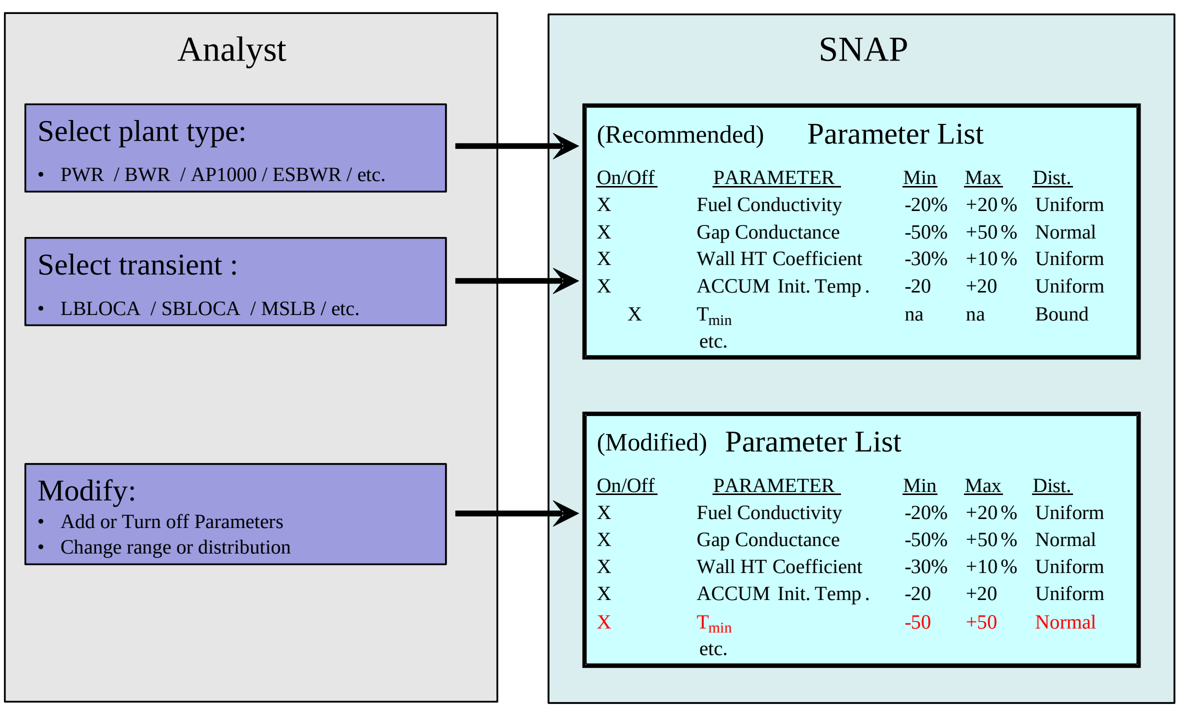

Figure 2. SNAP UQ plug-in analysis procedure.

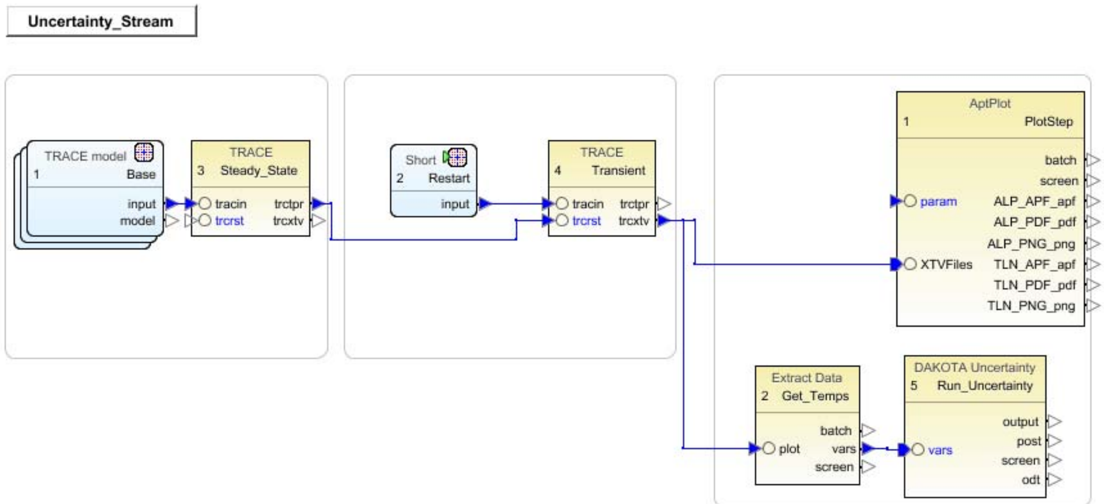

Figure 3. The SNAP UQ job-stream.

Westinghouse Experience in Licensing and Applying Best-Estimate LOCA Methodologies within the Industry: Past, Present and Future

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Abstract

Since the amendment of the 10 CFR 50.46 rule in 1988, Westinghouse has been developing and applying realistic or best-estimate methods to perform Loss of Coolant (LOCA) safety analyses for PWRs. Then main focus has been historically on a large break rupture postulated to occur in the cold leg of the reactor coolant system (RCS). Such scenario, called Large Break LOCA is still considered the design basis accident (DBA) by the current rule. A realistic analysis requires the execution of various realistic LOCA transient simulations where the effect of both model and input uncertainties must be accounted for by propagating these uncertainties throughout the transients. The outcome is typically a range of results with associated probabilities.

Westinghouse realistic Large Break LOCA methodology is based on the use of the WCOBRA/TRAC code. The methodology follows the Code Scaling, Applicability and Uncertainty (CSAU) methodology (Boyack et al., 1989)[1]. The original Westinghouse implementation of the CSAU methodology was approved by the NRC[2] in 1996 after an extensive review. The technique used to combine the uncertainties evolved over the years. In its original implementation Westinghouse methodology followed strictly CSAU where the use of response surface was suggested as a practical means to combine the various uncertainty components. A Monte Carlo sampling of uncertainties was applied to the surrogate model to obtain a probabilistic statement for the purpose of showing compliance with regulatory acceptance criteria. More recently the Westinghouse methodology[3] was updated using the non-parametric method which has become a standard within the industry since GRS, AREVA, GE, etc. adapted similar methods for combining the uncertainties. This paper will discuss some of advantages but also challenges encountered in applying these methodologies for licensing application.

Since its approval, Westinghouse Best Estimate Large Break LOCA Evaluation Model has applied to more than 75% of the Westinghouse PWR fleet in the US and also licensed for several international applications.

[1] Boyack, B. E., et al., 1989, "Quantifying Reactor Safety Margins," NUREG/CR-5249.

[2] Jones, R.C. (USNRC), Letter to Liparulo, N.J. (Westinghouse), 1996. Acceptance for referencing of the topical report WCAP-12945(P). Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis, 28 June.

[3] Nissley, M. E., Frepoli, C., Ohkawa, K., and Muftuoglu, K., 2003, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," Technical Report WCAP-16009-NP, Westinghouse Electric Company, Pittsburgh, PA.

Recently changes in the regulatory environment toward a risk informed approach combined with more efficient and demanding fuel power cycles, and utilization of margins put more emphasis in scenarios traditionally defined as Small and Intermediate Break LOCA. As a result, Westinghouse made several upgrades for the purpose of extending the EM applicability to smaller break sizes. Also the uncertainty methodology was upgraded to reflect a larger coverage of scenarios. The new EM is called Westinghouse Full Spectrum LOCA (FSLOCA^{TM}) Methodology and is intended to be applicable to a full spectrum of LOCAs, from small to intermediate break as well as large break LOCAs. A section in this paper provides a high level overview of the FSLOCA^{TM} and how it will be utilized in the future.

Keywords: Thermal hydraulics, LOCA, Evaluation Model, Safety Analysis

1. Introduction

The 1988 amendment of the 10 CFR 50.46 rule allowed the industry to use best-estimate evaluation models with realistic physical models for the purpose of analyzing Loss of Coolant Accidents (LOCA). A group of experts (referred to as the technical program group or TPG) under the sponsorship of the USNRC undertook to demonstrate that practical best-estimate plus uncertainty methods could be developed and applied in the licensing space. Shortly after its completion, the Code Scaling, Applicability and Uncertainty (CSAU) methodology and its demonstration were described in a series of papers appearing in Nuclear Engineering and Design by Boyack[4], Wilson[5], Wulff[6], Lellouche and Levy[7], and by Zuber and Wilson[8].

The methodology was reviewed by the technical community, and comments were published in Nuclear Engineering and Design (Letters to the Editors, by Wickett and Brittain[9], Hochreiter[10], Hofer and Karwat[11], Holmstrom and Tuomist[12], Sursock[13], Petrangeli[14], Shiba[15], and Yadigaroglu[16]). 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. The comments and questions raised were responded by TPG[17].

[4] Boyack, B.E., 1990, “Quantifying Reactor Safety Margins-Part 1: an overview of the code scaling, applicability, and uncertainty evaluation methodology,” Nuclear Eng. Des. 119, 1–15.

[5] Wilson, G. E., 1990, “Quantifying Reactor Margins – Part 2: Characterization of Important Contributors to Uncertainty,” Nuclear Eng. Des. 119, 17-31.

[6] Wulff, W., 1990, “Quantifying Reactor Safety Margins Part 3: Assessment and Ranging of Parameters,” Nuclear Eng. Des. 119, 33-65.

[7] Lellouche, G. S., and Levy, S., 1990, “Quantifying Reactor Safety Margins Part 4: Uncertainty Evaluation of LBLOCA Analysis Based on TRAC-PF1/MOD 1,” Nuclear Eng. Des. 119, 67-95.

[8] Zuber, N., and Wilson, G. E., 1990, “Quantifying Reactor Safety Margins Part 5: Evaluation of Scale-up Capabilities of Best Estimare Codes,” Nuclear Eng. Des. 119, 97-107.

[9] Wichett, A. J., and Brittain, I., 1992, “Quantifying reactor safety margins: what remains to be done,” Nucl. Eng. Des. 132, 405-408.

[10] Hochreiter, L.E., 1992. “Comments on ‘Quantifying Reactor Safety Margins’,” Nucl. Eng. Des. 132, 409–410.

[11] Hofer, E., Karwat, H., Werner, W., 1992, “Comments on ‘Quantifying Reactor Safety Margins’,” Nucl. Eng. Des. 132, 411–414.

[12] Holmstrom, H, Tuomisto, H.., 1992, “Comments on ‘Quantifying Reactor Safety Margins’,” Nucl. Eng. Des. 132, 415–416.

[13] Sursock, J., 1992, “Comments on ‘Quantifying Reactor Safety Margins’,” Nucl. Eng. Des. 132, 417–418.

[14] Petrangeli, G., 1992, “Comments on ‘Quantifying Reactor Safety Margins’,” Nucl. Eng. Des. 132, 419–422.

[15] Shiba, M., 1992, “Comments on ‘Quantifying Reactor Safety Margins’,” Nucl. Eng. Des. 132, 423.

[16] Yadigaroglu, G., 1992, “Comments on ‘Quantifying Reactor Safety Margins’,” Nucl. Eng. Des. 132, 425-429.

[17] Boyack, B.E., et al., 1992. TPG response to the foregoing letters-to-the-Editor. Nuclear Eng. Des. 132, 431–436.

Westinghouse development of the best-estimate methodology showed that many of these concerns could be addressed[18]. Acceptance of the methodology was achieved in 1996, after a rigorous review spanning three years. During this time, additional analysis and methodology developed during rigorous, extensive, and sometimes contentious discussions and reviews with the Advisory Committee for Reactor Safeguards ultimately would provide the basis for acceptance by the USNRC of the overall methodology. A summary of the technical review and the conditions of acceptance was issued by the USNRC[2]). In 1996, this was the first Best-Estimate (BE) LOCA evaluation model approved within the industry[19] and nowadays BE LOCA methods are extensively employed within the nuclear industry overall.

Since the amendment of the 10 CFR 50.46 rule in 1988, Westinghouse has been applying realistic or best-estimate methods to perform Loss of Coolant (LOCA) safety analyses for PWRs. Then main focus has been historically on a large break rupture postulated to occur in the cold leg of the reactor coolant system (RCS). Such scenario, called Large Break LOCA is still considered the design basis accident (DBA) by the current rule.

An overview of Westinghouse realistic Large Break LOCA methodology is presented by the author in other paper[20]. The methodology is based on the use of the WCOBRA/TRAC code[19]. Westinghouse acquired the COBRA/TRAC computer code which was originally developed at Pacific Northwest Laboratory by Thurgood et al. [21] for the US Nuclear Regulatory Commission (USNRC). COBRA/TRAC was the result of coupling the COBRA-TF [22] and the TRAC-PD2 [23] codes. Westinghouse continued the development of COBRA/TRAC through an extensive assessment against several Separate Effect Tests (SETs)[24] and Integral Effect Tests (IETs). The code was then renamed to WCOBRA/TRAC after a successful validation of the code.

A realistic analysis requires the execution of various realistic LOCA transient simulations where the effect of both model and input uncertainties must be accounted for by propagating these uncertainties throughout the transients. The outcome is typically a range of results with associated probabilities.

In its original form as approved in 1996, the uncertainty methodology was based on the response surface technique[19]. Subsequently the uncertainty methodology was significantly improved and streamlined by replacing the response surface technique with a direct Monte Carlo convolution of the uncertainty components. The new "Automated Statistical TReatment of Uncertainty Method" (ASTRUM)[25, 20] was approved by the US NRC in November 2004.

Westinghouse has applied the methodology extensively for licensing application in the US market as well as internationally, for operating plant as well as for the Design Certification of new plants (AP600 and AP1000). Currently BE LBLOCA safety analyses support more than 75% of W-design PWR fleet.

[18] Bajorek, S. M., et al., 1998, "Code Qualification Document for Best Estimate LOCA Analysis," Technical Report WCAP 12945 NP-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, Westinghouse Electric Company, Pittsburgh, PA.

[19] Young, M. Y., Bajorek, S. M., Nissley, M. E., and Hochreiter, L. E., 1998, "Application of code scaling applicability and uncertainty methodology to the large break loss of coolant," Nuclear Engineering and Design 186 (1998) 39-52.

[20] Frepoli, C., 2008, "Review Article - An Overview of Westinghouse Realistic Large Break LOCA Evaluation Model," Science and Technology of Nuclear Installations, Vol. 2008, Hindawi Publishing Corporation, Article ID 498737.

[21] Thurgood, M. J., et al., 1983, "COBRA/TRAC - A Thermal-Hydraulics Code for Transient Analysis of Nuclear Reactor Vessels and Primary Coolant systems," NUREG/CR-3046.

[22] Thurgood, M. J., et al., 1980, "COBRA-TF Development," 8th Water Reactor Safety Information Meeting.

[23] Liles, D. R., et al., 1981, "TRAC-PD2, An Advanced Best Estimate Computer Program for Pressurized Water Reactor Loss-of-Coolant Accident Analysis," NUREG/CR-2054, Los Alamos National Laboratories, NM.

[24] Paik, C.Y. and Hochreiter, L.E., 1986, "Analysis of FLECHT-SEASET 163 Rod Bundle Data Using COBRA-TF," NUREG/CR-4166.

[25] Nissley, M. E., Frepoli, C., Ohkawa, K., and Muftuoglu, K., 2003, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," Technical Report WCAP-16009-NP, Westinghouse Electric Company, Pittsburgh, PA.

Recently changes in the regulatory environment toward a risk informed approach combined with more efficient and demanding fuel power cycles, and utilization of margins put more emphasis in scenarios traditionally defined as Small and Intermediate Break LOCA. As a result, Westinghouse made several upgrades and added several new functionalities to the code, which has now been renamed to WCOBRA/TRAC-TF2, for the purpose of extending the EM applicability to smaller break sizes.

The new EM is called Westinghouse Full Spectrum LOCA (FSLOCA™) Methodology and is intended to be applicable to a full spectrum of LOCAs, from small to intermediate break as well as large break LOCAs. Section 4 in this paper provides a high level overview of the FSLOCA™ and how it will be utilized in the future.

2. Regulations, Regulatory Guides, Acceptance Criteria and its Compliance

A 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

The acceptance criteria above were established following an extensive rulemaking in 1973. The issue has been to demonstrate with high confidence that such limits are not exceeded would a LOCA event occur. The original regulations governing the methods and assumptions to be used in the analysis of the loss of coolant accident (LOCA) in LWR's, described in Appendix K of 10CFR 50.46, required conservative assumptions and prescriptive rules for the analysis. This resulted in very conservative estimates of the consequences of such accidents, forcing unnecessary restrictions on plant operation. At the time the intent of the regulation was 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; etc.

It was the recognition of this burden on plant operations that led the US Nuclear Regulatory Commission (USNRC), during the 1970's and 1980's, to sponsor an extensive international program to develop realistic physical models and supporting data to more accurately predict the evolution of the loss of coolant accident. The effort was both on the experimental side and analytical side (computer codes, evaluation models). The major contributor to this international development effort was the 2D-3D program which focused on multi-dimensional 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).

Westinghouse played a significant role in this effort; among others notably is the FLECHT program. Beginning in the mid 1970's, these test programs generated a vast amount of fundamental heat transfer data and reactor coolant system response data during the most critical period of the LOCA in a PWR. These data were used to develop improved physical models that were incorporated into advanced computer codes also being developed during this time period[26, 24].

The knowledge gained over the years is really what led the industry to consider a more realistic approach for the analyses of the LOCA scenario. Note that for the 1988 rule amendment (10 CFR

[26] Hochreiter, L. E., "FLECHT SEASET Program: Final Report", EPRI NP-4112, September 1986.

50.46) 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[1] was cited as an acceptable methodology framework.

More recently the CSAU approach was further formalized and endorsed in the Evaluation Model Development and Assessment Process (EMDAP) which was published in Standard Review Plan[27] and Regulatory Guide 1.203[28] in 2005. The EMDAP process is depicted in the flowchart of Figure 1.

2.1 The Evaluation Model Development and Assessment Process (EMDAP)

Similarly to the CSAU, the EMDAP is divided in four main elements. In Element 1, the scenario is characterized and the nuclear power plant broken down into relevant regions (e.g., fuel rod, core, lower plenum, etc.). 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 on the prioritization of code assessment which helps facilitate the decisions on physical model and methodology development.

The PIRT is a practical and useful method but its shortcomings were recognized since its introduction. In particular, the human judgment factor and the fact that knowledge gained is not always factored back into final documentation were seen as weakness. Soon after its introduction, the CSAU methodology was reviewed by the technical community. 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[9-17].

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, Reg. Guide 1.157[29] 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 has been definitely demonstrated by several successful implementations of the CSAU-based methodologies currently licensed and applied to safety analysis in the industry over the past three decades.

Elements 2, 3 and 4 describe a suitable process for the development and the assessment of the evaluation model (EM), sometimes referred to as Verification and Validation (V&V). 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.

[27] NUREG-0800, "Standard Review Plan - 15.0.2 Review of Transient and Accident Analysis Methods", December 2005.

[28] USNRC, "Transient and Accident Analysis Methods," Regulatory Guide 1.203, December 2005.

[29] USNRC, "Regulatory Guide 1.157 (Task RS 701-4), Best-Estimate Calculations of Emergency Core Cooling System Performance", May 1989.

One topic of debate in the V&V process is that there is an apparent incongruence between the 'attempt' of modelling thermal-hydraulic processed at a very fine scale with semi-mechanistic models while both PIRT, SET and IET are only able to characterize an aggregate of such sub-processes. First the code models are still empirical and based on a collection of correlations (closure relationships) which essentially are data fit and really not solving the fundamental physics underneath. Second, the data from the experiments is not even detailed enough to characterize all those sub-process at the same level of the current computer code describes them.

This 'gap' is resolved by indentifying macroscopic controlling parameters whose bias and uncertainty is shown to aggregate the uncertainty associated with various individual sub-process modelled by the computer code. At the end the code is shown to capture bias and uncertainty which results from comparing the code prediction of rather 'integral' SETs. The robustness of such approach often resides upon a detailed compensating error analysis which demonstrates that either the code is capable of predicting the right results for the right reasons, or, if distortions are found, they are determined to be in the conservative direction to help the safety case for the plant under study.

The key output from the EMDAP is to demonstrate the code's ability to correctly predict all the dominant physical processes during the transient which leads to the adequacy decision of the evaluation model overall. Another important output from EMDAP is the establishment of probability distributions and biases for the important contributors identified in the PIRT.

Once the EM is judge to be adequate for its intended purpose, the next step is to decide on method to model the plant and combine model and input uncertainties in order to generate a probabilistic statement that satisfy the intent of the rule (10 CFR 50.46).

The influence of the user on the results has been recognized as another potential source of uncertainty [30]. To eliminate such variability several engineering safeguards or procedures are considered as part of the methodology. A rigorous consistency between the noding utilized to model prototypical SET and IET and PWR is enforced. Calculation of plant-specific inputs and set up of initial and boundary condition 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 practical. Steady-state criteria are established to minimize variability of initial conditions. Following these procedures, consistency with the code assessment conclusion is ensured and "user effects" are virtually eliminated.

2.2 Best-Estimate Plus Uncertainty (BEPU) within a Statistical Framework and its Challenges

The key step in a realistic analysis is the assessment of uncertainties associated with physical models, data uncertainties, and plant initial and boundary condition variability. 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 […]*".

The development of the uncertainty methodology is probably the most straightforward of all the CSAU elements. Once the dominant contributors and their probability distributions are properly

[30] Aksan S. N., D'Auria F., Staedtke H, "User Effects on the Thermal-hydraulic Transient System Codes Calculations" *J. Nuclear Engineering & Design*, Vol 145, Nos. 1&2, 1993.

identified and quantified and 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, several well established methods are available to perform the uncertainty propagation step. The choice of method is basically a practical one, controlled by a trade-off between a stable estimator (in a statistical sense) and expenses incurred in performing and documenting several computer calculations. The methods utilized evolved over the last two decades.

The CSAU originally suggested the use of response surfaces methods. 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 (transfer function) between let's say the PCT and a set of values for the uncertainty attributes. Finally these response surfaces are used in a Monte Carlo simulation to generate the output distribution (PCT Cumulative Distribution Functions (CDF) for instance).

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[19] was based on the use of response surfaces. The final embodiment of the response surface technique as part of the original 1996 Westinghouse LBLOCA evaluation model was cumbersome and significantly more sophisticated than what the original CSAU developers had envisioned.

Another 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.

Ideally the uncertainty propagation model could be solved with a direct Monte Carlo method. The implementation of the method is straightforward where it simply requires sampling of the input distributions *n* times, then by using the computer code directly generating *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 for instance).

Several years ago, this approach was considered impractical due to the number of calculations involved (on the order of thousands). This may not be true today, however. While there are still several issues to be resolved 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 could be considered today.

A more practical approach that has become widely used in the industry in the last decade uses a direct Monte Carlo method for the uncertainties convolution but to infer probabilistic statements with reduced sample sizes. These methods are based on non-parametric order statistics procedures. Instead of attempting in obtaining information with regard to underneath probability distribution function (PDF) of the measure (say PCT), the PDF is ignored (thus distribution-free) and non-parametric statistics is used to determine a bounding value (tolerance limits) of the population with a given confidence level.

Non-parametric methods started to be applied in realistic calculations of LOCA/ECCS analysis in the late 90's[31]. Although, there are some conceptual similarities, most of these methods, started to be employed in Europe in the late 90's by Glaeser et al.[32,33,34]. In US, both AREVA-NP in 2003[35] and

[31] Wickett, T. (editor), et al., 1998, "Report of the Uncertainty Methods Study for Advanced Best Estimate Thermal Hydraulic Code Applications," NEA/CSNI/R(97)35.

[32] Glaeser, H., "User effects in application of thermal-hydraulic computer codes", paper presented at IAEA Specialists Mtg on User Qualification for and User Effect on Accident Analysis for NPPs, Vienna, 1998.

Westinghouse in 2004[20] have developed NRC-licensed Best-Estimate LOCA Evaluation Models based on the use of these methods. Other applications seen in the industry are the Extended Statistical Method (ESM) by AREVA/EDF in France[36] and the GE's application to non-LOCA events[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.

Westinghouse embodiment of the methodology is based on Guba et al.[38], which required the extension of the sample size beyond the 59 runs suggested by Wilks[39] to bound a multi-variate population. Accordingly to Guba[37] the general formula applicable to one-sided populations with multiple outcomes (p>1) is the following:

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

The number of runs can be found solving the equation for N, by substituting a $\gamma = 0.95$ and $\beta = 0.95$, and $p = 3$. The number of computer runs, N is found equal to 124. This method was implemented in the Westinghouse Realistic Large Break LOCA evaluation model, also referred to as "Automated Statistical TReatment of Uncerainty Method" (ASTRUM)[25]. The ASTRUM Evaluation Model and its approach to the treatment of uncertainties was approved by the US NRC in November 2004.

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[40].

From a licensing stand point, the challenges within the industry was that regulatory guides (both in the US and oversea) 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 and other international regulators as acceptable. 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 US NRC to demonstrated

[33] Glaeser, H., Hofer, E., Kloos, M., and Skorek, T., 1998, "GRS Analyses for CSNI Uncertainty Methods Study (UMS)," in CSNI Report NEA/CSNI/R(97)35, Volume 1, June 1998.

[34] Glaeser, H., 2000, "Uncertainty Evaluation of Thermal-Hydraulic Code Results," Proc. Int. Mtg. on Best-Estimate.

[35] Martin, R. P. and O'Dell, L. D., 2005, "Framatome ANP's Realistic Large Break LOCA Analysis Methodology", Nucl. Eng. Des., Vol. 235, pp. 1713-1725.

[36] Sauvage, J.-Y., Keldenich, 2005 "ESM-3D: A Fully-Extended Statistical Method Based on CATHARE 3D for Loss of Coolant Accident Transient Analysis", Proceedings of ICONE-13 International Conference on Nuclear Engineering, May 16-20, 2005, Beijing, China.

[37] Bolger F. T., Heck C. L. and Andersen J. G. M., "TRACG Statistical Method for Analysis of Anticipated Operational Occurrences," International Congress on Advanced Nuclear Power Plants (ICAPP) 2002, Hollywood Florida, USA, June 9-13, 2002.

[38] Guba, A., Makai, M., Lenard, P., 2003, "Statistical aspects of best estimate method-I," Reliability Engineering and System Safety, Vol. 80, 217-232.

[39] Wilks, S. S., 1941, "Determination of Sample Sizes for Setting Tolerance Limits," The Annals of Mathematical Statistics, Vol. 12, pp. 91-96.

[40] Frepoli C., Oriani L., "Notes on the Implementation of Non-Parametric Statistics within the Westinghouse Realistic Large Break LOCA Evaluation Model (ASTRUM)", Proceedings of ICAPP '06, Reno, NV USA, June 4-8, 2006, Paper 6257.

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 phylosophy. It is recognized that many other layers of conservatisism 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.

In practice there are many ‘unresolved’ challenges with these methods. Regardless the different embodiments that can be found within the industry, the general approach is to rely on the minimum sample size and use the extreme case (rank k=1) as upper tolerance limit that statistically bounds a prescribed fraction of the population (typically 95%) with a high confidence (typically 95%). There is nothing wrong mathematically in the formulation. The issue is that the extreme case is subject to large variability. This presents two problems:

1) The estimator is affected by the seed selected during the Monte Carlo procedure. This requires the applicant to develop ‘robust’ seed (or re-seed) procedures that provides confidence to the regulator that the exercise is performed in fairness.
2) When we aim to a 95/95 upper tolerance limit, we are willing to accept that we can have 5% probability of being in error. In other words there is 5% probability that the estimate is actually lower than the 95% of the population (unknown). The heart of the matter is that we do not know by how much we are under-predicting. This presents some challenges to some regulators. The problem is somewhat alleviated within the Westinghouse embodiment (ASTRUM) because the procedure aims on a 95/95 for three variables (max of 124 instead of max of 59). In that case, on a single output the probability of underestimation is reduced to 0.2%.
3) It is often not practical to perform parameter sensitivities, or regressions analyses with small sample sizes. The sample size is often insufficient relative to the magnitude of the spread of the data in the sample.

Many of these issues are alleviated by simply increasing the sample size.

3. Next Generation of Westinghouse Realistic LOCA Evaluation Model: The Full Spectrum LOCA (FSLOCA™) Methodology

Westinghouse has historically maintained separate evaluation models (EM) for analyzing design-basis small and large break loss-of-coolant accidents (LOCA). This is due to the differences in the physical phenomena of importance for different break sizes, and the challenges in developing a single computer code which is robust enough to cover the entire break spectrum. Advances in computational tools and the use of statistical methods have also tended to be focused on the large break LOCA, as it has traditionally been more limiting from a design perspective than the small break LOCA.

Safety analysis of Small Break LOCA scenario has always used the conservative or bounding Evaluation Models as stipulated in Appendix K of 10 CFR 50.46 rule. However, regulatory initiatives to risk-inform the 10 CFR 50.46 rule, combined with operational initiatives that call for increased margins to support power uprates and improved operational flexibility, have increased the interest in improved and more realistic analytical capabilities for the small and intermediate break portion of the LOCA spectrum. Furthermore, while historically the focus in LOCA analysis have been on assessing system performance in term of Peak Clad Temperature (PCT) reached during the postulated transient, more emphasis is now given to the Maximum Local Oxidation (MLO) and its impact in maintaining a coolable geometry during a LOCA. At the same time, the MLO during a small break LOCA can 'artificially' challenge the limits for certain classes of Pressurized Water Reactors (PWRs) when the analysis is performed with conservative Appendix-K methods. Finally, data-driven best-estimate plus uncertainty methods are now preferred both by regulators and industry overall since they provide a more realistic representation of the physics and phenomena involved in such analyses.

In order to respond to these needs, it was desirable to have the capability to have a single analytical method, or EM, that covers the entire spectrum of break sizes, in a realistic fashion. In 2005 Westinghouse initiated a development program to fulfill this objective which resulted in extending the code applicability to smaller break sizes such that the same code can be applied to a full spectrum of LOCAs, from small to intermediate break as well as large break LOCAs. In order to properly model the Small Break LOCA scenario the code was subject to a significant amount of changes which led to the creation of the advanced WCOBRA/TRAC-TF2 computer code. The development of WCOBRA/TRAC-TF2 started by combining the 3D module of the current WCOBRA/TRAC MOD7A, Rev. 7 with the TRAC-P V.5.4.8. More detailed on the development of the new code can be found in several recent papers by the authors[41,42,43,44,45,46,47,48,49].

The development of the new evaluation model, called Westinghouse Full Spectrum LOCA (FSLOCA™), was just completed and submitted to the US NRC for their review and approval in November 2010.

The scenario being addressed by the FSLOCA methodology is a postulated loss of coolant accident that is initiated by an instantaneous rupture of a reactor coolant system (RCS) pipe. The break size considered for a split break ranges from the break size at which the break flow is beyond the capacity of the normal charging pumps up to a size equal to the area of a double ended guillotine rupture. The term 'Full Spectrum' specifies that the new EM is intended to resolve the full spectrum of LOCA scenarios which result from a postulated break in the cold leg of a PWR

The FSLOCA uncertainty methodology is also based on a direct Monte Carlo sampling of the uncertainty contributors. The process is overall similar to the approved methods – ASTRUM.

[41] Ohkawa, K., 2007, "Development of Homogeneous Non-equilibrium Relaxation Critical Flow Model with Non-condensable Gas," Paper-151, The 12th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-12), Pittsburgh, Pennsylvania, USA, September 30 – October 4.

[42] Frepoli, C. "Need of a Coherent and Consistent Phenomena Identification and Ranking Table (PIRT) To Address Small, Intermediate and Large Break LOCA in PWRs", Transaction American Nuclear Society, Winter meeting and Nuclear technology Expo, November 12-16, 2006, Albuquerque, NM.

[43] Frepoli, C., Ohkawa, K. and Nissley, M., "Development of WCOBRA/TRAC-TF2 Computer Code: Coupling of the 3D Module (COBRA-TF) with the 1D Module of TRAC-PF1/MOD2", Proceedings of the 17th International Conference on Nuclear Engineering ICONE17 July 12-16, 2009, Brussels, Belgium, USA

[44] Frepoli, C., Ohkawa, K. "The Development of a Realistic LOCA Evaluation Model Applicable to the Full Range of Breaks Sizes: Westinghouse Full Spectrum LOCA (FSLOCA™) Methodology"., The 14th International Topical Meeting on Nuclear Reactor Thermalhydraulics, NURETH-14 Toronto, Ontario, Canada, September 25-30, 2011.

[45] Liao, J., Frepoli, C., Ohkawa, K., "Condensation in the Cold Leg As Results of ECC Water Injection During a LOCA: Modeling and Validation". The 14th International Topical Meeting on Nuclear Reactor Thermalhydraulics, NURETH-14 Toronto, Ontario, Canada, September 25-30, 2011.

[46] Petkov, N., Frepoli, C., Ohkawa, K., "WCOBRA/TRAC-TF2 Simulation of ROSAIV/LSTF Natural Circulation Test ST-NC-02". The 14th International Topical Meeting on Nuclear Reactor Thermalhydraulics, NURETH-14 Toronto, Ontario, Canada, September 25-30, 2011.

[47] Shockling, M., Frepoli, C., Ohkawa, K., "Best-Estimate LOCA Analysis over the Full Spectrum of Postulated Breaks Using Westinghouse FULL SPECTRUM™ LOCA Methodology". The 14th International Topical Meeting on Nuclear Reactor Thermalhydraulics, NURETH-14 Toronto, Ontario, Canada, September 25-30, 2011.

[48] Liao, J., Frepoli, C., Ohkawa, K., "Horizontal Stratified Flow Model for the 1-D Module of WCOBRA/TRAC-TF2: Modeling and Validation". The 14th International Topical Meeting on Nuclear Reactor Thermalhydraulics, NURETH-14 Toronto, Ontario, Canada, September 25-30, 2011.

[49] Frepoli, C., Ohkawa, K., et al., "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM™ LOCA Methodology)," WCAP-16996-NP, Revision 0, November 2010.

However some of the implementation details have been re-worked to properly address some concerns relative to the sample size and the need of providing an adequate coverage and consideration of all break sizes which are considered by the Full Spectrum LOCA EM.

FSLOCA^{TM} development followed the Evaluation Model Development and Assessment Process (EMDAP) which is outlined in the Regulatory Guide (RG) 1.203 and the Standard Review Plan (SRP) discussed in the NUREG-0800 (2005).

4. Summary and Conclusions

Since the amendment of the 10 CFR 50.46 rule in 1988, Westinghouse engaged in three decades of continuous development and application of realistic or best-estimate methods to perform Loss of Coolant (LOCA) safety analyses for PWRs in the industry. Then main focus has been historically on a large break rupture postulated to occur in the cold leg of the reactor coolant system (RCS). Such scenario, called Large Break LOCA is still considered the design basis accident (DBA) by the current rule. However recent interest in Intermediate and Small Break LOCA scenario was the driver behind further developments over the last five years which resulted in the new evaluation model, called Westinghouse Full Spectrum LOCA (FSLOCA^{TM}) Evaluation Model recently completed and submitted to the US NRC for their review and approval in November 2010.

The framework of these methods has been the CSAU, more recently reformulated in the Evaluation Model Development and Assessment Process (EMDAP) by the USNRC.

This paper presented an historical review of the advances. Many lesson learned during the extensive use of these methods for licensing application within the industry have been captured within the methodology.

However many challenges still lie ahead and there is significant room for further improvement. We believe these methods still retain significant conservatism in their model and how they are applied for nuclear power plant safety analysis. Also applications of approved methods are valid only within the constraints of the class of plants and scenarios for which they are certified. Therefore, as new designs and new scenarios are identified, this requires further development and assessment.

The area of uncertainty, particularly the use of non-parametric order statics has been a success story within the industry; however both applicants and regulators are aware of its shortcomings briefly discussed in this paper.

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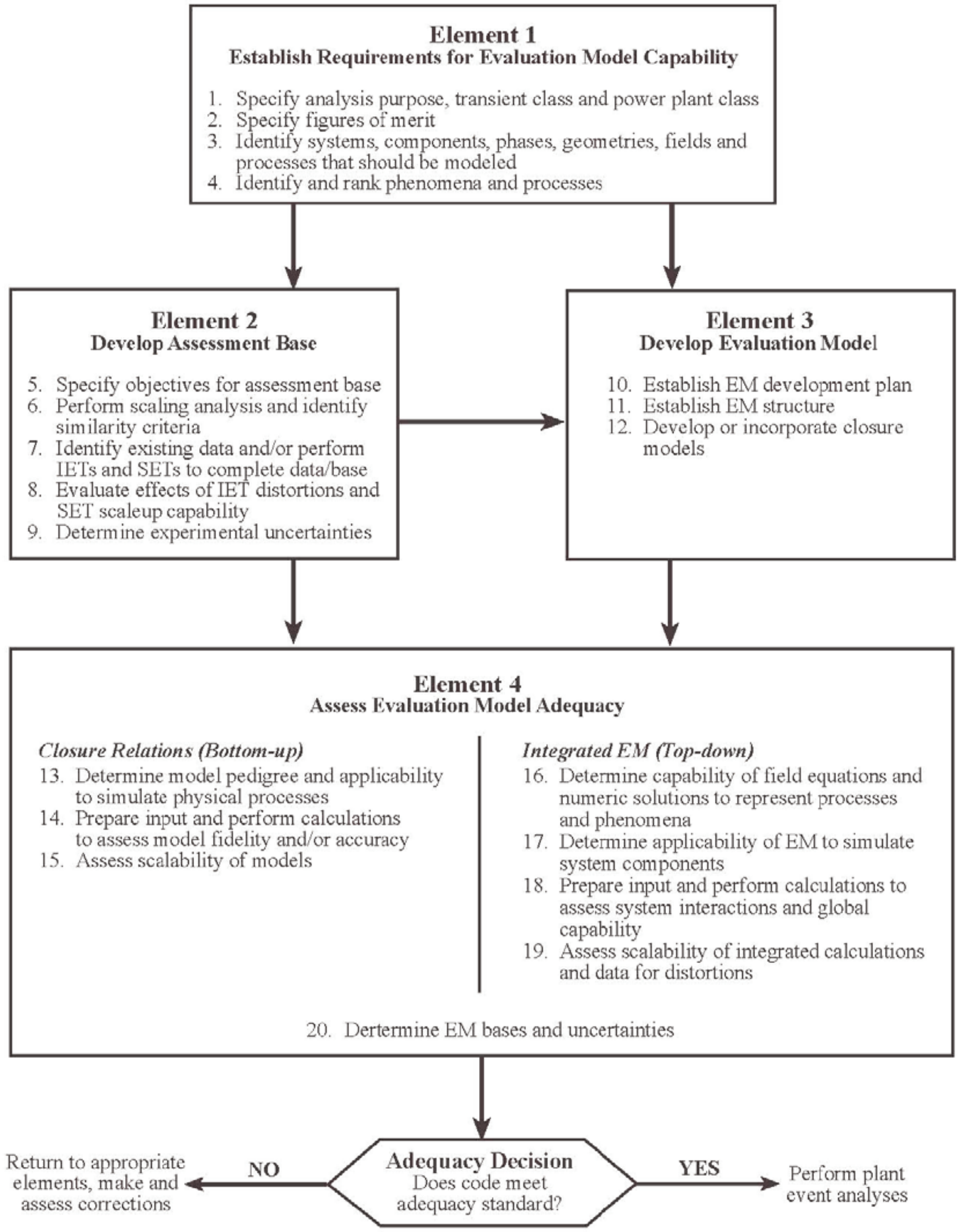

Figure 1 – EMDAP (Reg. Guide 1.203)

RELAP5/MOD3.2 Sensitivity Analysis Using OECD/NEA ROSA-2 Project 17% Cold Leg Intermediate-break LOCA Test Data

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Abstract

An experiment simulating a PWR intermediate-break loss-of-coolant accident (IBLOCA) with 17% break at cold leg was conducted in OECD/NEA ROSA-2 Project using the Large Scale Test Facility (LSTF). In the experiment, core dryout took place due to rapid drop in the core liquid level before loop seal clearing (LSC). Liquid was accumulated in upper plenum, steam generator (SG) U-tube upflow-side and SG inlet plenum before the LSC due to counter-current flow limiting (CCFL) by high velocity vapor flow, causing further decrease in the core liquid level. The post-test analysis by RELAP5/MOD3.2.1.2 code revealed that cladding surface temperature of simulated fuel rods was underpredicted due to later major core uncovery than in the experiment. Key phenomena and related important parameters, which may affect the core liquid level behavior and thus the cladding surface temperature, were selected based on the LSTF test data analysis and post-test analysis results. The post-test analysis conditions were considered as ‘Base Case’, for sensitivity analysis to study the causes of uncertainty in best estimate methodology. The RELAP5 sensitivity analysis was performed by changing the important parameters relevant to the key phenomena within the ranges to investigate influences of the parameters onto the cladding surface temperature. It was confirmed that both constant C of Wallis CCFL correlation at the core exit and gas-liquid inter-phase drag in the core, as parameters that need to consider for the evaluation of safety margin, are more sensitive to the cladding surface temperature than other chosen parameters.

1. Introduction

The occurrence frequency of the pipe break has been found to depend on the pipe size such that complete rupture of a smaller pipe or non-piping component is more likely to occur than double-ended guillotine break (DEGB) of a larger pipe, through the investigation on pipe integrity during long-term operation and for life extension of light water reactors (LWRs). Frequency of DEGB of main piping in the primary system of pressurized water reactors (PWRs), leading to large-break loss-of-coolant accident (LOCA), would then be quite low. The consideration of complete rupture of the largest branch pipe connected to the main piping should become relatively more important than ever in risk informed regulation-relevant safety analyses (Tregoning et al., 2008).

The United States Nuclear Regulatory Commission (2005) proposed risk informed changes to LOCA technical requirements, Appendix K to 10 CFR Part 50, that intermediate-break LOCA (IBLOCA) is chosen as a design basis event (DBE) for the assessment of emergency core cooling system (ECCS) effectiveness. Many test facilities have produced experimental data on small-break LOCA (SBLOCA) for code assessment and development. The experimental data, however, have been scarcely obtained for IBLOCA, although the thermal-hydraulic responses should differ from those for SBLOCA and LBLOCA. For the evaluation of safety margin in the case of IBLOCA in a realistic manner, detailed thermal-hydraulic data prepared by systematic tests simulating IBLOCA are desirable to understand the phenomena and to validate analysis methods including best-estimate computer code (e.g., RELAP5 code).

An experiment was conducted for OECD/NEA ROSA-2 Project with the Large Scale Test Facility (LSTF) (The ROSA-V group, 2003), simulating a PWR 17% cold leg IBLOCA (Takeda et al., 2011). The LSTF simulates a Westinghouse-type four-loop 3423 MW (thermal) PWR by a full-height and 1/48 volumetrically-scaled two-loop system. An upward long nozzle was mounted flush with the inner surface of cold leg to simulate DEGB of ECCS nozzle. Single-failure of diesel generators related to flow rates of both high pressure injection (HPI) and low pressure injection (LPI) systems as well as total failure of auxiliary feedwater were assumed.

Post-test analysis for the LSTF experiment was performed by RELAP5/MOD3.2.1.2 code (Carlson et al., 1990) to validate the predictability of the code to be used as a platform for best estimate plus uncertainty (BEPU) study (Wilson and Boyack, 1998; Yamaguchi et al., 2009). Key phenomena and related important parameters, which may affect the core liquid level behavior and thus the cladding surface temperature, were selected based on the LSTF test data analysis and post-test analysis results. The post-test analysis conditions were considered as ‘Base Case’, for sensitivity analysis to study the causes of uncertainty in best estimate methodology. The RELAP5 sensitivity analysis was conducted by varying the important parameters within the ranges to investigate influences of the parameters onto the cladding surface temperature. This paper describes major observations in the LSTF experiment during the cold leg IBLOCA with the RELAP5 post-test analysis and sensitivity analysis results.

2. OECD/NEA ROSA-2 Project

The JAEA started OECD/NEA ROSA-2 Project in 2009 following the ROSA Project (Nakamura et al., 2008) to resolve issues in thermal-hydraulic analyses relevant to LWR safety by using the LSTF of JAEA. Eighteen organizations from 14 NEA member countries have joined the ROSA-2 Project to date. In particular, the ROSA-2 Project intends to focus on the validation of simulation models and methods for complex phenomena such as multi-dimensional mixing and stratification of highly safety relevance for thermal-hydraulic transients in DBEs and beyond-DBE transients.

The experimental program is defined to provide valuable and broadly usable database to achieve the objectives agreed among participants. The ROSA-2 Project consists of six LSTF experiments that mainly include two groups of experiments which are IBLOCAs and steam generator tube rupture (SGTR) accidents. IBLOCA with a break at either hot leg or cold leg of PWR is dealt with as one of important safety issues. A SGTR accident induced by main steam line break is simulated further as well as a SGTR to evaluate amount of fission product (FP) release and to consider better accident management (AM) actions to minimize the FP release. New collaboration with other OECD/NEA Project: PKL-2 is also conducted focusing on the effectiveness of core exit temperature as an important indicator to start the AM actions.

3. ROSA/LSTF

The LSTF simulates a Westinghouse-type four-loop 3423 MWt PWR by a two-loop system model with full-height and 1/48 in volume. The reference PWR is Tsuruga Unit-2 of Japan Atomic Power Company (JAPC). **Figure 1** shows the schematic view of the LSTF that is composed of pressure vessel, pressurizer (PZR) and primary loops. An active steam generator (SG) with 141 full-size U-tubes (inner diameter of 19.6 mm each), primary coolant pump, hot and cold legs are included in each loop. The hot and cold legs, 207 mm in inner diameter, are sized to conserve the volumetric scaling (2/48) and the ratio of the length to the square root of the diameter to simulate the flow regime transitions in the horizontal legs (Zuber, 1980).

The LSTF core, 3.66 m in active height, consists of 1008 electrically heated rods in 24 rod bundles to simulate the fuel rod assembly in the reference PWR. Axial core power profile is a 9-step chopped cosine with a peaking factor of 1.495. The radial power profile is achieved by providing three different power rod bundles (high, middle and low) with a maximum peaking factor of 1.51 for high-power rod bundle. The LSTF initial core power of 10 MW corresponds to 14% of the volumetric-scaled (1/48) PWR nominal core power because of a limitation in the capacity of power supply. The core power after the test initiation is then kept constant at 10 MW for a little while before the core power starts to follow pre-determined decay power curve.

4. LSTF Experiment and Code Analysis Results

4.1. LSTF Test Conditions

The 17% break was simulated by using a nozzle, upwardly mounted flush with the cold leg inner surface, in the broken loop without PZR, as shown in **Fig. 2**. The break simulation nozzle is 41 mm in inner-diameter (d) and 492 mm in straight portion length (L). The ratio of L to d is designed to be 12 to avoid significant influences of the length onto mass flux through the nozzle (Henry, 1968). The nozzle flow area corresponds to 17% of the volumetrically-scaled cross-sectional area of the reference PWR cold leg.

The experiment was initiated by opening a break valve located downstream of the break nozzle at time zero. The PZR pressure, fluid temperatures in hot and cold legs during initial steady-state conditions were 15.5 MPa, 598 K and 562 K, respectively, according to the reference PWR conditions. The LSTF core power was maintained at the initial value of 10 MW for 18 s after a scram signal until the scaled PWR core decay power dropped to 10 MW. The LSTF core power started to decay afterwards according to the specified core power. To obtain prototypical initial fluid temperatures with this core power, core flow rate was set to 14% of the scaled nominal flow rate. Initial secondary pressure was raised to 7.3 MPa from nominal value of 6.1 MPa to limit the primary-to-secondary heat transfer rate to 10 MW. Set point pressures for opening and closure of SG relief valves (RVs) are 8.03 and 7.82 MPa respectively, referring to the corresponding values in the reference PWR. The experiment assumed loss of off-site power concurrent with scram and total failure of auxiliary feedwater.

As for ECCS conditions, high pressure injection (HPI), accumulator (ACC) and low pressure injection (LPI) systems were actuated in the intact loop only to simulate the ECCS line break. Further assumption was single-failure of diesel generators related to both the HPI and LPI flow rates.

4.2. Major Phenomena Observed in the Experiment

The LSTF test results are shown in **Figs. 3 through 11** as a comparison with RELAP5 post-test analysis results. Break flow turned from single-phase liquid to two-phase flow in a very short time after the break. Rather large size of break caused a fast primary depressurization. The SG secondary-side pressure increased rapidly up to about 8 MPa after the closure of the main steam isolation valves following the scram signal, causing that the SG RVs were opened for about 30 s. The primary pressure soon became far lower than SG secondary-side pressure. The SG U-tube liquid level started to decrease greatly due to the fast primary depressurization and steam flow from the inlet plenum soon after the break. The liquid level decrease in the SG U-tube downflow-side continued down to the crossover leg to cause loop seal clearing (LSC). Core dryout took place at about 35 s due to rapid drop in the core liquid level before the LSC. Liquid was accumulated in upper plenum, SG U-tube upflow-side and SG inlet plenum before the LSC due to counter-current flow limiting (CCFL) by high velocity vapor flow, causing further decrease in the core liquid level. Water remained on the upper core plate in the upper plenum due to CCFL because of continuous upward steam flow from the core. The HPI system was started almost simultaneously with the core dryout, but was ineffective on the core cooling because the injection flow rate was far smaller than break flow rate. The core bottom-up quench started after the incipience of ACC coolant injection at 4.51 MPa. However, a large temperature excursion appeared in the core, which induced actuation of automatic core power decrease to protect the LSTF core after the maximum cladding surface temperature exceeded pre-determined criterion of 958 K. The peak cladding surface temperature (PCT) of 978 K was observed at Position 6 (=about 0.4 m above the core center). The core power decreased down to 5% of the pre-determined power level as the PCT reached 978 K. Whole core was quenched due to reflooding.

4.3. RELAP5 Code Analysis

4.3.1. RELAP5 Analysis Conditions

Figure 12 shows a noding schematic of LSTF for RELAP5 analysis. The LSTF system is modeled in one-dimensional manner including pressure vessel, primary loops, PZR, SGs and SG secondary-side system.

The break unit connected to the top of the broken cold leg (**Fig. 2**) was modeled by a vertical pipe, horizontal pipe and oblique pipe between them to simulate corresponding facility configuration. The break nozzle in the vertical pipe was represented by five equal-volume nodes. The analysis employed RELAP5 critical flow model, which may correctly predict the discharge rate through the long break nozzle, with a discharge coefficient (Cd) of 1.0.

The core was represented by vertically-stacked nine equal-height volumes according to 9-step chopped cosine power profile along the length of the core. The radial power distribution was then given considering the peaking factor and the number of high-, mean- and low-power rod bundles.

Following Wallis correlation (Wallis, 1969) was applied to simulate CCFL at core exit, U-tubes and inlet plena of SGs;

$$j_G^{*1/2} + m j_L^{*1/2} = C \qquad (1)$$

where j^* is the non-dimensional volumetric flux. Subscripts G and L denote gas and liquid phases, respectively. Constants of m and C for core exit, U-tubes and inlet plena of SGs were set to 1 and 0.75, respectively, obtained from LSTF experiment on CCFL at the SG U-tube inlet (Yonomoto et al., 1991). These values were selected as the constants in the post-test analysis considering a tendency that water remains on the upper core plate in the upper plenum, though flow channel structure of the core exit is different from that of the SG U-tube inlet.

Kumamaru et al. (1999) have pointed out that the RELAP5 code should overpredict gas-liquid inter-phase drag in the core. Larger inter-phase drag in the core results in overprediction of the core liquid level. In the post-test analysis, inter-phase drag in each node of core was thus reduced to 1/10. No core power control depending on the PCT was made in the post-test analysis. Other initial and boundary conditions were determined according to the LSTF test data.

4.3.2. Comparison of Post-test Analysis Results with Experimental Observations

The RELAP5 code roughly predicted the overall trend of the LSTF test as has been compared in **Figs. 3 through 11**. Significant drop in the cold leg liquid level started slightly later than in the experiment due to a bit later LSC. The primary pressure was overpredicted probably due to smaller steam discharge through the break because of overprediction of cold leg liquid level resulting in later ACC coolant injection, though break flow rate and HPI flow rate were predicted reasonably well.

Major core uncovery following significant drop in the core liquid level before LSC started slightly later than in the experiment. The core liquid level was underpredicted after around 50 s due to overprediction of the upper plenum liquid level probably because of coolant remain in the crossover leg upflow-side under effects of CCFL at the core exit. The cladding surface temperature was underpredicted due to later major core uncovery than in the experiment. Then the PCT was about 963 K at Position 5 (=core center). Liquid accumulation in the SG U-tube upflow-side and SG inlet plenum before LSC was well calculated, though with tendencies that the U-tube upflow-side was slightly overpredicted and the inlet plenum was slightly underpredicted.

5. Sensitivity Analysis of LSTF Experiment

5.1. Selection of Key Phenomena and Related Important Parameters

Key phenomena which may affect the core liquid level behavior and thus the cladding surface temperature were selected based on the LSTF test data analysis and RELAP5 post-test analysis results, referring to high-ranked PWR LOCA phenomena (Boyack and Ward, 2000). The followings were the key phenomena: critical flow at the break, decay heat and stored heat of the fuel rod, mixture level in the core, liquid accumulation in the upper plenum, SG U-tube upflow-side and SG inlet plenum due to CCFL, and steam condensation on HPI or ACC coolant in the intact cold leg. The key phenomena include liquid accumulation in the upper plenum, SG U-tube upflow-side and SG inlet plenum due to CCFL as phenomena peculiar to the cold leg IBLOCA.

The key phenomena were ranked as shown in **Table 1**. Critical flow at the break, decay heat of the fuel rod, mixture level in the core, liquid accumulation in the upper plenum and SG U-tube upflow-side were identified as high-ranked phenomena which may have great influences onto the core liquid level behavior and thus the cladding surface temperature. Stored heat of the fuel rod and steam condensation in the intact cold leg as well as liquid accumulation in the SG inlet plena were identified as medium-ranked phenomena which may have somewhat or little influences onto the core liquid level behavior and thus the cladding surface temperature.

Sensitivity analysis should be necessary for both the high- and medium-ranked phenomena. Important parameters relevant to the key phenomena were selected as shown in **Table 1**. The important parameters were classified into three groups: code's physical model, boundary condition and material property. The post-test analysis conditions with Cd of 1.0 and reduction of inter-phase drag to 1/10 in the core as well as application of the Wallis CCFL correlation to the core exit, U-tubes and inlet plena of SGs were considered as 'Base Case', for the sensitivity analysis. Coefficients of m and C for the CCFL correlation are then given as 1 and 0.75, respectively.

5.2. Selection of Ranges of Important Parameters

The ranges of the important parameters were selected referring to the values of 'Base Case', as shown in **Table 2**. As for the parameter conditions concerning the high-ranked phenomena, the range of the Cd was set to 1.0±0.05 to clarify influences of small difference in the Cd. The band of the core decay power was ±0.07 MW according to the measurement uncertainty. The range of the inter-phase drag in the core was 1/10 of value by RELAP5 original model (as 'Base Case') and 1 (as original value) to clarify influences of the inter-phase drag. The common range of the constant C of the Wallis CCFL correlation at the core exit and SG U-tubes was set to 0.75±0.1 to clarify influences of small difference in the coefficient C of the CCFL correlation. As for the parameter conditions concerning the medium-ranked phenomena, on the other hand, the bands of the thermal conductivity and heat capacity of the fuel rod were ±10% and ±1% respectively referring to databases by Boyack et al. (1989) and by Takeuchi and Nissley (2000). The range of the condensation heat transfer with Shah's model was set to 1±0.5 to clarify influences of large difference in the condensation heat transfer. The range of the constant C of the Wallis CCFL correlation at the SG inlet plena was set to 0.75±0.1, similar to those at the core exit and SG U-tubes. The bands of the ECCS injection conditions were decided based on the measurement uncertainty. The ranges of the injection pressure and temperature were thus 12.27±0.11 MPa and 310±2.4 K respectively for the HPI system, while those were 4.51±0.05 MPa and 320±2.3 K for the ACC system.

5.3. Sensitivity Analysis Results

The RELAP5 sensitivity analysis was conducted by varying the important parameters relevant to the key phenomena within the ranges. The sensitivity analysis conditions are the same as the post-test analysis conditions as 'Base Case', except for the parameter to be surveyed. Influences of the parameters onto the cladding surface temperature were investigated through **Figs. 13 through 18**. In the 'Base Case', the PCT was about 963 K at Position 5 (=core center).

In the case with Cd of 1.05, the PCT of about 990 K at Position 5 was higher and appeared earlier than in the 'Base Case' due to a bit larger break flow rate. In the case with specified core decay power plus 0.07 MW, the PCT of about 980 K at Position 5 was higher than in the 'Base Case'. In the case with specified thermal conductivity plus 10% of the fuel rod, the PCT of about 973 K at Position 5 was higher than in the 'Base Case'. In the case with original inter-phase drag different from 'Base Case', the cladding surface temperature at Position 5, where the PCT appeared, started to increase at about 45 s but the whole core was once quenched due to temporary recovery of the core liquid level and thus temporal overprediction of the core liquid level. The cladding surface temperature began to increase again after about 90 s due to the core liquid level decrease. The PCT of about 820 K was thus much lower than in the 'Base Case'. In the case with constant C 0.65 of the Wallis CCFL correlation at the core exit, the PCT of about 998 K at Position 5 was higher than in the 'Base Case' due to more significant CCFL effects. In the case with constant C 0.85 of the Wallis CCFL correlation at the SG U-tubes, the PCT of about 990 K at Position 5 was higher than in the 'Base Case' while it appeared after the SG U-tubes became empty of liquid.

On the other hand, the following parameters had little influences onto the cladding surface temperature: heat capacity of the fuel rod, condensation heat transfer with Shah's model and constant C of the Wallis CCFL correlation at the SG inlet plena as well as ECCS injection conditions (not shown). The sensitivity analysis revealed that the phenomena concerning these parameters should be reconsidered as the low-ranked phenomena.

Anyhow, it was confirmed that both the CCFL at the core exit and the reduction of gas-liquid inter-phase drag in the core affect the cladding surface temperature significantly. Both the constant C of Wallis CCFL correlation at the core exit and the inter-phase drag in the core, as parameters that need to consider for the evaluation of safety margin, were more sensitive to the cladding surface temperature than other chosen parameters. In the future work, further sensitivity analysis will be performed to investigate influences of combination especially for the parameters related to the high-ranked phenomena onto the cladding surface temperature.

6. Summary

An LSTF experiment was performed for OECD/NEA ROSA-2 Project, simulating a PWR 17% cold leg IBLOCA. The results of the LSTF experiment were compared with the post-test analysis results by the RELAP5/MOD3.2.1.2 code. Key phenomena and related important parameters, which may affect the core liquid level behavior and thus the cladding surface temperature, were selected based on the LSTF test data analysis and post-test analysis results. The RELAP5 sensitivity analysis was performed by changing the important parameters relevant to the key phenomena within the ranges to investigate influences of the parameters onto the cladding surface temperature. Obtained results are summarized as follows;

(1) Core dryout took place due to rapid drop in the core liquid level before loop seal clearing (LSC). Liquid was accumulated in upper plenum, steam generator (SG) U-tube upflow-side and SG inlet plenum before the LSC due to counter-current flow limiting (CCFL) by high velocity vapor flow, and enhanced the core liquid level decrease. The core bottom-up quench started after the incipience of accumulator coolant injection. However, a large temperature excursion appeared in the core, which induced actuation of automatic core power decrease to protect the LSTF core. The peak cladding surface temperature (PCT) of 978 K was observed at Position 6.

(2) The post-test analysis showed that the cladding surface temperature was underpredicted due to later major core uncovery than in the experiment. Then the PCT was about 963 K at Position 5. Critical flow at the break, decay heat of the fuel rod, mixture level in the core, liquid accumulation in the upper plenum and SG U-tube upflow-side were identified as high-ranked phenomena which may have great influences onto the core liquid level behavior and thus the cladding surface temperature. The post-test analysis conditions were considered as 'Base Case', for the sensitivity analysis. As a result, both constant C of Wallis CCFL correlation at the core exit and gas-liquid inter-phase drag in the core, as parameters that need to consider for the evaluation of safety margin, were more sensitive to the cladding surface temperature than other chosen parameters.

Acknowledgements

This paper contains findings gained through the OECD/NEA ROSA-2 Project. It is noted that the value axis of all the data plot of LSTF test and RELAP5 analysis is normalized properly due to the agreement of the OECD/NEA ROSA-2 Project. Authors are also grateful to the Management Board of the ROSA-2 Project for their consent to this publication.

Authors would like to thank to Messrs. M. Ogawa and A. Ohwada of Japan Atomic Energy Agency for performing the LSTF experiments under collaboration with members from Nuclear Engineering Co. (NECO) as well as Miss K. Toyoda of IT-TEC Inc. for manipulating the experimental data.

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Table 1 Selection of key phenomena and related important parameters

Component	Key Phenomenon	Rank*	Important Parameter	Classification
Break	Critical flow	H	Break discharge coefficient (Cd)	Physical model
Fuel rod	Decay heat	H	Core decay power	Boundary condition
	Stored heat	M	Thermal conductivity of fuel rod	Material property
			Heat capacity of fuel rod	
Core	Mixture level	H	Gas-liquid inter-phase drag in core	Physical model
Upper plenum	Liquid accumulation due to CCFL**	H	Constant C of Wallis CCFL correlation at core exit	
SG U-tubes		H	Constant C of Wallis CCFL correlation at SG U-tubes	
SG inlet plena		M	Constant C of Wallis CCFL correlation at SG inlet plena	
Cold leg / HPI or ACC system	Steam condensation on HPI or ACC coolant	M	Condensation heat transfer with Shah's model	
			Injection pressure of HPI system	Boundary condition
			Injection temperature of HPI system	
			Injection pressure of ACC system	
			Injection temperature of ACC system	

*H; High-ranked phenomenon, M; Medium-ranked phenomenon **Phenomenon peculiar to cold leg IBLOCA

Table 2 Selection of ranges of important parameters

Important Parameter	Range
Break discharge coefficient (Cd)	1.0±0.05
Core decay power	Specified±0.07 MW*
Thermal conductivity of fuel rod	Specified±10%
Heat capacity of fuel rod	Specified±1%
Gas-liquid inter-phase drag in core	1/10, 1
Constant C of Wallis CCFL correlation at core exit	0.75±0.1
Constant C of Wallis CCFL correlation at SG U-tubes	0.75±0.1
Constant C of Wallis CCFL correlation at SG inlet plena	0.75±0.1
Condensation heat transfer with Shah's model	1±0.5
Injection pressure of HPI system	12.27±0.11 MPa*
Injection temperature of HPI system	310±2.4 K*
Injection pressure of ACC system	4.51±0.05 MPa*
Injection temperature of ACC system	320±2.3 K*

*Measurement uncertainty

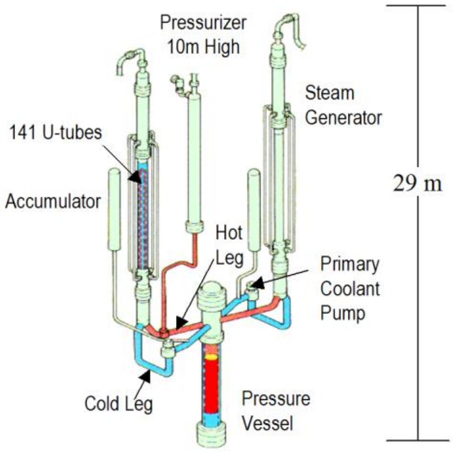

Fig. 1 Schematic view of ROSA/LSTF

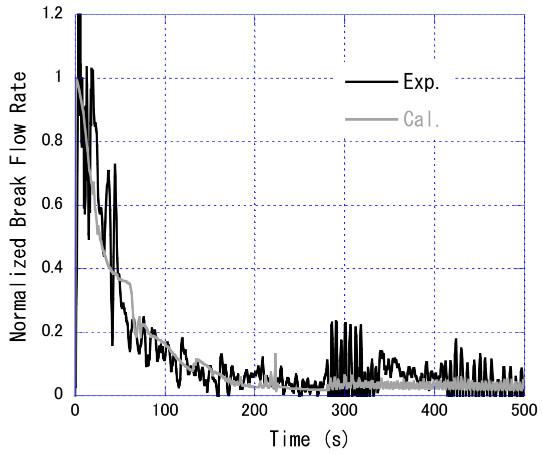

Fig. 4 LSTF and RELAP5 results for break flow rate

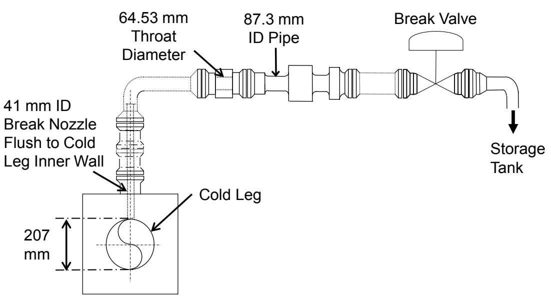

Fig. 2 Schematic view of LSTF break unit

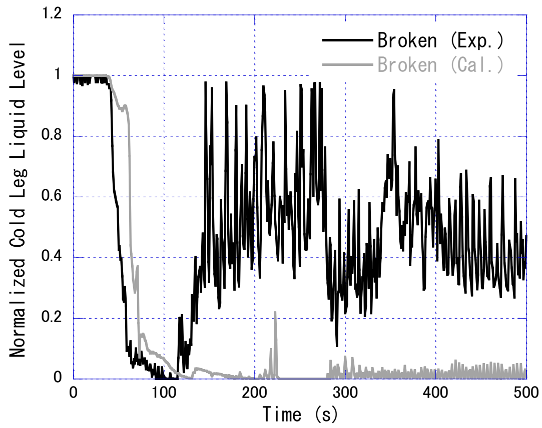

Fig. 5 LSTF and RELAP5 results for broken cold leg liquid level

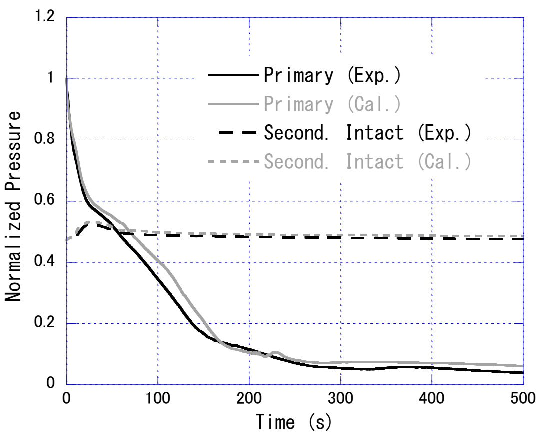

Fig. 3 LSTF and RELAP5 results for primary and secondary pressures

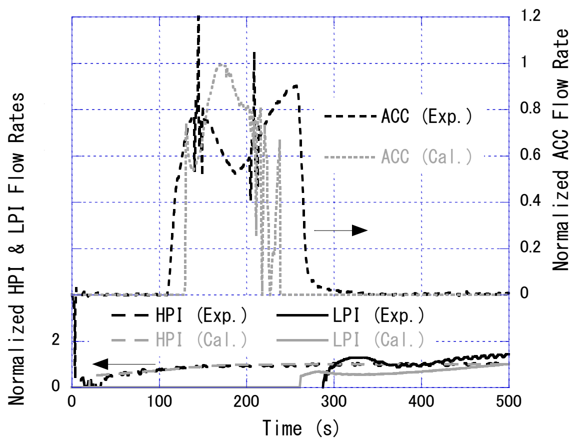

Fig. 6 LSTF and RELAP5 results for flow rates of intact ECCS

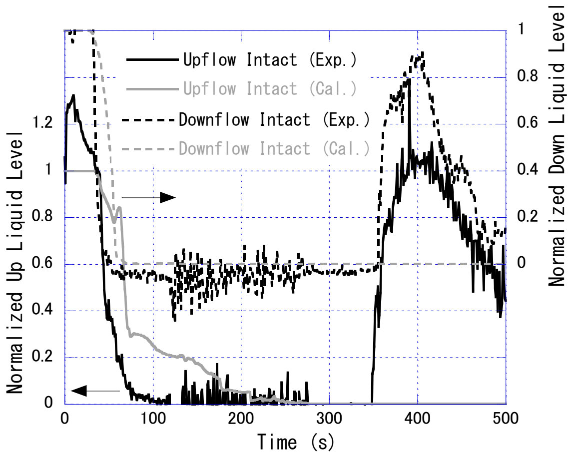

Fig. 7 LSTF and RELAP5 results for intact crossover leg collapsed liquid level

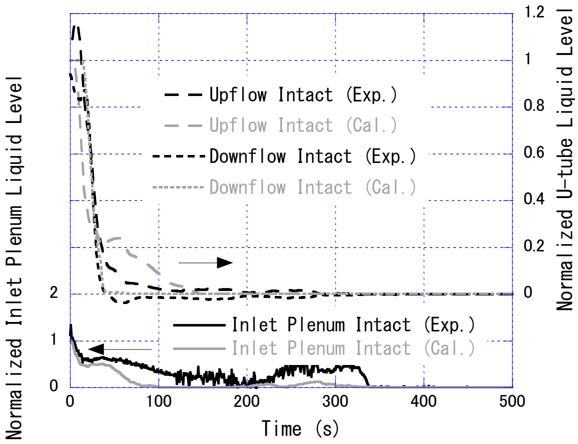

Fig. 8 LSTF and RELAP5 results for collapsed liquid levels in inlet plenum and U-tube of intact SG

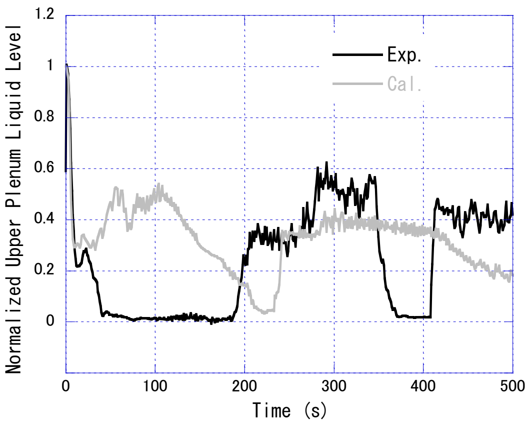

Fig. 9 LSTF and RELAP5 results for upper plenum collapsed liquid level

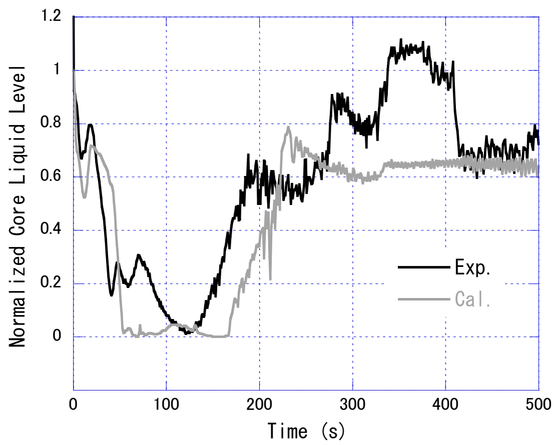

Fig. 10 LSTF and RELAP5 results for core collapsed liquid level

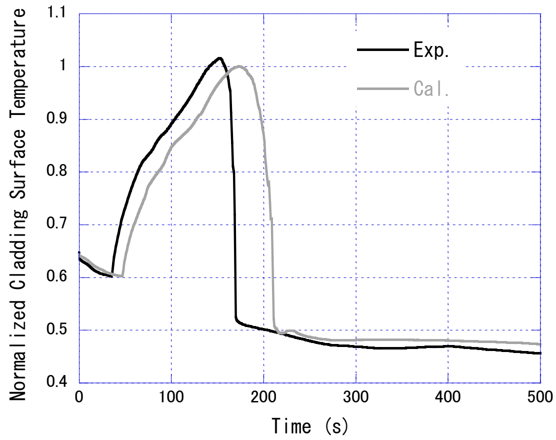

Fig. 11 LSTF and RELAP5 results for cladding surface temperature

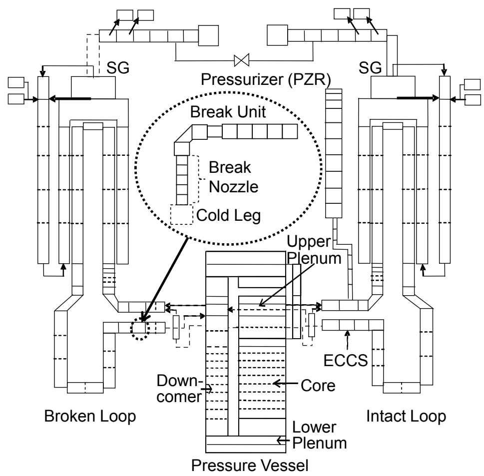

Fig. 12 Noding schematic of LSTF for RELAP5 analysis

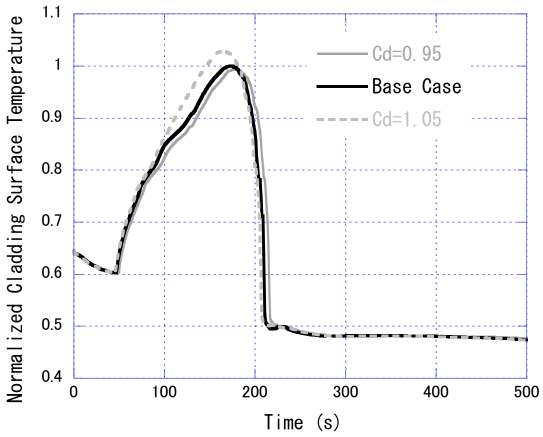

Fig. 13 Influences of break discharge coefficient onto cladding surface temperature

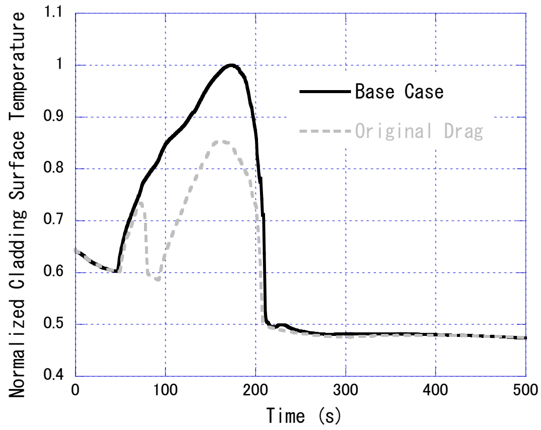

Fig. 16 Influences of inter-phase drag in core onto cladding surface temperature

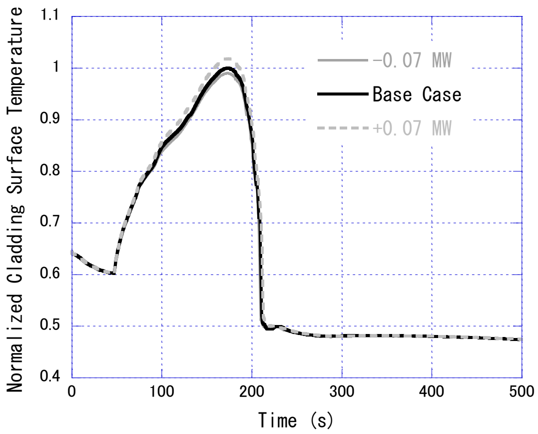

Fig. 14 Influences of core decay power onto cladding surface temperature

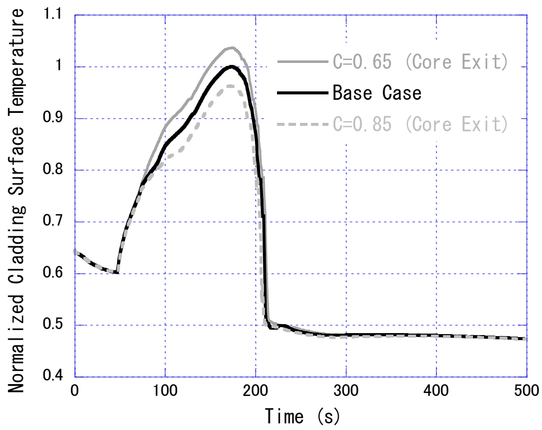

Fig. 17 Influences of Constant C of Wallis CCFL correlation at core exit onto cladding surface temperature

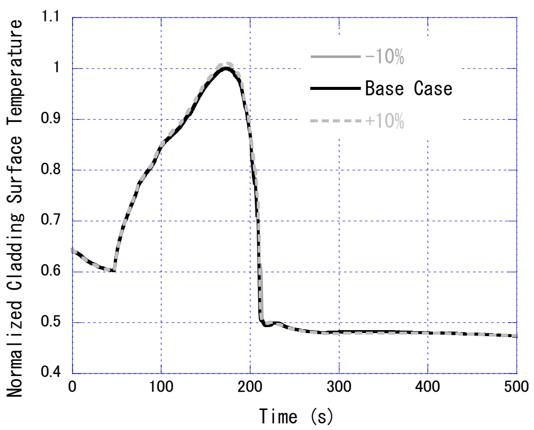

Fig. 15 Influences of thermal conductivity of fuel rod onto cladding surface temperature

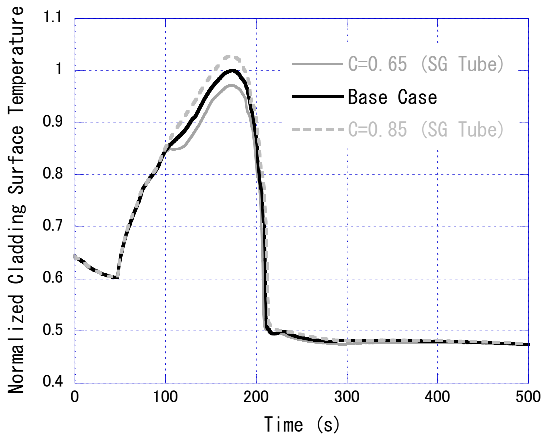

Fig. 18 Influences of constant C of Wallis CCFL correlation at SG U-tubes onto cladding surface temperature

Towards an Industrial Application of Statistical Uncertainty Analysis Methods to Multi-physical Modelling and Safety Analyses

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Abstract - Since 1980's, Tractebel Engineering (TE) has being developed and applied a multi-physical modelling and safety analyses capability, based on a code package consisting of the best estimate 3D neutronic (PANTHER), system thermal hydraulic (RELAP5), core sub-channel thermal hydraulic (COBRA-3C), and fuel thermal mechanic (FRAPCON/FRAPTRAN) codes. A series of methodologies have been developed to perform and to license the reactor safety analysis and core reload design, based on the deterministic bounding approach. Following the recent trends in research and development as well as in industrial applications, TE has been working since 2010 towards the application of the statistical sensitivity and uncertainty analysis methods to the multi-physical modelling and licensing safety analyses. In this paper, the TE multi-physical modelling and safety analyses capability is first described, followed by the proposed TE best estimate plus statistical uncertainty analysis method (BESUAM). The chosen statistical sensitivity and uncertainty analysis methods (non-parametric order statistic method or bootstrap) and tool (DAKOTA) are then presented, followed by some preliminary results of their applications to FRAPCON/FRAPTRAN simulation of OECD RIA fuel rod codes benchmark and RELAP5/MOD3.3 simulation of THTF tests.

1. Introduction

The use of best estimate codes and methods is necessary to meet the increasing technical, licensing and regulatory requirements for major plant changes (e.g. steam generator replacement), power uprate, core design optimization or cycle extension, as well as Periodic Safety Review.

Since 1980's, Tractebel Engineering (TE) has developed and applied a deterministic bounding approach to FASR accident analyses[1], using the best estimate system thermal hydraulic code RELAP5/MOD2.5 and the sub-channel thermal hydraulic code COBRA-3C. In this approach, the corrective methods are used to overcome the relevant known code deficiencies and to bound uncertainties, and the worst initial and boundary conditions, as well as the licensing conservatisms are deterministically combined. Later on in the 2000's, it has been extended to accident analyses using coupled neutronic and thermal hydraulic codes package RELAP/PANTHER/COBRA[2].

[1] J. Zhang, "The Tractebel Deterministic Bounding Approach to Accident Analysis," *Proc. OECD Exploratory Meeting of Experts on Best Estimate Calculations and Uncertainty Analysis*, 13-14 May, Aix-en-Provence, France (2002).

[2] J. Zhang and C. R. Schneidesch, "Development, qualification and application of the coupled RELAP5/PANTHER/COBRA code package for integrated PWR accident analysis", *IAEA Technical Meeting on Progress in Development and use of coupled Codes for Accident Analysis*, Vienna (Austria), 26-28 November, (2003).

This deterministic bounding approach has been applied to most of the existing non-LOCA accident analyses, and has been accepted by the Belgian Safety Authorities. While proving to be a cost-effective approach for licensing, this approach involves a number of un-quantified conservatisms, and hence a loss of licensing margins to the acceptance criteria[3].

Therefore, following the recent trends in research and development (R&D) as well as in various industrial applications, TE has been working since 2010 towards the application of statistical sensitivity and uncertainty analysis method to multi-physical modelling and licensing safety analyses. In this approach, the code bias and uncertainties for relevant models are first quantified via code verification and validation (V&V) or defined, and a simple and robust uncertainty analysis method based on the non-parametric order statistic tolerance limit or the bootstrap method is used to statistically treat uncertainties related to plant/core/fuel Initial and Boundary Conditions (IC/BC), models and modelling options.

The final objective is to apply the non-parametric order statistic method to develop a TE best estimate plus statistical uncertainty analysis methodology (BESUAM) for reloads safety evaluation and safety analysis, which will allow TE to provide more economic and flexible core design and more effective margin management for our customers. It will also allow TE to keep in pace with the changing regulatory requirements and the industry development trends.

In this paper, the TE multi-physical modelling and safety analyses capability is first described, followed by the proposed TE best estimate plus statistical uncertainty analysis method (BESUAM). The chosen statistical sensitivity and uncertainty analysis method (non-parametric order statistic tolerance limit or bootstrap) and tool (DAKOTA) are then presented, followed by some preliminary uncertainty analysis applications and results (FRAPCON/FRAPTRAN simulation of OECD RIA fuel rod codes benchmark and RELAP5/MOD3.3 simulation of THTF tests). The perspectives for future development are also given.

2. TE multi-physical modelling and safety analysis capability

2.1. TE Multi-physical code package

The TE multi-physical code package mainly consists in the following codes:

- The 3 dimensional (3D) nodal neutronic code PANTHER, which has been used for core reload design and neutronic safety analysis under the license agreement with British Energy since the 1990's;
- The best estimate system code RELAP5/MOD2.5 (and recently MOD3.3), which has been used as the principle code for nuclear power plant system thermal-hydraulic analysis under the agreement with USNRC since the 1980's;
- The sub-channel thermal hydraulic analysis code COBRA-3C, which has been used for core reload design and safety analysis since the 1980's;
- The fuel rod thermal mechanic codes FRAPCON/FRAPTRAN, which has been used for fuel design verification and safety analysis under the agreement with Battelle since 2001.

In order to obtain an integrated plant system thermal-hydraulics and neutron kinetics package, TE uses the external dynamic coupling of two existing codes, RELAP5 and PANTHER, by means of the TALINK interface developed by the former British Energy (BE). In this approach, both client codes perform in parallel their calculations as separate processes, while the TALINK interface controls the data transfers between the two processes (Figure 1).

Optionally, a dynamic coupling between PANTHER and COBRA-3C_TE has been developed, via the socket interface. An additional interface between PANTHER code and the sub-channel thermal-

[3] J. Zhang, "Perspectives on the Application of Statistical Uncertainty Analysis Methods," *Proceedings of the OECD/NSC Workshop on Uncertainty Analysis in Modelling (UAM-2006)*, Pisa, Italy, 28-29 April, (2006).

hydraulic analysis code COBRA-3C was developed in order to calculate online the Departure from Nucleate Boiling Ratio (DNBR) during coupled RELAP5-PANTHER transient calculations.

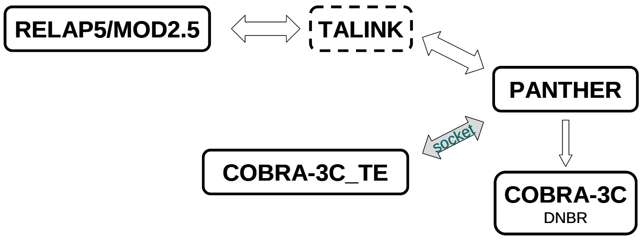

Figure 1. The TE coupled code package.

The TE coupled code package has been qualified for licensing applications[4]. It has been applied to develop accident analysis methodologies for main steam line break (MSLB)[5] and feedwater line break (FLB). Currently, the coupled PANTHER-COBRA code package is being used to develop a methodology for verification of Rod Ejection Accident (REA) analysis, using FRAPCON/FRAPTRAN for fuel rod behaviour analysis.

2.2. TE deterministic bounding approach to non-LOCA accident analysis

The TE deterministic bounding approach to non-LOCA accident analysis has been described in details in reference[1]. In this approach, corrective methods are used to overcome the relevant known code deficiencies and to bound code uncertainties, and the worst initial and boundary conditions, as well as the licensing conservatisms are linearly combined.

The development of such a methodology for accident analysis requires a large number of parametric sensitivity studies to identify the most limiting initial and boundary conditions, and needs a high level of expertise for setting-up the plant model and to assess the results. This bounding treatment of uncertainties is consistent with the Belgian regulatory requirements.

Moreover, since most of the non-LOCA accident analysis methodologies using RELAP5 code adapt the decoupled approach for the core conditions, the analysis results involved often too large un-quantified margins, due to:

- simplistic approximations for asymmetric accidents with strong 3D core neutronics - plant thermal hydraulics interactions;
- additional penalties introduced from incoherent initial/boundary conditions for separate plant and core analyses.

The TE deterministic bounding approach has been also extended to accident analysis methodology using the TE coupled code package[3; 5].

3. TE Best Estimate plus Statistical Uncertainty Analysis Method (BESUAM)

Following the recent trends in research and development as well as in industrial applications, TE plans to move towards the application of statistical sensitivity and uncertainty analysis method to the multi-physical modelling and licensing safety analyses.

As shown in Figure 2, the proposed TE best estimate plus statistical uncertainty analysis method (BESUAM) for multi-physics accident analysis will be based on six basic elements:

1. the current licensing basis (FSAR);

[4] C. R. Schneidesch and J. Zhang, "Qualification of the Coupled RELAP5/PANTHER/ COBRA Code Package for Licensing Applications," *Proc. of the Int. Meeting on Updates Best-Estimate Methods in Nuclear Installation Safety Analysis (BE-2004)*, Washington, D.C., USA, (2004).

[5] J. Zhang, et al, "Coupled RELAP5/PANTHER/COBRA Steam Line Break Accident Analysis in Support of Licensing Doel 2 Power Uprate and Steam Generator Replacement," *Proc. Int. Meeting on Updates Best-Estimate Methods in Nuclear Installation Safety Analysis (BE-2004)*, Washington, D.C., USA, (2004).

2. the applicable licensing rules (10CFR50, RG, SRP, …), as well as industry standard and codes (ANSI, ASME, ANS, …);
3. a Best Estimate frozen version of code, qualified through a wide scope of developmental verification and validation, independent code assessment, as well as TE participation in the international standard problems and benchmarks;
4. a realistic standard plant/core/fuel model with integral neutronics, hydrodynamics, heat structures, control logics and trips components, built according to generic and specific code user guidelines and experience from code assessment including specific real plant transient data.
5. a quantification of the coupled code package bias and uncertainties for relevant models;
6. a simple and robust uncertainty analysis method to statistically treat uncertainties related to plant/core/fuel Initial and Boundary Conditions (IC/BC), models and modelling options.

Figure 2. The TE Best Estimate plus Statistical Uncertainty Analysis Method (BESUAM)[6].

[6] V. Bellens, "Assessment of the Coupled PANTHER COBRA Code Package for Development of a Rod Ejection Accident Analysis Methodology," Master Thesis, Belgian Nuclear Higher Education Network, Belgium, August (2010).

The proposed BESUAM approach consists of nine steps:
1. a well-defined accident scenario, consistent with the initial design basis and current licensing basis analyses;
2. a well-defined acceptance criterion, also consistent with current licensing basis analyses;
3. an identification of key physical phenomena, for which the coupled code package and the standard pant and core model are shown to be applicable for their simulation, through Best Estimate calculations in a wide range of operational and transient conditions during the code verification and validation;
4. an identification of relevant input parameters including plant/core/fuel initial conditions, and boundary condition and model uncertainties;
5. a definition of uncertainty distribution for all relevant input parameters;
6. a random sampling of all relevant input parameters, leading to the definition of N cases according to the non-parametric tolerance limit theory;
7. preparation and run of N P/C specific transient calculations for each set of input parameters (each case);
8. a statistical uncertainty analysis of the N calculations;
9. a definition of the LIcensing Case (LIC) that envelopes the results of the N runs (including all uncertainties), which must satisfy the acceptance criteria.

It is expected that this statistical approach will remove the need for numerous parametric sensitivity calculations and a linear combination of all uncertainties, as the uncertainties on initial plant, core and fuel states are propagated simultaneously through the sampled N cases to obtain the final licensing case. This should greatly increase the efficiency of the accident analysis for complicated accidents such as REA.

However, this approach requires considerable efforts to identify, quantify and define the ranges and distributions of all relevant uncertainty attributors, and needs a robust and transparent uncertainty and sensitivity analysis method and tool.

4. Choice of the uncertainty analysis method and tool

4.1. The uncertainty analysis methods

4.1.1. The non-parametric order statistics method

Among all the available uncertainty analysis methods, the non-parametric order statistics method is so-far the most widely used in the nuclear safety analysis. In this method, the thermal-hydraulic, neutronic and thermal mechanic computer codes are treated as “black boxes”, and the input uncertainties are propagated to the simulation model output uncertainties via the code calculations with sampled input data from the known distributions (as shown in Figure 3).

The method consists in the following steps:
(1). All relevant uncertain parameters for the codes, plant modelling schemes, and plant operating conditions are identified;
(2). Any dependencies between uncertain parameters are quantified or defined, and the variation ranges and/or probabilistic distribution functions (PDFs) for each uncertain parameter are quantified or defined, based on engineering judgment and experience feedback from code applications to separate and integral effect tests and to full plants simulation.
(3). The uncertainty ranges of input parameters are randomly and simultaneously sampled N times by a simple Monte Carlo simulation, according to the combined subjective probability distribution of the uncertain parameters.
(4). Code calculations are performed using the sampled N sets of parameters as inputs. By performing code calculations using variations of the values of the uncertain input parameters, and consequently calculating results dependent on these variations, the uncertainties are propagated through the calculations, and the calculated results include the total uncertainty at the defined probability content (quantile or percentile) and confidence level.

(5). The calculation results are ranked by order and a certain rank is selected as the estimator of the output of interest (i.e., selected key safety variables or figures of merits, such as peak-cladding temperature or PCT) at the defined probability content and confidence level. Statistical evaluations are also performed to determine the sensitivity of input parameter uncertainties on the uncertainties of key results (parameter-importance analysis).

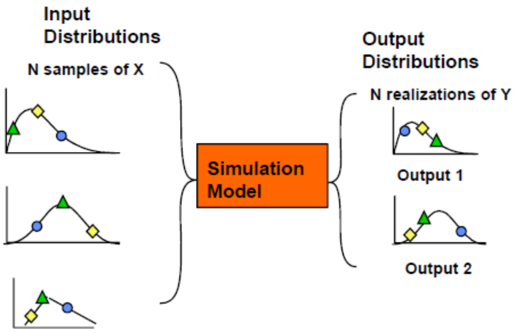

Figure 3. The uncertainty analysis method by propagation of input uncertainties[7].

The order statistics method needs to select a reasonable number of input uncertainty parameters and associated range of variations and possible distribution functions for each one. Selection of parameters and their distribution must be justified.

The number of code calculations (N) is determined by the requirement to estimate the probability content or tolerance (quantile)-confidence level interval for the calculation output results of interest. It is currently a common practice to rely on the non-parametric tolerance limits procedure to determine the minimum sample size. The so-called Wilks' formula[8] is used to determine the minimum number of calculations needed for deriving the one-sided tolerance limits (with top rank as the upper limit):

$$1 - \gamma^N = \beta \qquad \text{[Eq. 1]}$$

or the two-sided tolerance limits (with top rank as the upper limit, and lowest rank as the lower limit):

$$1 - \gamma^N - N\,(1-\gamma)\,\gamma^{N-1} = \beta \qquad \text{[Eq. 2]}$$

Where $\beta \times 100$ is the confidence level (%) that the maximum code result will not be exceeded with the probability $\gamma \times 100$ (%) (quantile) of the corresponding output distribution, which is to be compared to the acceptance criterion. A more general formulation was given by Guba and Makai[9].

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. N is representing the number of calculations such that the maximum calculated value in the sample is an upper-bound statistical tolerance limit. As an example, for a 95th/95th percentile ($\gamma = \beta = 0.95$), a minimum number of N=59 calculations should be performed for the single-sided, and 93 for the double-sided tolerance limit.

The non-parametric tolerance limits are used since nothing should be known about the distribution of the random variable except that it is assumed continuous. Moreover, the number N of code runs is independent of the number of the selected input uncertain parameters, but only depending on the tolerance limit quantile and on the desired confidence-level. The number of code runs for deriving sensitivity measures is also independent of the number of input parameters.

[7] Sandia National Laboratories, "DAKOTA Uncertainty Quantification," SAND 2009-0511P, (2009).

[8] S. S. Wilks, "Determination of Sample Sizes for Setting Tolerance Limits," *The Annals of Mathematical Statistics*, Vol. 12, pp. 91-96 (1941).

[9] A. Guba, M. Makai, and L. P´al, "Statistical aspects of best estimate method—I," *Reliability Engineering and System Safety*, vol. 80, no. 3, pp. 217–232, (2003).

This method is very robust and simple to implement, which makes it extremely interesting for licensing applications to nuclear safety analyses. Several such methodologies have been licensed and currently applied in the industry for best estimate large-break LOCA analysis[10-11].

However, it has been shown that this method may lead to rather conservative results (in particular when using the first rank) and variability (e.g., outliers). One way to improve the method is to increase the number of code calculations, and take higher ranks as the estimators for the given probability content (quantile) and confidence level[12]. However, it may be limited by the requested large calculation efforts in case of complex coupled code systems.

Due to its simplicity, robustness and transparency, this method will be implemented for most of the TE intended applications. However, an optimised number of calculations will be determined to improve the accuracy (see Section 5.1.1).

4.1.2. The bootstrap method

The use of a single order statistic (top rank or higher order) is subject to several limitations. First, it does not explicitly use potential information contained in other order statistics. Secondly, it may be subject to variability. Therefore, an alternative procedure based on confidence intervals obtained by the Bootstrap method[13] has been suggested to improve the accuracy[14; 15; 16].

The basic idea of bootstrap is to generate an empirical distribution for the calculation output results of interest by re-sampling the original samples with numerous replacements [12]. The method consists of:

(1). Random sampling repeatedly the calculation output results of interest to form Bootstrap samples of the same size as the original data sample.

The number of bootstrap samples can be as few as 100 for parameter estimation and up to more than 1000 for confidence interval estimation, and should be increased until the standard deviation error is acceptable.

The elements of each Bootstrap sample are randomly chosen from the original data, with replacements. Thus, a particular sample data point may be chosen several times or perhaps not at all in any particular bootstrap sample.

(2). The parameter of interest is then evaluated from each of the bootstrap samples generated.

(3). The numerous bootstrap replicates of the parameter can be used to estimate a probability distribution for the parameter. This is an estimate of the parameter sampling distribution, and from this distribution, confidence intervals may be approximated. Section 5.2.3 develops some situations that occur when replicating the probability distribution of a selected output.

The Bootstrap method allows one to assess the accuracy and uncertainty of estimated parameters from limited samples (e.g., the N Wilks' samples from order statistics) without any prior assumptions about the underlying distribution (e.g., the sample need not to be necessarily normally distributed). It also

[10] K. Muftuoglu, K. Ohkawa, C. Frepoli, M.E. Nissley, "Comparison of Realistic Large Break LOCA Analyses of a 3-Loop Westinghouse Plant Using Response Surface and Statistical Sampling Techniques," *Proceedings of ICONE (International Conference on Nuclear Engineering)-12*, Arlington, Virginia (Washington D.C.), U.S.A., April 25-29, (2004).

[11] R. P. Martin and L.D., O'Dell, "Framatome ANP's Realistic Large Break LOCA Analysis Methodology", *Nucl. Eng. Des.*, Vol. 235, pp. 1713-1725 (2005).

[12] A. de Crécy, et al., "The BEMUSE Programme: Results of the First Part Concerning the LOFT L2-5 Test," *Proceedings of 14th International Conference on Nuclear Engineering*, Miami, Florida, USA, 17-20 July, (2006).

[13] B. Efron, R.J. Tibshirani, "An introduction to the Bootstrap"; Chapman and Hall, (1993).

[14] J. JOUCLA and P. PROBST, "Rank Statistics and Bootstrap: A More Precise Evaluation of the 95th Perecentile in Nuclear Safety LB-LOCA Calculations," *Proceedings of 14th International Conference on Nuclear Engineering*, Miami, Florida, USA, 17-20 July, (2006).

[15] E. Zio and F. DiMaio, "Bootstrap and Order Statistics for Quantifying Thermal-Hydraulic Code Uncertainties in the Estimation of Safety Margins," *Science and Technology of Nuclear Installations*, Article ID 340164, (2008).

[16] C. Frepoli and I. Satish, "A Comparison of Non-Parametric Tolerance Limits with Linear Combinations of Order Statistics in Safety Analysis," *Proceedings of ICAPP 2011,* Nice, France, May 2-5, (2011).

allows assessing parameters that may not be expressed mathematically in simple terms (e.g., the median of a set of data).

It has been shown that, with the increase of computational power allowing running several hundreds of calculations, bootstrap method gives more precise results compared to the simple order statics using Wilks' formula[14; 15]. However, a more recent study[16] showed that the bootstrap estimators perform very similar to the Wilks' estimator. Depending on the selected implementation of the procedures (e.g., the selection of order statistics that defines the upper bound or the selection of the smoothing parameters), bootstrap may sometimes fail to ensure that the probability of under-estimating the quantile of interest is below the target (5%).

In this paper, TE will also compare the Guba&Makai's estimators and the bootstrap estimators, based on the application to the RELAP5 simulation of the THTF tests (see Section 5.2.3).

4.3. The sensitivity and uncertainty analysis tool

The DAKOTA (Design Analysis Kit for Optimization and Terascale Applications) code has been developed by the Sandia National Laboratory[17]. As shown in Figure 4, it provides a flexible, extensible interface between simulation codes and iterative analysis methods, via DAKOTA input files and executables.

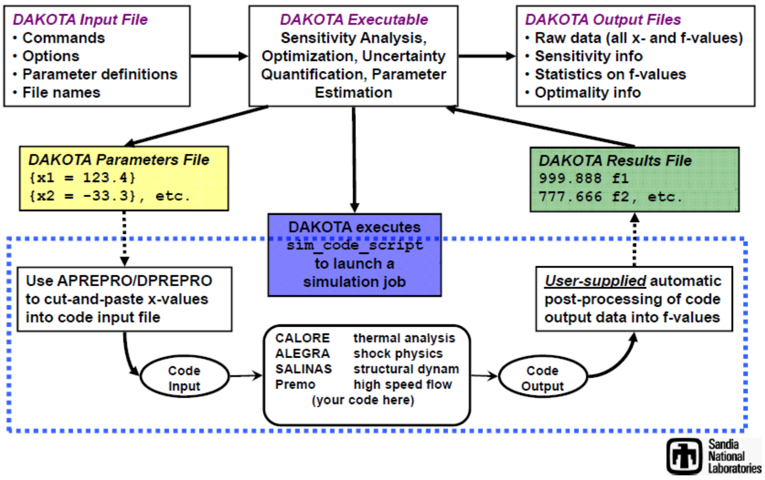

Figure 4. The DAKOTA uncertainty/sensitivity analysis process[18].

Among others, DAKOTA contains algorithms for uncertainty quantification (UQ) with sampling (Monte-Carlo or Latin Hypercube), epistemic uncertainty methods (Second order probability or Dempster-Shafer theory of evidence), and sensitivity analysis. These capabilities may be used on their own or as components within advanced strategies.

For the applications presented in this paper, the input uncertainty parameters ranges and distributions, as well as the uncertainty analysis method and number of samples are defined in the *DAKOTA Input File*. Based on the sampled or assigned input uncertainty parameters in the *DAKOTA Parameter File*, various scripts have been developed to create the code input files, to execute the simulation jobs, and to collect the code calculation output data into the *DAKOTA Results File*. The *DAKOTA Exec*utable will then perform the requested statistical uncertainty and sensitivity analysis, and provide the information in the *DAKOTA Output Files*.

[17] B. Adams, et al., "DAKOTA, A Multilevel Parallel Object-Oriented Framework for Design Optimization, Parameter Estimation, Uncertainty Quantification, and Sensitivity Analysis: Version 5.1 User's Manual," Sandia Technical Report SAND2010-2183, Updated Version 5.1, December (2010). (http://www.cs.sandia.gov/dakota)

[18] B. Adams, "DAKOTA 5.0 and JAGUAR 2.0 Capability Overview," SAND 2008-7901P, Sandia National Labs. (2008).

5. Preliminary applications and results

5.1. Application of the non-parametric tolerance limits to OECD RIA fuel codes benchmark

5.1.1. OECD RIA fuel codes benchmark

OECD/NEA is organising a benchmark of fuel rod codes based on 4 RIA experiments performed in the NSRR and CABRI test reactors[19]. The objective is to assess the capability and accuracy of the existing fuel rod codes to simulate fuel behaviours and evaluate the proposed safety criteria for the design basis RIA.

TE participates in this benchmark using the FRAPCON/FRAPTRAN codes. In addition the simulation of the test fuel rods according to the specifications, TE has applied the tolerance limit uncertainty and sensitivity analysis method to one of the test rods (CIP3-1)[20].

The CIP3-1 is a RIA test that will be performed in the CABRI reactor with pressurised water loop, at a pressure of 155 bars and an inlet temperature of 280 °C. The rodlet used for this experiment has been refabricated from a fuel rod irradiated for five cycles in the Vandellós 2 PWR nuclear power plant.

5.1.2. FRAPCON/FRAPTRAN models and assumptions

FRAPCON3.4 code is used to simulate the base irradiation of the fuel rod, based on the specifications. The base irradiation has been performed on a real fuel rod in a nuclear power plant but RIA experiments have been performed on rodlet taken from these rods. The base irradiation for the test rodlet is thus performed with some adjustment in the input parameters. The calculation results (burnup, corrosion thickness) are then compared with the available measured data, in order to validate the FRAPCON input model and assumptions.

FRAPTRAN1.4 is used to calculate the transient behaviour of the fuel rodlet during the RIA test, using the base irradiation obtained with FRAPCON3.4.

A base case with nominal values for all input parameters is defined, and the default models in both FRAPCON/FRAPTRAN codes are chosen in the calculations. The "best estimate" calculation results will be compared with those made by other organizations using various fuel rod codes, in order to identify their differences and to propose improvements in the modelling.

5.1.3. Application of the non-parametric tolerance limits method

The tolerance limit uncertainty and sensitivity analysis method based on the Wilks' formula is then applied to one of the test rods (CIP3-1). The objective is to test the applicability of the method to fuel rod behaviour calculations, and to determine the optimal number of calculations to obtain an acceptable accuracy of the Wilks' estimator.

The tolerance limit method is chosen because it provides a more realistic range of variation of the results based on physical considerations and on a well known and robust mathematical tool.

Following the approach presented in §4.1.1, all input parameters and the models that may be subject to uncertainties are first chosen based on the engineering judgement. Some sensitivity analyses are then performed on the individual input parameters of the code in order to determine the most influential ones.

For each of the identified key uncertainty input parameters, a mean value, a standard deviation and a range of variation (lower and upper limit values) as well as the distribution type must be defined, as summarized in Table 1.

For the current application, a normal distribution has been assigned to all the considered input parameters. For the geometrical parameters, this is justified by the fact that a lot of measurements

[19] M. Petit et al, "WGFS RIA fuel rod codes benchmark specifications." NEA/CSNI, OECD, (2011).

[20] J. Zhang, T. Helman, Z. Umidova and A. Dethioux, "Statistical Uncertainty Analysis of CIP3-1 Simulation with FRAPCON/FRAPTRAN - Preliminary Results and Suggestions," presented at the *OECD/WGFS 1st Seminar on RIA fuel rod codes benchmark*, Paris, 19-20 September, (2011).

have been done during the fabrication. Their standard deviation has been taken as the half of the maximum of the absolute value of the difference between their nominal value and their upper or lower bound.

Table 1. Input uncertainty parameters for statistical uncertainty analysis of CIP3-1 FRAPCON/FRAPTRAN simulation

Input uncertainty parameter	Mean	Standard deviation	Lower bound	Upper bound	Distribution
Thermal conductivity model	0	1	-2	2	Normal
Thermal expansion model	0	1	-2	2	Normal
Fission gas release model	0	1	-2	2	Normal
Fuel swelling model	0	1	-2	2	Normal
Cladding creep model	0	1	-2	2	Normal
Cladding corrosion model	0	1	-2	2	Normal
Cladding hydrogen uptake model	0	1	-2	2	Normal
Multiplicative factor on the temperature history during base irradiation	1	0,00355	0,9929	1,0071	Normal
Multiplicative factor on the power history during base irradiation	1	0,02	0,96	1,04	Normal
Multiplicative factor on the power pulse	0,92976	0,0186	0,89257	0,96695	Normal
Coolant inlet enthalpy (J/kg) during the transient	1232080	5080	1221920	1242240	Normal
Cladding outside diameter (m)	0,0095	0,000019	0,009462	0,009538	Normal
Cladding inside diameter (m)	0,008357	0,000019	0,008319	0,008395	Normal
Dish radius (m)	0,002475	0,0000625	0,00235	0,0026	Normal
Fuel density (%)	95,5	0,75	94	96,5	Normal
Pellet diameter (m)	0,008192	0,000006	0,00818	0,008204	Normal
Cladding roughness (µm)	0,6355	0,31725	0,001	1,27	Normal
Fuel roughness (µm)	1,6005	0,79975	0,001	3,2	Normal
Cold plenum length during base irradiation (m)	0,029531	0,000884	0,0278	0,0301	Normal

The coolant inlet temperature as well as the power history (for both the base irradiation and the pulse) is measured values. They are thus dependent on great number of independent random variables (i.e. the factors influencing a measure are numerous and random) and the central limit theorem shows that a variable which is dependent on great number of independent random variable also has a normal behaviour. Their mean has been taken as the value provided in the specification and their standard deviation has been taken as the standard measurement error considered in a nuclear power station.

Finally the parameters on the models in FRAPCON have also been chosen as normal for the same reasons as for the coolant temperature and power history. Since they bias the model to the number of standard deviation corresponding to the number assigned to them, their mean is zero and their standard deviation 1. The range of all these variables has been chosen as being their means plus or minus two times their standard deviations.

The impact of the number of calculations and distributions of the input uncertainty parameters are also studied.

5.1.5. Results and discussion

For the benchmark purpose, the double-sided tolerance limit is used in order to define the lower and upper bound of the calculated output values for the whole RIA transient. The objective is to demonstrate if the experimental data or the calculated mean values are well bounded by them.

From [Eq. 2], for a probability of 95% at a confidence level of 95%, the minimum number of simulations is determined as N=93, if we take the top rank as the upper bound and the lowest rank as the lower bound. The so-defined lower and upper bound values, as well as the best estimate values for the enthalpy increase after steady-state at peak power node (PPN) are shown in Figure 5.

Also shown in the Figure 5 are the results with uniform distribution for all input uncertainty parameters. It is clear that the calculated best estimate value is well bounded by both lower and upper limits, and that the uniform distribution tends to give larger differences between the lower and upper limits.

As discussed earlier, for safety demonstration purpose, one has to verify the compliance of a safety criterion. In this case, it is important to determine an upper bound on a safety parameter of unknown continuous distribution. According to the single-sided tolerance limit theory, the minimum number of simulations is N=59, if we take the top rank as the upper bound. However, it has been shown that the calculated upper bound with the minimum sample size and order statics (the Wilks' estimator or quantile) is often too imprecise.

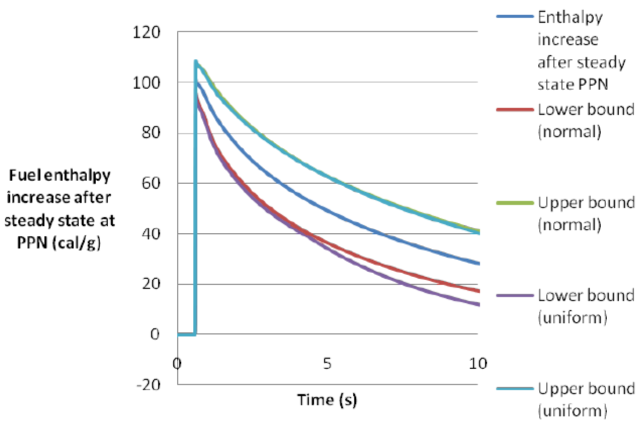

Figure 5. The statistical uncertainty analysis results for CIP3-1 simulation.

Intuitively, the precision on the calculated quantile increases with the number of simulations. Indeed, let's consider the fraction of the sampled values of the output parameter (Z/N) that is inferior to the real quantile. It is clear that if N becomes very large, the sampled population is near the real population; and if Z/N becomes close to the desired fraction (β), the calculated quantile is nearer to the real quantile (or empirical quantile, which can be found from the DAKOTA output files by using the position of the quantile as being the closest integer from 0.95N)[21].

This can be demonstrated by comparing the Wilks' quantile and the empirical quantile for the enthalpy increase after steady-state at PPN as shown in Figure 6. It clearly shows that the gap between the Wilks' and empirical quantile decreases with the number of simulations as expected.

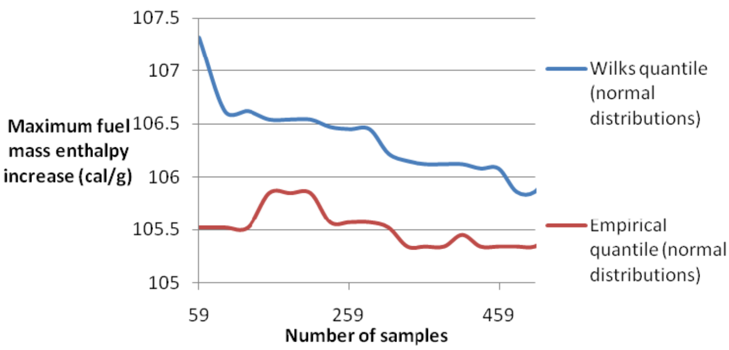

Figure 6. Comparison of the Wilks' quantile and empirical quantile for CIP3-1 simulation[21].

This fact allows enabling a cut-off criterion on the optimal number of calculations applicable for the safety demonstration. The number of run simulations is obviously dependent on the accuracy that one wishes to achieve. However, a greater number of simulations imply a higher computer time cost.

A stopping criteria based on those two considerations could be proposed as follows[21]:

- An absolute maximum number of simulations (N_{max}) is decided for an acceptable accuracy, depending on the cost in computer time per simulation.
- If this N_{max} is reached and sufficient accuracy has not been achieved, the last calculated quantile value is kept. If a sufficient accuracy has been achieved, the calculation stops.

The proposed way to determine this "acceptable accuracy" is to consider that the gap between the empirical quantile and the Wilks' quantile is sufficiently low and to verify if the variation in the

21 T. Helman, "Application of the statistical uncertainty analysis method to simulation of Reactivity Insertion Accidents with FRAPCON3.4/FRAPTRAN1.4," Master Thesis, Free University of Brussels, Belgium, September (2011).

Wilks' quantile is not too fast. This could be done by considering a certain number of the previously calculated quantiles and consider the variation between them.

The impact of the input parameters distributions is also studied. In order to do that, all the distributions in Table 1 have been changed to uniform distributions with the same variation range. The results on the maximum mass fuel enthalpy increase for both cases are compared in Figure 7.

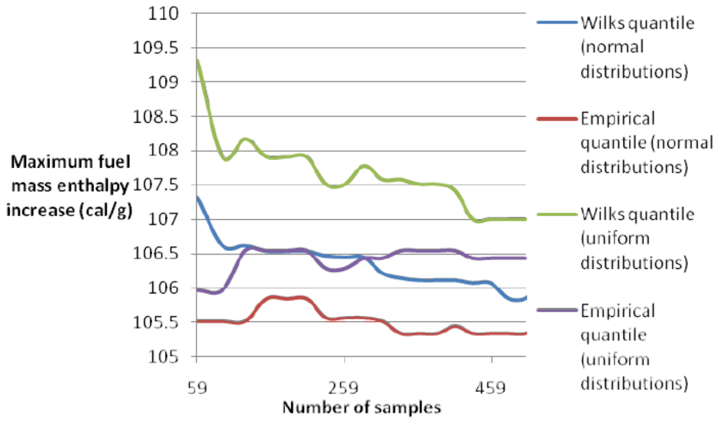

Figure 7. Impact of input uncertainty parameters distribution for CIP3-1 simulation[21].

It can be observed that both the Wilks' quantile and the empirical quantile increase when uniform distributions are used for the input uncertainty parameters. This is what was expected since the uniform distribution favours the sampling of extreme values of the input uncertainty parameters, which is equivalent to increase the uncertainties on the input parameters, and hence leads to the higher quantiles which are the uncertainties on the output parameters.

The impact of the input parameters distributions is thus important. Their distributions should therefore be chosen carefully and more study is needed to determine them precisely (at least for the consequent input uncertainty parameters).

The DAKOTA code calculates also the Pearson's correlation between the input and output parameters. This correlation is used to determine the correlation coefficient not between the parameters themselves but rather between the ranks of the parameters allowing determining if there is a monotone relation between them.

The input uncertainty parameters having a strong Pearson's correlation should be carefully defined due to their importance. The input uncertainty parameters having a weak Pearson's correlation (typically less than 0.3) with every output parameter can be eliminated (or simply keep at best estimate), after verification on the scatter plots that there is not another type of relation between the considered input uncertainty parameters and the output parameters (such as a sinusoidal relation). The scatter plot in Figure 8 below shows that the calculated energy input in the fuel has a strong relation with the sampled multiplicative factor on the power pulse (Pearson's correlation coefficient is approximately equal to 0.94).

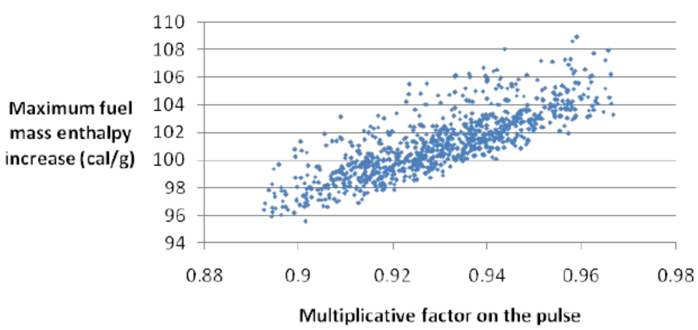

Figure 8. Impact of the power pulse multiplier on CIP3-1 simulation uncertainty[21].

However, the scatter plot in Figure 9 shows that there is effectively no dependency between the fuel enthalpy increase and the cold plenum length during base irradiation. It can thus be considered as having a weak influence, and hence can be eliminated as one input uncertainty parameter.

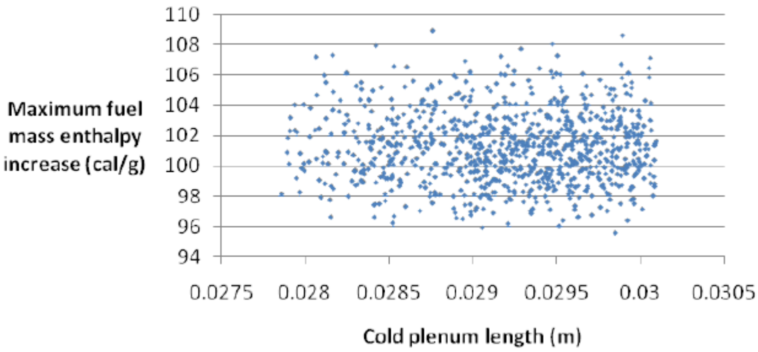

Figure 9. Impact of the cold plenum length on CIP3-1 simulation uncertainty[21].

The above results show that the DAKOTA sensitivity analysis by Monte Carlo is a simple and powerful tool to determine the importance of the input parameters in statistical uncertainty analysis. It is much easier and faster than the sequential sensitivity studies performed to define the input uncertainty parameters.

5.2. Application of the non-parametric tolerance limits and bootstrap methods to RELAP5 simulation of THTF tests

5.2.1. THTF tests

The experiments were performed in the Thermal Hydraulic Test Facility (THTF), an electrically heated thermal hydraulic test loop. Two sets of experiments were run; the first to obtain in steady state conditions both void fraction and uncovered core heat transfer data (tests 3.09.10 I through N) and the second to obtain only void fraction data (tests 3.09.10 AA through FF)[22; 23].

The THTF core is a 3.66-m heated length, 64-rod tube bundle. The flow is injected directly in the lower plenum and do not pass through the downcomer, which is isolated from the primary circuit.

The selected test for the experiment developed in Section 5.2.3 and 5.2.4 is Run 3.09.10 L, which is part of the uncovered bundle heat transfer assessment tests.

Table 2 shows the test initial and boundary conditions.

Table 2. THTF Uncovered Bundle Test 3.09.10 L Conditions

Test	***p*** **[bar]**	T_{inlet} **[K]**	***G*** **[kg/m²·s]**	***W*** **[kg/s]**	***Q*** **[kW]**	Q_{loss} **[%] / [kW]**
3.09.10L	75.2	461	29.1	0.181	476	2% / 10kW

5.2.2. RELAP5 models and assumptions

Figure 10 presents the RELAP5/Mod3.3 THTF model, in which the heat structure #310 (Radial Node 5) represents the simulated fuel rod (electric heater).

The RELAP5/Mod3.3 simulation results are shown in Figure 11, in comparison with the experimental data. The tendency is well predicted, but the wall temperatures at the top are slightly under-estimated.

22 T.M. Anklam, et al., "Experimental investigations of uncovered-bundle heat transfer and two-phase mixture level swell under high pressure low heat flux conditions", NUREG/CR-2456, USNRC, April (1982).

23 T.M. Anklam and R.F. Miller, "Void fraction under high pressure, low flow conditions in rod bundle geometry", *Nuclear Engineering and Design*, 75, pp. 99-108, (1982).

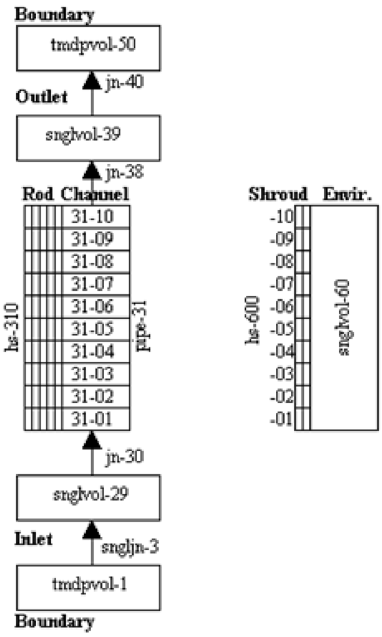

Figure 10. RELAP5/Mod3.3 model for THTF.

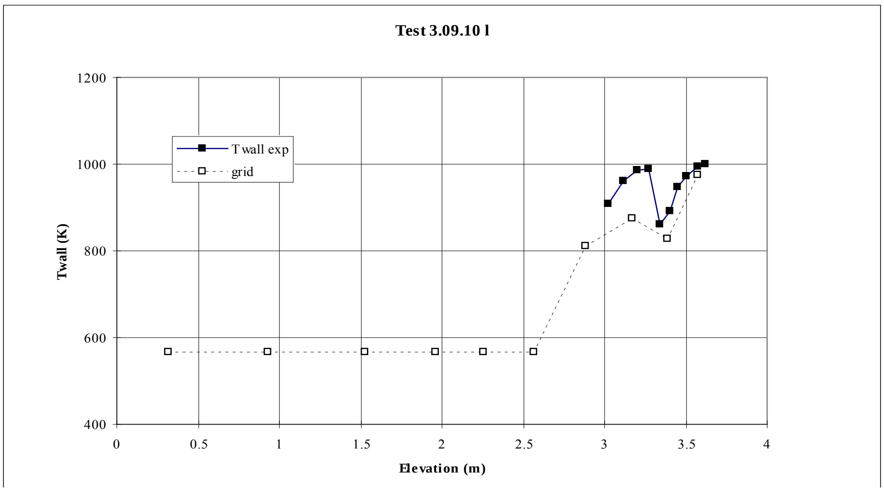

Figure 11. Run 3.09.10 L axial cladding temperatures and RELAP5/Mod3.3 predictions.

5.2.3. Application of the non-parametric order statistics method

The objective is to compare the predictions of different non-parametric estimators, for a given quantile at a certain confidence level, upon an arbitrary surrogate safety criterion surveying a parameter; the real aim though, is comparing the goodness of the methods when estimating the 95th quantile at a 95% confidence level, against a "true" quantile estimated from a very large sample. The safety criterion is not important per se for this purpose, but just defining a surveyed parameter which will be "Temperature at the Cladding Node 9".

The experiments are performed by sampling 4 parameters {inlet mass flow, inlet/outlet pressure, flow temperature and fouling factors (at the grid spacers in heat structure #310 nodes 6 and 9)}, according to their uncertainty distribution functions. Those variables are passed automatically by DAKOTA onto an input deck, which is run and its results collected. This operation is replicated for as many samples generated.

Four batches of 59, 153, 1000 and 4000 samples are generated by DAKOTA. The kth rank estimators (Guba&Makai's estimator) for those three samples are obtained by applying [Eq. 3], while the 95th

quantiles are obtained on the basis of a truncated order-statistic [Eq. 4] (guaranteeing that the estimator is in the upper tail and therefore bounds the target quantile):

[Eq. 3]

[Eq. 4]

Where β=0.95, γ=0.95, N is the sample size, and k and t the ranks of the order statistic estimators. According to [Eq. 3] or [Eq. 4], the estimator is chosen from the ordered sample output:

$\alpha(1)<\ldots<\alpha(i^{th})<\ldots<\alpha(t^{th})<\ldots<\alpha(k^{th})<\ldots<\alpha(N^{th})$.

Section 4.1.1 discussed the implementation of non-parametric order statistics method in safety analysis, which is a very conservative and robust procedure when a given confidence level on a measure needs to be assured. It is self evident that the Guba&Makai's estimator will over-predict the 95th quantile a 5% of the time, and as the population is enlarged Guba&Makai's kth estimator will converge towards the 95th quantile from [Eq. 4]. However, a large population is difficult to generate from a production point of view. Therefore it is interesting to use bootstrap to exploit all information available in a reduced size sample.

5.2.4. Application of the bootstrap method

Two examples of the bootstrap method are presented here. The first is based on the quantile estimation by percentile smoothing, the second is based on the reconstruction of the 95th quantile density.

Given a sample size N=153, the bootstrapping procedure is applied, generating *n* resamples of N elements, with replacement. The number of resamples (*n*) was first fixed to 100, nonetheless higher values of n need to be invoked however when estimating farthest quantiles. Another trial with a resample size of n=200 was generated to check whether we got stable quantile estimates with a slightly larger bootstrap resample size.

Percentile Smoothing

This method, as developed previously[24; 25] and recently recalled in[15], consists of using a single order statistic with linear combinations of order statistics with weights derived from a Gaussian kernel [Eq. 7]. The normalized weights [Eq. 6] are applied to a linear combination of order statistics α(i) to yield in the estimator for a target quantile P_{95} [Eq. 5]:

[Eq. 5]

[Eq. 6]

[Eq. 7]

The only parameters remaining to set in [Eq. 6] is p = 0.95 (centering the weights on the 95th quantile) and in [Eq. 7], the kernel bandwidth, h, which is fixed here as 1/N as suggested in[15].

These weighting are applied to each of the *n* samples, yielding in *n* estimates of the target quantile P_{95}. It is then possible to calculate the empirical standard deviation of the replications (σ), and an estimate of the upper confidence bound estimate of the 95th quantile by an L-estimate:

$P_{95/95} = \{P_{95}\} + 1.645\ \sigma$ [Eq. 8]

where $\{P_{95}\}$ is the original sample smoothed 95th quantile.

24 M. Wand, M. C. Jones, "Kernel Smoothing", Chapman and Hall, (1995).

25 S.J. Sheather and J. S. Marron, "Kernel Quantile Estimators" *J. American Statistical Association*, 85, (1990).

The L-estimate is assumed valid in this case by the central limit theorem applied to the 95th quantile distribution. However, it could also be claimed that a valid representation of the estimated quantile would be:

$P_{95/95} = <P_{95}> +1.645\ \sigma$ [Eq. 9]

where $<P_{95}>$ is the average calculated from the resamples smoothed P_{95}'s.

Non-Parametric Density Estimation

This method disregards the nature of the density function of a given random variable, but estimates this directly from the data. The simplest way to approximate a density function is by its histogram

or alternatively:

[Eq. 10]

where $\{x_1, x_2\ldots, x_N\}$ are observations, and w(t, h) are some appropriate weighting functions. These w-weighting functions are of the form and K is a kernel which determines the shape of the weighting function (function of x, $\{x_1, x_2\ldots, x_N\}$ and of the bandwidth *h*). For this experiment, a Gaussian kernel is used. An appropriate choice of h, suffices to say, is the most important factor in the reconstruction, rather than the kernel shape itself; for this experiment, it will be fixed as , where δ is denoted as the observations standard deviation[26].

Let us use the same information as above, coming out from the bootstrapped *n* resamples: from them, *n* P_{95} quantiles based on [Eq. 4] are determined which will be our observations: $\{x_1, x_2\ldots, x_n\}$.

By using [Eq. 10], the probability density function of the 95th quantile can be reconstructed, and from it, its average ($<P_{95}>$) and its standard deviation (σ) can be estimated in order to produce an L-estimate, identical to that of the percentile smoothing: $P_{95/95} = <P_{95}> +1.645\ \sigma$.

However, there is a property that yet needs to be commented: it would be highly desirable that the estimator has the same support as the real distribution f(x). In other words, if *x* is a random variable which takes values restricted to an interval [a, b), then the support of f(x) is [a, b) and as well needs to be restricted to [a, b), so that for any x such $x<a$ and $x\geq b$. This bump can be simply overcome by reconstructing the density of a variable $y = Z(x)$, which has unbounded support (by an appropriate change of variable Z), and reversing the variable change afterwards. More sophisticated methods are available.

It is precisely this property which seems to render not fructuous this approach: there is virtually no information about the support [a, b) of f(x). Using arbitrary supports {i.e., [0, ∞)} for a temperature density reconstruction, for instance, results in the kernel assigning non-zero, though extremely small probabilities to unphysical values. It can be argued that in a LOCA the clad temperature will never fall below the ECCS temperature, or that the maximum clad temperature would never be below the clad temperature at the transient onset.

While it still could be possibly be defended a lower bound in some cases, leaving the upper bound unassigned would lead to a crude overestimation. In any case, it would be rather more interesting to no restricting the upper limit and investigating the associated probabilities of extreme values from a risk-informed standpoint.

Just for comparison sake, the estimation based on kernel regression and percentile smoothing is presented in Section 5.2.5. For this application, two supports were chosen; the first one [1010, ∞) was

26 B. W. Silverman, "Density Estimation", Chapman and Hall, (1986).

chosen by selecting a lower limit slightly smaller than the smallest of the *n* P_{95}'s observed. The second one [1010, 1220] selected a much larger upper limit than any of the *n* P_{95}'s collected.

5.2.5. Results and discussions

Table 3 presents the results of different sizes (N), by using Guba&Makai's estimator and an order statistic for estimating the 95th quantile [Eq. 3] and [Eq. 4]. As expected, the Guba&Makai's estimator [Eq. 3] and the truncated estimator from [Eq. 4] converge at a very large population size (4000 runs).

Table 3. Estimation of the 95th quantile by order statistics.

Sample size (N)	**59**	**153**	**1000**	**4000**
k **(k$^{th – order\ statistic}$ largest value) [Eq. 3]**	1	4	39	200
Sample 95th quantile/95 confidence – rank *k* estimator (K)	1179.8	1161.4	1150.2	1137.3
t **(t$^{th – order\text{-}statistic}$ larger value) [Eq. 4]**	2	6	48	202
Sample 95th quantile – rank *t* estimator (K)	1164.6	1151.4	1142.7	1136.8
Sample smoothed quantile {P_{95}}	1156.7	1143.6	1139.8	1137.4

The last section explained the use of two different quantile estimators in order to exploit the information available in one sample. The results from the approaches exposed in §5.2.4 are presented in Table 4, in which the difference between using <P95> or {P95} is very small.

Table 4. Resample size: n= 100, N= 153 (sample 1).

Method	**Reconstruction (Variable change)**		**Percentile smoothing**	
Support (K)	**[1010, ∞)**	**[1010-1220]**	N/A	
<P_{95}> (K)	1144.5	1144.2	1141.9	1143.6 (*)
σ (K)	16.5	16.2	14.5	
<P_{95}> + 1.645 σ	1171.6	1170.8	1165.7	1167.5
(*) Corresponds to {P_{95}} from Table 3.				

The resample size *n* is increased to 200 in order to investigate the stabilization of the estimators and the results are presented in Table 5.

Table 5. Resample size: n= 200, N= 153 (sample 2).

Method	**Reconstruction (Variable change)**		**Percentile smoothing**	
Support (K)	**[1010, ∞)**	**[1010-1220]**	N/A	
<P_{95}> (K)	1142.6	1142.5	1140.6	1143.6 (*)
σ (K)	15.4	15.4	13.0	
<P_{95}> + 1.645 σ	1167.9	1167.8	1162.0	1165.0
(*) Corresponds to {P_{95}} from Table 3.				

The estimators have seemingly saturated, and the percentile smoothing (with <P_{95}>) approaches to the Guba&Makai's estimator (see Table 3, N=153). This, by comparison, confirms the robustness of Guba&Makai's estimator, as stated in Section 4.1.1.

The investigation performed herein does not suffice to conclude upon the variability of the estimators, which is directly related to the variance and function of the sample size. The bootstrap estimators are similar in nature, though the reconstruction methods would require more information than what is available to the analyst and therefore seem not valid for safety analyses purposes.

6. Conclusions and perspectives

Following the recent trends in research and development as well as in industrial applications, the TE best estimate plus statistical uncertainty analysis method (BESUAM) has been proposed, based on the chosen non-parametric order statistic or bootstrap methods, and a sensitivity and uncertainty analysis tool (DAKOTA).

The proposed method has been applied to some multi-physical simulations in order to demonstrate the robustness of the order statistic method and the DAKOTA tool.

The order statistic uncertainty analysis method has been used to the FRAPCON and FRAPTRAN simulation of a RIA test (CIP3-1). It has been shown that:

- This method provides upper and lower bounds on the chosen parameters that can be used to quantify the accuracy of the RIA fuel rod code simulation results.
- A better accuracy of the calculated quantile can be obtained by increasing the run number.
- A stopping criterion based on the maximum computer time cost acceptable and on the desired accuracy of the quantile has also been proposed.
- The feasibility of sensitivity analysis based on the sampled calculations is also shown.

The order statistic and the bootstrap uncertainty analysis methods have been used to the RELAP5/MOD3.3 simulation of a THTF test. It has been shown that:

- The order statistic (Guba&Makai's) estimator overpredicts the 95th quantile at reduced sample number, and will converge towards the true 95th quantile at higher sample numbers.
- The bootstrap estimator using the percentile smoothing (with $<P_{95}>$) approaches to the Guba&Makai's estimator, which confirms the robustness of the order statistics.
- The bootstrap estimators using different approaches are similar in nature, but the reconstruction method would require more information than what is available to the analyst and therefore seem not valid for safety analyses purposes.

The non parametric order statistic methods are robust and flexible tools to investigate extreme quantiles; they have been successfully applied in this paper to two problems of different nature: fuel rod thermal mechanics and system thermal hydraulics. No knowledge of the output distribution is needed in any of the methods used, with the exception of the density reconstruction method requiring the output support data.

As part of the TE's efforts to mastering the uncertainty analysis in multi-physics simulations, TE will further apply the non-parametric order statistic method and the DAKOTA tool to the OECD/NEA/NSC UAM benchmark phase II on fuel physics (II-1), neutronics (II-2) and bundle thermal hydraulics (II-3). The preliminary application to COBRA-3C_TE simulation of UAM II-3 [27] shows promising results.

The final objective is to apply the proposed BESUAM method for reloads safety evaluation and safety analysis. This, of course, still requires considerable efforts to identify, quantify or define the ranges and distributions of all relevant uncertainty attributors. A practical approach to separating and treating aleatory and epistemic uncertainties is also needed.

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The authors are also grateful to OECD/NEA for allowing publication of the preliminary results using the data provided within the CSNI (WGAMA and WGFS) and NSC (EGUAM) projects.

[27] A. Dethioux, F. Van Humbeeck, J. Zhang and C. Schneidesch, "Preliminary Uncertainty Analysis of UAM Exercise II-3 (Bundle Thermal-Hydraulics) with DAKOTA," presented at the *OECD LWR UAM-5 Workshop*, Stockholm, Sweden, 13-15 April (2011).

Experience in KINS on Best Estimate Calculation with Uncertainty Evaluation

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In the present paper, experience of Korea Institute of Nuclear Safety (KINS) on Best Estimate (BE) calculation and uncertainty evaluation of large break loss-of-coolant accident (LBLOCA) of Korean Pressurized Water Reactor (PWR) with various type of Emergency Core Cooling System (ECCS) design is addressed. Specifically, the current status of BE code, BE calculations and uncertainty parameters and related approaches are discussed. And the specific problem such as how to recover the difficulty in treating the uncertainty related to the phenomena specific to ECCS design (e.g., upper plenum injection phenomena) is discussed. Based on the results and discussion, it is found that the present KINS-REM has been successfully developed and applied to the regulatory auditing calculations. Need of further study includes the improvement of reflood model of MARS code, uncertainty of blowdown quenching, and reconsideration of the unconcerned model and fuel conductivity degradation with burnup.

1. Introduction

Best Estimate (BE) calculation with uncertainty evaluation of large break loss-of-coolant-accident (LBLOCA) has been increasingly applied to the licensing applications such as fuel design change, power uprate, steam generator replacement, and licensing renewal of nuclear power plants (NPP) since the KEPCO Realistic Evaluation Method (KREM) was approved for the Westinghouse 3-loop cold leg injection plants in Korea at 2002 [1]. Until 2010, the method has been expanded, improved and approved to apply to the various type of emergency core cooling system (ECCS) including Upper Plenum Injection (UPI), and Direct Vessel Injection (DVI) [2].

To support the licensing review of Korea Institute of Nuclear Safety (KINS) on those matters and to confirm the validity of licensee's calculation, regulatory auditing calculations have been conducted. For this aspect, a general and systematic BE methodology including uncertainty evaluation has developed, which aims to calculate the peak clad temperature (PCT) following a LBLOCA at the 95 percentile probabilistic upper limit in 95 percentile confidence level. The method, KINS Realistic Evaluation Model (KINS-REM), adopted the most advanced system thermal-hydraulic codes, RELAP5 and MARS-KS, and the internationally available experiment data base [3]. To obtain the PCT that reflected all the uncertainty contributors, a certain number of the code runs are conducted, which of each involves models and input parameters adjusted to represent the ranges and statistical distribution of the specific models of interest. The number of code runs is determined based on the non-parametric statistics and random sampling scheme [4].

Although basic framework of the KINS-REM was developed in general sense, the specific parts of the method should be further developed and improved through the application to calculations of actual plants having a various type of emergency core cooling system (ECCS) design.

The present paper is to discuss how to apply the KINS-REM to the actual problem, specifically regarding thermal-hydraulic phenomena, BE code, BE modeling, uncertainty parameters and related approaches for the various type of ECCS design. Also, the future direction to complement the current methodology is suggested to evaluate the important safety issue such as fuel conductivity degradation with burn-up.

2. Description of KINS-REM

The KINS-REM was illustrated in Fig. 1[4]. The method was composed of three elements subdivided by fifteen steps. Through the element 1, the important phenomena affecting the primary safety criteria (PSC) such as PCT are identified over the important components and the sub-phases of the given accident scenario for the selected plant based on PIRT process. Also the major models which are required to justify the applicability of the frozen code are identified for the highly-ranked phenomena. During this element, the deficiencies and/or inability of the code in simulating each specific phenomena are also identified, which needs additional resolutions to compensate the deficiency at element 3.

In element 2, the capability of the code is determined by the calculations of the selected SET and IET having well-established database and documentation describing the target phenomena and data quality. In this element, it is determined if the noding and modeling scheme to simulate the experiment facility and the specific experiment is the acceptable one. This modeling scheme should be consistent with the plant modeling scheme. (Step 7)

The code accuracy to predict the specific phenomena (e.g., critical flow) can be obtained from the comparisons of the code calculations with SET and IET during the element 2. Such a code accuracy is considered by bias of the model, i.e., difference between the model and the experimental data, when calculating the plant response. In the course of this process, the effect of scale difference between experiment and the actual plant is taken into account. If the comparison can be done for PCT, the bias may be directly added to the final PCT at element 3. (Steps 10 and 11)

In this element, the uncertainty ranges of the models and correlations of the code important to PSC are determined through the comparison with the experimental data or available data. The ranges confirmed at the international program such as BEMUSE [5] can be reasonably used. (Step 9)

In the element 3, the combined plant responses considering the uncertainties of models and correlation of the code and plant parameters are determined through the 124 code runs. Sampled combinations of the values randomly sampled from the distribution and range of each model and parameter of importance pre-determined above are implemented into 124 sets of input. The number '124' was based on one-sided third order Wilks's formula for 95 percentile probability and 95 percentile confidence level [6].

Regarding some phenomena whose uncertainties cannot be statistically represented by the individual parameters, bias may be introduced such that the difference between the code and the relevant experiment is reasonably addressed to the effect on the plant PSC. The final PSC considering all the uncertainty can be obtained by combining them.

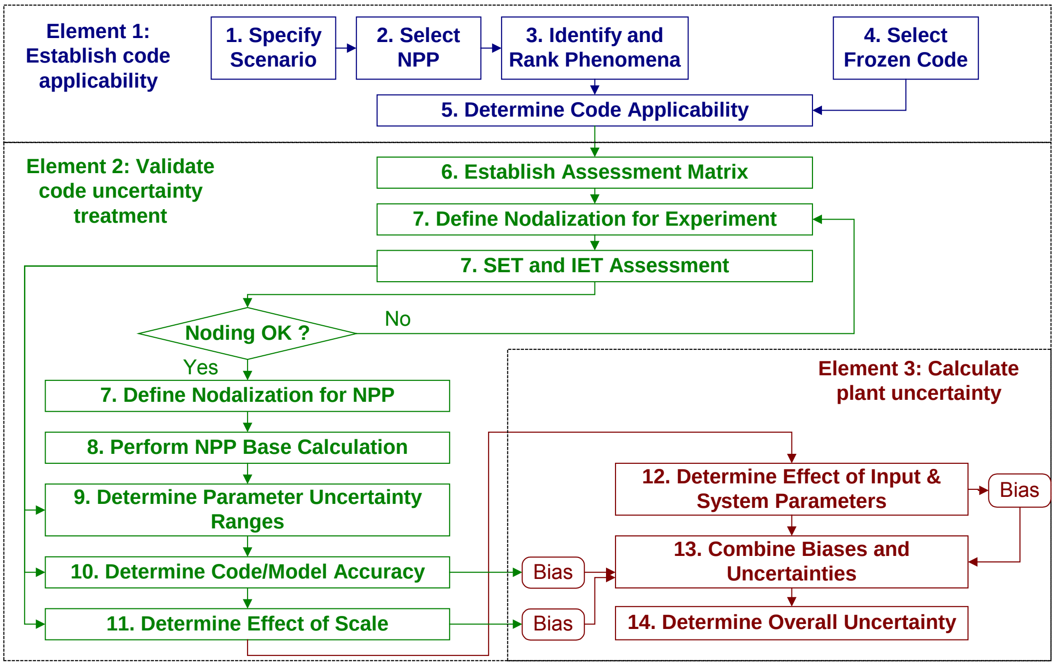

Fig. 1 Illustration of KINS-REM

3. Application of KINS-REM to Actual Plant LBLOCA

The KINS-REM described above has been applied several times to the independent analysis to support the licensing review such as fuel design change, steam generator (SG) replacement and construction permit of new NPP. Typical examples are as follows

Table 1. List of Cases Calculated by KINS-REM

Plant	Design	ECCS *	Application Type	Code	Ref
Kori 1	Westinghouse 2-loop	CLI/UPI	Licensing renewal	RELAP5/MOD3.3	[7]
Kori 2	Westinghouse 2-loop	CLI/DVI	Fuel design change	RELAP5/MOD3.3	[8]
YGN 3,4	Korean OPR1000	CLI/CLI	Methodology change	RELAP5/MOD3.3	[9]
SKN 3,4	Korean APR1400	DVI/DVI	Construction permit	RELAP5/MOD3.3 MARS-KS	[4]

*Note: *A/B* denotes the injected location of A for high pressure safety injection and one of B for low pressure safety injection.

As shown in the table, the plants have various types of ECCS design including low pressure direct vessel injection or upper plenum injection. Also the licensing submittals were prepared for the various purposes. Therefore, it was required to define the objectives of the independent audit calculation on each case in order to conduct BE calculation and uncertainty quantification suitable to the specific objectives. Key issue was how to perform the BE calculation, how to select parameters and how to determine the ranges of uncertainty specifically to the purpose.

3.1 Phenomena

The present methodology has a first step to define the important phenomena which is starting point of all the following steps. Phenomena during the blowdown phase of LBLOCA were well-known [10] and included fuel stored energy, break flow, depressurization and flashing, and core heat transfer, which are believed to be identical irrespective of ECCS design. Phenomena during refill/reflood can be differed by specific ECCS design and some of the blowdown phenomena such as gap conductance, decay heat, core heat transfer are still involved. ECC Bypass phenomenon was one of the most important contributors. Regarding the UPI, most of the existing reflood phenomena were involved and pool formation, local downward flow, and steam condensation at upper plenum were additionally identified based on the existing database [11]. For DVI, downcomer boiling at hot vessel wall and sweep-out by steam flow were identified as contributors to ECC Bypass during reflood phase [11, 4].

3.2 BE Thermal-hydraulic Codes

RELAP5/MOD3.3 code[12] and MARS-KS code [13] have been used in KINS. The MARS-KS code has been developed to keep the capability of RELAP5/MOD3 code and COBRA-TF code, to have 3-D hydrodynamic capability, and to have coupling capability with containment codes and 3-D neutronics codes. Accordingly, the same basic equations, numerical scheme and models/correlation are used in two codes but different code modular structure and data storage management. Especially, the reflood model of the MARS code was different from the one of the RELAP5/MOD3.3 (PSI model as default).

From the assessment of the RELAP5/MOD3.3 and MARS against the typical IET including LOFT and Semiscale,[12, 13] it was found that all the important blowdown phenomena were predicted well or a little conservatively and that the code applicability can be assured.

The code applicability for the refill and reflood phase can be differed by plant configuration of reactor vessel and ECCS design, thus, the plant-specific evaluation was needed. A few experiments applicable to the special injection rather than cold leg injection during reflood phase were available. Among them, one UPTF test was calculated to confirm the applicability of the REALP5 code for DVI, which showed that conservative result could be obtained [14]. From the assessment, we can assure the applicability of the RELAP5 code and modeling scheme and additional bias for the reflood phenomena was not needed. Regarding MARS code capability on reflood, some experiments were predicted well but some discrepancies was still found [13]. Thus, the improvement of the reflood model of the MARS code is ongoing.

3.3 BE Plant Calculation

It is important to model the location of ECCS injection and reactor vessel configuration since the plant response can be strongly influenced by the amount of net flow rate to the core considering the ECC bypass. Especially, it was found that large amount of water is in the upper head of reactor vessel of APR1400 and OPR1000, which may cause the top-down quenching when the water in upper head flows into the core through the upper guide structure and the guide tube after pressure of reactor coolant system decreases to a certain level during blowdown. Thus, modeling of the flow path between upper head and core was very important to simulate the behavior. The result indicated that most of water at upper head drained to the core and contributed to quench the core and to reduce significantly the PCT during the reflood period following a LOCA. However, such a behavior was not significant in Westinghouse 2-loop and 3-loop design.

During the reflood phase, plant response can be varied by the performance of LPSI of DVI plants, i.e., ECC bypass. This phenomena involves the complex combination of the inertia of injected water, driving force due to differential pressure between reactor coolant system and break, interaction by the steam generated in core, interfacial condensation, effect of hot downcomer wall, etc., Both RELAP5 and MARS codes have two-phase flow model to describe those phenomena and multiple channel modeling for the downcomer with crossflow junctions. Such a modeling scheme has been verified by MIDAS experiment simulation [13].

Regarding UPI plant, the upper plenum was modeled by two separated volumes connected with crossflow junctions. The hydraulic resistance at each flow path was adjusted to match the fraction of area and flow rate of downflow in the upper plenum which was estimated from the experimental data in various scales [11].

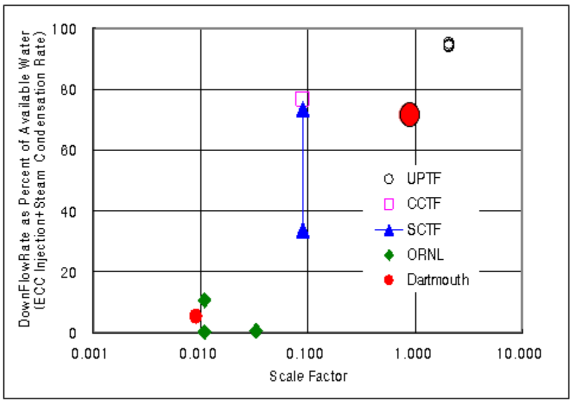

Fig. 2 Down flow fraction in UPI experiments

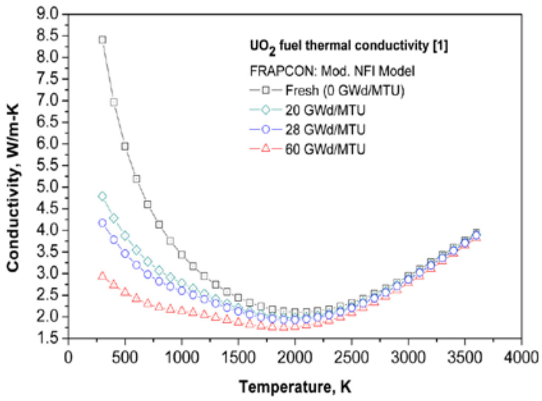

Fig. 3 Degradation of fuel conductivity

3.4 Uncertainty Parameters and Specific Approaches

Table 2 shows list of the important phenomena, phase and component of interest, parameters to be considered in the codes, method to treat, and ranges and distributions for the selected models and parameters. Some of the plant-specific parameters were determined from the plant Technical Specification.

In the KINS-REM, conservative limiting value was used for some parameters such as peaking factor, containment pressure, safety injection water temperature and flow rates to keep the conservatism instead of the effect of uncertainty, while uncertainties of other parameters were treated by statistical method. The table shows some models/parameters such as burst temperature and strain were not considered in KINS-REM, which was based on the insignificance of their impacts on PCT or difficulty in treating such parameters.

Regarding the uncertainty of the gap conductance model, a simple linear approximation of the roughness of fuel and clad as a function of linear heat generation rate was introduced and the uncertainty can be expressed through the coefficient of the linear equation [4].

Table 2. List of Cases Calculated by KINS-REM

Phenomena /sub-phenomena	Phase/ component	Models/parameters	Uncertainty Treatment	Distribution/ Range
Fuel stored energy	B/fuel	Gap conductance Peaking factor Dittus Boelter correlation Fuel conductivity Core power Burst temp/strain	Statistic Upper limit Statistic Statistic Statistic Not considered	0.4~1.5/N 0.606~1.39/N 0.847~1.153/U 0.98~1.02/N
Clad oxidation	R/fuel	Cathcart-Powell Cor.	Not considered	
Decay heat	R/fuel	ANS decay heat model	Statistic	0.934~1.066/N
Break flow	Both/break	Discharge coefficient	Statistic	0.729~1.165/N
2-ϕ pump performance	R/loop	2-ϕ head multiplier 2-ϕ torque multiplier	Statistic Statistic	0.0~1.0/U 0.0~1.0/U
ECC Bypass Condensation Entrainment CCFL Hot wall effect Sweepout Multi-D flow	R/ downcomer	 Wall condensation model Entrainment model CCFL model Wall-to-fluid HTC Interfacial drag model Crossflow resistance	Evaluate conservatism based on experiment data and add bias if needed	
Core heat transfer CHF Rewet Transition boiling Film boiling Nucleate boiling Reflood htc	Both /Core	 Groeneveld lookup table Rewet criteria Transition boiling HTC Bromley correlation Chen correlation Reflood htc package	 Statistic Not considered Statistic Statistic Statistic Statistic	 0.17~1.8/N 0.54~1.46/N 0.428~1.58/N 0.53~1.46/N As above
Void generation	B/core	Boiling model	Not considered	
2-ϕ frictional pressure drop	R/loop	2-ϕ friction multiplier	Not considered	
Effect of non-condensible gas		Gas transport	Not considered	
Steam Binding	U-tube, core	Dittus Boelter correlation	Bias if needed	
Containment pressure		Boundary condition	Conservative input	
UPI phenomena Pool formation local downward flow Condensation	R/Upper plenum	 Momentum equation. flow regime map, etc. Interfacial condensation	 Modeling pursuant to experiment, Bias if needed	
ECCS performance	ECCS	SIT pressure SIT temperature SIT water volume SIT flow resistance HPSI flow rates HPSI temperature	Statistic Statistic Statistic Statistic Lower limit Statistic	Lower and upper values from the plant Tech. Spec

Fuel conductivity as a function of temperature was originally taken from the MATPRO model[12]. Recently, it was noticed that the conductivity may be degraded as the fuel burnup increase, thus that the effect of conductivity degradation should be considered [15]. It indicated the initial fuel stored energy, i.e., may increase, thus, the blowdown PCT by burnup. Currently, this effect was not fully implemented into the present methodology but considered as additional bias. Further study is needed to consider the burnup effect.

Uncertainty of degradation of pump performance in two-phase state was considered for full range of head degradation multiplier (zero to one).

The APR1400 design has four safety injection tanks (SIT) having a fluidic device (FD). The role of the FD is to extend the duration of SIT discharge following a LBLOCA by separating high flow discharge through the standpipe and low flow discharge through the vortexing flow path. Since each flow rate may have a strong impact on PCT, thus their uncertainties should be considered. In the KINS-REM, two valves were used and the uncertainties of high-low flow rates were represented by K-factors at the valves. The range of K-factor was determined to cover the measured data. Fig. 4 shows a experimental data on flow rate and the calculated one with K-factors to cover the experimental data[4].

Regarding the uncertainty related to the ECC bypass, it is still difficult to represent the phenomena by the particular code models or parameters, thus, additional bias should be added if the code under-predicted the bypass ratio for the applicable experimental data. From the calculation of experiments, UPTF and MIDAS, it was found that both codes have over-predicted the ratio. Also the bypass ratio in the plant calculation was estimated by tracing the boron mass incoming from ECCS to the break, which indicated the over-prediction of the bypassed mass of ECCS (Fig.5) [4]. Thus, no further investigation of the uncertainty of ECC bypass has been made.

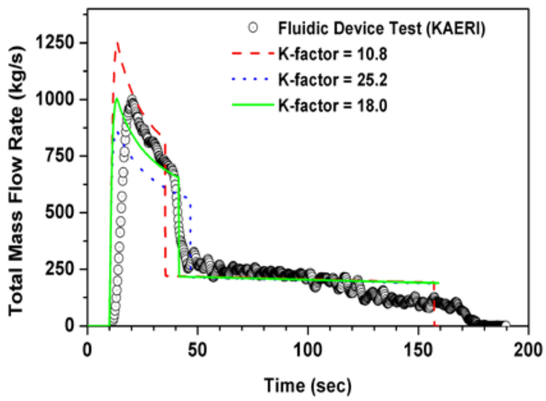

Fig.4 SIT Flow rates for VAPER experiment

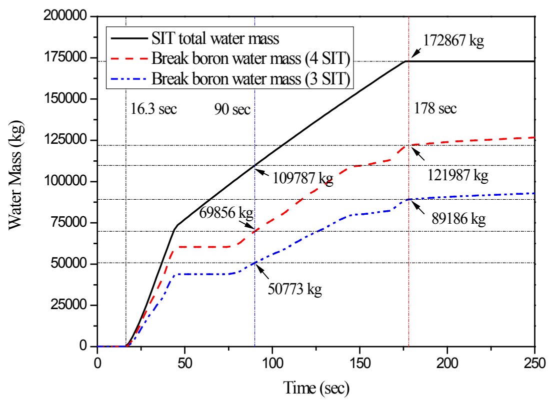

Fig.5 Calculated Mass of ECC Bypassed

Steam binding phenomena was similar to the ECC bypass. Based on the CCTF experiment data, the limiting condition due to the steam binding was the static quality to unity at the steam generator (SG) outlet plenum during the reflood phase[10]. It can be concluded the effect of steam binding was under-predicted if the code predicted it less than 1. In the KINS-REM, additional heating of SG u-tube was introduced to keep the quality of one, which led to an increase of the reflood PCT and a delay of final core quenching[4].

3.5 Results and Findings

Figures 6 to 9 show the cladding temperatures calculated from 124 code runs for 200% cold leg LBLOCA of Kori Unit 1, 40% cold leg break of Kori Unit 2, 200% cold leg break of YGN Units 3 and 4, 200% cold leg break of SKN Units 3 and 4, respectively. The 40% break for Kori Unit 2 was identified as limiting break from the break spectrum analysis. RELAP5/MOD3.3 code was used for those cases. For the SKN Unit 3 and 4, additional MARS-KS calculation was attempted as show in Fig. 9. From those results, the 3rd highest one could be obtained as PCT in 95/95 percentile.

As shown in those figures, one can find the followings:

(1) Two or three peak points of the cladding temperature responses were found for all the cases. Sometimes, the distinction of PCT's between blowdown and reflood was not clear, however, it can be judged by the condition that the core water level reaches core bottom.

(2) The dependency of blowdown PCT on the fuel stored energy (peaking factor, etc.) can be clearly observed. It means the possibility to delete some of the insignificant contributors concerning the blowdown PCT.

(3) Reflood PCT's of YGN Units 3, 4 and SKN Units 3, 4 are much lower than the blowdown PCT's, which is clearly due to the large amount of water in upper head and flow path through the guide tubes, as expected. However, to confirm the validity and uncertainty on such a blowdown quenching, further study is needed.

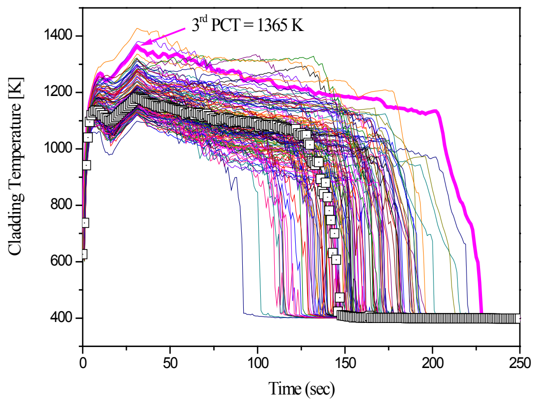

Fig.6 Clad Temperatures for Kori Unit 1

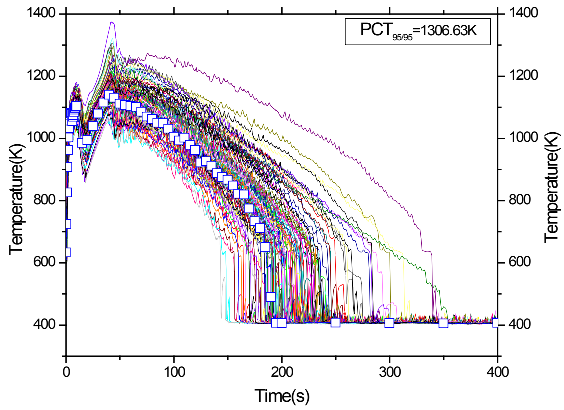

Fig.7 Clad Temperatures for Kori Unit 2

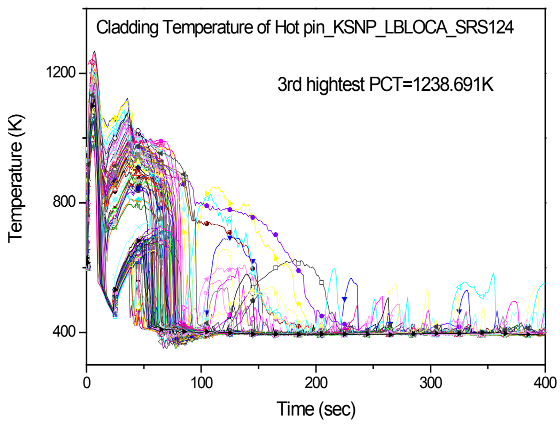

Fig.8 Clad Temperatures for YGN Unit 3 and 4

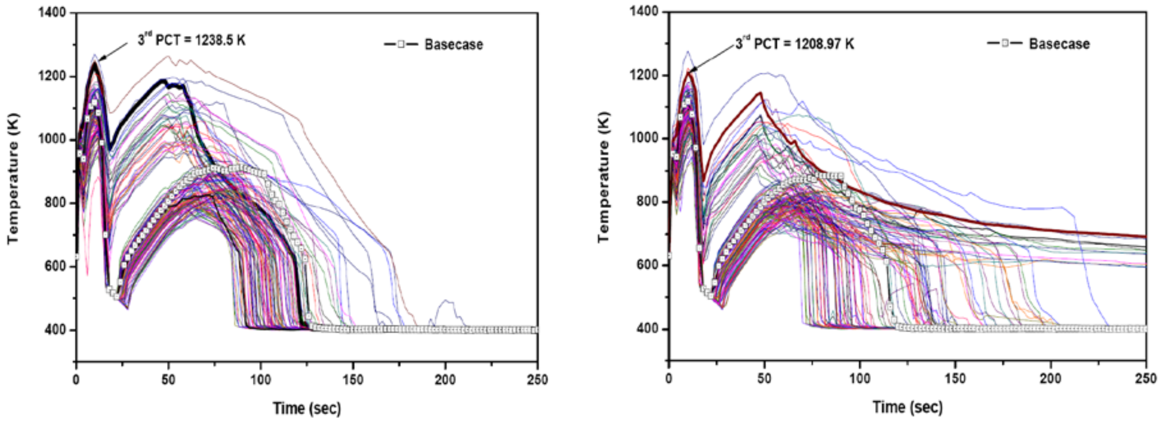

Fig.9 Clad Temperatures for SKN Unit 3, 4 (RELAP5 and MARS-KS)

(4) The significant difference in cladding temperature responses during reflood phase due to the difference of reflood model between RELAP5 code and MARS code was observed, it should be improved.

4. Concluding Remarks

In the present paper, experience of KINS on BE calculation and uncertainty evaluation of LBLOCA was addressed including various type of ECCS design. Specifically, the current status of BE code, BE calculations and uncertainty parameters and related approaches were discussed. Difficulty in treating the uncertainty related to upper plenum injection phenomena can be replaced by the conservative modeling scheme based on the experimental evidence. Based on the results and discussion, it is found that the present KINS-REM has been successfully developed and applied to the regulatory auditing calculations. However, further studies are needed for the improvement of reflood model of MARS code, uncertainty of blowdown quenching, and reconsideration of the unconcerned model and fuel conductivity degradation with burnup.

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Coupled Code Analysis of the Rod Withdrawal at Power Accident including Uncertainty Evaluation Using CIAU-TN Method

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ABSTRACT

A continuous uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal at power belongs to group of Reactivity Initiated Accidents (RIA). It will cause an increase in core heat flux and a reactor coolant temperature rise. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in departure from nucleate boiling (DNB) and/or fuel centreline melt. The accident can be DNBR or overpower limiting accident depending on initial power level and rate and amount of reactivity addition.

The Rod Withdrawal At Power (RWAP) accident was analyzed for NPP Krško to evaluate possible Resistance Temperature Detectors (RTD) bypass removal and introduction of thermowell for the average temperature measurement. The influence of different coolant temperature measurement delays to related protection system response and limiting system variables was studied first using point kinetics model as implemented in RELAP5/MOD3.3 code. The selected scenario (maximum insertion rate with rods in manual mode) has been re-calculated using RELAP5/PARCS coupled code. Core wide departure from nucleate boiling ratio (DNBR) calculation has been performed at the end of the coupled code calculation using COBRA based model to determine minimum DNBR for hot channel. In addition, FRAPTRAN based model can be used to determine behaviour of hot pin in fuel assembly experiencing peak power. The calculation included preparation of cross section data needed for neutronics part of coupled code. In order to assess available safety margins following such accident modified CIAU methodology has been applied to evaluate the uncertainty of coupled three-dimensional neutronics/thermal-hydraulics calculations.

1 INTRODUCTION

Uncontrolled RCCA bank withdrawal at power is RIA that causes an increase in core heat flux and a reactor temperature rise. Immediately after the uncontrolled RCCA withdrawal, the steam generator heat removal will lag behind the core power generation rate until the steam generator pressure setpoint of the relief or safety valves is reached. The unbalanced heat removal rate causes the reactor coolant temperature to rise. Since the RWAP is an overpower transient, the fuel temperatures rise until reactor trip occurs. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB and/or fuel centreline melt. In order to avoid core damage, the reactor protection system is designed to automatically terminate such transient before the DNB ratio falls below the limit value, or the fuel rod power density limit (kW/m) is reached. An additional concern may result due to temperature rise in the primary circuit that causes the coolant expansion into pressurizer and subsequent pressurizer level and pressure increase. In order to provide the necessary protection in the case of RWAP, the reactor protection system will actuate any of the following reactor trips: a) Power range high neutron flux, b) Power range high positive neutron flux rate, c) Overtemperature ΔT (OTΔT), d) Overpower ΔT (OPΔT), e) High pressurizer pressure, f) High pressurizer water level and g) Manual reactor trip.

For fast reactivity insertion rates, the steep nuclear power rise will result in actuation of high neutron flux or high neutron flux rate trips (signals: a and b) before core thermal flux and reactor coolant temperature rise significantly to actuate the respective trip signals (c through f). On the opposite, at low reactivity insertion rate, high neutron flux trip is not generated and nuclear power rises at a slower rate whereas the core heat flux lags slightly behind. Reactor coolant temperatures rise due to combined effects of: a) the increased core heat flux and b) the power mismatch between the core power and power transferred to the secondary side (until SG relief and/or safety valves open). The nuclear power increase rate is being reduced during the transient due to negative reactivity feedback (Doppler and coolant density). If the transient is not terminated before the core heat power and coolant temperatures significantly increase, the margin to DNBR may be compromised. In this case, the reactor protection is provided by either OTΔT trip signal aimed to protect the reactor core against low DNBR or by OPΔT trip.

The measured narrow range (NR) temperature signals are used in plant protection system (the setpoints for OPDT and OTDT reactor trip), as well as in a number of plant control systems (automatic rod control system, steam dump, pressurizer level control). NPP Krško is considering the replacing of the RTD bypass manifold system for the NR temperature measurement with fast-response thermowell-mounted (TW) RTDs. Fast acting TWs are embedded as part of a primary loop pipe wall and its response time is slower, due to thermal inertia of the additional metal mass attached to the RTD, than the response time of the directly immersed RTDs. On the other hand, for TWs, there is no delay due to loop transport or thermal lag. The major safety concern for the analyzed transient is that by replacing RTD bypass manifold system with TWs, the NR coolant temperature measurement response time increase may decrease the margin to DNB by increasing the time delay for OTΔT and OPΔT trip. The influence of the NR temperature delay on the results of the RWAP transient for various reactivity insertion rates for both RTD bypass and TW RTD configuration has been investigated, as well as related uncertainty for selected reference case.

2 RELAP5/MOD3.3 MODEL OF NPP KRŠKO

The RELAP5/MOD3.3 nodalization for NPP Krško, Figure 1, with explicit RTD bypass model has been developed at Faculty of Electrical Engineering and Computing (FER), [1], [2], [3] and [4]. The RELAP5 model has 521 thermal-hydraulic nodes, 553 junctions, 430 heat structures (with 2419 mesh points), and 684 control variables, with 199 variable and 208 logical trips, respectively. The model contains the models of the NPP Krško monitoring as well as protection and control systems, e.g., the detailed models of Safety Injection (SI) system, MFW and AFW system as well as control systems (automatic rod control, pressurizer pressure and level control, steam dump control and steam generator level control). The developed nodalization with RTD bypass lines included has enabled the accurate representation of current configuration for Narrow Range (NR) temperature measurement at the plant. For TW RTDs configuration, the measured RCS temperatures in the respective hot and cold legs are modeled with lag control component with appropriate time delay.

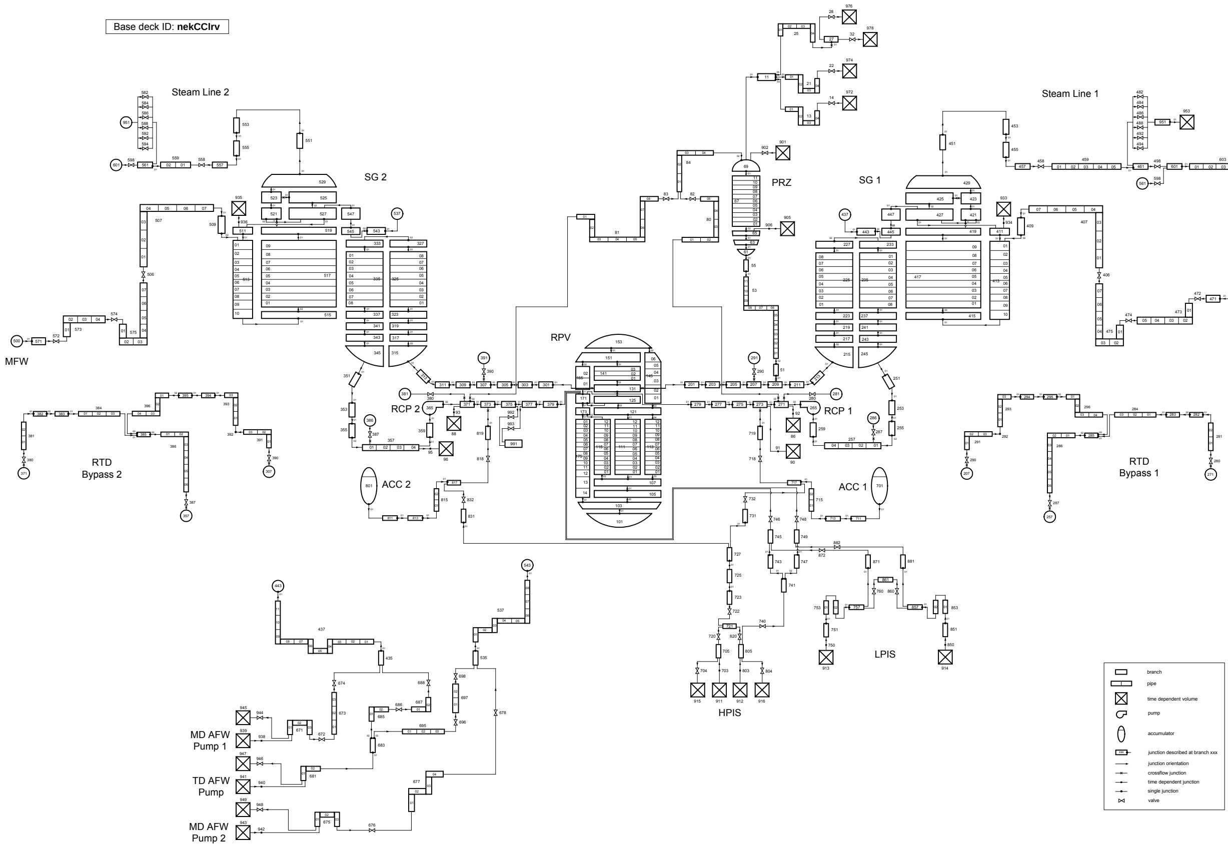

Figure 1: RELAP5 nodalization of NPP Krško with RTD bypass included

3 RELAP5/MOD3.3 ANALYSIS

Depending on initial and boundary conditions as well as reactivity insertion rates, the RCS temperature measurement delay can affect the minimum DNBR, when reactor trip is due to OTΔT reactor trip signal. The sensitivity analyses for RWAP at NPP Krško using RELAP5/MOD3.3 for different RCS temperature measurement delays were evaluated. Since RELAP5 is 1D code, it does not have the capability to calculate 3D parameter like the minimum local DNBR. Therefore, the RELAP5 analyses were performed to make an assessment of the influence of RCS temperature measurement delay for various reactivity insertion rates on physical parameters affecting the DNBR (e.g., core heat power, RCS average temperature and ΔT, fuel cladding temperature). Thereby, the initial and boundary conditions resulting in most adverse parameters influencing the DNBR could be identified. The analyses were performed for Beginning Of Life (BOL) cycle 24, and nominal power (nuclear power=1994 MW). Best-estimate initial and boundary conditions as well as availability of control systems with exception of the automatic rod control system (not credited) were assumed in analyses. Both RTD bypass and TW RTD RCS temperature measurement systems were addressed in the analyses. The base case analysis was for RTD bypass configuration representing the current RCS temperature measurement at NPP Krško, Figure 1. The analyses with an additional RCS temperature measurement delay (4, 6 and 8 sec) were performed only to make an assessment of the influence on RWAP outcome. For TW RTD measurement configuration, the measurement delay imposed to the calculated loop temperatures was 2 sec (base case), whereas in the sensitivity analyses additional delay times have been assumed (4 and 6 sec).

Three groups of RWAP analyses for both RTD bypass and TW RTD measurement configurations have been performed:

1) Reactivity insertion rate = 2.4 pcm/sec, delay = 0 sec (base case), 4, 6 and 8 sec
2) RCS temperature measurement without delay at reactivity insertion rates 2.4 pcm/sec (base case), 1, 10, 20, 40 and 80 pcm/sec.
3) Max. additional delay for RCS temperature measurement = 8 sec, at reactivity insertion rates 2.4, 1, 10, 20, 40 and 80 pcm/sec.

The base case assumed 100 % power, BOL and reactivity insertion rate of 2.4 pcm/sec. Since the Rod control system was assumed unavailable, the only negative reactivity is due to moderator and Doppler feedback. The total reactivity is positive and nuclear power as well as core power increase. This, together with the fact that the steam flow to the turbine is limited to its nominal value, leads to the imbalance between the heat production in the core and the heat transferred in the steam generators. The RCS temperatures rise, which in turn causes reactor coolant expansion, pressurizer insurge and RCS pressure rise. The pressurizer spray reduces the RCS pressure. Due to persistent PRZR spray flow, the pressurizer level rises. The pressurizer level control flow (CVCS charging flow) decreases and on the other side the pressurizer backup heaters are turned on (level error greater than 5 %). On the secondary side, the SG pressure rises, since there is an imbalance between the transferred power on the SG secondary side and the removed heat to the turbine. The main FW flow is following the steam mass flow as long as there is available pressure difference to obtain the desired flow. Along with the secondary side pressure rise, the feedwater pumps were not able to deliver the required flow and the decrease of SG secondary side inventory resulted. Due to secondary side pressure increase, the saturation temperature on the secondary side increases, too, and the imbalance between the primary and secondary power leads to further RCS coolant temperature increase. The RCS temperature increase is the major cause of OTDT setpoint decrease, since the pressure oscillates around its nominal value. On the other side, the RCS temperature difference rises due to increased power generation in the core. Finally, the OTDT signal trips the reactor and the turbine trip is actuated on reactor trip. Due to relatively slow reactivity insertion rate, the moderator and Doppler reactivity feedback have a significant impact on nuclear power. Thus, the maximum nuclear power was recorded before the OTDT reactor trip. The minimum DNBR was at a time of reactor trip. After reactor trip, the reduction of the coolant temperature and of the specific volume result thus causing an outsurge from the pressurizer, and the RCS pressure drop. Following the turbine trip, the steam dump (plant trip mode) valves open, whereas the SG relief and safety valves are not actuated.

4 COUPLED CODE ANALYSIS

In order to take into account possible local effects and to calculate DNBR values, coupled code based on RELAP5 mod 3.3 and PARCS 2.5 [5] was used in second part of the calculation. The coupled code uses usual explicit coupling scheme with calculation overlap within the core (the core is calculated by both system code and subchannel code). RELAP5 and coupled code share the same nodalization and assumptions except that split reactor vessel model was used in coupled code. The vessel is split in two halves to take into account influence of two loops (our standard approach for coupled code calculation) even though the expected calculation conditions are symmetric ones, both from point of view of core reactivity and loops thermal hydraulics.

The core is represented by 18 channels subdivided in 24 equidistant axial layers, Figure 2. Subchannel model uses one channel per fuel assembly and the same axial subdivision as RELAP5 part. Neutronics calculation was done with one calculation cell per fuel assembly.

Whole calculation procedure includes:

- Preparation of burnup and thermal hydraulics dependent neutron cross sections for material compositions present in the core,
- 3D depletion calculation including preparation of reactivity data for RELAP5 point kinetics and 3D burnup and history variables distributions for coupled code (coupled calculation is possible from any history or power level point),
- RELAP5 point kinetics calculation to verify assumptions and inputs and to explore main influence of changes in reactor coolant temperature measurements and time delays,
- Coupled code steady state calculation to determine boron concentration and initial RCCAs positions,
- Coupled code calculation for different RWAP withdrawal rates (state points for hot fuel assemblies are saved),
- Hot channel DNBR calculation for selected state points (state point includes all interface data needed to perform more detailed calculation from selected limiting point identified in coupled code run).

Coupled cases were initially performed only for original RTD bypass configuration for RWAP at full power. They were used to demonstrate calculation procedure and to test additional assumptions needed as compared to earlier performed point kinetics calculation. At full power D bank is only control rod bank that can be inserted into the core. Following bank D withdrawal cases were calculated:

- Case 1, withdrawal with speed of 39 steps/min (intended to be similar in effects to point kinetics 2.4 pcm/s case),
- Case 2, withdrawal with speed of 48 steps/min (max withdrawal rate when rods are in manual mode),
- Case 3, withdrawal with speed of 72 steps/min (max withdrawal rate when rods are in auto mode),
- Case 4a, to complete withdrawal within 10 s, with high neutron flux trip disabled,
- Case 4b, to complete withdrawal within 10 s, with high neutron flux trip enabled,
- Case 5a, to complete withdrawal within 2.35 s, with high neutron flux trip disabled (intended to be similar to point kinetics 80 pcm/s case, situation has no practical value as HFP case),
- Case 5b, to complete withdrawal within 2.35 s, with high neutron flux trip enabled
- Case 6, withdrawal with speed of 8 steps/min (min withdrawal rate when rods are in auto mode).

Direct comparison with reactivity insertion rates used in point kinetics calculation is not always possible. The reactivity worth of inserted D bank estimated in steady state was around 160 pcm. It is possible to determine time used for withdrawal so to reproduce required insertion rate on average, but due to changed feedbacks during transient calculation inserted reactivities were around 140 pcm or less (when interrupted by scram signal, bank being withdrawn started the fall into the core). Case 1 was terminated by OTDT trip. Cases 2, 3, 4b and 5b were terminated by OPDT trip. Cases 4a and 5a were terminated by high neutron flux in power range and case 6 has stabilized power increase without trip. Obtained nuclear power responses for different cases are shown in Figure 3.

Rather effective thermal hydraulic feedback is able to limit power increase especially for slow reactivity insertion rates. Corresponding core wide minimum DNBR values are shown in Figure 4. DNBR is calculated for each fuel assembly for average rod/channel configuration in COBRA. Only small decrease is present and it is terminated by reactor scram. All minimum DNBR values were above 3.

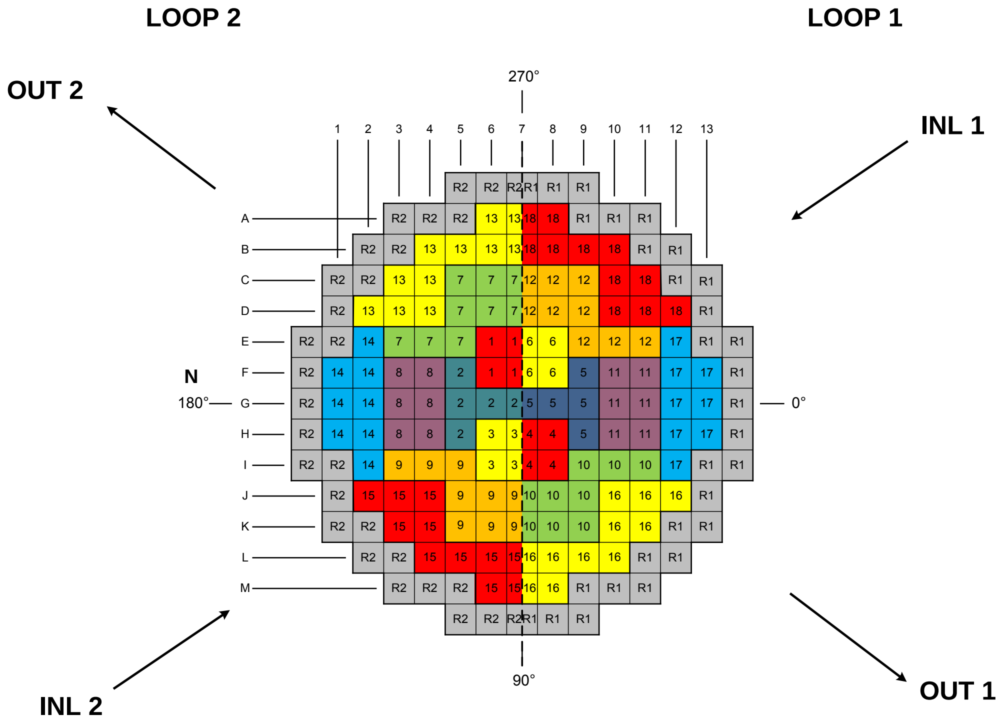

Figure 2: RELAP5 channels distribution in coupled code

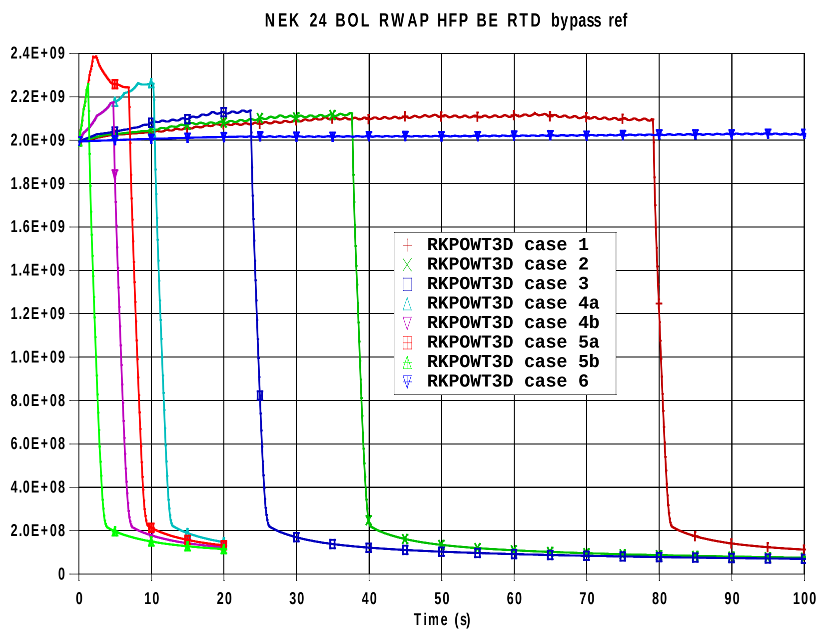

Figure 3: Nuclear power for different D bank withdrawal rates

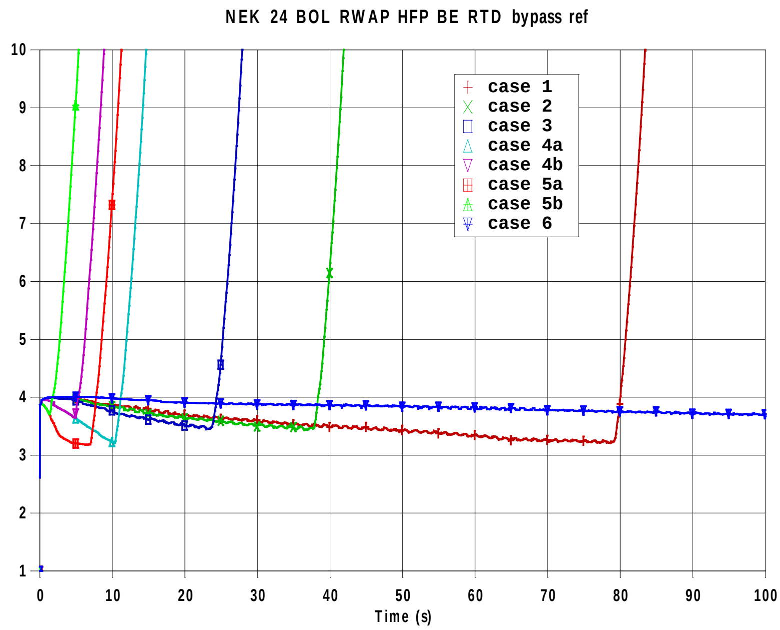

Figure 4: Core wide average rod minimum DNBR for different D bank withdrawal rates

5 UNCERTAINTY ANALYSIS OF THE RWAP CASE USING CIAU-TN

CIAU-TN [6] is the extension of CIAU method for the uncertainty evaluation of the 3D Thermal-Hydraulics/Neutron-Kinetics coupled code calculations. The capability to derive continuous uncertainty bands bounding the core power history and the axial and radial peaking factors distributions has been achieved and sample results related to the OECD/NRC PWR MSLB Benchmark [7] have been obtained. Although the full implementation and use of the procedure requires a database of errors that is not available at the moment, the results obtained gave an idea of the errors expected from the application of the present computational tool to problems of practical interest.

The status approach at the basis of CIAU methodology, implied the selection of new ‘driving’ quantities to take into account the thermal-hydraulics/neutron-kinetics feedbacks between the two codes and to characterize the regions of the phase-space (hypercubes) to which assign the uncertainty values. In order to achieve this extension, the number of ‘driving’ quantities has been increased by two units, as can be seen in Table 3. The total reactivity and the core average exposure constitute the two additional quantities able to characterize the series of subsequent statuses by which each transient scenario in NPP evolves. The ranges of variation for these variables and their subdivisions into intervals together with the new adopted subdivision for the total core power are identified in Table 3. The total number of hypercubes, resulting from the combination of all intervals of the eight variables, is increased to 311040 instead of the 8100 hypercubes used in CIAU.

In order to achieve the capability to predict the uncertainty affecting the 3D thermal-hydraulics/neutron-kinetics calculations, the uncertainty matrices and vectors listed in Table 4 had to be filled following a similar approach to the one pursued for the filling of CIAU time and quantity uncertainty vectors. The application of CIAU-TN is straightforward once these uncertainty matrices and vectors are available: each time the transient enters in one plant status, uncertainty values are picked up from the uncertainty database and continuous uncertainty bands are automatically generated and superimposed to the following three new additional ‘object’ quantities: a) Total core power history; b) Axial peaking factors distribution (Fz); c) Radial peaking factors distribution (Fx,y).

The dependence on the spatial position of the quantities b) and c) requires the development of specific tools in order to compare experimental (or reference) and calculated results to get accuracy values.

Notwithstanding CIAU-TN is able to perform the uncertainty evaluation of the local factors at each time step of the calculation, hereafter the attention will be focused only on the peaking factors distributions corresponding to the time instant (tp) when the core power peak occurs.

Table 1: Subdivision of the new driving quantities into intervals.

NEW DRIVING QUANTITIES	HYPERCUBE LIMITS					
	(1)	(2)	(3)	(4)	(5)	(6)
Total Reactivity (dk/k)	-0.40 ÷ -0.10	-0.10 ÷ -0.01	-0.01 ÷ 0.0	0.0 ÷ 0.003	0.003 ÷ 0.005	0.005 ÷ 0.007
Core Average Exposure (GWd/t)	0 ÷10	10 ÷ 20	20 ÷ 30	30 ÷ 40	-	-

Table 2: New uncertainty matrixes and vectors in CIAU-TN.

ID	DESCRIPTION	OBJECTIVE
DTUV	Detailed Time Uncertainty Vector	Uncertainty on the Time when the Power Peak occurs
DPUV	Detailed Power Uncertainty Vector	Uncertainty on the Quantity of the Core Power
APFUV	Axial Peaking Factors (Fz) Uncertainty Vector	Uncertainty on the Quantity of Fz
ZUV	Axial (Z) Position Uncertainty Vector	Uncertainty on Axial Position of the Maximum
RPFUV	Radial Peaking Factors (FR) Uncertainty Vector	Uncertainty on the Quantity of FR
XUV	Radial (X) Position Uncertainty Vector	Uncertainty on Radial (X) Position
YUV	Radial (Y) Position Uncertainty Vector	Uncertainty on Radial (Y) Position

The reference calculation for the CIAU-TN uncertainty application deals with the coupled RWAP scenario assuming withdrawal speed of 48 steps/min and disabled OPDT trip.

Consistent with the CIAU methodology, [6], global uncertainty values have been produced for the whole duration of the transient. It is important to notify that upper band of pressure remains below safety valves set value, Figure 5. Initial uncertainty of the core power is high, Figure 6, but as the OPDT has been disabled as an assumption, this does not interact with the overall scenario.

As the scenario is characterized by strong localized effects, of particular interest were uncertainties related to axial and radial power distribution at the time of maximum power and reactor trip, Figure 7 to Figure 10:

Core power:

- 23,7%

Fz distribution:

- time of maximum power: uncertainty value of about 24.6% ;
- time of OTDT trip: uncertainty value of about 29.5%;

Fxy distribution:

- time of maximum power: uncertainty value of about 69.6%;
- time of OTDT trip: average value of about 72.9%.

Notwithstanding the full implementation and use of the procedure requires a database of errors not available at the moment, the results obtained give an idea of the errors expected from the application of present computational tool to the problems of practical interest. It must be stated that largest uncertainties have been observed in the lower part of the core where highest peaking factors have been calculated. The results still indicate robustness to the localized effects. However it should be indicated that CIAU-TN hypercubes are not fully qualified for all core states so the discussed results are applicable only for qualitative discussion of uncertainty impact on the final outcome.

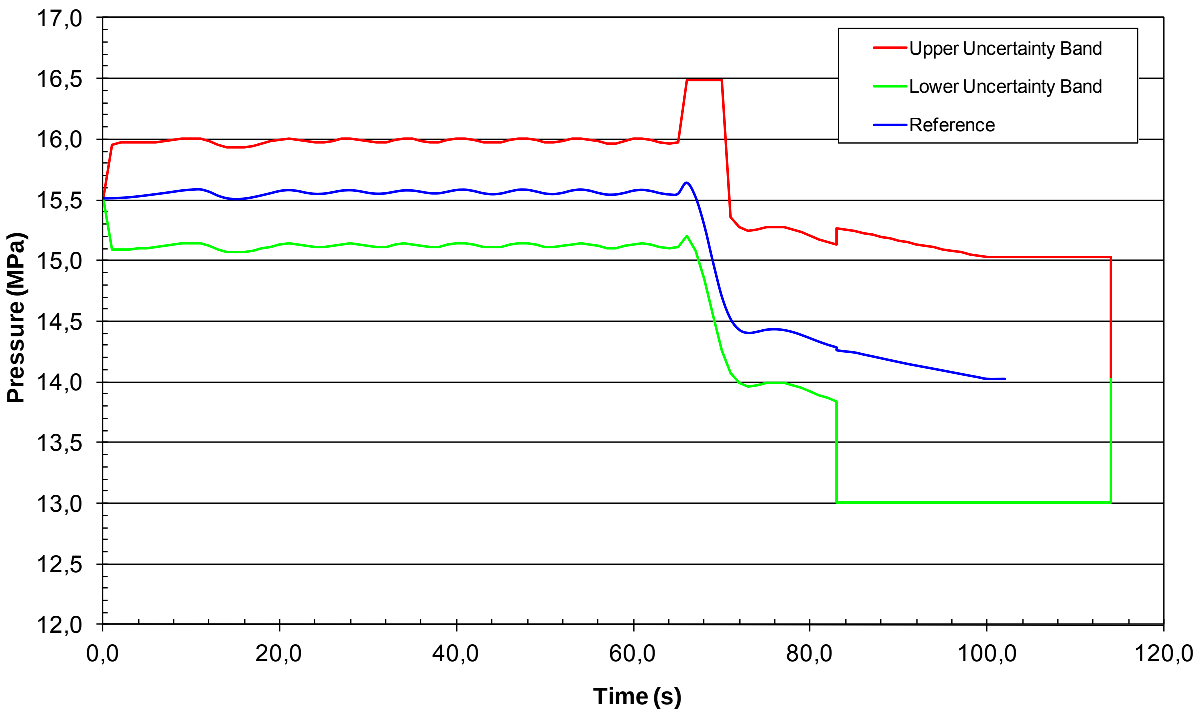

Figure 5: Primary pressure - derived uncertainty bands

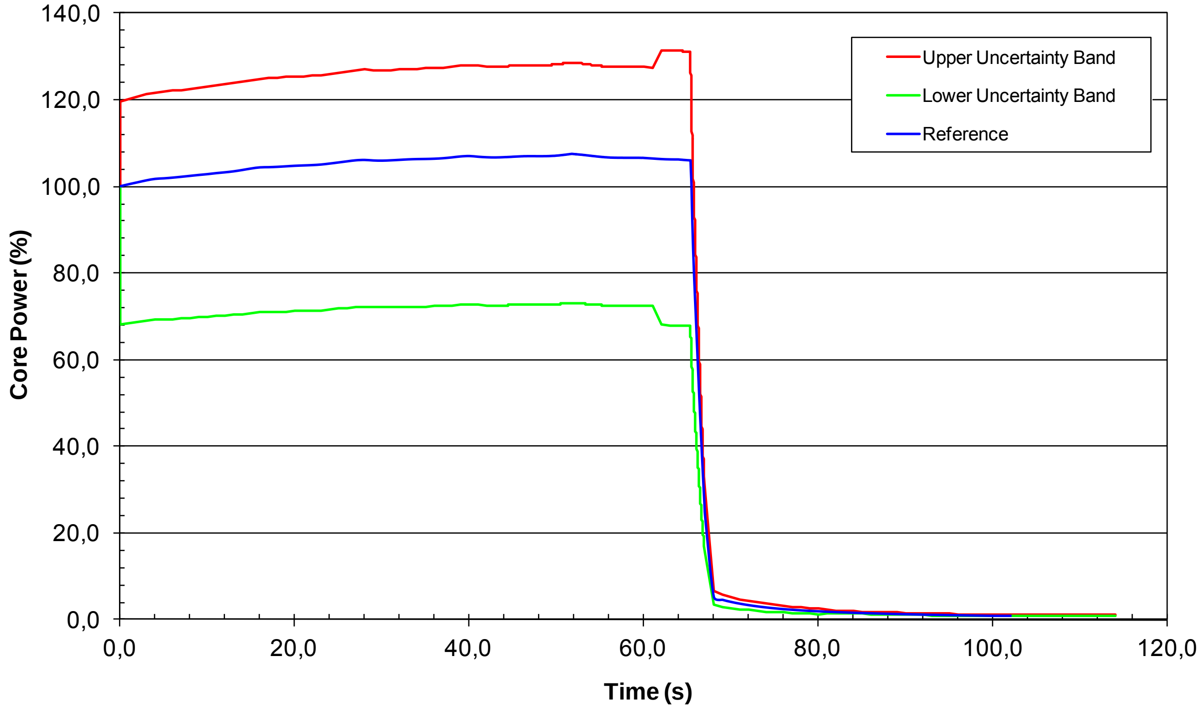

Figure 6: Core power - derived uncertainty bands

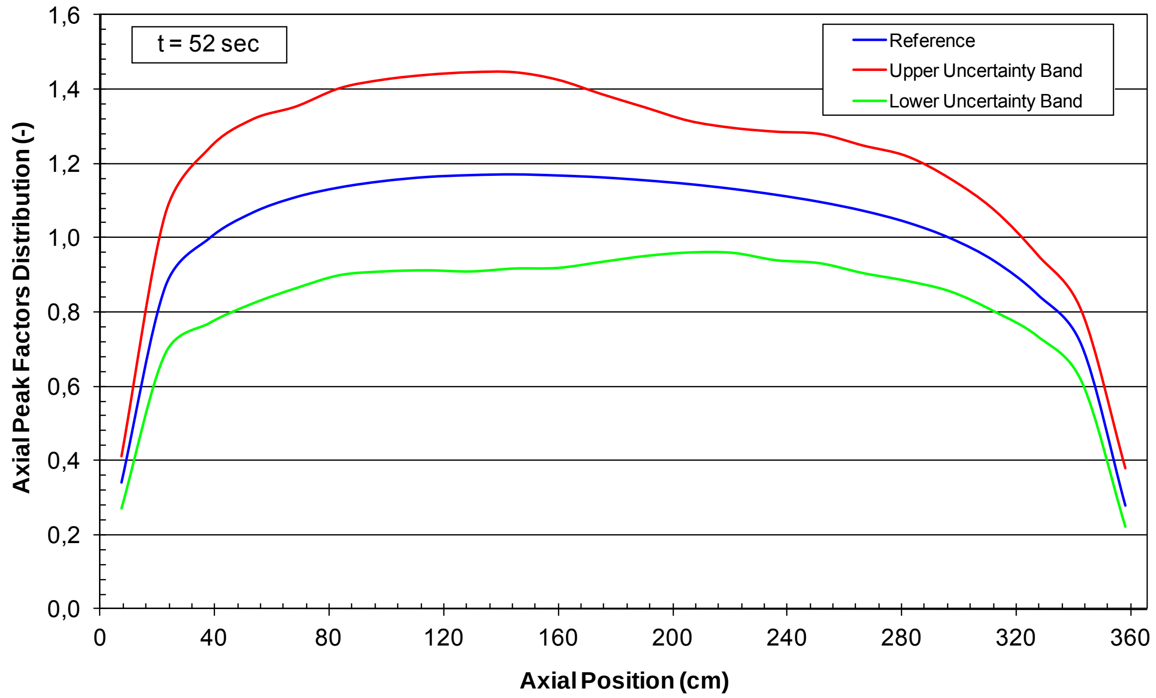

Figure 7: Axial pear power distribution at time of maximum power - derived uncertainty bands

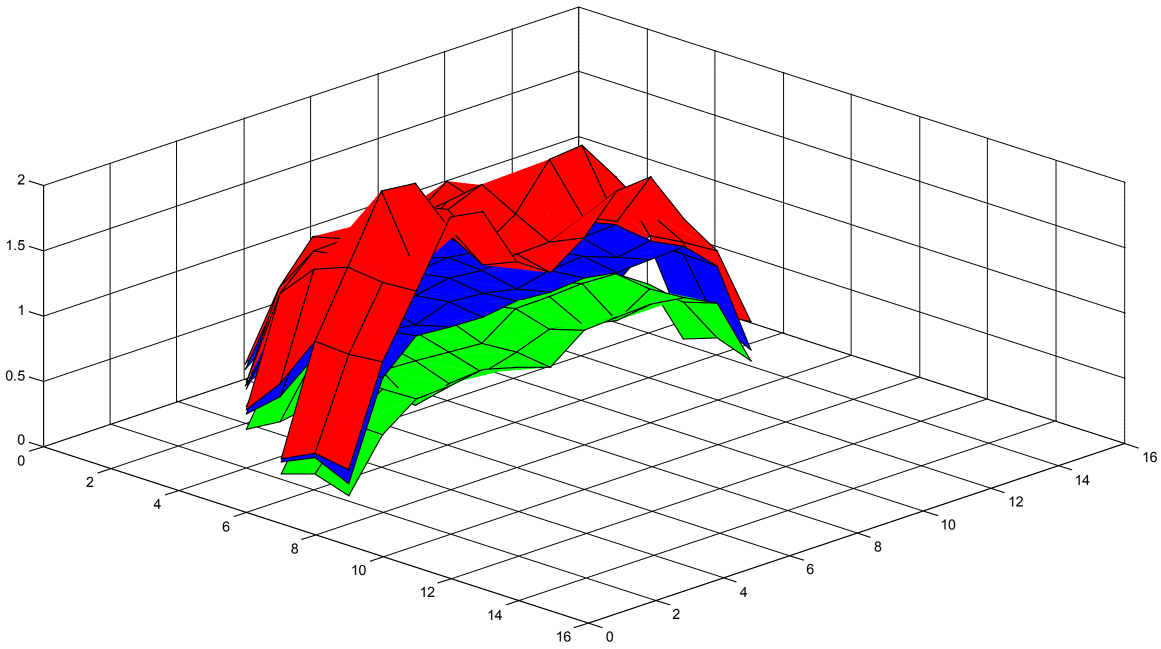

Figure 8: Radial peak power distribution at time of maximum power - derived uncertainty bands

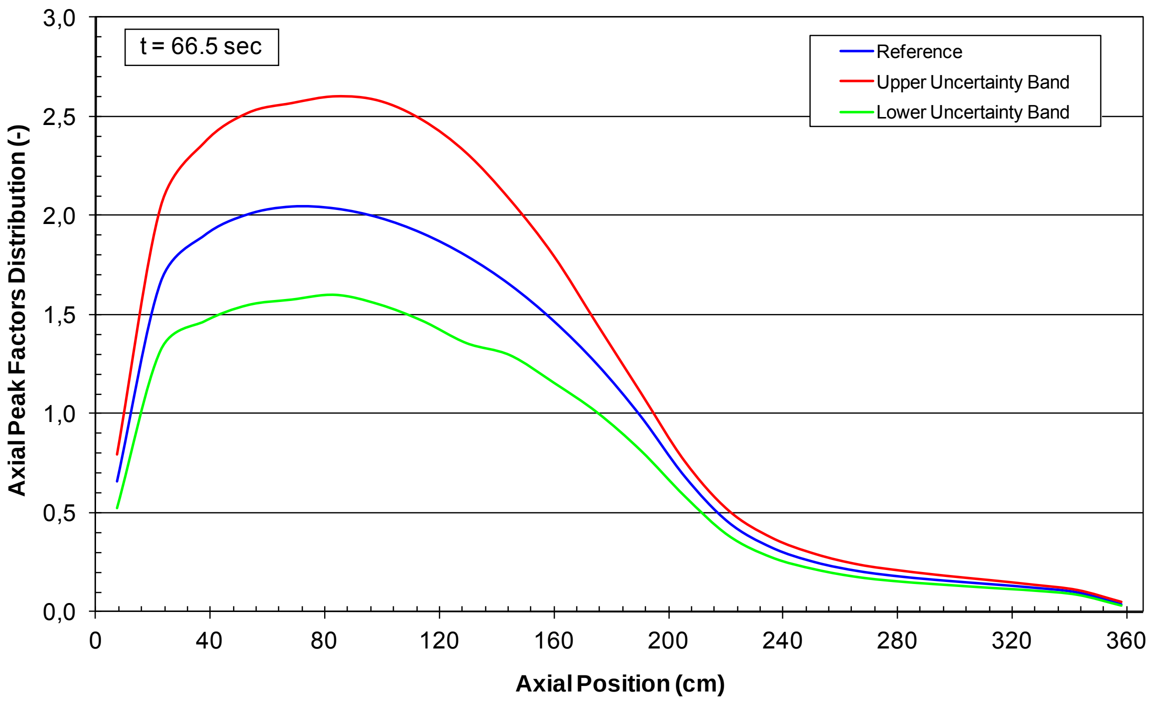

Figure 9: Axial peak power distribution after reactor trip (rods partially inserted) - derived uncertainty bands

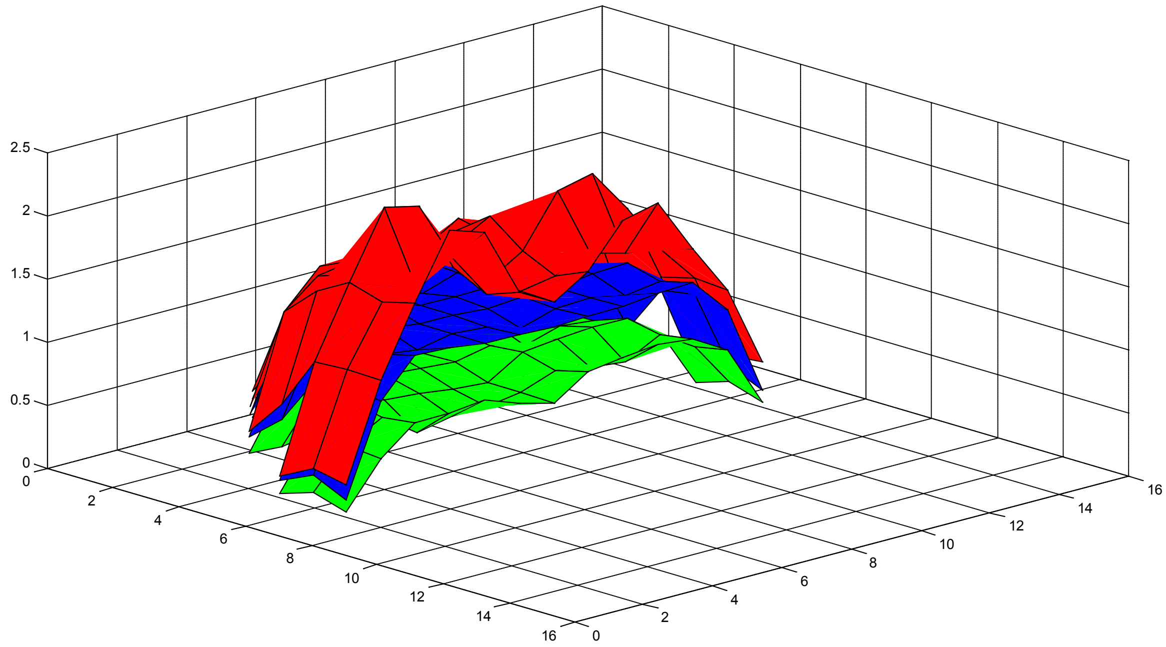

Figure 10: Radial peak power distribution after reactor trip (rods partially inserted) - derived uncertainty bands

6 CONCLUSION

Both point kinetics and coupled models were successfully tested and prepared for calculation. Only small influence of change in coolant temperature measurements and applicable delays was found in OPDT and OTDT protection setpoints. Based on performed analyses it can be concluded that time delays expected for RTD thermowells should not affect system response. Only small increases in peaking factors are present for withdrawal from full power and corresponding increase of fuel temperature is mild for the reference case. When the uncertainties calculated using CIAU-TN method are considered, the results remain within acceptable limits.

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Uncertainty Analysis for LBLOCA of Wolsung Unit-I Reactor using MARS Code

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Abstract

An application of the methodology of the best estimate (BE) calculation plus uncertainties has been done to estimate the number of rods of fission product release (NFR) which determines the amount of the fission product released for a large break loss-of-coolant accident (LBLOCA) in CANDU plants. For the purpose of the estimation MARS code, which has some CANDU specific models as BE code, has been applied to calculate system thermal-hydraulic response during the LOCA and transient state. In the modeling and simulation of CANDU NPP, 547 volumes, 580 junctions, and 508 heat structures were modeled, Assumptions in the modeling were that break occur at RIH and the size is 35% of the X-sectional area according to the FSAR of Wolsung NPP. As a result of single calculation, thermal-hydraulic response for 84 rods could be obtained. As a criterion on the estimation, maximum sheath temperature was used, and the parameters related to the critical flow at the break and other important heat-transfer related parameters were considered as well. The sheath temperature for 124 X 84 rods was traced to find the 3rd highest one by the uncertainty calculation. The 3rd highest NFR was found 1,554 in 95 percentile through investigating the distribution of the NFR.

1. Introduction

Till the end of 2011, there are 22 nuclear power plants in commercial operation in Republic of Korea which includes 18 PWRs and 4 PHWRs. The four units at the Wolsung site are 700 MWe class CANDU 6 pressurized heavy water reactors (PHWR), designed by Atomic Energy of Canada Limited (AECL)[1]. Wolsung Unit 1 which went into service in 1983, got refurbishment process started in 2009[2]. Pressure tube and calandria tube replacement is the main target of refurbishment.[3] After the refurbishment, the reactor will be able to operate for another 25 years. There are also some design improvements such as Emergency Core Cooling System, set points reset etc. were conducted. The FSAR (Final Safety Analysis Report) reflect re-analysis of accidents as well as implementation of new reactor core and design changes. Thus, there is a need to perform regulatory auditing calculations to evaluate the validity of new core and design changes

A loss of coolant by the rupture of inlet or outlet header in the heat transport system would result in rapid depressurization and voiding of the coolant in the core. Voiding in the core channel causes an increase in reactivity. This results in a rapid increase in power until one or both of the safety shutdown systems trip and shut down the reactor. Due to increasing power and decreasing coolant flow, fuel temperatures rise and fuel failures may result leading to fission product release.

Analysis is performed using MARS code to check if acceptance criteria are met for LBLOCA (Large Break Loss of Coolant Accident) with break size equal to 35% of inlet header flow area. Coolant from accumulator gets injected into the system once the pressure in the header falls below the set point. Then coolant from the medium pressure and low pressure ECCS (Emergency Core Cooling System) gets injected into the core. Due to degraded heat transfer in the core, the clad surface temperature exceeds melting point and may lead to fuel failure by clad coolant interaction.

Uncertainty analysis was carried out for several important parameters. There are a large number of uncertain parameters. But only ten parameters which related to heat transfer characteristics of nuclear fuel, break size and development time, initial oxide thickness, RCP's two phase performance, discharge coefficient and thermal non-equilibrium constant related to Henry-Fauske critical flow model are considered. Based on 3rd order Wilks assumption, for 95% fractile and 95% confidence, 124 numbers of configurations of sensitivity parameters are selected based on LHS (Latin Hypercube Sampling). Then 124 input decks were generated based on the selected parameters. After, performing 124 code runs, the third highest peak clad temperature (PCT) was obtained. Also the number of fuel rods failed(NFR) in all the cases is estimated based on the acceptance criteria.

2. Wolsung-I Nuclear Power Plant

Wolsung-1 622MWe PHWR (CANDU 6 Type) consists of the reactor core, Primary Heat Transport (PHT) system, secondary system, reactivity control system, reactor protection system, moderator system and reactor components such as calandria, end shield, dump tank, etc.

The reactor is heavy water moderated, pressurized heavy water cooled, natural uranium fuelled and horizontal pressure tube type reactor. Pressure tubes (PT) are located within the 380-calandria tubes (CT) and are supported by the end -shields. Each channel consists of 12 fuel bundles. There are 37 fuel rods in each bundle. The clad, PT and CT are made of zircaloy. The calandria tubes are separated from pressure tube by an annulus, which contain CO2. In the calandria vessel, heavy water is contained and acts as both moderator and reflector.

The PHT system consists of two loops. Each loop consists of two reactor inlet headers and two reactor outlet headers, 190 inlet and outlet feeders, two primary circulating pumps and discharge lines, primary side of steam generator (SG). The PHT system pressure is controlled by feed-bleed system, pressurizer system and safety valves. The two outlet headers of each loop are interconnected. Feed is to the suction of one pump in each loop and bleed is from discharge of same pumps in each loop. The pressurizer located on the outlet headers of each loop.

Secondary system consists of main and auxiliary feed water system and steam generators. The secondary system also consists of steam transfer piping, Steam Relief Valves (SRV), Atmospheric Steam Discharge Valves (ASDV), Condenser Steam Discharge Valves (CSDV), main steam headers, Main Steam Safety Valve (MSSV), turbine control system etc.

There are three types of reactivity control rods(coarse rods, fine rods, shut down rods) are equipped for the control or safe shutdown of reactor. The material of all the rods is Cd/H2O.

The ECCS consists of accumulator (high pressure injection), low and medium pressure injection systems and connected via valves to all the inlet and outlet headers of PHT.

3. MARS Code

The MARS code[4,5] is being developed by KAERI (Korea Atomic Energy Research Institute) for a multi-dimensional and multi-purpose realistic thermal-hydraulic system analysis of light water reactor transients. The backbones of MARS are the RELAP5/MOD3.2 and COBRA-TF codes that provide bases of System Analysis and 3-D Vessel Analysis modules of MARS respectively. The RELAP5 code is a versatile and robust code based on a one-dimensional two-fluid model for two-phase flows. The COBRA-TF code employs a three-dimensional, two-fluid, three-field model. Basically MARS code was developed for the simulation of PWR type vertical fuel channel models, recently some features of RELAP/CANDU like the horizontal core channels (CANCHAN) are developed and are used in the analysis of safety of CANDU reactors.

4. System Representation

The primary circuit, secondary circuit and ECC system are simulated using MARS. The total number of control volumes used in the model is 546 and total number junctions used are 574. The break which simulate LOCA is modelled as a servo valve and connects inlet header (IHD8) to containment atmosphere which is modelled as time dependent volume. The valve area signifies the break area. Henry-Fauske critical flow model is used. The time of valve opening and opening duration(assumed as linear) was modelled and their effect was included in uncertainty analysis.

4.1. Primary System

The Wolsong-1 NPP primary side consists of connecting pipes from or to the headers, core channels, steam generators(inlet/outlet plenum and u-tubes), pumps, pressurizer, feed-bleed system and degasser condenser system etc. The nodalisation used to MARS models are shown in Figures 1 and 2. In this study, full core was modelled as 3 quarter core channel group and 7 1/7 interest channel group. The numbers of channels and power distribution are listed in table 1.

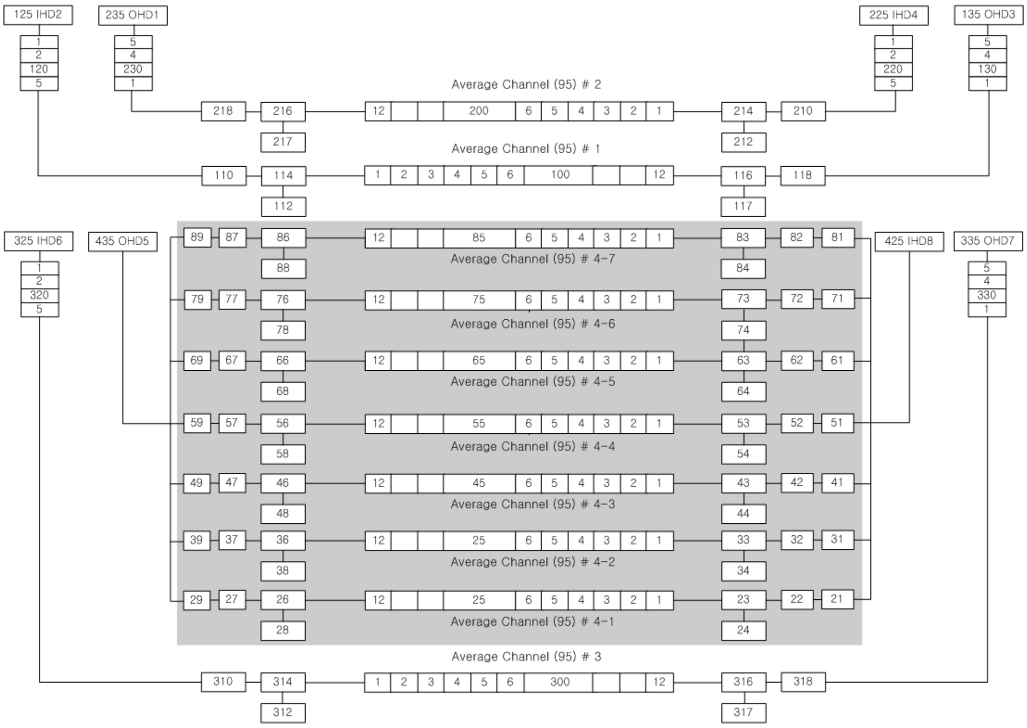

Figure 1. Primary System Nodalisation(Core and under Header Piping)

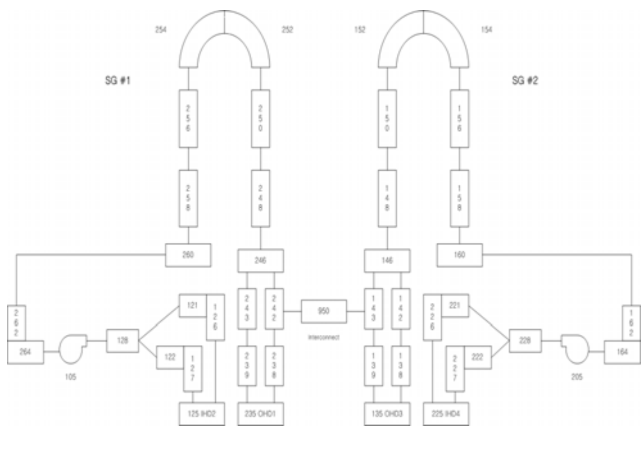

Figure 2. Primary System Nodalisation
(LOOP 1 and over Header Piping)

Table 1. Channel Group’s Conditions

Loop	Sub Channel No.	No. of Channels	Power (MWth)
1	1	95	528.298
1	2	95	527.439
2	3	95	527.450
2	4-1	12	77.427
2	4-2	12	78.266
2	4-3	11	72.806
2	4-4	14	89.560
2	4-5	16	70.499
2	4-6	15	74.168
2	4-7	15	65.584
Total		380	2111.497

As in figure 2, each header is modelled as branch component. The inlet header is connected to inlet feeders by junctions of the BRANCH. LOCA break is modelled on header 8. Inlet/outlet feeder was modelled as pipe component. The inlet feeders are connected to the end fittings by single junction. Each of the end fitting is split into two branches and one single volume component.

As in figure 1, each channel is modelled as a CANDU Channel component (CANCHAN) of twelve volumes. The cross-sectional flow area was calculated by subtracting the cross-sectional area of the fuel pins from the area of the channel. As in table 1, in loop 1, the 95 channels are clubbed into single CANCHAN component. In loop 2, in one pass the 95 channels are clubbed and in the other pass the 95 channels are clubbed into 7 CANCHAN components.

The fuel pins are simulated as heat structure components having 12 axial heat structures and 6 radial node points. 12 axial node represents horizontally positioned 12 fuel assembly. In each channel, all the fuel pins are grouped as single fuel rod. Symmetric boundary condition is provided for the fuel center. The right boundary of each heat structure is linked to the corresponding hydrodynamic volume of channel. The thermo-physical data for each material used in the pin is provided in tabular format as a function of temperature. Also the power of the fuel pin is provided in tabular format. The axial distribution of power in the fuel bundles are cosine shaped[10].

In each loop, the above header piping is modelled as seven PIPE components and the steam generator inlet and outlet plenum are modelled as SNGLVOL components. The primary pumps are modeled as pump components in MARS.

4.2. Secondary System

Each steam generator is modelled as three vertical SNGLVOL components capped by a separator component (Figure 3). A single volume acting as the top of steam drum is connected to the top of the separator. The SG Level control programme is also modelled. The feed water supply and condenser for SG are modelled as time dependent volume (TDV) with temperature and pressure as boundary condition respectively. The secondary system valves such as MSSV, ASDV, SRV, CSDV and the governor valve are modelled as valve components(servo, motor and trip valve).

4.3 ECCS

The ECCS consists of accumulator (acts as a high pressure injection), low and medium pressure injection systems. The ECCS is connected via trip valves to all the eight headers. The nodalisation scheme for the ECC system is shown in Figure 4.

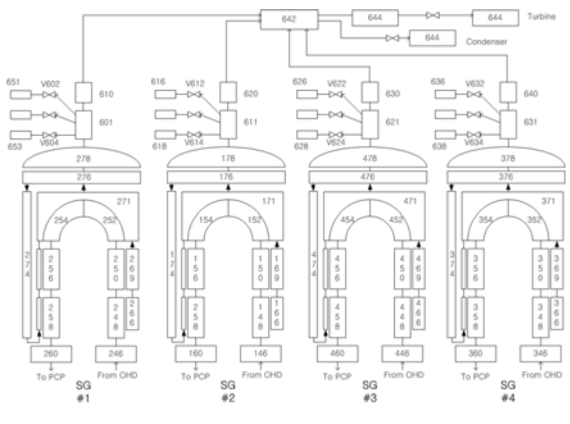

Figure 3. Secondary System Nodalisation

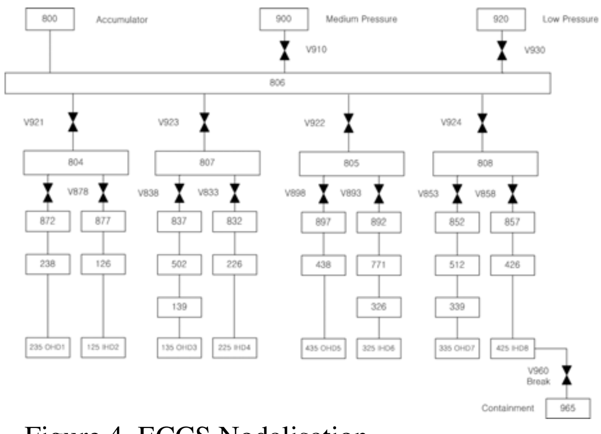

Figure 4. ECCS Nodalisation

5. Reference Running

5.1. Steady State

Using the system nodalisation and proper initial conditions, the steady state was achieved after a code run of hundreds of seconds in workstation. A fixed pressure of 9.89 MPa as in reference 22 is considered in the pressurizer for achieving steady state. The modelled steam generator level control system controls the level of SG to the actual SG Level using turbine back pressure, feed water flow and steam flow. After the steady state was achieved, all of main system parameters such as pressure, flowrates, temperatures are keep constant after about 200 seconds.

Among them, pressurizer pressure and main feedwater flows are shown in figure 5 and 6. With these system parameters, most of major conditions of the steady state are acceptably within the actual conditions of normal operation[10].

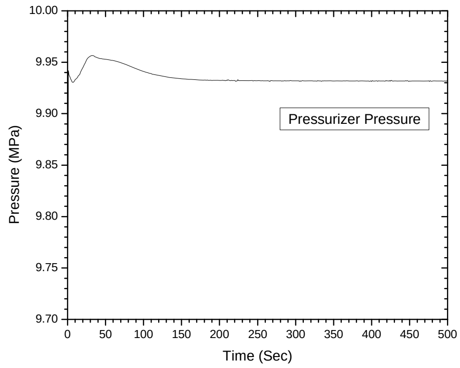

Figure 5. Steady State Pressurizer Pressure

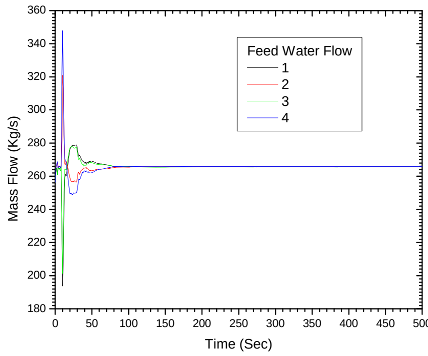

Figure 6. Steady State Main Feedwater Flow

5.2. LOCA Initiation

For the simulation of transient, this component along with the connected trip valve is deleted(using DELETE command). After reactor trips, the reactor power initially increase due to positive void coefficient of reactivity and then gradually decreases to decay power level after reactor trip. In CANDU design approach, each channel is separated, reactor kinetics in MARS or RELAP cannot be applicable and the reactor power is specified as a function of time in MARS table. The normalized reactor power for each nodalised channel (3 quarter core channel and 7 sub channels) is plotted in Figure 7. Different power transient data is provided for intact loop and broken loop due to difference in voiding.

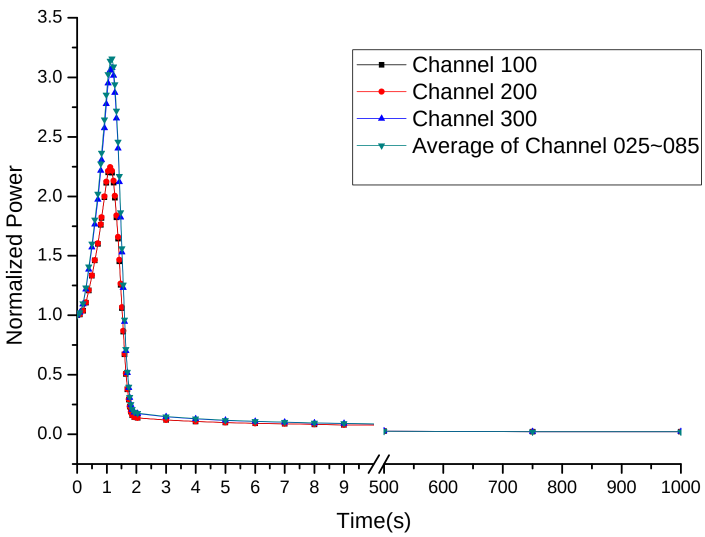

Figure 7. Normalized Channel Power during LOCA

5.3. Assumptions

The following are the assumptions made in the analysis.

a) The steady state power is 103% of the full power under normal operation.

b) The reactor trips immediately after initiation of LOCA.

c) When the value Max ,Min (OHD), Min (IHD) < 5.25MPa LOCA signal is generated.
d) Feed-bleed system and the pressurizer system are isolated after LOCA signal.
e) The PHT pumps trips after 5 sec after LOCA signal.
f) The governor valve closes and feed water stops after 5 sec from LOCA signal.
g) MSSVs are opened after 30 sec from LOCA signal.
h) Aux. feed water pump starts 3 minutes after main feed water pump trip.
i) Reactor regulating system is assumed froze.
j) High Pressure ECC Injection Signal is generated when Max ,Min (OHD), Min (IHD) < 3.62 MPa.
k) The intact loop is isolated when volume of water in accumulator decreases to 15.5 m^3.
l) The accumulator is isolated when the volume of water in accumulator decreases to 5.47m^3.
m) The medium pressure injection starts 1.5 minutes after the start of high pressure injection and stops when an amount of 200 m^3 of coolant is injected.
n) The low pressure injection starts when the medium pressure injection stops and continues till the end of the transient.

5.4. Results

Once the break valve opens, single phase liquid discharges through the valve, changing to two-phase flow when the inlet header pressure reaches the saturation corresponding to the fluid temperature. The sequence of events are :

Table 2. Sequence of events

Time (sec)	Event	Remark
0.0	35% RIH Break Occurs	The break valve opens
	Reactor Trips	
6.21	Peak Clad Temperature	1426 ^{0}K @channel055,Node7
9.22	LOCA signal generated	Max,Min(OHD),Min(IHD)<5.25MPa
	Feed & Bleed system isolated	
	Pressurizer system isolated	
	ECCS valves open	
14.226	Turbine rolls back.	Governor valve starts closing
	Main Feed water pump stops	
	PHT Pumps trip	
29.416	High Pressure Injection Signal generated	Max,Min(OHD),Min(IHD)<3.62MPa
39.232	MSSV open	
119.43	Medium pressure injection signal generated	
194.25	Auxiliary Feed Water Pump starts	
217.007	Medium Pressure Coolant injected into core	
312.84	High pressure injection stops	
638.77	Medium Pressure Injection stops	
	Low pressure injection starts	Continues till the end of the transient.

After the break is initiated, the pressure in the broken inlet header and the downstream channels decreases rapidly followed by depressurization of the other headers. During this process, rapid voiding of the coolant occurs in the downstream channels. This early rapid voiding in the fuel channels makes positive reactivity and causes reactor power increase.

The pressure of inlet and outlet headers are shown in Figures 9~10. After the break occurs, the pressure in the header decreases continuously. The pressure decreases faster in the broken loop than the intact loop. In the broken loop, the pressure of headers in the critical pass decreases faster than the other pass. After rapid pressure decrease, trip of PHT pumps, the crash cool down of SG and the trip of reactor along with the break lead to decrease, the speed of pressure decrease in primary system is lower than that of early. The pressure in the primary system stabilizes to pressure of medium pressure

ECC and then to low pressure ECC. There is slow rate of pressure reduction when high pressure ECC injected into the core.

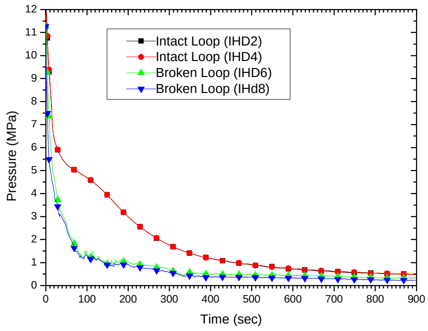

Figure 9. Inlet header pressure

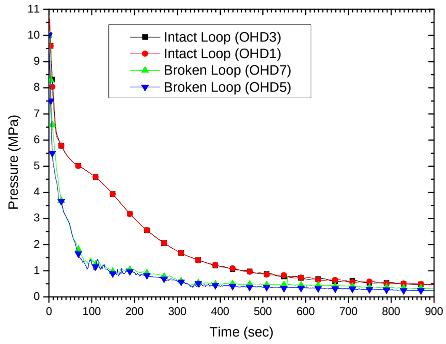

Figure 10. Outlet header pressure

As soon as the break occurs, the flow rate sharply increases to very high value due to high primary pressure. As in figure 11, sub-cooled liquid discharges through the break. Then it decreases rapidly due to rapid depressurization of primary system and due to formation of vapor in primary (two-phase flow). The time when the high pressure ECC injection starts, there is slow rate of break flow reduction and then the flow rises continuously till the high pressure injection stops.

In the broken loop, the flow in the non-critical pass decreases till the high pressure ECC injection starts. Then after it decreases slowly till medium pressure ECC starts when it increases.

The coolant temperature decreases after reactor trip. After then the PHT pumps trip, primary system flow decreases and the temperature increases by heat transfer loss to the secondary system. This heat increase in primary system continues till the high pressure coolant from accumulator which have low temperature water is injected into the core. There is a rise in temperature in the broken loop after the high pressure ECCS stops. The temperature in the intact loop is higher than the broken loop due to low depressurization. The temperature in the broken loop is lower due to depressurization of the circuit and formation of two-phase liquid in the core. Due to depressurization the coolant in the broken loop becomes two-phase.

As soon as break occurs, the quality in all the headers increases due to loss of coolant. The equilibrium quality of coolant at the inlet and outlet header is shown in Figure 12. The quality in all the headers drops sharply after cold water injection by high pressure ECCS. After the high pressure injection stops, the quality in inlet headers of broken loop increases and then decreases.

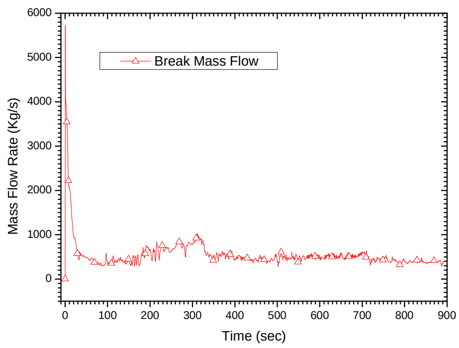

Figure 11. Break Flow Rate

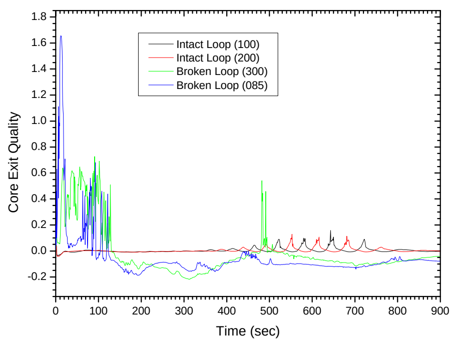

Figure 12. Outlet Header Quality

6. Uncertainty Analysis

The objective of safety analysis is to demonstrate that sufficient margins exist between the real values of important parameters and the threshold values at which the barriers against release of radioactivity would fail.[6] There are kinds of ways to define safety margins. Generally 1) either in absolute terms in relation to the expected damage to safety barriers or 2) in relation to acceptance criteria typically set up by the regulatory body was accepted.

In the current analysis, margins to acceptance criteria is considered. In a conservative approach the results are expressed in terms of a set of calculated conservative values of parameters limited by acceptance criteria where as in a BE approach the results are expressed in terms of uncertainty ranges for the calculated parameters. In the present analysis the input uncertainty is considered. There are a large number of uncertain parameters to be considered to fully represent analysis uncertainty. To make the analysis simple, only ten parameters are considered. They are listed in Table 3 with their distribution type, range.

Upper statistical tolerance limits are the upper □ confidence for the chosen □ fractile. The fractile indicates the probability content of the probability distributions of the code results. One can be □ % confident that at least □% of the combined influence of all the characterized uncertainties is below the tolerance limit. In this study 95-95(tolerance limit of the 95% percentile at a confidence level of 95%) was selected to compare with acceptance criteria. The number of code runs depends on the order of Wilks formula.[7,8] The order of Wilks' formula is expected to increase the required code runs but also to obtain a more accurate estimation of the uncertainty bounds. The Wilks formula for 2nd order was used to check uncertainty analysis.

The constrained Monte Carlo sampling scheme developed by McKay, Conover, and Beckman, is called Latin hypercube sampling(LHS).[9] LHS uses a stratified sampling scheme to improve on the coverage of the input space. In the present analysis, 124 samples were selected based on LHS and Wilks formula. Each sample consists of 10 values. The KINS in-house LHS program was used to generate uncertainty distributions of input parameters. That requires the mean value as well as the minimum and maximum value of each parameter along with their distribution type to be provided. The data provided in the input to LHS program are given in Table 3, and the output result to code run are given in Table 4.

Table 3. Data input to LHS Program

No	Variables	PDF	Min	Max
1	Initial Oxide Thickness	uniform	2.00E-06	4.50E-05
2	Cp Pellet Mult.	normal	0.9	1.1
3	Cp Clad Mult.	normal	0.9	1.1
4	K Pellet Mult.	normal	0.9	1.1
5	K Clad Mult.	normal	0.9	1.1
6	Pump Two Phase Mult.	uniform	0	1
7	Break Area Mult.	uniform	0.7	1.15
8	Break Develop Time.	uniform	0	1
9	Disc. Coef. for HF Crit. Flow	normal	0.9	1.1
10	Thermal non-Eq. const. for HF Crit. Flow	normal	0.1	0.18

Table 4. MARS input from LHS Program

No	Th. Ox.	Cp pellet	Cp clad	K pellet	K clad	PMP Mlt.	A break	T break	Cd	Cne
1	2.92E-05	0.971221	0.96745	0.994894	1.031247	0.173547	1.045828	0.358937	0.993123	0.135451
2	4.40E-05	0.962406	0.943264	0.963041	1.019421	0.411588	0.850418	0.570963	0.993239	0.155691
3	3.15E-05	0.999045	1.044174	1.015516	0.964521	0.771209	1.119055	0.901058	0.936834	0.131578
4	2.06E-05	0.996834	0.97774	1.021073	1.007909	0.181583	1.109579	0.231296	0.978049	0.16971
5	3.97E-05	1.005733	0.99653	0.955392	0.990567	0.836577	0.969681	0.959822	0.995166	0.131705
6	3.96E-05	0.97741	0.987822	0.971512	1.007233	0.736669	1.111351	0.457067	1.012038	0.144615
7	3.96E-05	1.013469	1.022497	0.991028	1.049413	0.508812	0.95764	0.902551	0.976986	0.137486
8	2.18E-05	1.040008	0.984789	1.077331	0.962522	0.781878	1.079683	0.899328	1.005041	0.134268
9	1.47E-05	1.014303	0.983103	1.014845	1.042251	0.746736	0.841023	0.806899	1.047783	0.152318
10	1.83E-05	1.041304	1.016778	0.96603	1.004636	0.945246	0.713894	0.414068	1.016961	0.14507

...	...	...	...	...	...	...	...	...	...	...
124	2.34E-05	0.999342	0.977623	1.001765	1.015436	0.285681	1.125241	0.480152	1.059111	0.137259

The output distributions of LHS program satisfies the input distribution conditions. Results of some parameters are presented as histogram in Figures 13 and 14. The x-axis represents the mid-point of the sampled intervals. It should be noted that there are 124 different values of input parameters. It is inferred from the two histograms that the two parameters are sampled reasonably.

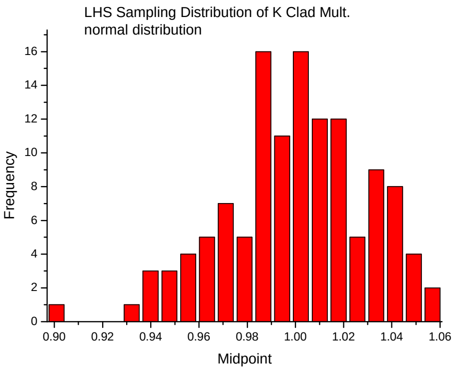

Figure 13. Input Distribution Clad K

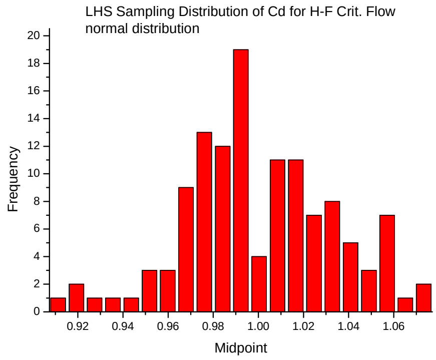

Figure 14. Input Distribution of H-F Discharge Coefficient

7. Acceptance Criteria

For the analysis of Wolsung-I LBLOCA, following criteria was adapted. Generally radiation, safety shutdown, fuel integrity, containment stability were used as a acceptance criteria of safety analysis. In this study fuel integrity was evaluated using the result of MARS. Fuel channel performance criteria was summarized as[11,12] :

1. Dynamic rupture
   i. Maximum Fuel Temperature << Fuel Melting Point (2840℃)
   ii. Maximum Clad Temperature << Fuel Clad Point (1760℃)
2. Fuel Channel Failure by Internal Overheating
   i. Pressure Tube Temperature << 600℃

Also ANL(Argon National laboratory) surveyed on the embrittlement mechanisms of Zr alloy and announced “even though all the possible mechanisms are not defined…[13] During the LOCA transient, the increase of oxygen accommodation in oxide layer by temperature increase can make beta-layer embrittlement”. If the clad temperature is higher than 1200 °C, beta-phase of Zr alloy can accommodate much oxygen. But after transient, during the temperature decrease excess oxygen makes embrittlement in the prior-beta-phase of Zr alloy.[14] In this study, as this criteria was not fully proved and did not applied in CANDU reactor, various temperature criteria was applied in the evaluation of fuel failure.

1. Maximum Clad Temperature << Fuel Clad Point (1000, 1100, 1200, 1300 ℃)

8. Results and Discussion

After 124 runs of MARS using sampled input combinations in table 4, peak centerline temperature and maximum clad temperature at every axial nodes of every channel are evaluated.

The effect of uncertainty of 10 input parameters(Table 3) are investigated. Only two parameters, fuel conductivity and break area shows distinct linear correlations(Figure 15, 16) and the other parameters have no clear correlations with maximum fuel/clad temperatures(Figure 17, 18). As in table 15, if the conductivity of fuel is low the heat transfer within the fuel was decreased and that makes the energy stored in the fuel increase. In figure 16, there are strong dependencies between break area and maximum PCT. That can be explained as the larger the break area, the more coolant to the containment.

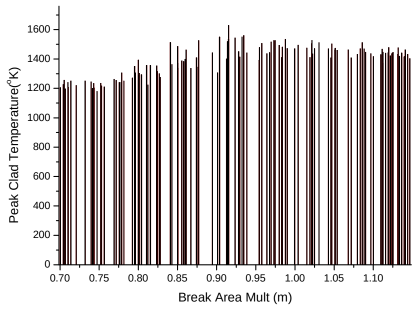

Figure 15. Effect of Break Area on PCT

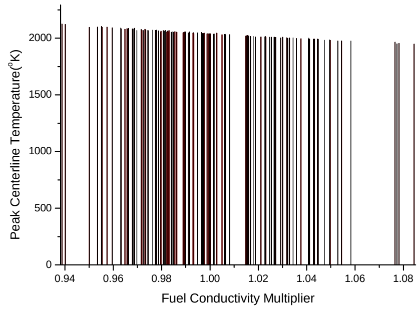

Figure 16. Effect of Fuel Conductivity on Maximum Fuel Temperature

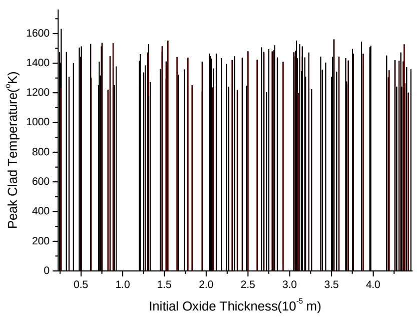

Figure 17. Typical Effect of Parameters on PCT

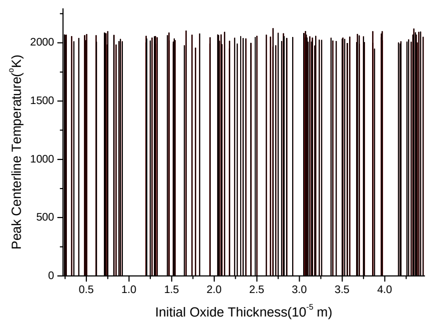

Figure 18. Effect of Fuel Conductivity on Maximum Fuel Temperature

The peak clad surface temperature (PCT) for all the 124 cases are shown in Figure 19 and the top ranked results are listed in Table 5. The highest PCT is 1544.35 °K for case 123. The third highest value is 1527.02 °K(case 52). Hence PCT95/95=1527.02 °K. This and even top case satisfies acceptance criteria on PCT. There are no melting in the clad material during the LOCA event.

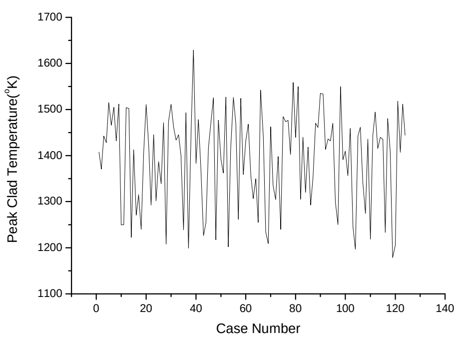

Figure 19. PCT Distributions

Table 5. Top Rank Result of PCT

Ranking	Case No	PCT(°K)
1	123	1544.35
2	13	1535.28
3	52	1527.02
4	30	1525.95
5	39	1522.61
6	79	1520.58
7	98	1511.17
8	95	1502.14
9	91	1496.83
10	70	1496.66

The peak centerline temperature for all the 124 cases are shown in Figure 20 and the top ranked results are listed in Table 6. The highest value is 2155.97 °K for case 23. The third highest value is 2136.34 °K(case 84). Hence the 95/95 value is 2136.34 °K. There is small variation between maximum and minimum fuel centerline temperature. In no case even in maximum case, the centerline temperature exceeded the acceptance criterion. Hence there is no fuel melting during the LOCA event.

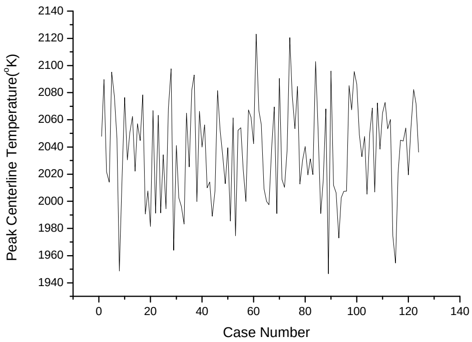

Figure 20. Peak Centerline Temperature Distributions

Table 5. Top Rank Result of Maximum Peak Centerline Temperature

Ranking	Case No	PCT(°K)
1	61	2155.97
2	74	2153.68
3	84	2136.34
4	28	2130.06
5	90	2128.92
6	5	2127.51
7	99	2126.49
8	37	2124.81
9	2	2123.86
10	70	2123.62

The number of rods of fission product release (NFR) was estimated using final acceptance criteria. Table 6 lists the number of ruptured bundle, where rupture happened, maximum PCT for top ranked cases' when acceptance criteria of rupture is 1200 °C. As in Table 1 and considering the axial power distribution in each channel is cosine like shape, most fuel failure occurs in 4-1 ~ 4-4 channels and center nodes(4~8) where power densities are high. There are 37 fuel rods in each channel, we can get the total number of NFR by 37 multiplied by the number of bundle. The highest value is 53 fuel bundle for case 53. The third highest value is 42 bundles(case 123). Hence the 95/95 value of NFR is 42*37=1554 rods. Exactly the same trends of failures are shown when 1000, 1100, 1300 °C of criteria was applied. There are 315, 171, 0 fuel bundles are failed for the acceptance criteria of 1000, 1100, 1300 °C.

Table 6. Result of Fuel Failure Evaluation(Acceptance Criteria : 1200 °C)

Case No.	**PCT(°K)**	**Ruptured Bundle No.**	**Failed Node(1: Failed, 0: Safe)**						
			4-1	**4-2**	**4-3**	**4-4**	**4-5**	**4-6**	**4-7**
13	1535.28	53	000000000000	000000000000	000000100000	000001110000	000000000000	000000000000	000000000000
39	1522.61	48	000001100000	000001100000	000000000000	000000000000	000000000000	000000000000	000000000000
123	1544.35	42	000000000000	000000000000	000000000000	000001110000	000000000000	000000000000	000000000000
52	1527.02	36	000000000000	000000000000	000001100000	000000100000	000000000000	000000000000	000000000000
30	1525.95	28	000000000000	000000000000	000000000000	000001100000	000000000000	000000000000	000000000000
70	1496.66	28	000000000000	000000000000	000000000000	000001100000	000000000000	000000000000	000000000000
76	1496.07	28	000000000000	000000000000	000000000000	000001100000	000000000000	000000000000	000000000000
88	1478.53	25	000000000000	000000000000	000000100000	000000100000	000000000000	000000000000	000000000000
5	1493.18	24	000000000000	000001100000	000000000000	000000000000	000000000000	000000000000	000000000000
58	1488.28	24	000000000000	000001100000	000000000000	000000000000	000000000000	000000000000	000000000000
66	1494.1	24	000000000000	000001100000	000000000000	000000000000	000000000000	000000000000	000000000000
79	1520.58	24	000000000000	000001100000	000000000000	000000000000	000000000000	000000000000	000000000000

9. Conclusions and Future Works

Using the MARS code, the uncertainty evaluation of important parameters on LBLOCA and the number of rods of fission product release (NFR) which determines the amount of the fission product released for a large break loss-of-coolant accident (LBLOCA) in CANDU plants were estimated.

As a result of evaluation, thermal-hydraulic response for all the modelled channels and axial nodes are obtained and then evaluated using the acceptance criteria. As a criterion on the estimation, maximum sheath temperature was used. The sheath temperature for all the channels and nodes are traced to find the 3rd highest one by the uncertainty calculation to get the 95-95 probabilistic values. In no case, the fuel centerline temperature and clad temperature exceeded the acceptance criterion. The 3rd highest NFR was found 1,554 in 95 percentile through investigating the distribution of the NFR.

But this study didn't estimate more important parameters such as reactor power distribution, reactor kinetics, power history of channels, trips, etc. that affect and change major accident sequence, additional evaluations are necessary to fully understand the uncertainty of thermo-hydraulic and other nuclear inputs.

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Advanced Approach to Consider Aleatory and Epistemic Uncertainties for Integral Accident Simulations

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Abstract

The use of best-estimate codes together with realistic input data generally requires that all potentially important epistemic uncertainties which may affect the code prediction are considered in order to get an adequate quantification of the epistemic uncertainty of the prediction as an expression of the existing imprecise knowledge. To facilitate the performance of the required epistemic uncertainty analyses, methods and corresponding software tools are available like, for instance, the GRS-tool SUSA (Software for Uncertainty and Sensitivity Analysis) /1/.

However, for risk-informed decision-making, the restriction on epistemic uncertainties alone is not enough. Transients and accident scenarios are also affected by aleatory uncertainties which are due to the unpredictable nature of phenomena. It is essential that aleatory uncertainties are taken into account as well, not only in a simplified and supposedly conservative way but as realistic as possible. The additional consideration of aleatory uncertainties, for instance, on the behavior of the technical system, the performance of plant operators, or on the behavior of the physical process provides a quantification of probabilistically significant accident sequences. Only if a safety analysis is able to account for both epistemic and aleatory uncertainties in a realistic manner, it can provide a well-founded risk-informed answer for decision-making.

At GRS, an advanced probabilistic dynamics method was developed to address this problem and to provide a more realistic modeling and assessment of transients and accident scenarios. This method allows for an integral simulation of complex dynamic processes particularly taking into account interactions between the plant dynamics as simulated by a best-estimate code, the dynamics of operator actions and the influence of epistemic and aleatory uncertainties.

In this paper, the GRS method MCDET (Monte Carlo Dynamic Event Tree) for probabilistic dynamics analysis is explained, and two application examples are briefly presented. The first application refers to a station black-out scenario. The other application is an analysis of the emergency operating procedure 'Secondary Side Bleed and Feed' which has to be applied after the loss of steam generator feed-water supply.

1 Introduction

The use of best estimate thermal-hydraulic and integral codes is generally recommended for calculating transients and accident scenarios in a deterministic safety analysis. Beside the option to apply best estimate codes with conservative input data, it is recently requested to use realistic input data allowing for a more precise specification of safety margins. This generally requires that all potentially important state of knowledge (epistemic) uncertainties are considered in order to get an adequate quantification of the epistemic uncertainty of the code prediction. Methods and tools supporting the performance of an (epistemic) uncertainty- and sensitivity analysis have been developed and meanwhile are state of the art like, for instance, the GRS-tool SUSA (Software for Uncertainty and Sensitivity Analysis) /1/.

However for a comprehensive safety analysis and risk-informed decision-making, it is essential that stochastic (aleatory) uncertainties are taken into account as well, not only in a simplified and supposedly conservative way but as realistic as possible. This requires a combination of deterministic and probabilistic models which is able to handle the interaction of plant behavior (physical process and system behavior), operator actions and aleatory uncertainties along the time axis.

Neglecting the probabilistic modeling of aleatory quantities and using conservative assumptions instead allows only for a very limited view on the dynamics of accident scenarios and leads to a significant completeness uncertainty with respect to the simulation results. If aleatory uncertainties are considered in a simplified manner under predetermined conservative assumptions the deterministic calculation provides only one single accident sequence (which might be rather unlikely to occur) out of the whole spectrum of potential sequences which may arise from the influence of aleatory uncertainties.

The extent to which deterministic and probabilistic models can provide insights to the plant behavior under accident conditions and contribute to improve decision making regarding plant safety depends on the level of detail of the applied models (model completeness), the way how the models interact and on whether assumptions are made as realistic as possible. To satisfy these requirements it is necessary to combine deterministic and probabilistic issues in a consistent manner.

At GRS the probabilistic dynamics method MCDET (Monte Carlo Dynamic Event Tree) was developed to address this problem. MCDET is implemented as a module system which can in principal be coupled with any deterministic dynamics code (like RELAP, MELCOR, or ATHLET) simulating the plant behavior under accident conditions. The combination of the MCDET modules with a deterministic code permits an integral simulation of complex dynamic processes particularly taking into account interactions of the plant behavior with epistemic and aleatory uncertainties.

In chapter 2 the probabilistic dynamics method MCDET is briefly explained to give an idea of its modeling capabilities and the working mechanism of its modules.

The MCDET modules were supplemented by a so-called Crew-module which enables calculating the dynamics of crew actions depending and acting on the plant dynamics as modeled in a deterministic code and on the uncertainties as considered in the MCDET modules. The Crew-module is described in chapter 3.

In chapters 4 and 5 two application examples of the MCDET- and Crew-modules are given to demonstrate what kind of results can be derived from the detailed analysis using the advanced methods. The first application refers to a station black-out scenario in a German PWR. The other application is an analysis of the emergency operating procedure ‘Secondary Side Bleed and Feed’.

2 The MCDET-method

The most straightforward numerical procedure to consider aleatory and epistemic uncertainties in transients and accident scenarios is the Monte Carlo (MC) simulation. It provides a random sample of sequences out of the (infinitely large) population of possible sequences. Each sequence is generated from a combination of values (randomly) drawn for the quantities that are subjected to uncertainty.

For quantifying the separate influence of aleatory and epistemic uncertainties, a nested two loop Monte Carlo simulation is applied, where for each sample elements for the epistemic uncertainties (outer loop) a Monte Carlo simulation is performed for the aleatory uncertainties (inner loop).

In order to be able to adequately account for values which might occur with a very low probability, the sample of simulated sequences must be sufficiently large. Therefore the computational effort of the MC analysis might be immense. Furthermore, Monte Carlo simulation requires the calculation of a complete sequence starting from the beginning to the end of calculation time.

One method which is able to reduce the computational effort is the Discrete Dynamic Event Tree (DDET) approach /3/. This approach organizes the computation of accident sequences according to the tree structure known from the conventional event tree analysis. A DDET evolves along the time axis according to the interaction of the dynamics as calculated by a deterministic code and the aleatory uncertainties which might influence the dynamics. An event which is subject to uncertainty can occur at any point in time along the time axis. All possible situations (system state changes) which might occur at a point in time - also those of very low probability - are calculated. They are combined with other situations which might occur in the further evolution of the dynamics. Result is a DDET where each new situation (branch of the tree) is calculated.

Different from Monte Carlo simulation the DDET-method avoids repeated calculations of situations shared by different sequences. A sequence is calculated from the point in time where a new situation starts. The past history of the sequence is given by another sequence from which the current sequence branches off.

One drawback of the DDET approach is that continuous variables like, for instance, the timing of events have to be discretized. A coarse discretization - providing a less accurate result - would generally be applied for accident scenarios with a large number of events that might occur at some (random) point along the time axis. A detailed time discretization would lead to a combinatorial explosion of the number of sequences because DDET methods generally consider all combinations of alternative dynamic situations which may arise at the discrete points in time. Of course, the introduction of a probabilistic cutoff criterion and of boundary conditions may help to limit the computational effort.

Another drawback of DDET approaches is that the accuracy of their results, which are derived from a more or less detailed discretization, is difficult to quantify. Compared to that, MC simulation or an appropriate combination of MC simulation and DDET approach would permit the application of principles from mathematical statistics to estimate the relevant quantities and to determine the accuracy of these estimates.

The MCDET (Monte Carlo Dynamic Event Tree)-method developed at GRS is a combination of Monte Carlo simulation and the Discrete Dynamic Event Tree approach. It is capable of accounting for any aleatory and epistemic uncertainty at any time during an accident scenario. Events characterized by a timing and system state change, each of which may be either deterministic or random and discrete, are generally treated by the DDET approach. MC simulation is generally used to handle events with a continuous random timing or system state change. MC simulation may also be used for events with many discrete alternatives for the timing and/or the system state change.

If the events of an accident scenario have only discrete alternatives for the timing or system state change, MCDET generally keeps track of all possible combinations of the discrete alternatives and produces - in combination with an appropriate deterministic code - one DDET. If, in addition, continuous random timings or system state changes have to be considered, MCDET uses MC simulation to take the corresponding continuous probability distributions into account and produces -

in combination with the deterministic code - a random sample of DDETs. Each DDET is constructed on condition of a randomly selected set of values from the continuous probability distributions.

MCDET can be identified as a special case from the class of so-called "variance reduction by conditioning" Monte Carlo simulation methods. Any scalar output quantity Y of a (deterministic) model h subject to aleatory uncertainties (stochastic events) can be represented as Y=h(**A**) with **A** being the set of all stochastic (aleatory) variables involved. **A** is then divided into two subsets $\mathbf{A}_d$ and $\mathbf{A}_c$ with $\mathbf{A}_d$ = subset of the specified discrete random variables treated by DDET approach and $\mathbf{A}_c$ = subset of all specified continuous random variables treated by MC simulation.

MCDET is implemented as a module system (set of routines) which can in principal be combined with any deterministic code. Each MCDET-routine performs a specific task of the probabilistic dynamics calculations. These tasks are, for instance, the initialization of epistemic and aleatory quantities for DDET generation, the simulation of values for the continuous aleatory variables, the definition of initial conditions for new dynamic situations (new DDET branches) at corresponding points in time (the branching points of a DDET) and storing of the conditions or the selection of the dynamic situation (DDET branch) to be calculated next.

The result of a MCDET-analysis is a sample of DDETs. Fig.1 provides a schematical illustration of the sample of DDETs in the time-event space. Timing and order of events might differ from one DDET to another due to the different values from MC simulation. For each sequence of events, the corresponding probability and the temporal evolution of any process quantity (considered in the deterministic analysis) is calculated. Probabilities can be calculated at any point in time. Hence, for each DDET, the conditional probability distribution for any process quantity can be given not only at the specified time end of the analysis, as it is indicated in Fig.1, but at any point in time. The probability distributions of a DDET are conditional on the values obtained from MC simulation.

The temporal evolution of a process quantity corresponding to each event sequence in the sample of DDETs is schematically illustrated in Fig.2.

Since the distributions derived for a DDET are conditional on the values obtained from MC simulation, the final probabilistic assessments are derived from the mean probability distributions over all DDETs. Each mean distribution can be given together with confidence intervals indicating the sampling error with which the corresponding mean value derived from only a sample of DDETs (generated on condition of only a limited selection of values for the continuous aleatory variables) may estimate the true probability. Thus, MCDET allows the application of principles from mathematical statistics to estimate probability distributions and to determine the accuracy of these estimates.

One important feature of the MCDET module system is, that it can in principal be combined with any deterministic dynamics code. A requirement for complex codes is the capability to generate restart files. MCDET offers a considerable scope for stochastic modelling. There is practically no limitation to model aleatory (and epistemic) uncertainty.

The MCDET module system has to be linked to a program scheduling the MC simulation, the DDET construction and the corresponding calculations of an appropriate deterministic dynamics code. The combination of the MCDET modules with a deterministic code calculates the sequences of a DDET and automatically provides for each sequence:

- the order of events,
- the time histories of dynamic quantities, and
- the probabilities at any point in time.

Post-processing programs are available to evaluate the huge amount of output data of an MCDET-analysis.

Fig. 1: Sample of DDETs in the time-event space. (S1- S3, HA and PC denote events subjected to aleatory uncertainty)

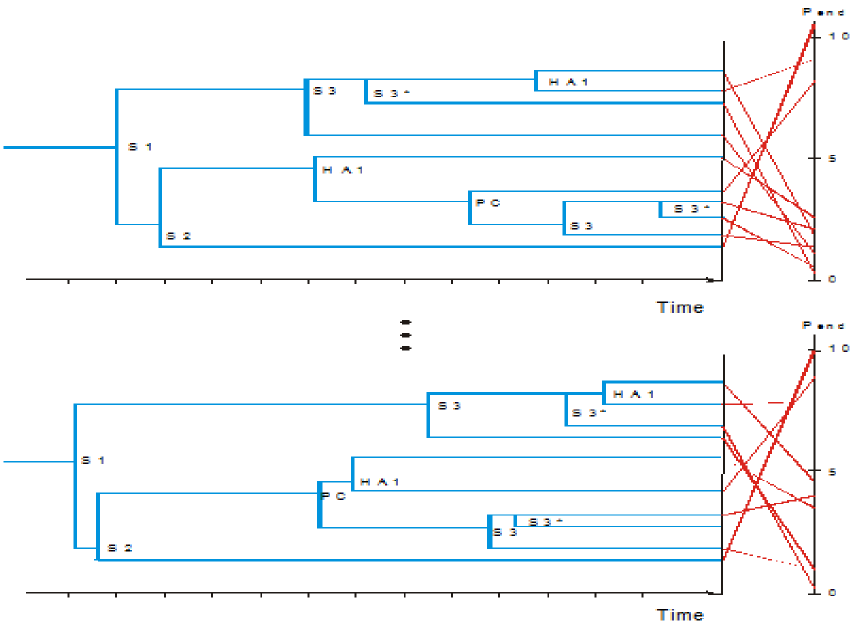
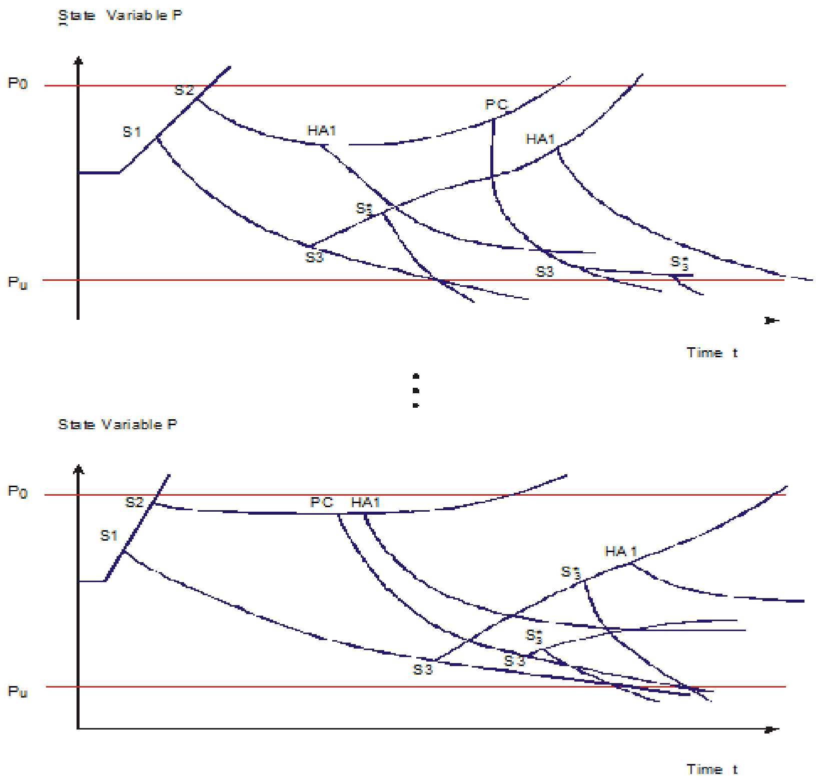


Fig. 2: Sample of DDETs in the time-state space. Temporal evolution of process quantity P according to the event sequences in the sample of DDETs in Fig. 1



3 The Crew-module

An extra Crew-module was developed to allow for calculating the dynamics of operator actions depending and acting on the plant behavior as modeled in the dynamics code and on aleatory uncertainties as considered in the MCDET-modules. Aleatory uncertainties which may affect the crew dynamics are, for instance, the execution times and consequences (success, errors of omission and commission) of human actions or the behavior of the technical system and the physical process. Vice versa, operator actions, performance shaping factors, the mental state or cognitive behavior of the operators can influence the aleatory uncertainty on the outcome of further actions. For instance, human errors may occur with a higher probability as a consequence of component failures which may cause a rise of the stress level of operators.

A crew as can be considered by the crew module may consist of a number of individuals performing tasks and communicating with each other. Each crew member can be responsible for a special task. There may be tasks which can be executed in parallel, and other tasks which can only be started, if some conditions are fulfilled, e.g. system and process conditions prescribed in the emergency procedure or the confirmation from another operator of a successfully accomplished task. Information on the process and system state is given by the respective control room equipment like, for instance, indicators or alarms. Indicators are read by the responsible operator and communicated to other crew members (supervisor, supervisor assistant).

Human actions may change the system state and a system state change may immediately affect the physical process. On the other side, the reaction of the process as indicated by the corresponding equipment in the control room may affect the actions of the crew. Other factors which may influence human actions are so-called performance shaping factors like stress, knowledge or ergonomics and the communication between crew members. For instance, the information from an operator, that he is not able to correctly finish his task, may increase the stress level of the supervisior, and this may increase the likelihood to omit necessary actions (e.g. instructions) or to commit incorrect actions.

The Crew-module is composed of a collection of input-files and routines. The input files include the information necessary to describe the process of human actions and the routines read and calculate the respective information.

The main tasks of the Crew-module routines are i) to read the information on alarms, basic actions and action sequences given in the corresponding input files, ii) to check for system and process conditions at each time step iii) to check, if relevant alarms are activated depending on the given state iv) to check, if an action sequence has to be activated and vi) to simulate a sequence of actions for which the conditions are fulfilled as a dynamic process.

The Crew-module is linked to the scheduler program which organizes the alternate calculations of the Crew-module, the deterministic dynamics code and the MCDET-modules. The combination of the MCDET-modules (Stochastic- and Crew-module) with a deterministic dynamics code allows an integral simulation of complex dynamic systems where human actions, the technical system, the physical process and aleatory uncertainties are the main interacting parts in the course of time.

4 Application 1: Station black-out scenario

A station black-out (SBO) scenario in a 1300-MW pressurized water reactor (PWR) of Konvoi type at nominal power was chosen as an illustration example. In this analysis MCDET was coupled with the integral code MELCOR. The transient is characterized by the total loss of power including emergency diesels and power from off-site systems.

At the beginning of the scenario the reactor is shut off automatically and the main coolant pumps of all 4 loops fail. The power supply provided by batteries is given. The automatically driven pressure control by the pressurizer relief valve and the two safety valves are principally given, but their opening and closing function, if demanded, are subject to aleatory uncertainty. The number of the demand cycle at which the respective valve would fail is sampled according to a probability distribution determined by the total number of demand-cycles (given through the process dynamics) and the probabilities for valve failure at the corresponding demand-cycles. The randomly selected demand cycle at which a valve fails determines the failure time of the respective valve. The failure mode ‘fail to open’ and ‘fail to close’ of each valve is another aleatory uncertainty.

The primary-side pressure release (primary side bleed) is initiated by the crew at some time after the corresponding system indicates that the core exit temperature exceeds 400°C and the fluid level in the reactor pressure vessel (RPV) passes below 7.5 m (plant-specific criterion). The time the operators need to execute the pressure relief is assumed to be a continuous random variable. During the pressure-release, each of the three pressurizer valves - which didn’t fail at an earlier point in time - may now fail with a probability depending on the number of cycles the corresponding valve has already been demanded for pressure limitation.

Whether the accumulators can inject their coolant inventory is an aleatory uncertainty. It depends on whether the source and additional isolation valves fail to open on demand. The time of power recovery is subjected to aleatory uncertainty as well. The application was performed on the assumption that the external power is restored not earlier than 5700 s and not later than 12 000 s (time end of simulation) after the initial event. Given that power is recovered, further aleatory uncertainties refer to the availabilities of the four high pressure and low pressure pumps of the emergency core coolant system (ECCS). The reconnection times of the ECCS trains were classified as epistemic uncertainties, because the uncertainty due to lack of knowledge was judged to be predominant in comparison to the aleatory uncertainty about the reconnection times.

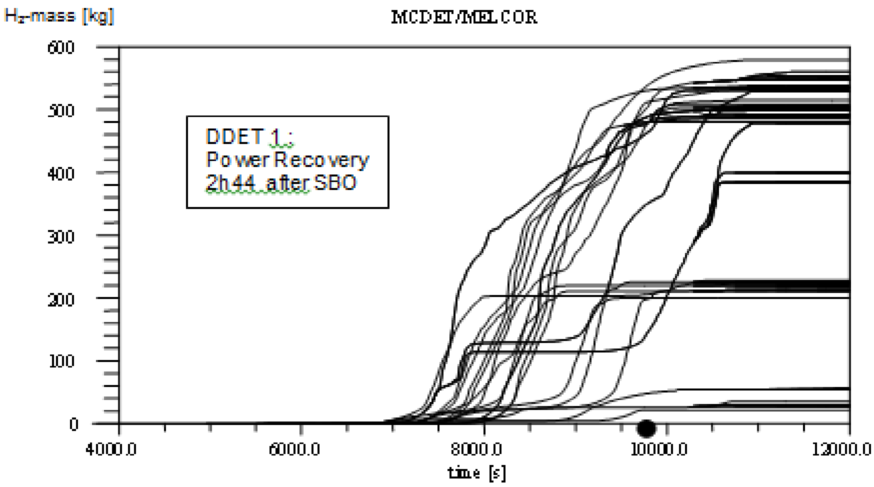
In addition to the aleatory uncertainties ca. 30 epistemic variables have been specified. All uncertainties (aleatory and epistemic) are described in reference /4/. Although MCDET is capable of accounting for epistemic uncertainties, in this analysis the epistemic uncertainties were represented only by the mean values of the corresponding probability distributions. Hence, the variability of the results arises exclusively from the specified aleatory uncertainties.

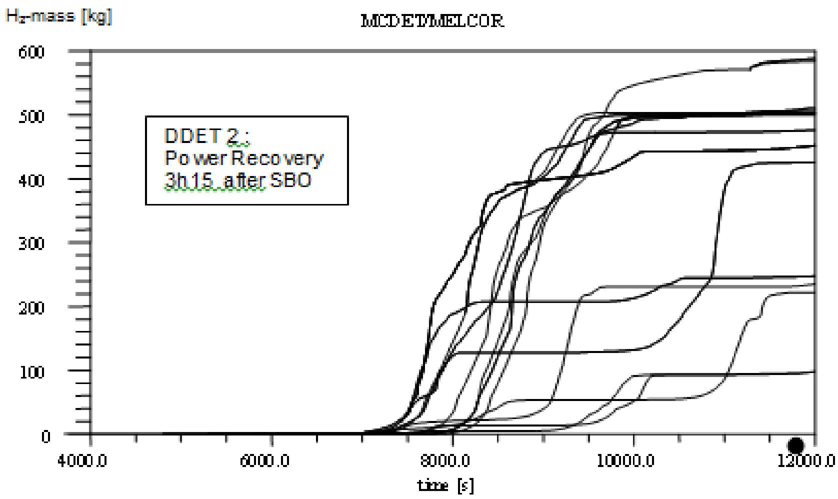
Of particular interest were the time histories of process quantities like the pressure in the RPV and in the containment, the core exit temperature, and the degree of core degradation as expressed by the UO2-melt mass and the hydrogen (H_2) mass generated. Because of the vast amount of generated output data and possibilities to analyze them, only a small selection of results is given.

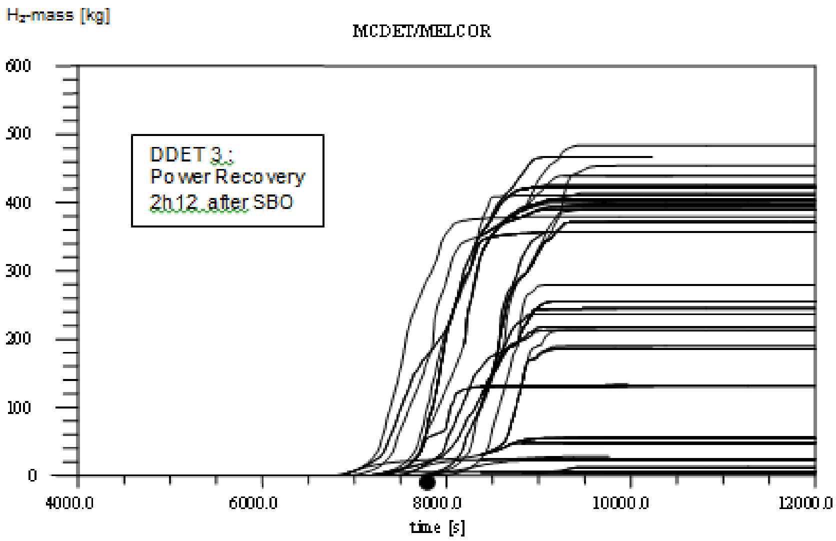
In Fig.3, three DDETs out of a sample of 50 generated DDETs are given. Each DDET shows a variety of time histories for the process quantity 'H_2-mass'. The variety within a DDET arises from the aleatory uncertainties which are handled by the DDET approach. The difference between the sampled DDETs arises from the aleatory uncertainties considered by MC simulation.

To each time history the respective sequence of events and the corresponding occurrence probabilities are attached. With this information, a very detailed analysis of the process behavior can be performed. It allows, for example, to filter out events which are responsible for critical process states, e.g., events that lead to a high H_2-mass at a relatively early point in time after the initiating event. This might help to identify so called 'cliff edge effects', where small variations in events lead to large differences in the process evolution. Or to determine event and time pattern which might indicate in advance that the process will lead to an undesired state within a given time range and with a high probability. This information might be useful to start preventing and mitigating actions as early as possible.

Fig. 3: Three DDETs (out of a sample of 50 generated DDETs) showing the time evolution of the process quantity 'total generated hydrogen mass (H_2-mass)'.



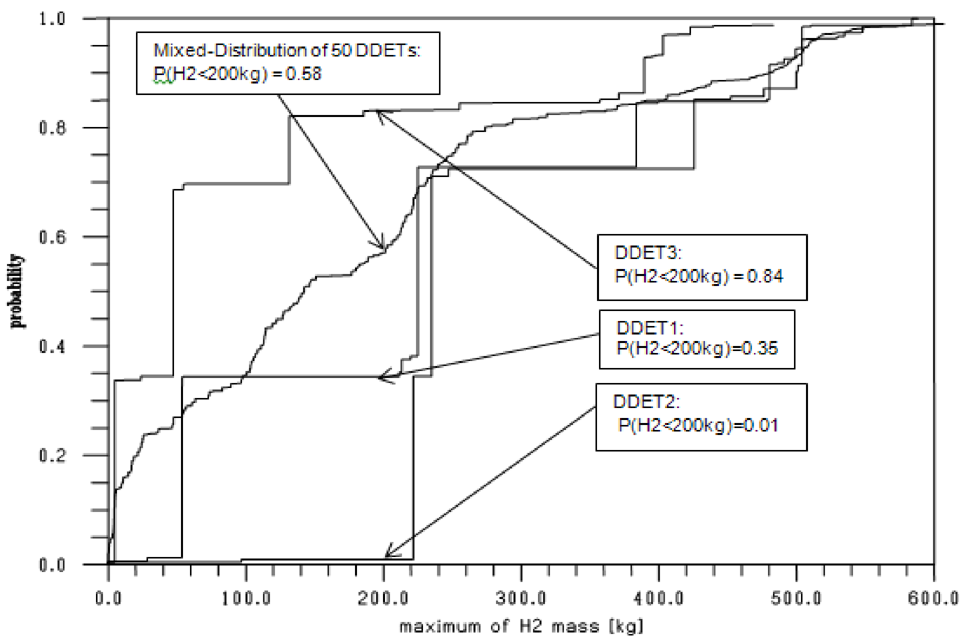




From the comparison of the 3 DDETs shown in Fig. 3 we can conclude, that in DDET 3 - where power recovery is about 1 h earlier than in DDET 2 – the H_2-mass production is lower than in DDET 2. But a quantification of the difference of the H_2-mass production between the DDETs is not possible if only the time histories are considered. For that reason the probabilities attached to each trajectory of a DDET have to be taken into account. Using these probabilities, conditional probability distributions of any process quantity can be calculated for each generated DDET.

In Fig. 4 the conditional probability distributions for the maximum of produced H_2-mass are given for DDET 1-3 and for the unconditional mean probability distribution over all 50 generated DDETs.

Fig. 4: Conditional probability distributions for DDET1-3 and mean distribution over all 50 generated DDETs for the maximum of produced H_2-mass



From Fig. 4 we can see that in DDET 2 – where power recovery is about 1 h later than in DDET 3 - the probability P(H_2-mass < 200 kg) for the H_2-mass not to exceed 200 kg is only 0.01 whereas in DDET 3 it is 0.84. In DDET 1, where power recovery is between DDET 2 and DDET 3, the respective probability P(H_2-mass < 200 kg) is 0.35. The probabilities indicate that in case of DDET 2 a much higher maximum of H_2-mass is produced than in DDET 3. The probability is about 99% that the maximum of produced H_2-mass exceeds 200 kg, whereas in DDET 3 this probability is only 16%. The mean probability P(H_2-mass < 200 kg) over all DDETs is 0.58.

The mean probability distribution (mixture of distributions of all 50 generated DDETs) of each quantity the corresponding (1- α)% confidence limits can be calculated with α = 0.01, 0.05,0.1 etc. The 90% confidence interval for the mean probability P (H_2-mass < 200 kg) = 0.58 is given by (0.5, 0.64). Similarly it is easily feasible to derive probability assessments for more complex dynamic situations. The probability assessments can be calculated for any process quantity considered in the deterministic model. A more detailed description of this application and the MCDET-method is given in /4/.

5 Application 2: ‘Secondary Side Bleed and Feed’

Emergency Operating Procedures (EOP) are part of the safety concept of NPPs. Their application is required if design based safety functions fail. Core damage states occur only if EOPs are not successful. The EOP ‘Secondary side bleed and feed’ (SBF) is to be employed in a German PWR to achieve the protection goal of steam generator (SG) injection after the loss of feed-water supply.

The EOP requires a crew of at least 8 operators which have to execute different tasks within and outside the control room. Crew actions have to be initiated as soon as the corresponding displays in the control room indicate that the water level of all four steam generators (SG) passes below 4m. In this application, it was assumed that all displays operate correctly, and that they immediately indicate the process condition. Diagnosis and decision problems were not assumed, because signals and criteria to start the EOP are clear.

The application example did not consider uncertainties in the behavior of the technical system and in the physical process. It is restricted to the aleatory uncertainties in the crew actions. Human Error Probabilities (HEP) concerning crew actions were derived from the ASEP method /5/, while the probability distributions of the random times needed to execute the actions were obtained by expert judgment. A more detailed description is given in /6/.

The deterministic code MELCOR was applied to simulate the system and process dynamics after the loss of feed-water supply. The combination of the Crew-Module and MELCOR is capable of deterministically handling the interactions between the plant dynamics and operator actions. Coupling the MCDET-modules to this combination provides a simulation tool which is able to account for the potentially important aleatory uncertainties.

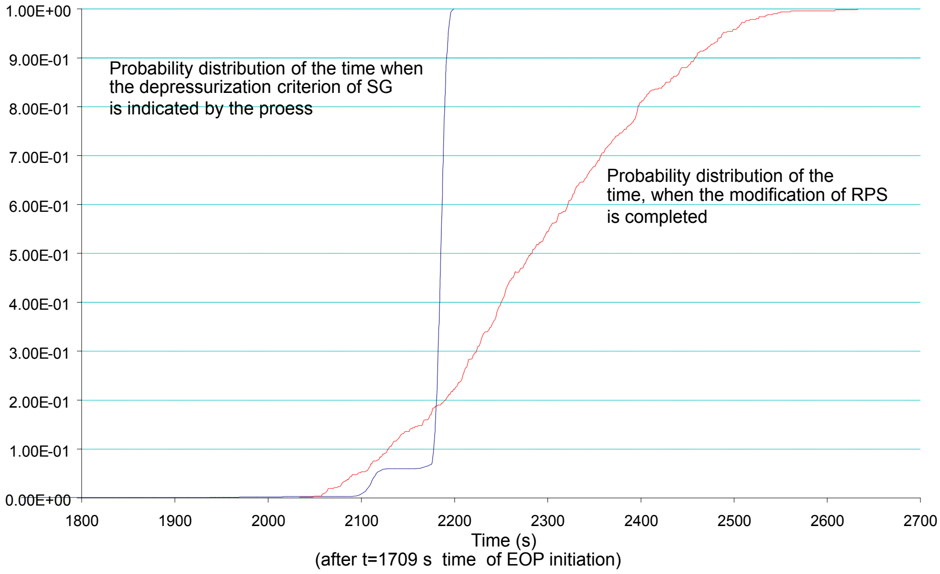
According to the MELCOR calculation, the EOP initiation criterion (water level of all four SGs < 4m) occurs at t=1709 s after the loss of feed-water supply. The first EOP task to be performed is the switching off of the reactor coolant pumps (RCP). Its successful execution would lead to a reduction of the thermal energy generated in the reactor. The mean (unconditional) probability of the RCP not to be switched off is 9.6E-4. The corresponding 95%-confidence interval is given by (3.0E-4; 1.6E-3). Depending on whether the RCP are switched off and, if so, when they are switched off, the system and process criterion to depressurize the SGs (depressurization criterion of secondary side) is reached sooner or later.

Besides switching off the RCP, other essential tasks of the SBF procedure are the task to modify some parts of the reactor protection system (RPS) and the task to pressurize the feed-water storage tank (FST) in order to use its water inventory for SG injection. According to the written procedure, the crew must perform the depressurization of the SG immediately after the corresponding criterion is indicated by the process. But the depressurization of the SG can only be performed, if the task of modifying the RPS has been accomplished. If the modification of the RPS is not yet finished and the depressurization criterion is given, the crew has to wait with the SG depressurization until the necessary modification of the RPS is completed. The task of pressurizing the FST for using its water inventory for SG injection can only be started if the modification of the RPS has been finished. If the situation is given that the modification of the RPS has been completed and the process criterion for SG depressurization is already given, the crew has to skip the pressurization on the FST and must depressurize the SG. In that case the water inventory of the FST cannot be used for SG injection.

According to the EOP, it is assumed that after accomplishing the modification of the RPS there is enough time for the crew to perform the task to pressurize the FST in order to use its water inventory before the system and process criterion for SG depressurization is indicated. However, the results of the MCDET-analysis indicate, that this assumption does not hold with a rather high probability.

Fig.5 presents the mean probability distributions of the time when the modification task of the RPS has been accomplished, and the time when the process criterion of the SG depressurization (depending on the time when the RCP have been switched off) is given. The times indicated in Fig.5 are related to t = 1709 s which is the time when the EOP initiation criterion occurs.

Fig. 5: Mean probability distributions of the occurrence time of the depressurization criterion and of the time, when the modification task of the RPS is accomplished.



From the given distributions in Fig.5 it can easily be seen, that with a probability of approximately 0.78 the task of modifying the RPS is completed later than the latest time when the criterion for SG depressurization is indicated by the process. This implies that the pressurization of the FST cannot be performed and therefore the water inventory of the FST cannot be used for SG injection with a very high probability. It is striking that neither technical failures nor human errors, which would lead to the omission of FST pressurization, are responsible for that situation. The only reasons are time effects resulting from the interaction between system and process dynamics, operator performance and the aleatory uncertainties relating to operator performance.

6 Conclusion

The extent to which deterministic and probabilistic models can provide insights to the plant behavior under accident conditions and contribute to improve decision making regarding plant safety depends on the level of detail of the participated models (completeness), the way of how the interaction between the models is handled and on whether assumptions are made as realistic as possible. To satisfy these requirements it is necessary to combine deterministic and probabilistic issues in a consistent manner. This is realized with the probabilistic dynamics method MCDET which is a combination of MC simulation and the DDET approach. MCDET is implemented as a module system that can in principal be coupled and operate in tandem with any deterministic dynamics code. The tandem allows for a realistic modeling of the mutual dependencies between the behavior of the technical system, the physical process, and aleatory (as well as epistemic) uncertainties. It provides a representative spectrum of accident sequences together with corresponding probabilities and, therefore, is appropriate for risk-informed decision making.

An important issue of safety assessments of NPPs is the analysis of operator performance in response to the plant dynamics. To achieve a more realistic modeling in this context, interactions between operator performance, system behaviour and the physical process must be taken into account in the course of time. For that reason, an extra Crew-module was developed. It can simulate operator actions as a dynamic process. Combining the Crew-module with an appropriate deterministic code allows for handling the interactions between the plant dynamics and operator actions. Coupling the MCDET-modules to this combination provides a simulation tool which is able to account for all important uncertainties affecting the plant-crew dynamics.

The application of the probabilistic dynamics method MCDET provides both i) a consistent integration of probabilistic issues into the deterministic evaluation of the plant behavior and ii) a more detailed insight into the plant behavior. Therefore, it might be particularly important for risk-informed decision making and licensing.

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Safety Margin Assessment (SM2A): Stimulation for Further Development of BEPU Approaches?

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Abstract

During recent years, many nuclear power plants underwent significant modifications, e.g. power up-rating. While compliance with all the deterministic acceptance criteria must be shown during the licensing process, the larger core inventory and the facts that the plant response might get closer to the limits after a power up-rate, suggest an increase of the core damage frequency (CDF) and other possible risk indicators. Hence, a framework to quantitatively assess a change in plant safety margin becomes very desirable.

The Committee on the Safety of Nuclear Installations (CSNI) mandated the Safety Margin Action Plan expert group (SMAP) to develop a framework for the assessment of such changes to safety margin. This framework combines PSA and the analytical techniques developed in BEPU. CSNI then mandated the SM2A expert group to especially explore the practicability of the SMAP framework. This pilot study was completed end of 2010.

An increase of the (conditional) probability of exceedance for a surrogate acceptance limit (PCT) indicating core damage was successfully evaluated for the selected sequences from several initiating event trees, and it was found that only a restricted number of sequences need to be analyzed.

Based on the insights gained from this study, areas of methodology improvement have been identified and related proposals for further R&D work will be discussed.

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1. Background

Potential erosion of safety margins was identified as an area for further research by the Committee on Nuclear Regulatory Activities (CNRA) already in 1998. In December 2003, the Committee on the Safety of Nuclear Installations (CSNI) mandated the Safety Margin Action Plan (SMAP) expert group to develop a methodology that allows for integral assessment of the changes of overall plant safety resulting from concurrent changes to plant modifications and/or operations/conditions. Such a framework was outlined in the final report of the SMAP expert group [1], the main ideas being:

- Safety margin is related to the frequency of exceedance of the relevant acceptance criteria.

- The SMAP framework combines probabilistic and deterministic analysis approaches. It first identifies a set of sequences (risk space) that shall be included in the deterministic analysis. This means that the risk space is systematically examined, using detailed event trees to define the sequences to be analyzed with best-estimate tools including uncertainty analysis, so-called Best-Estimate plus Uncertainty (BEPU) approach.

- Increments in frequencies of exceedance of the acceptance criteria related to selected safety variables are used as indicators of safety margin changes (erosion or increase).

A follow-up Task Group was initiated by CSNI in June 2007 and given the mandate to carry out a pilot study that applies the SMAP framework for the evaluation of the possible change in safety margins which would result from implementing newly proposed USNRC rulemaking related to LOCA analysis. In addition, the task group was directed to concentrate its efforts on assessing the practicability of the SMAP framework.

The Task Group on Safety Margin Assessment and Application (SM2A) actually started to work in January 2008 and completed in October 2010, with the following Group of participating countries: Czech Republic, Finland, France, Germany, Japan, Korea, Mexico, Spain, Switzerland, The United States and with the participation of the IAEA as observer. CSNI approved its final report [2] during its meeting in December 2010.

2. Short Review of SMAP Framework

The introduction of the concept of safety margins in traditional engineering is a way to allow for the “known unknowns” as well as to also accommodate “unknown unknowns” viz. challenges not foreseen during the design process. The nuclear industry adopted this concept from the beginning, in particular, for assessing the capabilities of the plant design by analyzing so-called Design Basis Transients and Accidents (DBT&A, referred to as DBA). A set of well defined enveloping scenarios, classified into a few frequency classes, are chosen as design basis for the protection and a set of safety variables are used as indicators of challenges to the protection barriers. For this limited set of design basis scenarios class-specific acceptance criteria (also called safety limits) are specified in terms of extreme values of the safety variables.

Worldwide experience on the operation of nuclear power plants showed that the exclusive analysis of DBA to assess plant safety is not adequate. Some events, especially the TMI accident, suggest the relevance of complicated scenarios that belong to the out-of-design category. Another important lesson learned from this experience was that the impact of operator actions must be taken into account. To gauge the relative importance of the many possible sequences, a detailed evaluation of their frequencies is necessary.

Probabilistic Safety Assessment (PSA) techniques implement these new aspects of the safety analysis but they have only been applied to the assessment of severe accidents and their consequences.

1 *SMAP Task Group on Safety Margins Action Plan (SMAP) - Final Report*, NEA/CSNI/R(2007)9, July 2007.

2 *Safety Margin Assessment and Application - Final Report*, NEA/CSNI/R(2011)3, 2011.

Extensive plant modifications, such as life extensions, power up-rates and others, could possibly imply a reduction in existing margins and, therefore, the overall plant safety can be reduced even though the acceptance criteria of the DBA analyses are still fulfilled. The preservation of enough safety margins becomes an important question, which cannot be answered properly with either an analysis of DBA alone or with PSA alone.

The analysis framework put forward by SMAP [1] consists of a combination of traditional deterministic (analysis of DBA) and probabilistic (PSA) techniques. The use of event and fault trees, similar to those of PSA, is combined with simulation techniques, typical of DBA analyses, in order to quantify the exceedance frequencies of the acceptance limits that define the safety objectives being assessed. These exceedance frequencies are then used as risk indicators that characterize the overall plant safety. When applied to the analysis of plant modifications, a significant increment in the exceedance frequency of some acceptance limit represents an indication of significant erosion in safety margins. The quantification of frequencies of exceedance can also provide useful insights to the identification of plant safety degradation mechanisms as a consequence of plant modifications.

From the point of view of traditional DBA analysis techniques the SMAP proposal consists of extending the limited set of DBA to the almost complete set of all the credible plant sequences.

In the general case, both the safety variable value resulting from an accident sequence and the barrier strength associated to a safety objective are described by probability density functions (PDFs). Standard load-strength analysis techniques that were developed in the context of reliability engineering can then be applied. The distribution of the safety variable represents the load function and the distribution characterizing barrier strength represents the strength function. According to this load-strength scheme, the concept of "margin to damage" in a particular sequence is defined as probability that the load remains below the strength. The margin to damage is often referred to as safety margin in the literature and is given by a convolution integral over the two distributions. Its complement, i.e., the conditional failure probability is the one needed for risk space quantification of damage states and formulated as convolution integral yields [1]:

$$p(L > S) = \int_0^\infty f_S(S) \left[\int_S^\infty f_L(L) dL \right] dS \qquad (1)$$

In nuclear engineering, probability distributions for barrier strength are frequently not available. They are replaced by acceptance limits - limiting values– of safety variables that are specified by the nuclear regulators. In this case, the expression of the conditional failure (limit exceedance) probability reduces to:

$$p(L > S_L) = \int_{S_L}^\infty f_L(L) dL \qquad (2)$$

where SL is the safety (acceptance) limit.

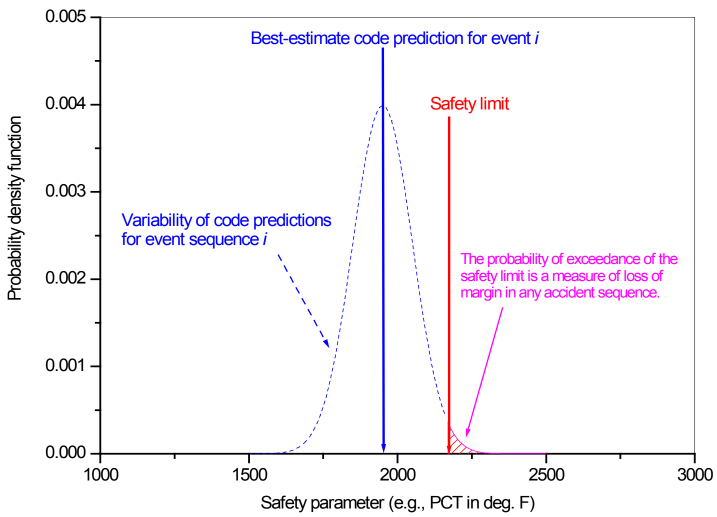

Figure 1: Conditional probability of safety limit exceedance

SMAP proposes to include all those limits usually addressed in DBA analyses. (The consideration of additional limits - if necessary - would be straight forward.) The accident scenarios composing the Risk Space can be described using Event Tree / Fault Tree techniques typical of PSA. However, each limit can be exceeded in a sequence with a probability that can be calculated with uncertainty analysis techniques. This conditional probability of limit exceedance is the fraction of the sequence frequency that actually contributes to the limit exceedance frequency. A fraction of success sequences (branches) have in fact a contribution to failure; and reversely a fraction of failure sequences have a contribution to success. Put in these terms, classical PSA (level 1) most often represents the particular case where the only addressed limits are those that represent transitions to severe accidents and where the conditional probability of limit exceedance can take only values 0 or 1.

3. Adaptation of the SMAP Framework

SM2A represents the first large-scale application of the SMAP framework. It is therefore not surprising that some aspects of the framework had to be revised and adapted in view of this application. The main items are described below.

3.1 Use of a bounding safety limit as a substitute for the barrier strength distribution function

In the general case, the conditional probability of barrier failure in an accident sequence results from the convolution of the load and the strength probability density functions, as described before. The safety (acceptance) limit is specified by the regulator. If the safety variable exceeds this limit; the system is assumed to be put in an unacceptable state. Therefore, as the SM2A pilot study aims at supporting the safety margin assessment in a nuclear regulatory setting, the adoption of a safety limit instead of the barrier strength function is evident.

3.2 Selection of Peak Clad Temperature as the only safety variable for analysis

The strong time constraints imposed on the SM2A expert group made to decide to limit to the evaluation of a single safety variable: PCT was selected, using 1204 ºC as the only safety limit for this variable. Exceedance of this limit was taken as indication for core damage. Furthermore, PCT is a safety variable already used as risk criterion in existing PSAs.

3.3 Reliability models

Failure probabilities of equipment and operators performing protective actions are essential elements of both PSA and safety margin assessment (viz. in the SMAP framework). There is, however, an important difference. The possible configurations of an event tree header are qualified as successful or unsuccessful based on the result of the simulation of the particular transient belonging to the sequence, contrary to the level 1 PSA where failure or success is determined based on enveloping analysis for a predefined configuration of the safety systems and/or operator actions. SMAP event tree headers therefore represent discrete system configurations (e.g. 3 systems or 4 systems) and/or operator actions rather than the minimal required ones to ensure safety functions (e.g. 3 out of 4), as generally used in PSA. This means that PSA fault trees need some reworking before they can be applied for SMAP applications.

Another adaptation of the reliability models was needed for SM2A due to the consideration of uncertainties in the timing of operator actions. Some of the events that can be included in a fault tree or directly in a sequence occur at unpredictable times. In the PSA approach, it is generally assumed that these events occur at some particular time or within an allowable interval and the assigned probability does not depend on the specific time of occurrence. In the SMAP framework, these times are considered as uncertain parameters.

As the available information on the PSA for Zion NPP was very limited, the analysis was done at PSA sequence level in many cases, without exploring all the possible configurations of safety systems. In some few instances lacking information had to be supplemented with “typical” values derived from PSA studies of other reactors.

3.4 Treatment of uncertainties

It is suggested in the open literature that a proper treatment of uncertainties supporting risk-informed decisions should distinguish epistemic from aleatory uncertainties - e.g. [3] - since the former can be reduced (to some extent) and the latter cannot. The SMAP final report [1] recommends making this distinction, i.e. separating epistemic and aleatory uncertainties, and refers to the nested two-loop calculation: The inner loop deals with aleatory uncertainties whereas the epistemic uncertainty would be dealt with in the outer loop. Other authors, however, are of the opinion that such a distinction is not necessary and that uncertainty is in any case a question of degree of belief, whatever the uncertainty type is (epistemic or aleatory), e.g. [4].

Because the simulated plant response is comparable in nominal and up-rated reactor conditions, the same uncertainties can be assumed applicable to both. This minimizes the impact onto the evaluation of ΔCDF not distinguishing between the two types of uncertainties. In fact, the SM2A calculations did not separate between the two types of uncertainties. As a consequence, the SM2A results are limited to the expected values of the frequency of exceedance for base and up-rated conditions and to the difference between these.

3.5 Conditional exceedance probability header added to event trees

When assessing ΔCDF, the PSA approach is in principle to sum the CDF frequencies (failure branches) for the reference and up-rated cases and to take their difference. This simple approach is improved in the SMAP framework because a fraction of each success branch may in fact include failure when uncertainties are taken into account. This fraction can be impacted by the plant changes. In addition, it has to be remembered that in many recent PSAs, the core damage sequences are not mutually exclusive; in these cases, a simple addition of the frequencies is not correct. A possible resolution to this problem is to add a so-called “thermal-hydraulic” header which quantifies the fraction of the success branches. Ref. [5] introduces this idea. In some of the SM2A analyses, e.g. the LBLOCA and MBLOCA analyses, this approach was hence adopted.

4. Step-by-Step Description of the SM2A Procedure

3 M.E. Paté-Cornell, “Uncertainties in risk analysis: Six levels of treatment”, *Reliability Engineering and System Safety* **54** (1996), pp. 95-111.

4 Robert L. Winkler, “Uncertainty in Probabilistic risk assessment“, *Reliability Engineering and System Safety* **54** (1996), pp. 127-132.

5 H.F. Martz, L.L. Hamm, W.H. Reer, P.Y. Pan, “Combining mechanistic best-estimate analysis and Level 1 probabilistic ri sk assessment”, *Reliability Engineering and System Safety* **39** (1993), pp. 89-108

The SMAP framework was implemented with the following steps:

1. screening of fault/event trees and sequence model
2. revision of available event trees
3. specification of the safety variables
4. transient simulation, taking uncertainties into account, and
5. quantification of change of safety margins.

4.1 Screening of scenarios and revision of available event trees (steps 1 and 2)

The first step of the SMAP framework consists in screening of seqeunces in order to identify the ones to be analyzed with transient simulations. It is expected that only a rather limited set of sequences will at the same time be impacted by the postulated plant modification and significantly contribute to the exceedance frequency (above a threshold value arbitrarily chosen as 10^{-7}/yr for the SM2A exercise). Furthermore, many of the sequences are expected to clearly be core damage or success sequences in both nominal and up-rated conditions; they do not need to be evaluated.

The available event trees must be revised in order to provide a basis for the deterministic consequence analysis. This means:

- Headers that represent so-called minimum requirements (e.g. “minimal 1 out of 2 systems available”) may need to be broken up into several headers, each indentifying a configuration that can be unambiguously represented in a deterministic simulation tool.
- The availability of systems needs to be critically reviewed in consideration of the phenomenology of the transient, e.g. Large LOCA,: A break in the cold leg may lead to the loss of the injection of one accumulator. Therefore, the maximum number of accumulators available needs to be reduced by one according to the specific sequence (and eventually preliminary deterministic analysis).
- Evidently, a re-quantification of the event tree may be required after such review.

4.2 Specification of the safety variables (step 3)

For the selection of the safety variable and the corresponding acceptance criterion, consideration should first be given to the decision problem to be addressed and second to the required complexity of the analysis versus the necessary level of accuracy for the evaluation.

4.3 Deterministic scenario evaluation (transient simulation) (step 4)

The Best Estimate Plus Uncertainty (BEPU) approach is adopted to evaluate the probability of the safety variable above the acceptance criterion. BEPU evaluations are now widely applied worldwide, see e.g. OECD/NEA BEMUSE [6]. Access to a sufficiently detailed and well validated plant model is crucial.

4.5 Quantification of change of safety margins (step 5)

A change of safety margins is expressed by a change of the frequency to exceed the acceptance criterion of interest, e.g. PCT. The conditional probability of exceedance evaluated with transient simulation tools will be combined with the scenario frequencies evaluated with PSA tools. This can be achieved by including the conditional probability of exceedance as a new (so-called thermal-hydraulic) header into the event trees.

6 A. de Crécy et.al., “Uncertainty and sensitivity analysis of the LOFT L2-5 test: Results of the BEMUSE programme”, *N uclear Engineering and Design* **238** (2008) pp. 3561-3578.

5. Approach and Implementation of the SMAP framework

As an initial step, the SM2A participants agreed that an appropriate reference case would be a hypothetical power up-rate of 10% for a typical PWR. The ZION NPP was chosen as reference plant with the following rationale: (1) The ZION NPP is in permanent shutdown, thus no implications to licensing issues are expected; (2) The ZION NPP does not have any directly comparable sister plants still in operation; (3) A major part of the BEMUSE programme of the CSNI working group on Analysis and Management of Accidents (WGAMA) is based on LOCA analysis for this plant, and therefore, plant models are available for various thermal-hydraulic analysis codes; (4) A PSA exists for ZION NPP, although its documentation as available to the participants is rather limited and its level of detail falls a bit short of today's standards for PSA.

At the beginning, there was consensus among the SM2A members that at least two safety variables should be analyzed, namely peak clad temperature, (PCT) for low frequency high damage risks and departure from nucleate boiling ratio (DNBR) for high frequency low damage risks. However, the structure of the event trees strongly depends on the particular acceptance limit whose exceedance frequency is being calculated. Moreover, even the initiating events to be considered and, therefore, the number and nature of the event trees could change. For this reason, it was considered necessary to limit the scope of the SM2A exercise in such a way that changes to the available PSA event trees were minimized. The Group therefore decided to only analyze the impact of the 10% power up-rate on the PCT limit (1 204°C) exceedance frequency.

It was expected that only a rather limited set of scenarios will at the same time be impacted by the postulated 10% power up-rate and reveal a significant contribution to exceedance frequency (above a threshold value). Sequence frequencies were compared with the threshold value, regardless of their associated consequence (either Failure or Success). In the frame of the SM2A exercise, the threshold has been arbitrarily chosen to $1.E^{-7}$/yr with respect to PCT acceptance criterion as a surrogate to Core Damage. Then, this requires quantifying sequence frequencies, refining the event tree modelling as necessary to make each PSA sequence correspond to a unique non-ambiguous simulation scenario. Following this procedure, 21 sequences were selected as summarized in Table 1.

Each of the selected sequences was simulated using best-estimate methods and uncertainty (BEPU) (except for the L CC/SW sequence for which a "Damage Domain" approach was adopted). For these simulations, a rather simplified power up-rating scheme was assumed with unchanged cold leg temperatures and unchanged coolant pump capacity and using a simplified (three-ring) core model. In addition, and as far as BEPU simulations are concerned, a core of uncertain parameters as well as their uncertainty distributions and ranges were agreed upon and adapted to the calculated sequences by the participants. All these calculations focus on the evaluation of the PCT as surrogate of the Core Damage before power up-rate (at 100% power) and after power up-rate (at 110% power). The change of safety margin is then reflected in each analyzed scenario in the change of the conditional probability to exceed 1 204°C. The conditional probabilities of exceedance were thereupon combined with the sequence frequencies evaluated by PSA tools to provide the integral values of the limit exceedance frequency.

PSA Initiating Event	Org	# Seq.	Transient Simulation Tool	Org	Uncertainty Tool	Sampling Method
LBLOCA	EDF	4	CATHARE / ATHLET	EDF/NRI	OPEN-TURNS	Monte Carlo (MC)
MBLOCA	PSI	2	TRACE	PSI	SUSA	MC and biased MC
SBLOCA	PSI	1	TRACE	PSI	SUSA	MC and biased MC
Loss of Offsite Power + Seal LOCA	CNSNS	2	CATHARE	IRSN	SUNSET	MC
Main Steam Line Break + Loss of Feed Water	STUK	1	ATHLET	GRS	SUSA	MC
Steam Generator Tube Rupture	KAERI / KINS	3	MARS	KAERI	MOSAIQUE	MC
Turbine Trip	JNES	5	RELAP5	JNES	Ad-hoc	Latin Hyper Cube Sampling
Loss of Main Feedwater	NRC	2	TRACE	US NRC	MOSAIQUE	MC
Loss of Service Water / Component Cooling Water	CSN	1	MAAP	CSN	Ad-hoc	Uniform Grid

Table 1: Simulation and Uncertainty tools applied for SM2A pilot study

6. Results and their significance - lessons learned

The SM2A study addressed the impact of a substantial plant modification on safety margins, considering a range of safety-relevant scenarios. It implemented the proposed SMAP framework for the first time and on a large scale and demonstrated the capability to evaluate the plant modification's impact on safety margin by estimating the change of the CDF as well as revealing the change of the distribution of a safety variable. In addition, the SM2A study represents one of the first and largest-scale applications of BEPU calculations. The campaign addressed nine types of accident scenarios and included hundreds of calculations.

Please note: Generic conclusions concerning a change in safety margin as a consequence of a power up-rate of 10% cannot be derived because the SM2A exercise represents but a simplified pilot study.

The main results are given in Table 2 below. From the table, it can be seen that changes in CDF were found to be larger than the assumed cut-off criterion of 10^{-7}/y for 3 scenarios. Two of them include an operator intervention. The next four scenarios showed changes in CDF that were below the 10^{-7}/y criterion. Among them are large LOCA, loss of offsite power + seal LOCA and steam line break + loss of FW.

In this particular study, significant differences have been noted between the LBLOCA and MBLOCA results; these have been evaluated and explained:

- The LBLOCA initiator frequency was reduced (~2 10^{-9}/r.y compared to the 6.3 10^{-6}/r.y in NUREG/CR-4550) and is also based on a lower break size limit of 14" whereas [7] used a lower limit of 6".

7 M. B. Sattison, K. W. Hall, "Analysis of Core Damage Frequency: Zion, Unit 1, Internal Events", NUREG/CR-4550 Vol.

- In addition, the LBLOCA frequency was updated based on [8]. This reduced the LBLOCA frequency from 9.4 10^{-4}/r.y to 4.8 10^{-7}/r.y

- Furthermore, the stochastic treatment of the LBLOCA break size in the range 14"-30", with double ended guillotine break (DEGB) assumed above 30" further decreased the LBLOCA CDF. Only 1 (of 150) cases led to exceedance of the PCT limit and this case was one of the DEGB, which only made up 15% of the cases. Therefore, the contribution of sequences related to recirculation failure falls far below the cut-off frequency of 10^{-7}/r.y and was therefore not considered in this study.

- In contrast, the MBLOCA calculation applied a limiting break size value (10") for the range 2"-10". The related ΔCDF reflects the effect of the power up-rate on the sequence dynamics and on the time to switch to recirculation.

For the sequences with small conditional probability of exceedance, a much larger number of calculations may be needed in order to estimate this probability with a sufficient level of confidence. Several pragmatic solutions were proposed by the participants in order to reduce this number to a feasible value.

7, (1990).

8 R. Tregoning, L. Abramson, P. Scott, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process, NUREG-1829 Vol 1. (2008).

Table 2 Overview of impact on CDF, conditional exceedance probability and ΔCDF by scenario

Impact on CDF/PCT (1)	Scenario	Section	Sequence frequency (/r.y)	Conditional exceedance probability 100% power	Conditional exceedance probability 110% power	CDF 100% power	CDF 110% power	ΔCDF
Increase > 1E-7/r.y	Loss of offsite power + loss of FW	5.2.1.2	2.7E-5	0.104	0.122	2.81E-6	3.29E-6	4.8E-7
	Loss of service water/component cooling water	5.2.3	1.88E-3			1.34E-6	1.46E-6	1.18E-7
	MBLOCA (2)	5.2.5	9.4E-4	0.123	0.286	8.39E-7	1.64E-6	8.01E-7
Increase << 1E-7/r.y	Loss of offsite power + seal LOCA	5.2.1.1	1.7E-6	-	-	-	-	<<1.E-7 (3)
	Steam line break + loss of FW (4)	5.2.2	2.3E-7 4.35E-11	0.34	0.81			<<1.0E-7
	LBLOCA (5)	5.2.4	4.8E-7			2.49E-9	2.52E-9	2.8E-11
	SBLOCA	5.2.6	1.61E-5	-	-	-	-	<<1.E-7 (3)
ΔCDF No increase. Only a shift of PCT	Turbine trip	5.2.9	?	0.0	0.0			0.0
No ΔCDF No shift of the PCT	Loss of main feedwater	5.2.7	1.0E-7	0.0	0.0			0.0
	SGTR	5.2.8	2.793E-7	3.33E-3	3.33E-3	9.30E-10	9.30E-10	0.0

1. The participant's contributions of the individual analyzed sequences are reported in the Appendix 5.
2. For MBLOCA, the conditional exceedance probabilities reported apply to the sequence conditioned on the timing of recirculation (stratified analysis). The stratum of interest (between definite failure and definite success) has a conditional probability of 5.24E-3; in other words, the conditional exceedance probability applies to a sequence with probability 4.9E-6.
3. No uncertainty evaluation performed, ΔCDF supposed very low.
4. Sequence frequency 2.3E-7/r.y is to be multiplied by the conditional probability of loss of 1 centrifugal charging pump, 1 safety injection signal & 1 power operated relieve valve (the only scenario which leads to different exceedance probability and contributes to "delta core damage frequency").
5. For LBLOCA, the distribution of the break size is explicitly treated in the analysis (SBLOCA and MBLOCA use limiting break sizes).

From the SM2A study, a number of lessons can be learned, the most important ones being:

1. The application of the SMAP framework requires intensive and close interactions between the probabilistic and transient simulation experts. A quick look at the SMAP steps could suggest that sequence selection is primarily a PSA issue and the transient simulation is a subsequent step. However, it was found that the expertise in transient simulation was essential in the sequence selection and that probabilistic expertise was also important in the transient simulation step, both to a much greater degree than initially anticipated.

2. In current NPP PSAs, the event trees typically represent thousands of sequences. However, the evaluation of changes to safety margin does not require analyzing all of these sequences as changes in the monitored safety variable are to be expected for only a small subset of the sequences. Therefore, the efficient identification of this subset becomes very important. In the SM2A study, a simple cut-off criterion for the sequence-frequency 10^{-7}/yr was applied, which led to a focus on those sequences where the magnitude of the change of safety margin could be a significant contributor to the overall change of safety margin.

   Due consideration must also be paid to the screened-out sequences. It must be shown that their possible cumulative contribution stays within reasonably “small” bounds.

3. The sequences most affected by a plant modification and contributing to a change of the CDF are, in many cases, not the dominant contributors of the baseline PSA. The PSA dominant contributors may include failure events and probabilities that are not or negligibly affected by the plant modification; as a result, the SM2A calculations of these sequences would not be expected to yield a significant contributor to the change of the CDF. (At the same time, the PSA dominant contributors would be expected to remain dominant contributors also in the up-rated plant.)

4. Related to the additional sequence selection criterion, it was also found that triangular or uniform interval distributions should not be adopted for time delay distributions for application in the SMAP-framework. As these types of distribution truncate the distribution tail(s), an underestimation of ΔCDF is the likely result. Therefore, the selection of the time delay distributions becomes quite crucial.

5. The intricate combination of PRA techniques with transient simulation may require delineating sequences with only “discrete” system configurations that can be represented one-to-one in the simulations. Furthermore, thermal-hydraulic modeling considerations will influence the delineation of the event sequences: The injection of one of the redundant safety injection systems may have to be assumed lost because the injected coolant can leave the primary system through the assumed break before reaching the core, depending on the assumed break location. Hence, one injection system has to be assumed failed due to this effect even if all ECCS are assumed available physically. (Note that modern PSAs may already consider this type of interaction between probabilistic analysis and transient simulation.)

   This lesson can readily be generalized: it is possible that an increased number of redundant trains of systems need to be available after the plant modification to properly mitigate the consequences. For this reason, minimal configurations must be carefully checked against possible modifications of the number of minimally required safety systems also after plant modification.

6. The introduction of a header for the conditional probability of exceedance (so-called “TH” header) into the different sequences can be useful for a straight-forward aggregation of the frequency of exceedance for a set of sequences analyzed, using the same PRA-tool that was applied for establishing the event tree. This additional header enables one to introduce a supplementary branch in the event tree corresponding to the fraction of a success branch which

is in fact a failure branch when taking into account TH uncertainties, and this "TH" failure branch could be particularly impacted by the plant modification. It should be noted that for quantification this method works only if the assumption of mutually exclusive event sequences can be justified.

7. Particular attention has to be paid to sequences that include a time delay for a recovery action or for a repair. These sequences could be impacted directly by the plant modification, e.g., due a change in the available time window, or indirectly, as a result of the more realistic modeling of the sequence used in SM2A. This lesson is based on the SM2A Zion-specific results, where such sequences contribute to the more visible risk increase:

   - LOSP: LOOP followed by AFWS failure and no power recovery before core damage.
   - MBLOCA: MBLOCA followed by SI success but recirculation failure (operator action).
   - LOSS of ESWS/CCWS: Seal LOCA and no recovery operator action before core damage.

   Consequently, the probability distributions used to characterize such time delays require particular attention.

8. Some of the analyses demonstrated how a point-estimate calculation (performed with the best-estimate values of the parameters) and an uncertainty analysis could lead to different conclusions for a sequence. For instance, the nominal power point-estimate calculation for the Steam Line Break showed no exceedance of the PCT criterion while the uncertainty analysis resulted in a conditional probability of exceedance of 34%.

7. Conclusions and recommendations

In terms of technical feasibility of the SMAP framework, the following conclusions can be drawn:

1. An increase of the (conditional) probability of exceedance for a surrogate acceptance limit (PCT) indicates that core damage was successfully evaluated for the selected sequences from several initiating event trees.

2. During the SM2A study, several areas of refinement or even enhancements to the SMAP framework were identified:

   – A better sequence screening process needs to be elaborated that provides for a sound bases for the screening criteria.

   – It was found that some of the event trees of the existing PSA had to be reformulated (expanded) in order to represent only scenarios with a configuration of the systems that can be directly represented in transient simulation codes.

   – The introduction of dedicated headers for the conditional probability of exceedance (so-called TH headers) for taking into account TH uncertainties can ensure a consistent aggregation of the results from several sequences of different initiating event scenarios.

   – While the final SMAP report indicated separate treatment of aleatory and epistemic uncertainties as very desirable, no workable procedure for computationally intensive problems was proposed. The SM2A group did not work on this topic. While the treatment of the different types of uncertainties remains an issue of significant controversy in the open literature, the group judged that a systematic separate treatment is not a requirement for the current application focusing on <u>changes</u> of safety margin. Accordingly, the

SM2A results consist of expected values for the changes, in most cases without an uncertainty distribution for the change of the safety margin.

3. An obvious key conclusion from this exercise is that a successful application of the SMAP framework needs an intensive interaction and cooperation between the PSA and transient simulation specialists from the very beginning of the study.

4. The modelling of human actions has been found to be of particular importance, as the sequences related to scenarios that require a time delay to perform a recovery action or a repair correspond to the more visible risk increase. In particular, the more detailed analysis performed with the SMAP framework requires improvement regarding the estimates of both the time window available (through the BEPU transient analysis calculations) and the time for response or recovery (in determining the response/recovery time distributions).

5. The effort required for performing a consistent study for assessing the impact of a significant plant modification on safety margin has been estimated to a few person-years, assuming the availability of the analytical infrastructure including computer codes, a modern PSA and plant specific input models, statistical tools and adequate computing and engineering resources. However this estimate needs to be confirmed taking into account the number of assumptions which were made, and the open issues which remain. In any case, such an effort is hence only warranted for significant plant modifications with high relevance to safety.

The following recommendations can be derived from the results of this study:

1. The group of experts managed to build a bridge between analysts working with deterministic and transient analysis methods and those working with probabilistic methods. A suitable framework for such collaborations needs to be identified.

2. A number of limitations of the current SM2A approach have been identified. They should be addressed in future work. The most notable of these are:

   – Consideration of several safety variables and acceptance limits; therefore involving a wider spectrum of transient simulation codes would have to be included: 3D-kinetics, fuel behavior, containment behavior or possibly severe accident codes.

   – When considering several safety variables and acceptance limits, an appropriate PSA is needed, by modifying an existent study or adding specific developments.

   – There is a need for development of specific sampling or discretization methods for addressing uncertain delay times for a more efficient coverage of the problem space.

   – Development of effective evaluation strategies for the probability of exceedance via biased sampling strategies is required.

   – Development of compute-effective treatment for both aleatory and epistemic uncertainties is required.

   – Integration of dynamically dependent probabilities (e.g., probabilities of delay times) into the simulation models is needed.

As the application of the (revised) SMAP framework involves a large scale BEPU campaign, improvements to the framework can possibly also stimulate new approaches to BEPU:

1. The consideration of several barriers and safety variables requires adopting different codes, e.g. a system thermal-hydraulic code chained to a containment code or to a severe accident code. This asks for an efficient BEPU-procedure for chained codes.

2. On the background of the large scope of an evaluation of plant safety margin changes, the development of computational tools should allow for automating the analysis to a high degree. Obvious candidates are discrete dynamic event tree (DDET) tools to address the automatic identification of sequence branching points and the respective sequence simulations. Its combination with BEPU may well pose new challenges: With the perturbation of the uncertain parameters, transition from one sequence to another one may occur, e.g. by reaching trip levels. Therefore, adequate number of simulation cases (as stipulated by the Wilks formula) allocated to an individual sequence must be ensured. This requires reconsidering the sampling strategy in the space of the uncertain parameters and the different sequences if one does not want to simply rely on a brute force approach and running cases until adequate number of cases have accumulated for each single sequence.

3. The evaluation of scenarios of long duration could benefit from employing a set of plant (input) models with graded levels of modeling detail for different phases of the transient: Instead of simply relying on the ever increasing compute power, a set of models staged in modeling detail would help to not only reduce the computational effort, but also would reduce the amount of data generated. Furthermore, restrictions on the time step size could possibly be lessened (in case the numerical solution scheme is not implicit in time).

4. Compared to the range of the phenomena that needs to be considered for design-basis accident evaluations, safety margin evaluations have much larger demands as they are concerned with a much larger range of phenomena. In this respect, a significant problem arises: Is the respective experimental basis adequate for the proper code validation and - related to it - the uncertainty assessment for the extended range of phenomena? As a consequence, collection of (evaluated) experimental data into publicly available data bases (as e.g. the criticality handbook of NSC) appears as an important desideratum if not a necessity. Good starts are seen in the SET and IET collections of CSNI CCVM databases, and the new OECD projects all are providing modern data sets that after a protection period become publically available.

   Engineering judgment, ideally combined with a systematic expert solicitation process, could fill the gaps in uncertainty information where experimental evidence is lacking. Likely, it will be an important source of uncertainty characterization especially for phenomena occurring during low-frequency sequences.

8. Acknowledgments

This paper mainly reports the results elaborated by the SM2A expert group and documented in it's final report [2]. Therefore, credits belong to all the participants in the SM2A expert group, and the members of the writing group of the final report deserve especial mentioning: Vinh N. Dang (PSI), Jeanne-Marie Lanore, Pierre Probst (IRSN), Javier Hortal (CSN) and Abdallah Amri (NEA).

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Combining Insights from Probabilistic and Deterministic Safety Analyses in Option 4 from the IAEA Specific Safety Guide SSG-2

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Abstract

In 2009 the International Atomic Energy Agency (IAEA) has published in their Safety Standards the IAEA Specific Safety Guide SSG-2 on Deterministic Safety Analyses for Nuclear Power Plants (NPP). This Guide addresses four options for the application of deterministic safety analyses. The first option using conservative codes/models and conservative initial and boundary (I&B) conditions has been used since the early days of civil nuclear power and is still used today. The Option 2 is using realistic codes/models but with conservative I&B conditions, and widely used world-wide. Option 3 consists of realistic codes/models and realistic I&B conditions and therefore need also to consider the associated uncertainties. It is today known as Best Estimate plus Uncertainty (BEPU) option.
Option 4 is not developed in the SSG-2 document. It is only indicated that Option 4 is an attempt to combine insights from probabilistic safety analyses with deterministic approach. In Options 1 – 3 availability of safety systems is based on conservative assumptions whereas in Option 4 the availability of safety systems is derived using probabilistic means.

This paper explains in more detail the approach proposed for Option 4, recognizing the fact that this option is still a research option and will remain so for some time.

1. Introduction.

Recently, the International Atomic Energy Agency (IAEA) has produced guidance on the use of deterministic safety analysis for the design and licensing of nuclear power plants [1]. This guidance addresses four options for the application of deterministic safety analysis as summarised in the table below.

Table 1. Options for combination of a computer code and input data.

Option	Computer code	Availability of systems	Initial and boundary conditions
1. Conservative	Conservative	Conservative assumptions	Conservative input data
2. Combined	Best estimate	Conservative assumptions	Conservative input data
3. Best estimate	Best estimate	Conservative assumptions	Realistic plus uncertainty; partly most unfavourable conditions[a]
4. Risk informed	Best estimate	Derived from probabilistic safety analysis	Realistic input data with uncertainties[a]

[a] Realistic input data are used only if the uncertainties or their probabilistic distributions are known. For those parameters whose uncertainties are not quantifiable with a high level of confidence, conservative values should be used.

The conservative Options 1 and 2 have been used since the early days of civil nuclear power and are still widely used today. However, the desire to utilise the current understanding of the important phenomena and to maximise the economic potential of nuclear plants without compromising their safety has led to many countries using Option 3, i.e. using best estimate codes and data together with an evaluation of the uncertainties.

Using the Best Estimate Plus Uncertainty (BEPU) approach for the analysis of a particular Design Basis Accident (DBA) leads to a distribution of code predictions/results for the most limiting value of the safety variable (for example the peak clad temperature during a transient). This distribution is a consequence of uncertainties in the initial and boundary conditions as well as in the computer model.

On the other hand, the distribution of failures values of the safety variable i.e. value of the variable where barrier fails, is a consequence of the fact that failures are random and our knowledge of the precise phenomena that can cause the failure is limited.

If these distributions do not significantly overlap, there will be a negligible failure probability for the analysed DBA. If only best estimate values were used, there would be an apparent margin as illustrated in Figure 1. However, if load and capacity distributions overlap there could be a significant failure probability while the apparent margin is maintained.

A common solution in licensing analyses is to replace each probability distribution by a bounding value of the safety variable. For the capacity distribution, a lower bound is set at or below the start of non-negligible failure probability. This value is called *safety limit* and, in many cases, it is the licensing criterion itself.

For the load distribution, Options 1 and 2 replaced the distribution by a single value obtained with conservative assumptions, with the hope that this value would be an upper bound of the actual distribution. In Option 3, however, the upper bound is set at a high percentile of the distribution, typically 95, covered with a high confidence level, typically 95%.

In any case, the upper bound of the load must be below the licensing acceptance criterion, i.e., below the lower bound of the capacity. The distance between these two values is called *licensing margin*, as shown in Figure 1.

In order to illustrate the way in which the available licensing margin is expected to increase as one goes from Option 1 to Option 3, we compare the results of the upper bound of the uncertainty band for Option 3 with the results of calculations using Options 1 and 2. This is illustrated in Figure 2.

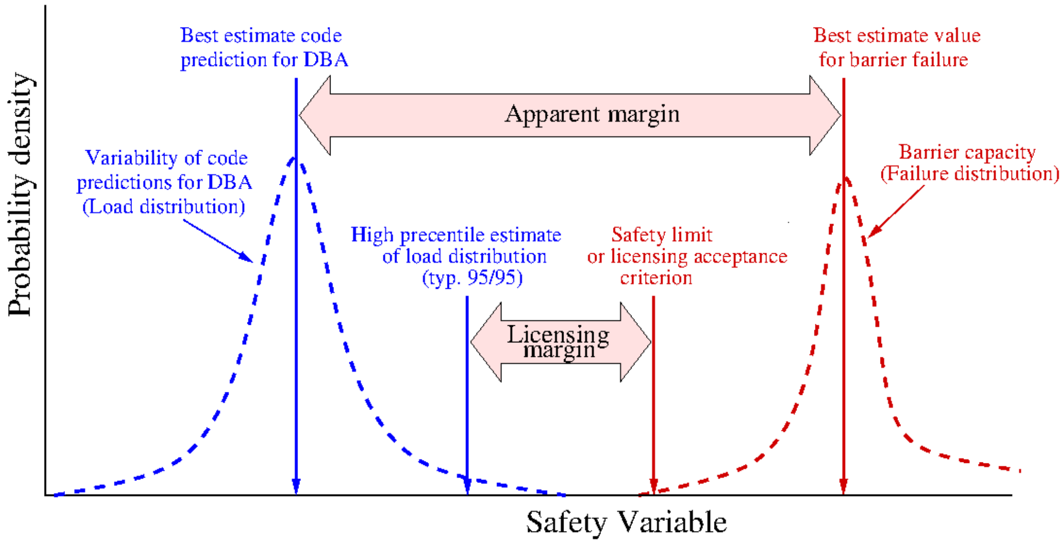


Figure 1: Probability density functions for load and strength /capacity

Option 3 includes a detailed evaluation of the uncertainties, and therefore, several calculations are performed to obtain the distribution of the safety variable and the uncertainty range. The Safety Guide SSG 2 recommends that the value that should be compared with the acceptance criterion is the value that encompasses 95% of the range of uncertainty with a 95% confidence level (95/95 value). This is illustrated in Figure 3.

Thus, changes can be made to the plant provided that the 95/95 value does not exceed the acceptance criterion.

The common feature of Options 1 to 3 is that conservative assumptions are made about the availability of redundant trains of safety related systems. Deterministically, some trains are assumed to be available and others are not. For example, for a sequence such as a Loss Of Coolant Accident (LOCA), where the main protection is provided by safety injection pumps, the typical approach in Option 3 is to assume that only one pump is available (for simplicity, we assume one pump per train).

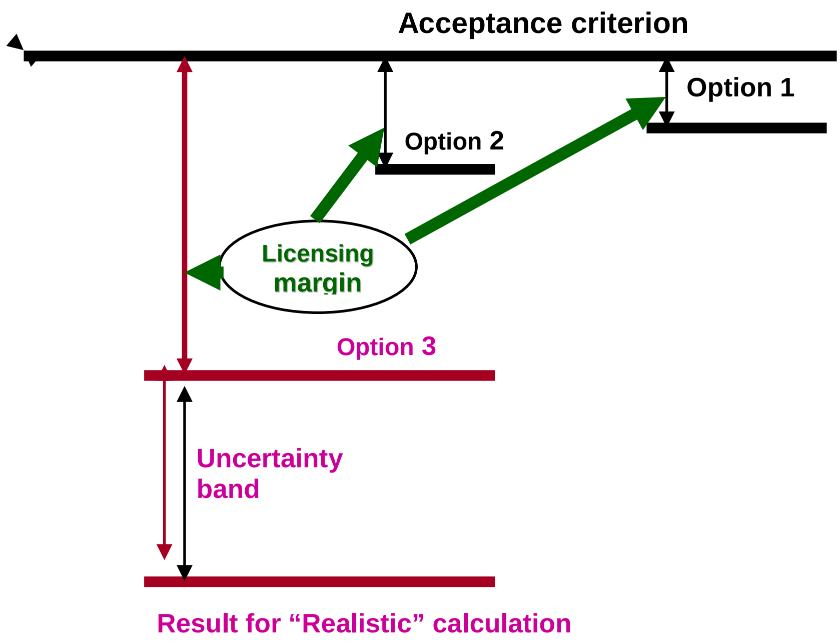


Figure 2: Illustrative Licensing Margins for Different Options

The purpose of Option 4 is to provide a further level of flexibility, by using probabilistic arguments, to take credit for the probability that a system which was deterministically excluded in Options 1 to 3 is actually available.

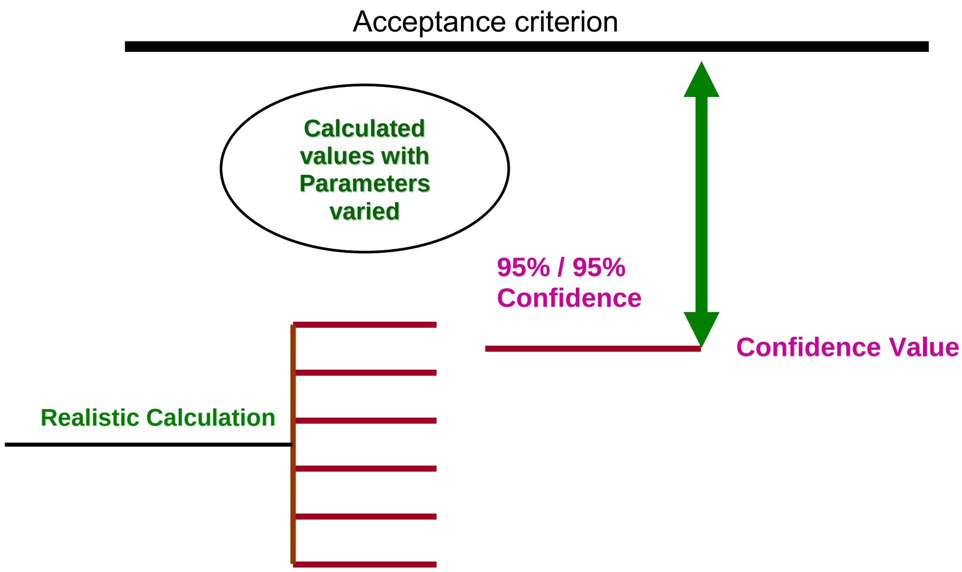


Figure 3: Illustrative Design Margin for Option 3

The philosophy of Option 4 is already embedded in the USA and German guides ANSI/ANS 51.1 [2] and KTA-SG -47 (Draft) [3]. These guides allow the use of different acceptance criteria for different sequences starting from the same Postulated Initiating Event (PIE) if their frequencies are significantly different and under some specified conditions (Table 2).

Table 2. ANSI/ANS 51.1 Methodology for Determining the Plant Condition of an event

PC of IO	PC Assuming CO > 0.01/IO	PC Assuming a CO ≤ 0.01/IO or SF	PC Assuming both CO > 0.01/IO and a SF	PC Assuming both a CO ≤ 0.01/IO and SF
2	2	3	3	4
3	3	4	4	5
4	4	5	5	5
5	5	5	5	5

PC: Plant Condition
IO: Initiating Occurrence
CO: Coincident Occurrence (or combination of coincident occurrences)
SF: Single Failure
CO > (or ≤) 0.01/IO means that the CO has a probability of occurrence, concurrently with the IO that is > (or ≤) than 0.01

An illustration of acceptance criteria that are related to the frequency of the sequences following each Postulated Initiating Event is given in Figure 4 below (from ref 2):

Figure 4 shows that, as in previous approaches to licensing analysis, the dose limit increases as the transient frequency decreases. However, the classification of a transient is not determined only by the frequency of the initiating event but also by possible additional events such as coincident events or single failures.

Even though this curve relates to the radiological dose limits, it can also be related to acceptance criteria such as Departure from Nucleate Boiling Ratio (DNBR), Boiling transition, number of failed rods, peak clad temperature applicable to different plant conditions, etc.

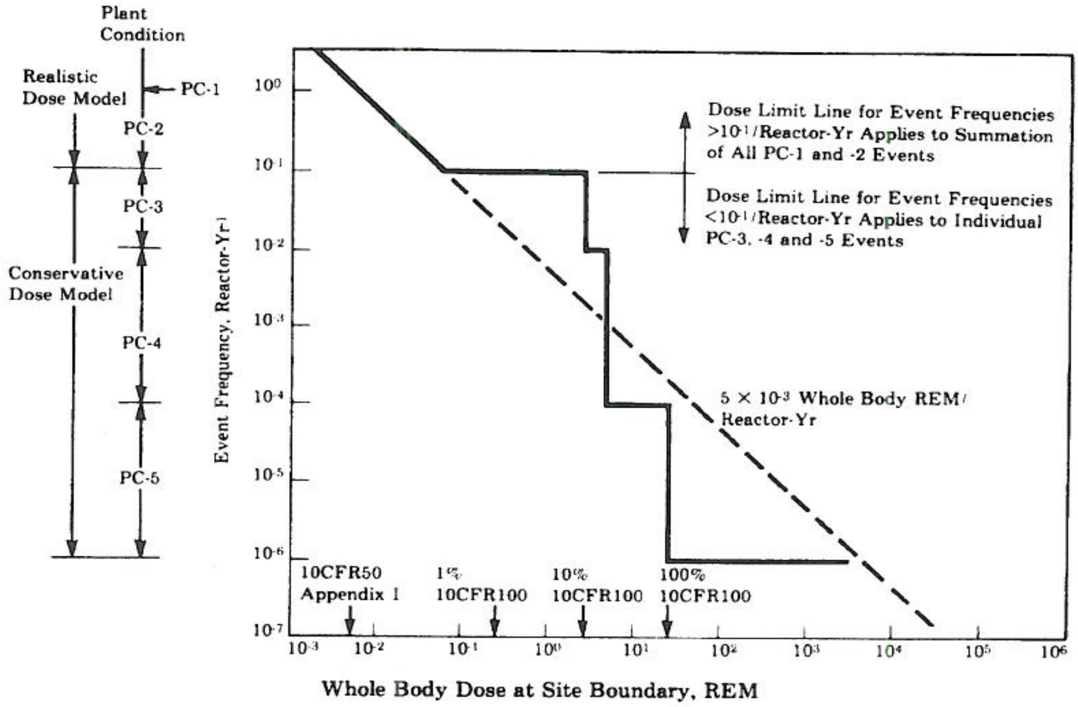
2. Option 4: definition and analysis.

At the present time, Option 4 is on the drawing board, pending of further development of both regulatory and technical aspects. It is premature to analyze the potential regulatory implications in detail, so we remain at the level of compatibility with the applicable fundamental safety principles as established in [4], in particular principle 8: “prevention of accidents” where “defence in depth” plays a pivotal role i.e.[4, § 3.31]*”defence in depth ensures that no single technical, human, or organizational failure could lead to harmful effects, and that the combinations of failures that could give rise to significant harmful effects are of very low probability”*. Similarly, [4, § 3.30] states the need to *“ensure that the likelihood of an accident having harmful consequences is extremely low, measures have to be taken:*

- To prevent the occurrence of failures or abnormal conditions (including breaches of security) that could lead to such a loss of control;

- To prevent the escalation of any such failures or abnormal conditions that do occur;

- …”.



Offsite Radiological Dose Criteria for Plant Conditions

Best-Estimate Frequency of Occurrence (F) Per Reactor Year	Plant Condition (PC)	Offsite Radiological Dose Criterion
Normal Operations	PC-1	10 CFR 50, App. I
$F \geq 10^{-1}$	PC-2	10 CFR 50, App. I
$10^{-1} > F \geq 10^{-2}$	PC-3	10% 10 CFR 100
$10^{-2} > F \geq 10^{-4}$	PC-4	25% 10 CFR 100
$10^{-4} > F \geq 10^{-6}$	PC-5	100% 10 CFR 100

Figure 4: Dose Limit for Whole Body at Site Boundary

In the following paragraphs, it will be explained what Option 4 is and the interplay of its different elements. Also, its potentiality will be analysed. In a separate section some examples of application are described.

As already indicated, the purpose of Option 4 is to integrate the use of probabilistic arguments for system availability into the DBA licensing requirements. This very general statement suggests that the transition from Option 3 to Option 4 is, to some extent, analogous to the transition from Option 2 to Option 3. Option 3 was intended to have a measure of uncertainties and consequently to remove excessive conservatisms from Option 2. This was done by allowing replacement of conservative values of some model parameters, initial or boundary conditions by realistic though uncertain values. As a consequence, and in order to ensure that "the likelihood of an accident having harmful consequences is extremely low", uncertainty analysis was required resulting in the fulfilment of acceptance criteria with a high degree of certainty (95/95). This, in turn, implies that some unlikely (typ. <5% probability) combinations of parameter values result in plant transients initiated by a PIE that violate the acceptance criteria of the class to which the PIE belongs. Or in other words, an escalation of the accident consequences is marginally acceptable, that according to the above mentioned IAEA SF.1. § 3.30 [4] should have an extremely low likelihood.

Option 3 still retains important conservative assumptions, most notably, those regarding to system availability. In fact, in Options 1 to 3, a DBA is defined not only by a PIE but by a combination of a PIE, a single failure and possibly other requirements such as loss of off-site power. The class where a DBA is classified has been traditionally the class of its initiating event [8] and the applicable acceptance criteria have been those defined for that class

Following the path of the progression from Option 2 to Option 3, it seems that an obvious progression towards Option 4 would be to include the availability of safety systems in the uncertainty analysis. This way, a DBA would be defined by a PIE only and the probability of exceeding the class acceptance limits would be calculated not only from the probabilities of uncertain elements used in Option 3 but also from the probabilities of system configurations. If the acceptance criterion (e.g., 95/95) is fulfilled, the analysis of the DBA would be acceptable. Note that the allowed 5% probability of limit exceedance may include single failure sequences.

Full application of Option 4 will then allow for relaxing the single failure criterion in cases where the event sequence has a very low probability but, on the other hand, it will provide the possibility of including multiple failures with high likelihood of occurrence.

Option 4, as proposed in this paper, includes this extension of Option 3 to system availability, but it is not limited to that. In order to maintain an acceptable level of Defence-in-Depth (DiD), additional requirements should be imposed for that fraction of cases with or without single failure of one system, going beyond the acceptance criteria corresponding to their PIE. The basic idea is that the consequences of such transient should be limited.

By allowing additional failures, coincident or subsequent to the initiating event, occurring with probabilities $0 < P < 1$, new sequences appear starting from the PIE and the frequency of each sequence can be calculated from the frequency of the initiating event and the probabilities of the additional failures. Consequently, a DBA is no longer a single sequence and the frequency of an accident sequence differs from the frequency of the PIE from which it starts.

The application of Option 4 to a particular initiating event is explained below. In this description, we will use the term "*sequence*" for referring to an ordered set of events e.g. systems actuations, starting from the PIE. A sequence does not correspond to a single simulation. Within each sequence the uncertainty analysis is performed by selecting different combinations of values of the uncertain parameters and each combination of parameter values will result in different simulation results. We will refer to a particular choice of parameter values, as a simulation run of the sequence.

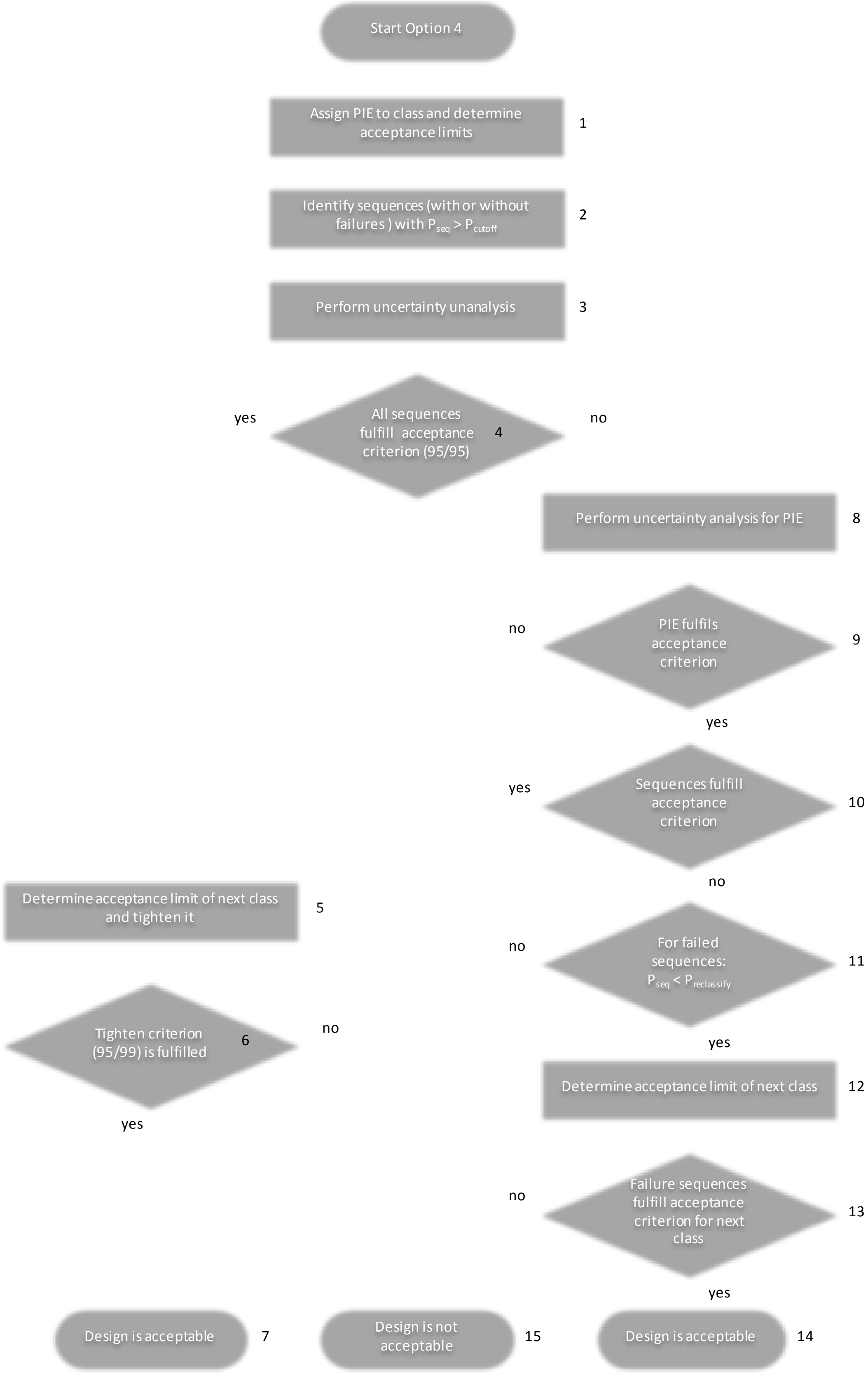


Figure 5: Flow chart of Option 4

The flow diagram of Figure 5 illustrates the application of Option 4. First, (block 1), the initiating event is classified in some accident category as in Options 1 to 3 and the acceptance limits of that category are identified. The sequence with no additional failures is also included in the category of the initiating event.

The extended uncertainty analysis is then performed in the following way:

1. Identify possible sequences starting from the initiating event, with or without additional failures, whose conditional probability is above a specified value P_{cutoff}. Calculate those probabilities (block 2).
2. Perform the uncertainty analysis of each sequence as in Option 3 (block 3).

If ALL the sequences fulfil the acceptance criterion (typically 95/95), it is guaranteed that the acceptance criterion for the PIE is also met. Still, we impose additional requirements to ensure that the 5% of the possible sequence simulation runs which is allowed to exceed the acceptance limits does not result in harmful consequences i.e. cliff edge effect. This corresponds to the "yes" exit of the question block 4. Blocks 5 and 6 indicate a possible way to implement these additional requirements. It consists of taking the acceptance limits of the next class and tightening them to get intermediate limits between the two classes. Then, from the uncertainty analysis, demonstrate with a high degree of confidence (e.g., > 95%) that the probability of remaining below the tightened limit is very high (e.g., > 99%).

For example let's consider a PIE of moderate frequency with an acceptance limit on the expected number of failed fuel rods below 0.1%. Even if the uncertainty analysis shows that more than 95% of the possible sequence cases fulfil this acceptance limit, it has to be shown that the remaining possible cases where the acceptance limit is exceeded, still have limited consequences. If the acceptance limit of the next class is 10% of failed fuel rods, we tighten it and define an intermediate limit of, say, 6%. Then, the uncertainty analysis is revisited and extended if necessary, to show that the probability of having a sequence case with more than 6% of failed rods is very low (<1%). A confidence level higher than 95% has to be ensured.

Going back to Figure 5, it could happen, however, that one or more sequences with additional failures violate the 95/95 (or whatever) criterion. We will call them "damage sequences". In any case, we need to check that the acceptance criterion for the PIE, i.e., for all the sequences together, is still fulfilled. The uncertainty analysis is performed for the PIE in block 8. The result is then compared to the acceptance criterion in block 9, and it is decided whether the design should be improved ("no" output of block 9 going to block 15) or the analysis of individual sequences is performed ("yes" output of block 9).

Each identified sequence is then individually considered. Block 10 discriminates between damage and non-damage sequences. Non damage sequences are analyzed against the tightened acceptance limits of the next class as in the case of absence of damage sequences (Block 5).

For damage sequences, we go through the "no" option to block 11 where the conditional probability of the sequence is compared with a reclassification criterion named $P_{reclassify}$. If the sequence can be reclassified (output "yes") the acceptance limits of the next class are identified and the uncertainty analysis of the sequence is revisited to check for compliance of the 95/95 criterion with respect to the acceptance limits of the next class. Note that the usual single failure requirement is equivalent to setting $P_{reclassify}$ to 0 for single failure sequences. In any case, reclassification of single failure sequences is only allowed to the next category as much.

If ANY reclassified damage sequence does not fulfil the acceptance criterion 95/95 with respect to the acceptance limits of the new class, the design is unacceptable and some modifications need to be done (output “no” of block 13). If ALL the reclassified sequences comply with the requirements of the new class, the design is considered provisionally acceptable. A final check should be performed to ensure that the contribution of reclassified sequences (starting from any PIE) to the cumulative frequency of events pertaining to each class is not excesive. Only in this case the provisional results become conclusive.

3. Use of Option 4

Once Option 4 basics have been exposed, some examples of application are presented. The first one deals with an application in the context of a power uprate which accordingly to Option 4 will require of a plant modification to recover compliance with the original design requirements.

The second example addresses the relation of Option 4 to the proposed amendment to rule 10CFR 50.46 [6] to allow for less prescriptive treatment of breaks on the reactor coolant pressure boundary beyond a so called transition size break.

Finally the potentiality of option 4 to analyze the design basis extension in a structured way is briefly presented for the case of an external event.

3.1 Adoption of compensatory measures following a power uprate.

Option 4 would be used when a required change to the plant results in an acceptance criterion being exceeded with the assumptions that were made for the Option 3 analysis but the acceptance criterion can be met when account is taken of the availability of safety systems on probabilistic grounds.

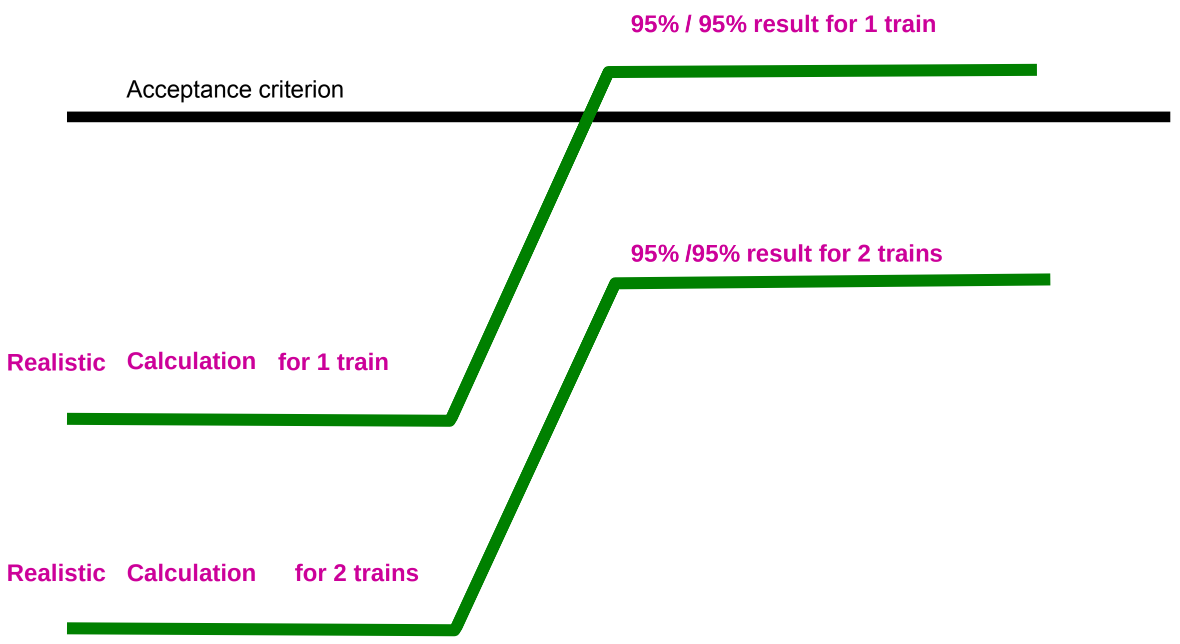


Figure 6: Illustration of the Application of Option 4

Figure 6 above (based on [2]) illustrates an example where plant modifications have been made such that the Option 3 calculation results in the acceptance criterion being exceeded. In the example of a LOCA, only one train is assumed to be available.

This situation would be acceptable if the frequency associated with this sequence is low enough for the resulting damage/dose to remain less than the damage/ dose limit shown in Figure 4.

This can be demonstrated as follows. In Option 3, only one train is assumed to be available, the second one is considered to be in maintenance and the third one is assumed to be unavailable to satisfy the single failure criterion. The probability of one train being available is therefore 1 and the probability of two trains being available is 0. The sequence could be represented by X1 in Fig.7, indicating that for the frequency of the sequence in the range of PC-4 the acceptance criterion is fulfilled.

After modification, the consequences of the sequence are higher than the acceptance criteria for PC-4, moving from X1 to X2.

In Option 4, without changing the requirements for one train under maintenance, both probabilities of having only one or two trains available are calculated and result in values between zero and one. When multiplying the frequency of the initiating event with the unavailability of anyone of two trains of the safety system, the frequency of the sequence decreases compared to that from Option 3 (in Fig. 7 we move from X1 to X3).

After the modification, point X3 moves to the point X4 in the same way as X1 moved to X2 using Option 3. However in this case X4 is within the acceptable area of PC-5 i.e. consequences are below the acceptable damage/dose limit for that category.

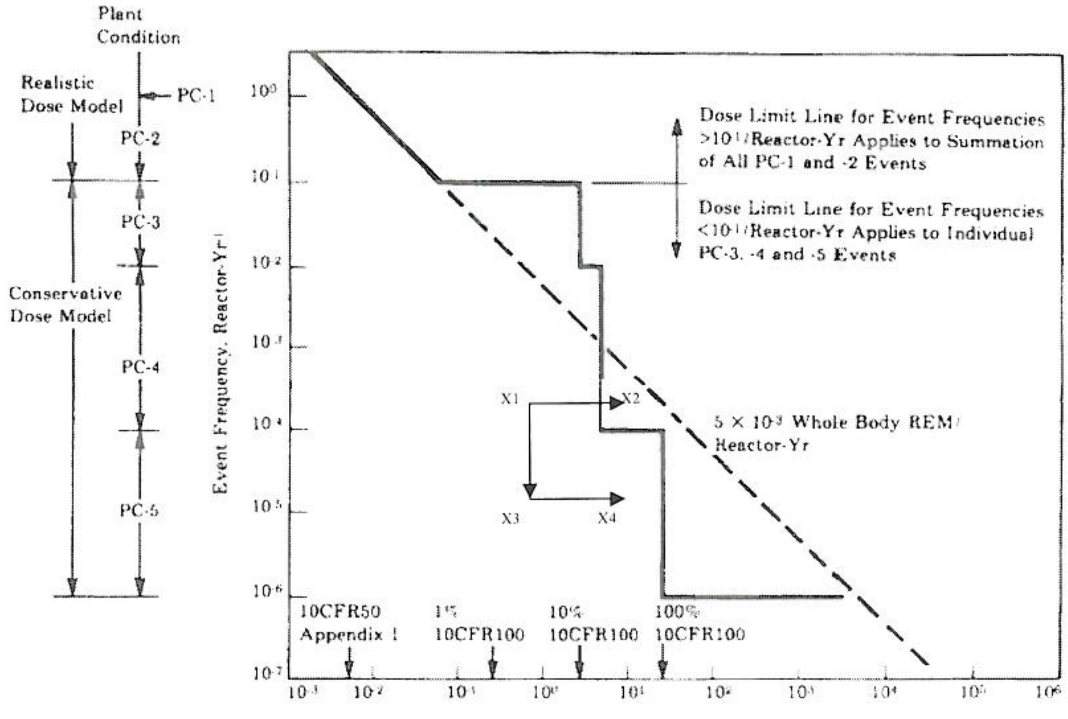


Figure 7: Illustration of an Acceptable Case where an Acceptance Criterion is Exceeded

By multiplying the frequency of the transient with the unavailability of the safety system, no account has been taken of the single failure criterion, which under Option 3 requires the most limiting failure to be considered. Therefore compensation is required in the form of hardware solutions/additions or by using tools and principles from a risk-informed decision making process. In addition, in order to preserve defence-in-depth, the analysis needs to be repeated taking into account the next most unfavourable single failure and demonstrating that the original acceptance criteria are met. The next most unfavourable failure can be determined by looking at the event tree for the initiating event that is being analysed. The event tree will consist of different sequences that can be allocated according to their frequencies to different event classes (plant conditions in Fig. 7). Different event classes have different acceptance criteria. From this event tree, sequences with failures which can be considered as single failures can be identified (see a hypothetical event tree in Fig. 8). The calculations can be repeated with consecutive single failures in order of their importance. These are obtained by performing Option 1, 2 or 3 analyses assuming different single failures and the one with the least safety margin is the most unfavourable one.

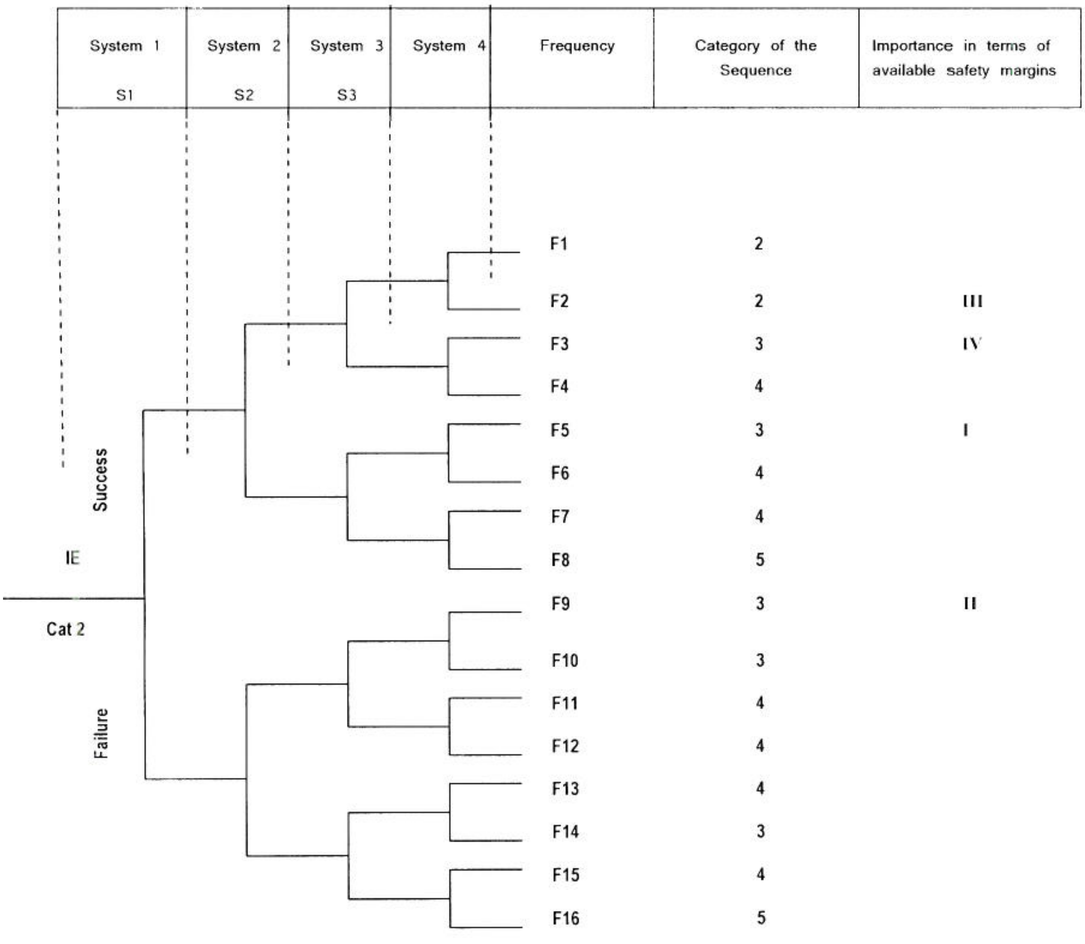


Figure 8: Hypothetical Event Tree

Figure 8 illustrates a hypothetical event tree for an initiating event whose frequency corresponds to Category 2, which is the equivalent of Plant Condition 2 in Figure 7. From Figure 8, it can be seen that a single failure is associated with the sequences that have frequencies f2, f3, f5 and f9. In a purely deterministic approach, all of these sequences must fulfil the acceptance criteria corresponding to the category of the initiating event (Category 2) in order to comply with the single failure criterion. In Figure 8, however they belong to Categories 2 or 3 depending on their frequencies.

The next stage is to rank the four identified sequences with single failures using Option 1, 2 or 3 calculations in terms of the available safety margins and, in the example, the single failure associated with f5 is found to be the most limiting, followed by f9, f2 and f3. Using Option 4, the single failure criterion for f5 is relaxed by applying compensating measures and it is demonstrated that the acceptance criteria for Category 3 are met. It must be shown that Sequence f2 meets the acceptance criteria for Category 2 but it is also necessary to verify that when the single failure criterion is applied to sequences f3 and f9, the acceptance criteria for Category 2 are still met, in spite of the fact that their frequencies correspond to Category 3. If one of them fails to do so, the process can be repeated as for f5 with additional compensatory measures if feasible but, in any case, the acceptance criteria for Category 2 must be met for the single failure corresponding to the sequence f2.

An example of compensating measures that could be used to compensate for not complying with the single failure criterion is illustrated by the following example where, in the event of a loss of main heat sink after a power uprate, the single failure criterion was not met and compensation was provided by adding an additional auxiliary feed water system. In simple terms, after the power uprate, the plant was not able to remove the residual heat using a single train of the main feed water system as it was before the power uprate. To compensate for this, an auxiliary feed water system (which by itself also does not have the capacity to sufficiently remove the residual heat) is added.

In this simplified example, the loss of the main heat sink (e.g. by a turbine trip) is the initiating event. Then, the plant is in Plant Condition 2 (abnormal operation). To manage this initiating event, the shutdown system (system function S) and one of two feed water pumps is required (system function A). This addresses the single failure of one feed water pump. If system functions S and A are available, the plant stays in State 2 (consequence OK2 – acceptance criteria DNB of one fuel rod).

If both feed water pumps fail, primary feed and bleed is required. This is considered to be Plant Condition 3 (design base accident), because it is equivalent to a LOCA via the pressuriser safety valve. Therefore the sequence results in consequence OK3 and the acceptance criterion is the number of fuel rods experiencing DNB.

The event tree for the loss of the main heat sink with the original state of the plant is shown in Figure 9.

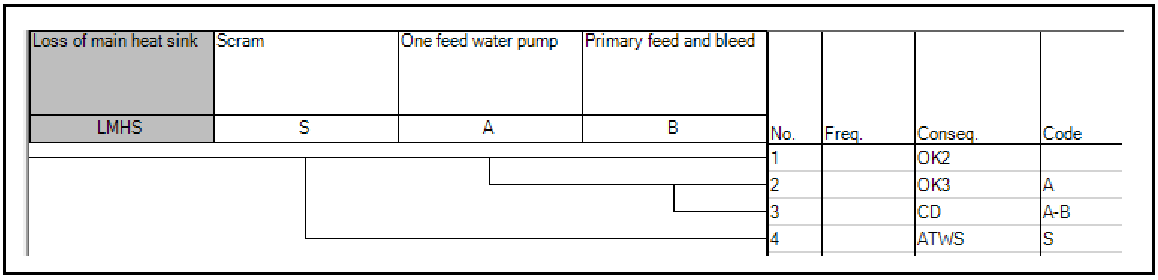


Figure 9. Event tree of the loss of main heat sink in the original plant state

In this example, if there is a power upgrade and both pumps are available, the plant remains in Condition 2 (OK 2) if the main heat sink is lost as shown by Sequence 1 in Figure 10.

However, if the single failure criterion is applied and only one feed water pump is available, the residual heat is too large to be removed at the beginning of the transient, because one pump is not capable of feeding enough water into the steam generators. Core damage is avoided by opening and closing the steam pressuriser safety valve several times until a thermal equilibrium is reached, but the consequences are now those for Plant Condition 3 (OK3) instead of Plant Condition 2 (OK2), as shown by Sequence 2 in Figure 10, and **the single failure criterion has not been met**. (Actually the heat is removed by releasing the steam via a relieve valve or a safety valve, which had to be modelled separately in the event tree but are omitted here for simplicity.)

At this point, Option 4 can be used. By multiplying the frequency of the initiating event with the probability that only one pump is available i.e. with the frequency of Sequence 2 in Fig. 10, the resulting frequency is low enough for the acceptance criteria for Plant Condition 3 (OK3) to apply (see Figure 11). This results in a new plant condition OK3 for which different acceptance criteria apply. In OK2, releases to the atmosphere are the same as for normal operation whereas in OK3, releases (dose in rem or Sievert) are larger due to opening/closing of the pressuriser safety valve and ventilation of the containment.

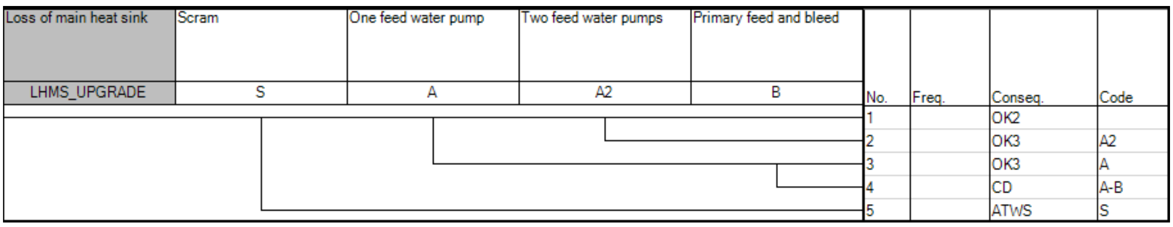


Figure 10 Event tree of the loss of main heat sink after the power upgrade

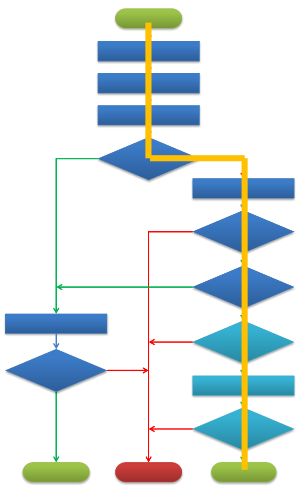


Figure 11: Applying Option 4 sequence 2 can be reclassified, if its frequency is low enough. Then it meets the acceptance limits of the next higher class.

If the acceptance criteria for Plant Condition 3 are met, the consequence/frequency requirements are met but there might still be a need to address the single failure criterion.

In actual fact, the frequency of the initiating event has been multiplied by the conditional probability of a single failure. The resulting frequency of such a sequence is sufficiently low so that Plant Condition OK3 is acceptable (in spite of the violation of the single failure criteria). Nevertheless, if the sum of all sequences leading to the state of OK3 is too high, compensating measures as described below in Figure 12 are necessary.

To compensate for violating the single-failure (of system function A), an auxiliary feed water system might be added to the plant **which is not capable of removing the residual heat on its own.** This is shown as System AA in Figure 12.

Now, with the new auxiliary system, the plant is constrained to be in Condition 2 (OK 2) even if one pump fails (Sequence 2 in Fig. 12) **and the single failure criterion is met**.

This new system also reduces the frequency of Sequence 2 in Figure 9 (code A2), which has now become Sequence 3 in Figure 12 (code A2-AA), because the failure of AA is also necessary for this sequence to occur. Thus, the conditional probability for transferring from Plant Condition OK2 to plant state OK3 for the initiating event being considered is reduced.

The frequency of Sequences 2 to 4 in Figure 8 have not changed (which are equivalent to Sequences 3 to 5 in Figure 10 and Sequences 4 to 6 in Figure 12).

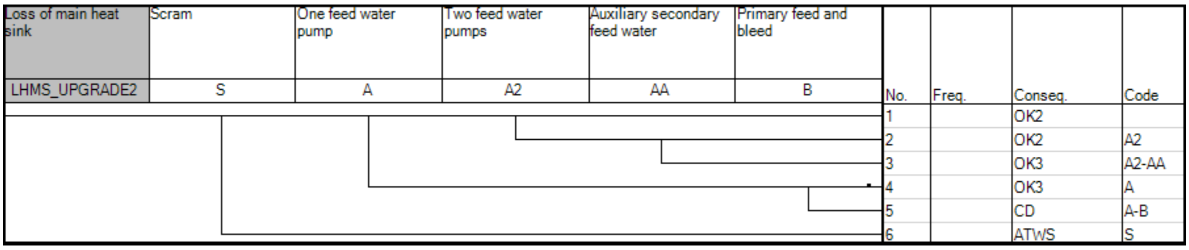


Figure 12 Event tree of the loss of main heat sink after the power upgrade and the addition of an auxiliary feed water system.

3.2 Option 4 and LOCA redefinition

There is currently a proposal for introducing risk-informed changes to loss-of-coolant accident (LOCA) technical requirements in the USA [6]. It is proposed that licensees may choose to fulfil risk-informed requirements instead of the former requirements based on conservative or BEPU analyses. The proposed regulation divides the break size spectrum into two regions, by the so-called “Transition Break Size” (TBS). Breaks larger than TBS are assumed to be much less frequent than those lower or equal than TBS.

This is equivalent to splitting the category 4 for LOCAs into two new categories, say 4a and 4b. Breaks below TBS (category 4a) maintain the same analysis requirements and the same acceptance criteria as in current category 4. Breaks larger than TBS (category 4b), on contrary, are termed as “beyond design basis accidents” and can be analyzed without assuming neither worst single failure nor concurrent loss of offsite power (LOOP). Furthermore, their acceptance criteria are less strict than those for 4a, but still directed to maintain the ability of mitigating the accident consequences.

This proposal is partially equivalent to Option 4, as described in the present paper. A large LOCA (i.e. belonging to category 4b) followed by a single failure or LOOP can be reclassified if the failure probability is low enough. But, being 4b the highest category of accidents, the sequence would be included in the “residual category” where no acceptance limit is imposed.

In Figure 13 (left) this is mapped onto the procedure of Option 4. The sequences belonging to category 4b are reclassified (block 11) and then fulfil the acceptance criterion of the next category (“residual category”) because there are no acceptance limits which could be violated (block 13).

On the other hand, the new rule does not seem to allow the reclassification of a LOCA lower than TBS (category 4a), followed by a highly improbable single failure or a LOOP, to the category 4b. In fact, sequences of this type are forced to stay in the 4a category, and should comply with the well known acceptance criteria of the current regulation. This corresponds to setting $P_{reclassify}$ to zero (see Figure 13 (right)).

A fully adoption of Option 4 would lead to a different proposal. All breaks, including those below the TBS followed by single failure or LOOP might be allowed to reclassify to the higher category if $P < P_{reclassify}$ (see Figure 14 (left)). Even if category 4a fulfils the acceptance criteria at the prescribed level (95/95) single calculations violating those limits should be required to fulfil the acceptance criteria of category 4b, or a tightened version of them, essentially meaning that ability of mitigating the LOCA consequences must be maintained (see Figure 14 (right)).

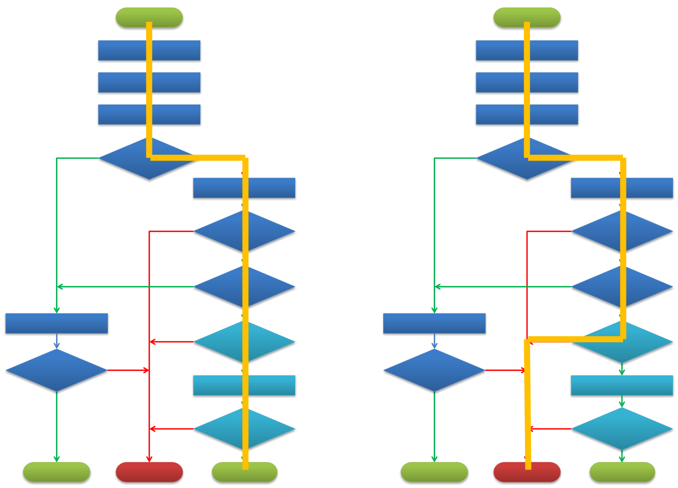


Figure 13: Current procedure for LOCA mapped to Option 4 (left for LOCA above TBS, right for LOCA below TBS). For LOCA below TBS, $P_{reclassify}$ is effectively 0.

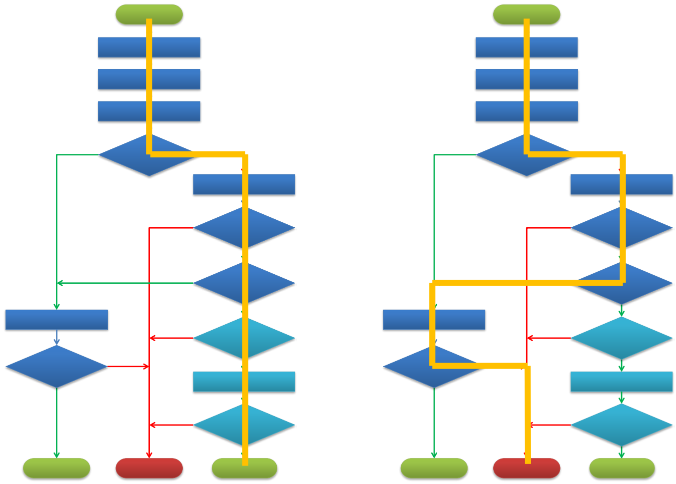


Figure 14: Option 4 applied to LBLOCA. Sequences with single failure can be reclassified if $P_{seq} < P_{reclassify}$ (left). All sequences fulfilling the acceptance criterion for their class have to also fulfil the tightened criterion for the next higher class otherwise the design has to be changed (right).

3.3 Option 4 and design extension

Under WENRA (Western European Nuclear Regulators Association) reactor safety reference levels [7, app. F], design extension is understood as "*measures taken to cope with additional events or combination of events, not foreseen in the design of the plant. Such measures need not involve application of conservative engineering practices but could be based on realistic, probabilistic or best estimate assumptions, methods and analytical criteria*".

A list of types of events to be analysed for design extension as a minimum, if not already considered in the design basis includes:

- Anticipated transient without scram (ATWS)
- Station Black Out (SBO)
- Total loss of feed water
- .....

The objective being "*to minimise as far as reasonably practicable radioactive releases harmful to the public and the environment in cases of events with very low probability of occurrence*".

As it is mentioned above, there is an ample margin of flexibility with regard to the methodology to be used when addressing the issue. It is not rare to observe the use of a pure best estimate calculation making use of somewhat realistic assumptions on initial and boundary conditions.

The problem reduces to check whether the current design is able to cope with a given type of event and so to comply with the applicable acceptance criteria, which can be "*modified in relation to the conservative criteria used in the analysis of the design bases events*", or in another case to develop some design change or mitigation strategy to introduce a new level of defence.

Beyond design basis accidents (BDBA) are characterised by the occurrence of multiple failures which can either be independent and random, or consequential with origin on internal or external events above the severity considered in the NPP design.

Under option 4 a comprehensive analysis of a given type of event is performed that necessarily takes into consideration the associated event sequences and probabilities of failure upon demand for actuations including the failure to fulfil its mission as a result of the transient dynamics (driven by automatic actuation of systems or through the exercise of emergency operating procedures).

In the following paragraphs it is shown how option 4 method can be easily extended to analysis of BDBA.

Use of option 4, and implicitly WENRA approach, in BDBA would require the definition of categories and acceptance criteria for each scenario in terms of barriers' performance and fission products releases (for early and late).

These categories would have strict requirements on the robustness of the design (DiD principle) or mitigation provisions in order to limit to the minimum the probability of accident escalation and hence dose impact on the population. Note that under this scheme, BDBA analysis leads to a reclassification process.

Probability of being in certain BDBA category (defined in terms of consequences), is obtained from the probabilities of escalation ($P_{0\rightarrow 1}$, $P_{0\rightarrow 2}$, ...). Also note that, consistent with WENRA approach, the use of probability of escalation allows decoupling the robustness analysis from the frequency of the type of event.

To simply illustrate how Option 4 would work, let's assume a credible flooding scenario, without regard to its frequency, where the water level is progressively increasing and thus affecting different systems and components. NPP design provisions include the analysis of internal and external events up to a severity level where safe shutdown and long term coolability capability is surpassed. As the flooding progresses beyond that level of severity i.e. transits from DBA to BDBA, the cutsets of the systems needed to attain a safe shutdown will eventually turn failed. If no countermeasure is available a severe accident scenario will develop and consequently different damage thresholds will be crossed with some time dependant probability of escalation.

Analysis of such scenario under Option 4 will require the use of both Probabilistic Safety Assessment (PSA) tools and BEPU. As mentioned before, PSA is connected to external/internal events through the cutsets of systems required to cope with the developing scenario, and it is this connection, together with use of Emergency Operating Procedures (EOP) and Severe Accident Management Guidelines (SAMG) that the dynamic of the accident will evolve. Time evolution of systems availability, together with the use of BEPU method, i.e. Option 4, will help to quantify the robustness of the preventive and mitigatory actions.

The damage categories for this case are predefined (category 1 core damage, category 2 core damage plus early release, category 3 core damage plus late release). Now we need to calculate the probability of escalation into another damage category. The quantification of this probability of escalation can only be rigorously made by means of option 4 or similar approach.

The acceptable probability of escalation determines the maximum failure probability of the prevention or compensatory action. In case of the above flooding example and consequent SBO, it will determine the applicable "design criteria".

For a systematic application of this analytical approach it is required to establish classification categories for BDBA and the setting of acceptance criteria based on probability of escalation. This allows the evaluation of the robustness of the available preventive or mitigatory actions needed to exclude the escalation into another category with a given probability (DiD assessment).

4. Conclusions

A conservative approach is still used for licensing purposes in most countries. Some countries, however, have moved to allow the use of the Combined or even BEPU approach i.e. the use of Options 2 or 3. Option 4 is still a "research" option and will remain so for some time.

Option 4 integrates the system availability and in particular the single failure criterion into a risk informed design basis safety analysis. A possible strategy for the analysis of a given PIE using Option 4 is the following:

1. Total conditional exceedance probability for all analyzed sequences must be less than a low percentage (typ., <5%) with a high degree of confidence (typ., >95%).
2. Individual sequences where the exceedance probability is higher than 5% can be reclassified to a higher class if the probability of the additional failures is low enough. However, reclassification of single failure sequences is only allowed to the next class.
3. For any non-reclassified sequences (including the nominal one), no case (i.e., no computer run) exceeding the acceptance limit of the class is allowed to exceed a tightened acceptance limit of the next class (to be determined).
4. Finally, for the reclassified sequences, the exceedance criterion (<5%/>95%) must be fulfilled for the acceptance limits of the new class.

As an extension of Option 3, the proposed Option 4 would tighten the requirements for those sequences runs that do not fulfill the 95/95 acceptance criterion and in doing that exclude any cliff-edge effects.

On the other end, it would allow certain sequences runs having low enough probability of occurrence to be reclassified into the next higher class, where they would be compared to the acceptance limits of that class or some tightened acceptance limits, if appropriate.

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OECD/CSNI Workshop, Barcelona (Spain), Nov.16-18, 2011

 # Invited #

Extension of BEPU methods to Sub-channel Thermal-Hydraulics and to Coupled Three-Dimensional Neutronics/Thermal-Hydraulics Codes

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Abstract

The principles that support the risk-informed regulation are to be considered in an integrated decision-making process. Thus, any evaluation of licensing issues supported by a safety analysis would take into account both deterministic and probabilistic aspects of the problem. The deterministic aspects will be addressed using Best Estimate code calculations and considering the associated uncertainties i.e. Plus Uncertainty (BEPU) calculations. In recent years there has been an increasing demand from nuclear research, industry, safety and regulation for best estimate predictions to be provided with their confidence bounds. This applies also to the sub-channel thermal-hydraulic codes, which are used to evaluate local safety parameters. The paper discuses the extension of BEPU methods to the sub-channel thermal-hydraulic codes on the example of the Pennsylvania State University (PSU) version of COBRA-TF (CTF).

The use of coupled codes supplemented with uncertainty analysis allows to avoid unnecessary penalties due to incoherent approximations in the traditional decoupled calculations, and to obtain more accurate evaluation of margins regarding licensing limit. This becomes important for licensing power upgrades, improved fuel assembly and control rod designs, higher burn-up and others issues related to operating LWRs as well as to the new Generation 3+ designs being licensed now (ESBWR, AP-1000, EPR-1600 and etc.). The paper presents the application of Generalized Perturbation Theory (GPT) to generate uncertainties associated with the few-group assembly homogenized neutron cross-section data used as input in coupled reactor core calculations. This is followed by a discussion of uncertainty propagation methodologies, being implemented by PSU in cooperation of Technical University of Catalonia (UPC) for reactor core calculations and for comprehensive multi-physics simulations.

1. Introduction

The principles that support the risk-informed regulation are to be considered in an integrated decision-making process. Thus, any evaluation of licensing issues supported by a safety analysis would take into account both deterministic and probabilistic aspects of the problem. The deterministic aspects will be addressed using Best Estimate code calculations and considering the associated uncertainties i.e. Plus Uncertainty (BEPU) calculations. The main tools for performing design and safety analyses of nuclear power plants (NPPs) are computer codes for simulation of physical processes. The current trends in nuclear power generation and regulation are to perform design and safety studies by "best-estimate" codes that allow a realistic modeling of nuclear and thermal-hydraulic processes of the reactor core and the entire plant behavior including control and protection functions. These best-estimate results should be supplemented by uncertainty and sensitivity analysis.

Realistic methods are referred to as "best-estimate" calculations, implying that they use a set of data, correlations, and methods designed to represent the phenomena, using the best available techniques. As compared to the traditional conservative calculations (in which the results obtained are interpreted with additional safety margins) the best-estimate evaluations result in increased plant operation flexibility and improved performance. The regulations also require that the uncertainty in these best-estimate calculations be evaluated. For this reason, it is necessary to identify and quantify all the sources of possible input uncertainties, and then apply comprehensive methodologies to propagate these uncertainties through the nuclear design and safety analyses procedure to the predictions of output parameters of interest.

Different uncertainty and sensitivity methodologies have been developed and utilized in nuclear reactor analysis. Two main approaches have been adopted in the current practice:

a. Statistical methods, which are utilizing statistical sampling based on statistical propagation and processing. In these methods in order to incorporate uncertainties into the process usually many runs of the computer code are required and the result is a range of results with associated probabilities;

b. Deterministic methods, which are using sensitivity analysis based on first-order perturbation theory. The perturbation theory has been developed in order to extend its applicability to estimate higher order variations; however, the computational overhead of higher order perturbation theory is often overwhelming and do not justify the development effort required for their implementation.

In nuclear engineering applications the tendencies have been towards using of deterministic methods for cross-section uncertainty propagation in multiplication factor (k_{eff}) predictions for criticality neutronics calculations while the statistical methods have been applied to thermal-hydraulics and coupled thermal-hydraulics/neutronics calculations.

In recent years there has been an increasing demand from nuclear research, industry, safety and regulation for best estimate predictions to be provided with their confidence bounds. This applies also to the sub-channel thermal-hydraulic codes, which are used to evaluate local safety parameters. The paper discuses the extension of BEPU methods to the sub-channel thermal-hydraulic codes on the example of the Pennsylvania State University (PSU) version of COBRA-TF (CTF).

The paper also presents the application of Generalized Perturbation Theory (GPT) to generate uncertainties associated with the few-group assembly homogenized neutron cross-section data used as input in reactor physics core calculations. Cross-section uncertainties are one of the important input uncertainty sources in reactor core analysis and coupled neutronics/thermal-hydraulics calculations.

The use of coupled codes supplemented with uncertainty analysis allows to avoid unnecessary penalties due to incoherent approximations in the traditional decoupled calculations, and to obtain more accurate evaluation of margins regarding licensing limit. This becomes important for licensing power upgrades, improved fuel assembly and control rod designs, higher burn-up and others issues related to operating Light Water Reactors (LWRs) as well as to the new Generation 3+ designs being licensed now (ESBWR, AP-1000, EPR-1600 and etc.). The extension of BEPU methods to coupled three-dimensional neutronics/thermal-hydraulics codes requires higher than linear order uncertainty analysis techniques capable of treating the non-linear thermal-hydraulic feedback phenomena as well as approaches allowing for combination of different input sources of uncertainties, and computationally efficient in dealing with large size of input parameters often associated with realistic reactor models. The paper discusses uncertainty propagation methodologies, being implemented by PSU in cooperation of Technical University of Catalonia (UPC) for reactor core calculations and for comprehensive multi-physics simulations.

2. Uncertainty Quantification of Sub-Channel Thermal-Hydraulic Calculations

In the past few decades the need for improved nuclear reactor safety analyses has led to a rapid development of advanced methods for multidimensional thermal-hydraulic analyses. These methods have become progressively more complex in order to account for the many physical phenomena anticipated during steady state and transient LWR conditions. In safety analyses, there are two families of thermal-hydraulic codes that are commonly used. Each family was developed for solving specific transients with a particular degree of detail in the simulation.

System codes were developed for modelling the entire primary system, including heat exchangers, pumps and other components, yet they use a relatively coarse nodalization that introduces some inaccuracies in the analysis. In the system codes coolant channels typically represent lumped regions of core. Sub-channel codes apply a finer discretization to the LWR core, and therefore, reduce some of these inaccuracies but with reduced flexibility and at a higher computational cost. They focus on mass exchange between “sub-channels” on pin-by-pin scale, and can use arbitrary geometry. Channels can represent multiple scales. The sub-channel codes are usually applied for core/vessel modelling and utilize inlet and outlet are boundary conditions. Advanced thermal-hydraulic sub-channel codes are widely used for best-estimate evaluations of nuclear reactors safety margins of LWRs.

As mentioned above the sub-channel codes are used for fuel bundle and reactor core thermal-hydraulic analysis and safety evaluation. The results of sub-channel code predictions are subject to some uncertainties. Input uncertainties in sub-channel calculations can arise from model limitations, approximations in the numerical solution, nodalization, homogenization approaches, imperfect knowledge of boundary and initial conditions, scaling effects, user effects, programming errors, and compiler effects. The various sources of uncertainties can be classified as:

a. Operational uncertainties;

b. Geometry uncertainties;

c. Modeling uncertainties;

d. Code uncertainties.

These input uncertainties are propagated through sub-channel simulations to determine the uncertainty in fuel temperature prediction (when the fuel rod model is utilized), which is related to the Doppler feedback prediction, as well as the uncertainty in predicting moderator parameters, which are related to the moderator feedback prediction, such as moderator temperature, density, and void fraction. Uncertainty analyses consider some or all of the various sources of uncertainties and calculate the uncertainties of thermal hydraulic code predictions. In other words, uncertainty analyses determine how a code propagates the corresponding input uncertainties. The improved and validated PSU version of the thermal-hydraulic sub-channel code COBRA-TF - CTF is used in conjunction with the GRS uncertainty analysis methodology [1]. The GRS methodology has been previously applied to wide range of problems but not to sub-channel thermal-hydraulic calculations. It is an efficient statistical uncertainty methodology, based on order statistics, and in this application is combined with the well-known Phenomena Identification and Ranking Table (PIRT) process.

The operating conditions, such as the reactor power and coolant flow rate, vary over time. These are also called as boundary condition effects, as they are used to define the main operating conditions of the core. These variations are small during steady-state conditions but still carry some uncertainty that will propagate to the output parameters. Example of uncertainties in boundary condition effects in sub-channel calculations is given in Table 1 based on data provided by the OECD/NRC Pressurized Water Reactor (PWR) PSBT benchmark [2]. The Probability Density Function (PDF) is the assumed distribution function for each parameter.

Table 1. Estimated accuracy and PDFs of the operational parameters

Quantity	Accuracy Range	PDF
System pressure	±1%	Normal
Flow	±1.5%	Normal
Power	±1.0%	Normal
Inlet fluid temperature	±1.0 °C	Flat

The geometry of the bundle and core affects the way coolant flows and other parameters pertinent to sub-channel simulations. The components of the bundle, such as the fuel rods and channel spacers, carry some uncertainty in their manufactured tolerances. The outer cladding diameters as well as the surface roughness of the cladding are the only tolerances from the fuel rod that should be considered since the flow is only exposed to this portion of the rod. The example of such manufacturing tolerances for a PWR bundle are given in Table 2, which is based again the OECD/NRC PSBT benchmark data:

Table 2. Manufacturing tolerances for a PWR bundle

Item	Tolerance
Heater rod diameter	0.02 mm
Heater rod displacement	0.45 mm
Flow channel inner width	0.05 mm
Flow channel displacement	0.20 mm
Power distribution	3 %

The effects of these tolerances are important to quantify because tolerances are built into everything that is manufactured and there is almost no way to eliminate this uncertainty from a nuclear reactor analysis. Other uncertainties can actually be improved but the manufacturing uncertainties are accepted as inherent and fixed. There can be also a physical distortion of the bundle or the test assembly that occurs when the system is in operation due to the heat that causes thermal expansion of the various components. The system will lose some accuracy as is it simplified in order to be used as code input, since it is difficult and impractical to exactly model every detail of the reactor core exactly as it is. There are also uncertainties associated with axial and radial power distributions of the fuel rods, which are input to sub-channel codes. The use of computer codes for analysis of the sub-channel thermal-hydraulics of a reactor is based on approximations that are determined from interpolations of known data. The correlations are mostly valid over a range of temperatures but lose accuracy near either side of these ranges.

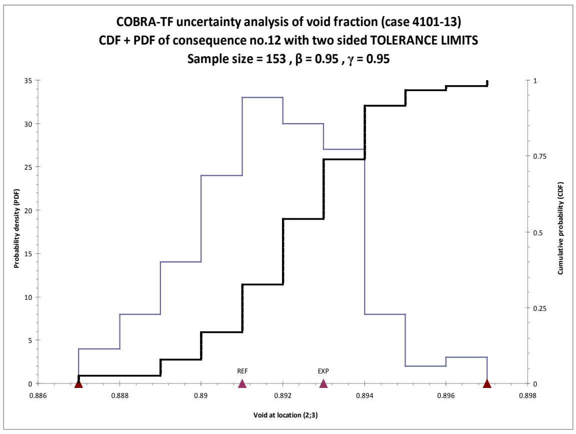
As it has been previously discussed, there are various groups of uncertainties that can affect the desired output values. The parameters that have the largest effect are the most important to study due to their significance on the results of the test cases. The boundary condition effects are expected to be largest with variations in the coolant mass flow rate and the power. Of the geometry effects, those with the largest anticipated importance are the sub-channel area and the heated perimeter. The hydraulics parameters are built into the codes and are grouped under code uncertainties; the important ones are the mixing coefficient and the equilibrium distribution weighting factor in the void drift. Another code uncertainty with large expected effects on the output is the interfacial friction factor. The input uncertainty parameters in CTF sub-channel calculations have been ranked using the PIRT approach [3] and the OECD/NRC BFBT benchmark [4]. The PIRT is a useful framework for identifying and ranking significant phenomena. PIRT is a guide for considering which phenomena are significant for uncertainty analysis. The comprehensive BFBT experimental database leads one step further in developing modeling capabilities by taking into account the uncertainty analysis in the benchmark. The uncertainties in the input data (boundary conditions) and geometry as well as in the code models are the main sources of "errors" in the simulated results which are of importance to be compared with the measurement uncertainties. Therefore, uncertainty analysis has been performed for the void distribution in a Boiling Water Reactor (BWR) bundle. The chosen uncertainty input parameters with their suggested accuracies and PDFs are given in Table 3. The selection of uncertainty input parameters was guided by the PIRT. Exit pressure, inlet flow rate and temperature, and bundle power are selected as boundary conditions uncertainty parameters. The sub-channel flow area represents the geometry uncertainty parameters. As modeling uncertainty parameters the single-phase mixing coefficient, the two-phase mixing multiplier, the equilibrium distribution weighing factor in the void drift model, the nucleate boiling heat transfer coefficient, and the flow regime dependent interfacial drag coefficients are analyzed. The accuracy of the boundary conditions and geometry parameters are provided by the BFBT benchmark specifications, while the accuracies of the modeling parameters are the published uncertainties of the corresponding correlations used in CTF.

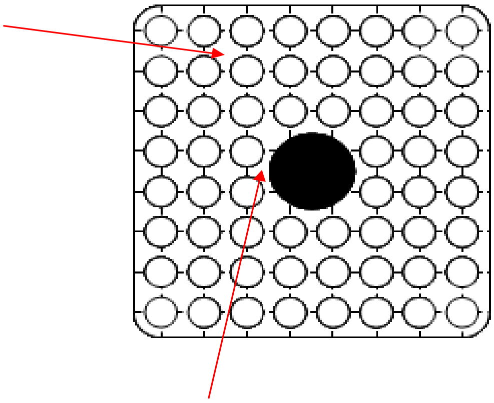
Table 3. Input uncertainty parameters

Parameter	Accuracy	PDF
1. Pressure	± 1 %	Normal
2. Flow Rate	± 1 %	Normal
3. Power	± 1.5 %	Normal
4. Inlet Temperature	± 1.5 C	Flat
5. Subchannel Area	± 0.5 %	Normal
6. Single-phase mixing coefficient	$2\sigma = \pm$ 42 %	Normal
7. Two-phase multiplier of the mixing coefficient	$2\sigma = \pm$ 24%	Normal
8. Equilibrium distribution weighing factor in the void drift	$2\sigma = \pm$ 14 %	Normal
9. Nucleate boiling heat transfer coefficient	$2\sigma = \pm$ 24 %	Normal
10. Interfacial drag coefficient (bubbly flow)	$2\sigma = \pm$ 32 %	Normal
11. Interfacial drag coefficient (droplet flow)	$2\sigma = \pm$ 26%	Normal
12. Interfacial drag coefficient (film flow)	$2\sigma = \pm$ 36 %	Normal

The developed calculation model with CTF for high-burnup 8×8 BWR assembly consists of 80 sub-channels, 40 axial nodes, and 140 transverse connections between sub-channels (gaps). Selected results of the performed uncertainty and sensitivity analysis for test cases 4101-13 (see Figure 1), 4101-69 (see Figure 2), and 4101-86 (see Figure 3) of the BFBT benchmark database are shown in this paper. The GRS methodology was applied in this study with two-sided tolerance intervals and the sample size N is selected to be 153 (for better statistics as compared to the minimum required 93) for β=0.95 and γ=0.95.

The observed discrepancies between predicted by CTF and measured void distribution are believed to be due to two major factors: the previously indicated CTF tendency of void fraction over-prediction, and asymmetrical void measurements in regions with otherwise quite symmetrical power load. Nevertheless, it can be seen that the largest deviations are in the corner sub-channels and in the sub-channels connected to the water rod or, in other words, in fluid volumes bounded by unheated walls. The operational conditions for each case are given in the tables. The three cases being analyzed spanned over operating conditions presented in a BWR.





Void Fraction (%)	Pressure (MPa)	Flow Rate (t/h)	Power (MW)
86.8	1.224	55.01	4.46

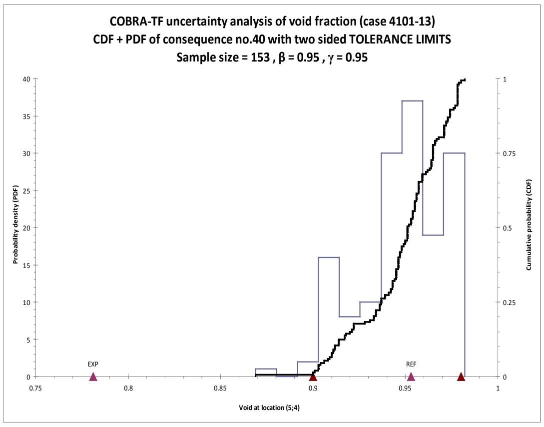
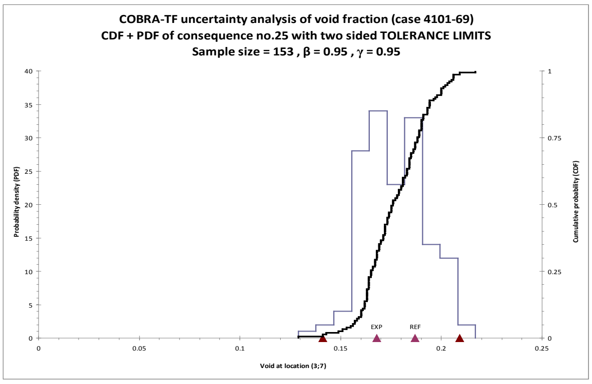
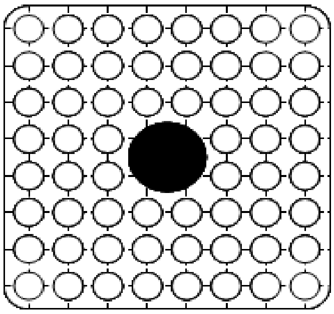


Figure 1. Uncertainty analysis of case 4101-13





Void Fraction (%)	Pressure (MPa)	Flow Rate (t/h)	Power (MW)
18.2	8.638	10.08	0.23

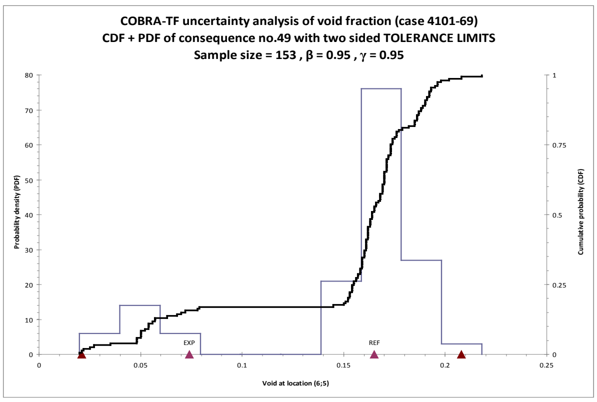
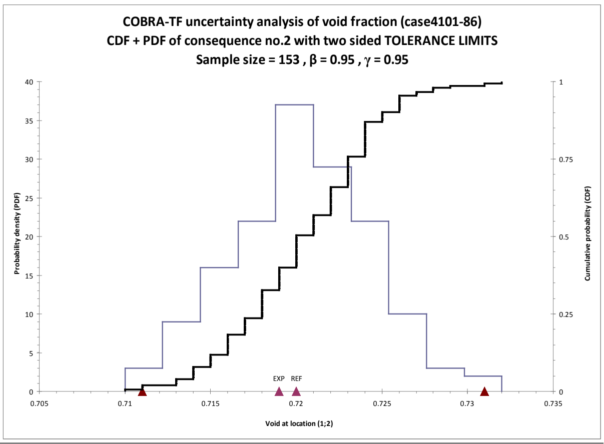
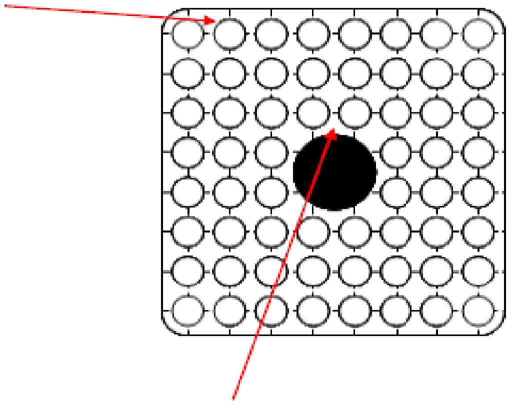


Figure 2. Uncertainty analysis of case 4101-69





Void Fraction (%)	Pressure (MPa)	Flow Rate (t/h)	Power (MW)
69.8	8.705	54.59	4.62

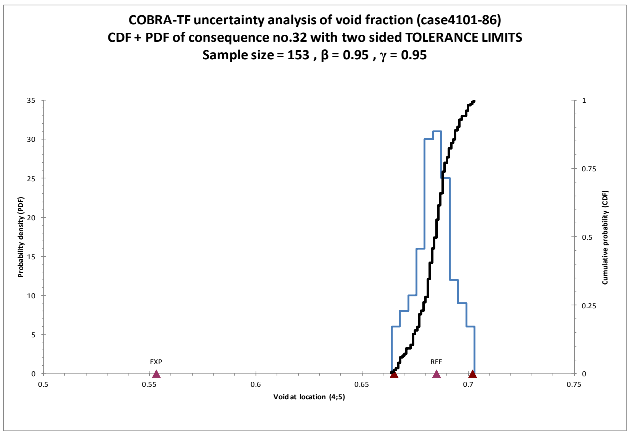


Figure 3. Uncertainty analysis of case 4101-86

3. Propagation of Uncertainties in Coupled Three-Dimensional Neutronics/Thermal-Hydraulic Calculations

Understanding the uncertainties in key output reactor core parameters associated with coupled core simulation is important in regard to introducing appropriate design margins and deciding where efforts should be directed to reduce uncertainties. The extension of BEPU methods to coupled three-dimensional neutronics/thermal-hydraulics codes requires higher than linear order uncertainty analysis techniques capable of treating the non-linear thermal-hydraulic feedback phenomena, as well as approaches allowing for combination of different input sources of uncertainties, and computationally efficient in dealing with large size of input parameters often associated with realistic reactor models.

Uncertainties in predictions of key reactor core parameters associated with LWR core simulation occur due to input data uncertainties, modeling errors, and numerical approximations in neutronics, thermal-hydraulic and heat transfer models. Input data for core neutronics calculations primarily include the lattice averaged few group cross-sections.

In the current established calculation scheme for LWR design and safety analysis the lattice averaged (homogenized) few-group cross-sections are an input to core calculations. The few-group cross-section uncertainties (few-group covariance matrix) are obtained using the SCALE-6.0 44-group covariance matrix as input to the TSUNAMI-2D sequence with GPT in SCALE 6.1[5]. The obtained few-group cross-section uncertainties then can be propagated to uncertainties in evaluated stand-alone neutronics core parameters.

Lattice physics responses of interest include k_{inf}, two-group homogenized cross-sections, assembly discontinuity factors, kinetics parameters, and power peaking factors for pin power reconstruction. Each non-k response can be written as a response ratio and requires GPT calculations. These responses include Capture (Σ_{c1}, Σ_{c2}), Fission (Σ_{f1}, Σ_{f2}), Neutron-production ($\nu\Sigma_{f1}$, $\nu\Sigma_{f2}$), Diffusion coefficient (D_1, D_2) and Downscatter ($\Sigma_{1\rightarrow 2}$).

Table 4. Assembly k_{inf} values and associated uncertainties

Reactor type	Case	k	%Δk/k	Major contributor
BWR	HZP	1.11	0.50	^{238}U (n,γ)
	HFP (40% void)	1.08	0.56	^{238}U (n,γ)
PWR	HZP	1.41	0.46	^{238}U (n,γ)
	HFP	1.39	0.47	^{238}U (n,γ)
VVER	HZP			
	HFP	1.31	0.76	^{238}U (n,n')
GEN III	Type 1 (UOX 4,2% ^{235}U)	1.25	0.49	^{238}U (n,γ)
	Type 2 (UOX 4,2% ^{235}U+ $UO_2Gd_2O_3$ 2,2% ^{235}U)	1.12	0.49	^{238}U (n,γ)
	Type 3 (UOX 3,2% ^{235}U+ $UO_2Gd_2O_3$ 1,9% ^{235}U)	0.96	0.53	^{238}U (n,γ)
	Type 4 (MOX)	1.07	0.97	^{238}U (n,n')

One can define 9-dimensional response vector $\mathbf{R} = [\Sigma_{a1}, \Sigma_{a2}, \Sigma_{f1}, \Sigma_{f2}, \nu\Sigma_{f1}, \nu\Sigma_{f2}, D_1, D_2, \Sigma_{1\rightarrow2}]$ for two-group assembly homogenized cross-sections and obtain a corresponding covariance matrix in which the diagonal elements are the % standard deviations, while off-diagonal elements are the correlation coefficients.

The obtained results for different LWR types and cases, shown in Tables 4 and 5 and Figure 4, indicate the following tendencies:

a. Group 1 (fast) cross-section uncertainty is~2-3 times larger than Group 2 (thermal) cross-sections uncertainty;

b. Uncertainty Contributions:

- A major contributor to Group 1 (fast) cross-sections is U-238 inelastic scattering;
- U-238 inelastic scattering uncertainty is quite large;
- 40% void (and higher) exhibit larger k_∞ due to harder flux spectrum.

c. Uncertainty (Correlation) Contribution:

- U-238 inelastic scattering uncertainty is quite large, and dominates correlation coefficient.

The next step is to propagate the few-group cross-sections uncertainties in stand-alone core neutronics calculations. PSU and UPC are exploring the so-called hybrid approach similar to the efforts performed elsewhere [6] in which the SCALE sequence is combined with front end perturbation engine based on statistical sampling. TSUNAMI-2D provides a generalized GPT method within SCALE-6.1 to generate covariance matrices for assembly averaged cross-sections (as described above). The stochastic method is used to propagate these uncertainties in core calculations to the output parameters of interest. This method is named as the two step method. For core calculations, normally cross sections of different fuel assemblies are necessary. Generally, there will be correlations between cross sections of different fuel assemblies (in particular, if these are similar). The complete covariance matrix (for all cross sections of all fuel assemblies) has to be determined. Another option is to perform statistical sampling at the level of multi-group cross-sections and associated uncertainties (multi-group covariance matrix). In this all the correlations are taken into account implicitly. This method is called the random sampling. The primary difference between the random sampling and two-step methods are the moment at which the perturbations are applied and the origin of the covariance matrix.

Table 5. Two-group cross-section uncertainties

Reactor Type	Case	Responses (%ΔR/R)								
		Σ_{c1}	Σ_{c2}	Σ_{f1}	Σ_{f2}	$\nu\Sigma_{f1}$	$\nu\Sigma_{f2}$	D_1	D_2	$\Sigma_{1\rightarrow 2}$
BWR	HZP	1.20	0.51	0.68	0.32	0.98	0.45	0.84	0.13	1.01
	HFP (40% void)	1.29	0.55	0.73	0.32	1.01	0.45	0.92	0.16	1.23
PWR	HZP	1.29	0.82	0.36	0.32	0.51	0.44	0.88	0.15	1.21
	HFP	1.29	0.82	0.36	0.32	0.51	0.44	0.88	0.15	1.21
VVER	HZP									
	HFP	1.25	0.63	0.72	0.31	1.05	0.44	0.92	0.12	1.17
GEN III	Type 1 (UOX 4,2% ^{235}U)	1.30	0.62	0.37	0.32	0.57	0.45	0.91	0.15	1.25
	Type 2 (UOX 4,2% ^{235}U+ $UO_2Gd_2O_3$ 2,2% ^{235}U)	1.29	0.46	0.38	0.32	0.58	0.45	0.91	0.15	0.91
	Type 3 (UOX 3,2% ^{235}U+ $UO_2Gd_2O_3$ 1,9% ^{235}U)	1.27	0.37	0.50	0.33	0.76	0.45	0.90	0.15	1.24
	Type 4 (MOX)	1.40	0.75	0.44	0.63	0.77	1.09	0.98	0.16	1.47

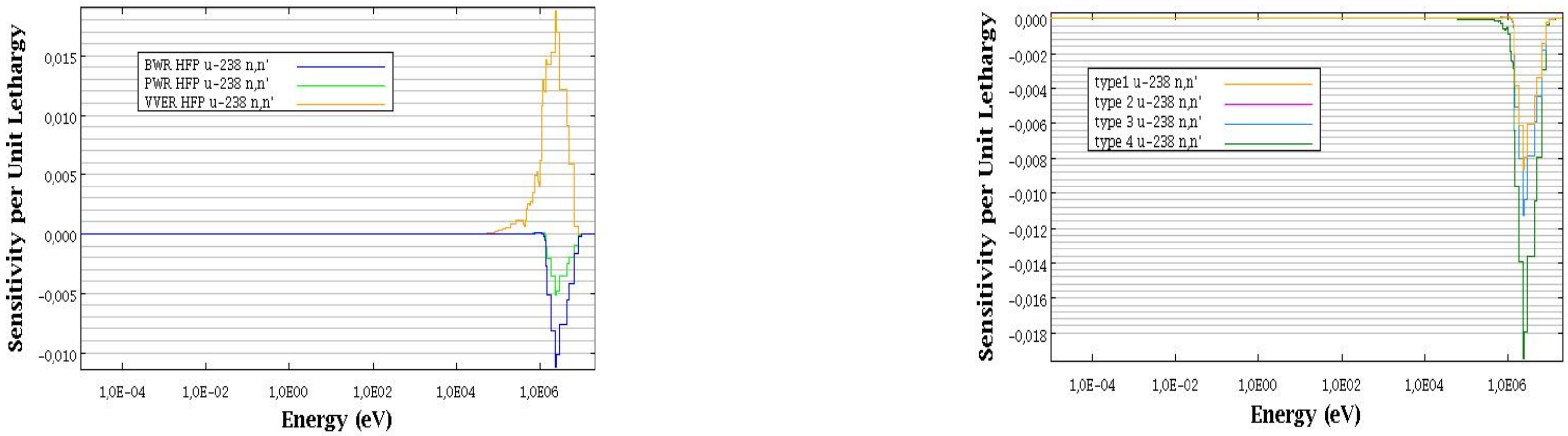


Figure 4. Response Sensitivity to U-238 n,n'

Sensitivity analysis and uncertainty quantification of tightly coupled reactor physics thermal-hydraulic analyses require the development of not only first but also higher (at least second) order sensitivities. Analytic sensitivity methods (direct differentiation and adjoint methods) can be used in conjunction with an implicit solution approach to develop numerically efficient methods for addressing the issue of high dimensionality in uncertainty quantification in reactor simulations.

A computational model can be considered as a function (frequently termed response function) $\mathbf{y} = \mathbf{f}\ (\mathbf{x})$ where $\mathbf{x}$ is a vector quantifying the input state such as material properties, initial and boundary conditions, and geometry and $\mathbf{y}$ is a vector quantifying the model output, such as the solution field, or metrics of solution field at critical regions. For computationally expensive models, re-running the analysis may be time prohibitive and an approximate representation of the system response $\mathbf{f}\ (\mathbf{x})$ is used. One approach is the moment method where a Taylor expansion from a nominal input state $\mathbf{x}_0$ is used [7]. Using first order derivatives approximates the system response $\mathbf{f}\ (\mathbf{x})$ by a hyper plane. For non-linear systems, second or even higher order terms may be needed. An implementation of the moment approach in neutronics analysis is presented in [8]. An alternate approach in approximating the system response $\mathbf{f}\ (\mathbf{x})$ is to employ global models such as in Polynomial Chaos Expansion (PCE) and Stochastic Collocation (SC) methods [9], which employ orthogonal and interpolation polynomial bases respectively, and in global reliability analyses [10], which adaptively refine Gaussian process models. In order to address the scaling for each of these global methods in systems with highly dimensional input state $\mathbf{x}$, gradient-enhanced versions can be employed using implicit or explicit approaches on structured or unstructured grids.

Analytic methods for sensitivity analysis of non-linear and transient problems have been developed since the 70's [11]. When implicit methods are utilized to solve the non-linear systems, analytic sensitivity methods can be very efficient. A non-linear system can be denoted as a residual vector $\mathbf{R}(\mathbf{u}) = \mathbf{0}$, where $\mathbf{u}$ is the unknown system response. Using the Newton-Raphson method, the system is iteratively computed as:

$$\mathbf{u}^{I+1} = \mathbf{u}^{I} - \left[\frac{d\mathbf{R}}{d\mathbf{u}}(\mathbf{u}^{I})\right]^{-1} \mathbf{R}(\mathbf{u}^{I}) \tag{1}$$

where $d\mathbf{R}/d\mathbf{u}$ is the Jacobian. For large systems, more than 90% of the computational time is devoted in computing and evaluating the Jacobian. In a sensitivity analysis, the response function $\mathbf{f}\ (\mathbf{x}) = \mathbf{f}\ (\mathbf{u}(\mathbf{x}), \mathbf{x})$ is expressed as a function of the system response $\mathbf{u}$ and input state $\mathbf{x}$. Differentiating $\mathbf{f}$ with respect to each input state component x_i results to:

$$\frac{d\mathbf{f}}{dx_i} = \frac{\partial \mathbf{f}}{\partial \mathbf{u}}\frac{d\mathbf{u}}{dx_i} + \frac{\partial \mathbf{f}}{\partial x_i} \tag{2}$$

Using the direct differentiation approach, the system response sensitivity $d\mathbf{u}/dx_i$ is computed by differentiating the residual equation:

$$\frac{d\mathbf{R}(\mathbf{u}(\mathbf{x}), \mathbf{x})}{dx_i} = \frac{\partial \mathbf{R}}{\partial \mathbf{u}}\frac{d\mathbf{u}}{dx_i} + \frac{\partial \mathbf{R}}{\partial x_i} = \mathbf{0} \Rightarrow \frac{d\mathbf{u}}{d\mathbf{x}_i} = -\left[\frac{\partial \mathbf{R}}{\partial \mathbf{u}}\right]^{-1} \frac{\partial \mathbf{R}}{\partial \mathbf{x}_i} \tag{3}$$

The sensitivity calculation of Equation (3) is performed after the iterative solution of Equation (1) using the same Jacobian. Thus, it only requires one back-substitution per x_i. Typically, 1% additional computational time is required per state variable *xi*. The total computational overhead is proportional to the number of state variables p (rank of $\mathbf{x}$). The adjoint sensitivity analysis method can be derived by augmenting the functional response function $\mathbf{f} = \mathbf{f}\ (\mathbf{u}(\mathbf{x}), \mathbf{x}) + \lambda^T\mathbf{R}$ by a Lagrange multiplier λ noting that $\mathbf{R} = \mathbf{0}$, which after differentiation results to:

$$\frac{df_k}{dx_i} = \frac{\partial f_j}{\partial \mathbf{u}}\frac{d\mathbf{u}}{dx_i} + \frac{\partial f_k}{\partial x_i} + \boldsymbol{\lambda}^T\left[\frac{\partial \mathbf{R}}{\partial \mathbf{u}}\frac{d\mathbf{u}}{dx_i} + \frac{\partial \mathbf{R}}{\partial x_i}\right] = \frac{\partial f_k}{\partial x_i} + \boldsymbol{\lambda}^T\frac{\partial \mathbf{R}}{\partial x_i} \text{ with } \boldsymbol{\lambda} = -\left[\frac{\partial \mathbf{R}}{\partial \mathbf{u}}\right]^{-T}\left(\frac{\partial f_k}{\partial \mathbf{u}}\right)^T \tag{4}$$

The adjoint sensitivity calculation of Equation (4) also uses the same Jacobian as the analysis. The total computational overhead is proportional to the number of response functionals $\boldsymbol{q}$ (rank of $\mathbf{f}$). Second order sensitivities can be obtained by a hybrid direct differentiation-adjoin method that results into $\boldsymbol{p} + \boldsymbol{q}$ additional back substitutions. Sensitivity analysis for transient and coupled non-linear systems using both direct differentiation and adjoint methods are derived in [12]. The approach has been implemented in thermal-mechanical problems in [13].

The current state-of-art coupling in reactor safety analysis performed with coupled neutronics/thermal-hydraulic calculations is based on an explicit time integration method which works well when power is not changing rapidly or when detailed spatial feedback is not important. However, for analysis such as Boiling Water Reactor (BWR) Turbine Trip [14] and BWR stability transients, where fast power response and detailed spatial kinetics models are important, the time step size is limited by the error with the explicit treatment of the feedback parameters. To overcome this limitation in this type of analysis a fully implicit coupling of code systems was developed in [15] and implemented into the coupled TRACE/PARCS code package. The fully implicit coupling involves forming the full Jacobian for all physics involved in the analysis and solving the entire nonlinear system.

The on-going developments involve generating first and second order sensitivities for coupled neutronics/thermal-hydraulic analyses and incorporating them in an uncertainty analysis of the OECD/NRC Peach Bottom Turbine Trip benchmark (PBTT) as part of the OECD LWR UAM benchmark [14]. The full scale thermal hydraulics model contains 68 components simulating the reactor vessel, core, coolant loops, and steam line. The thermal hydraulic model of the core is modeled using 33 channels connected to the vessel model. The neutronics model has one node per assembly in each radial plane with 26 axial nodes. The core model uses quarter core symmetry. Each neutronics node represents a particular fuel type. There are 435 unique fuel types in this model. The two group assembly-homogenized macroscopic cross sections are tabulated in a library for each fuel type for varying fuel temperatures and moderator densities. Nine different sets of nuclear data are tabulated for each fuel type, they are; diffusion coefficient, absorption macroscopic cross section, fission macroscopic cross section, nu-fission macroscopic cross section, scattering macroscopic cross section, assembly discontinuity factors for west and south faces, and detector flux ratios and microscopic cross sections. The Covariance Matrix for BWR homogenized two-group cross-sections are being calculated using SCALE 6.1. The Taylor based and gradient enhanced static collocation uncertainty analysis methods are being compared and then the best method will be implemented.

In summary, sensitivity formulations for fully coupled kinetic-thermal-hydraulic implicit analyses will be developed and implemented in the TRACE/PARCS code package. Following this implementation will be the demonstration of sensitivity analysis in the uncertainty quantification of power level, power distribution, void distribution, pressure, etc of OECD/NRC Peach Bottom Turbine Trip benchmark simulations.

4. Conclusions

The uncertainty and sensitivity analysis methods are being developed for comprehensive multi-physics simulations with nonlinear feedback mechanisms. These methods allow one to explore the full phase space of input parameters and to take the non-linearity of the model into account. There are parallel activities to establish comprehensive knowledge of uncertainties in input parameters and data (e.g., cross sections, correlations, dimensions, and compositions), as well as knowledge and understanding of sources and uncertainties and biases in analytic and numerical modelling approximations. The on-going international OECD Light Water Reactor Uncertainty Analysis in modeling benchmark establishes a framework for uncertainty propagation through multi-physics multi-scale calculations in order to compare different uncertainty analysis methods.

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Application of the Integrated Safety Assessment Methodology to Sequences with Loss of Component Cooling Water System.

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Abstract

The Integrated Safety Assessment (ISA) methodology, developed by the Consejo de Seguridad Nuclear (CSN), Spanish Nuclear Regulatory Body, has been applied to a thermo-hydraulic analysis of Zion NPP for sequences with loss of the Component Cooling Water System (CCWS). The ISA methodology allows obtaining the damage domain (the region where the PCT limit is exceeded) for each sequence of the dynamic event tree as a function of the operator actuations times (secondary side cooling and recovery of CCWS) and the time of occurrence of stochastic phenomena (seal LOCA), and computing from it the exceedance frequency by integrating the dynamic reliability equations proposed by ISA[1,2]. For every sequence, several data are necessary in order to obtain its contribution to the global exceedance damage frequency. These data consist of the results of the simulations performed with MAAP and TRACE codes that are inside of the damage domain and the time-density probability distributions of the manual actions and the time of seal LOCA occurrence. Reported results show an slight increment of the exceedance damage frequency for this kind of sequences in a power uprate from 100% to 110%.

1. Introduction

As a part of the collaboration between Consejo de Seguridad Nuclear (CSN), Universidad Politécnica de Madrid (UPM) and Indizen Technologies an analysis of sequences with loss of Component Cooling

[1] Izquierdo, J. M., Cañamón I., (2008). TSD, a SCAIS suitable variant of the SDTPD. Safety, Reliability and Risk Analysis. Proceedings of the ESREL, (pp. 163-171), Valencia, Spain.
[2] Hortal, J., & Izquierdo, J. M. (2008). Exceedance Frequency Calculation Process in safety Margins Studies. Spain, Madrid: Nuclear Safety Council (CSN).

Water System (CCWS) has been performed with MAAP[3] and TRACE[4] codes in the context of the NEA application exercise SM2A (Safety Margin Application and Assessment). Integrated Safety Assessment (ISA) methodology intends to consider all the relevant uncertainties (occurrence times and physical parameters) that could impact the considered safety or damage limit. Nevertheless for the sake of demonstrating the method within the SM2A project, the analysis has focused on the treatment of uncertain times since they were expected to be dominant and traditional uncertainty analysis methods could not be applicable. This analysis aimed at understanding the impact of the time of occurrence of the SLOCA, the time needed to recover the CCW system, the time to begin the secondary side depressurization as well as the availability of the accumulators in this kind of sequences. Results shown in this paper are focused on those obtained with TRACE code[5].

ISA methodology has been developed by the Modeling and Simulation (MOSI) branch of CSN, aiming to provide with an adequate method to perform a general uncertainty analysis, making emphasis in those sequences where some events occur at uncertain times or randomly. For a given safety limit or damage limit (PCT within this paper), the numerical result of this methodology consists of the damage exceedance frequency (DEF) for the sequences stemmed from an initiating event. This is done along with the delineation of the dynamic event tree (DET) and the identification of the damage domain (DD) of the sequences that contribute to the total DEF. The damage domain is defined as the region of the space of uncertain parameters of interest where the limit is exceeded. DDs have as many dimensions as the number of uncertainties involved in each sequence. In the case being analysed, DD are up to three-dimensional. In principle, DD dimensions are not limited (i.e., being n-dimensional in general; other examples of three-dimensional DD can be found in Izquierdo (2008)[6]. The UPM group has applied extensively this methodology in several projects[7][8][9]**[10]. ** ISA methodology introduces some differences with respect the classical Probabilistic Safety Analysis (PSA):

- Event tree headers in PSA are usually defined at safety function level, i.e., each header represents the successful achievement of a safety function. System success criteria are therefore needed to develop the header fault trees. In the ISA context, however, event tree headers represent hardware states (system trains working or not) or operator actions. ISA fault trees are used to calculate the probability of each system configuration, not to quantify failure probabilities.[11]
- In PSA event trees, header intervention (i.e., demand of a safety function) is decided on the basis of generic analyses. On the contrary, demand for header intervention in ISA is a simulation result. As a result the number of possible branches in a header is different in PSA

[3] MAAP User's group, 1994. "MAAP4, Modular Accident Analysis program for LWR Power plants". Computer Code User's Manual.

[4] NRC, 2008. Division of Risk Assessment and Special Projects "TRACE V5.0 USER'S MANUAL". NRC.

[5] Equivalent results obtained with MAAP code will be published in an extended paper, and have been also reported in the SM2A final report.

[6] Izquierdo, J.M., Cañamón, I., 2008 "Status report on dynamic reliability: Conclusions of the SDTDP/TDS methods development: Results of its application to the H2 combustion". Severe Accident Research Network (SARNET).

[7] Izquierdo, J.M. et al., 2008. "SCAIS (Simulation Code System for Integrated Safety Assessment): Current status and applications". Proc. ESREL 08, Valencia, Spain.

[8] Hortal, F.J., et al, 2010 "Application of the Damage Domain approach to the calculation of exceedance frequencies" 10th International Probabilistic Safety Assessment & Management Conference (PSAM10).

[9] Ibáñez, L., et al, 2010 "Damage Domain Approach as a Strategy of Damage Exceedance Computation", Proc. Int. Conf. Nuclear Energy for New Europe 2010, Portoroz, Slovenia.

[10] Queral, C. et al, 2010 "Application of the Integrated Safety Assessment Methodology to MBLOCA Sequences" The 8thInternational Topical Meeting on Nuclear Thermal-Hydraulics, Operation and Safety (NUTHOS-8) Shanghai, 2010.

[11] Due to limitations in the available information for the SM2A exercise about the PSA model, the only hardware configurations considered in this analysis are total failure and minimal configuration for PSA success. In a general case, however, there would be a separate header for each redundant train of safety systems or, alternatively, multi decision branching points with a branch for each system configuration.

and ISA. In PSA there are two branches for a header: failure or success, but ISA considers three possible branches for a header: demanded with failure, demanded with success and not demanded.

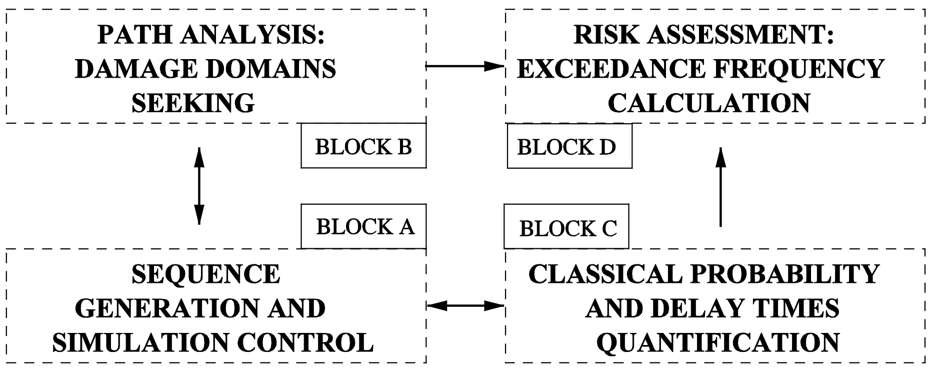
- The end state of a sequence is a discrete variable with two possible values: success or failure. In PSA event trees the end state of a particular sequence takes only one of these values. The end state of ISA sequences, however, is a random variable where each value has an associated probability. Success and failure probabilities are obtained from the sequence uncertainty analysis. PSA end states can be seen as a particular case of ISA end states where the only possible probability values are 0 or 1.

- In PSA a human action is failed if it is not performed within a pre-specified time interval (available time). An action delayed beyond the available time is treated as a non performed action. In ISA methodology, human actions are events occurring at uncertain times. A delayed action is still a performed action even if it is not able to avoid a damage condition (limit exceedance). As a consequence, a PSA success sequence, when analysed in the ISA context, may contain a non-empty damage domain resulting from excessive delays of protective actions.

A high level description of the methodology is given by the flow diagram of Figure 1 which shows the main methodology modules (blocks) and the overall interactions among modules; see also Izquierdo (2008)[7] for further more details:

Block A. The Sequence Generation module performs the simulation of reference DET, allowing to identify the candidate sequences with non trivial DD (success or damage for all conditions) to be analyzed in detail in the Path Analysis module (Block B).
Block B. The Path Analysis module takes each sequence of interest from block A (Sequence Generation), performing multiple simulations with different values of uncertain parameters and/or time delays (human actions or stochastic phenomena). Each such simulation, called a path, can end either in a success or damage state. Those paths ending in damage state are said to belong to the DD of the sequence.
Block C. The probability and delay times quantification module provides the necessary information to calculate in Block D (Risk Assessment) the probabilities and the contribution to DEF of each sequence of interest.
Block D. The Risk Assessment module calculates the DEF by integrating on the DD region, obtained from Block B (Path Analysis module), the frequency density function obtained from the probability distributions evaluated in Block C (Probability module).

Figure 1. ISA methodology general diagram.



The analyses can be iterated to precisely define the DD border, also taking the complexity of the TH code into account to limit computing time.

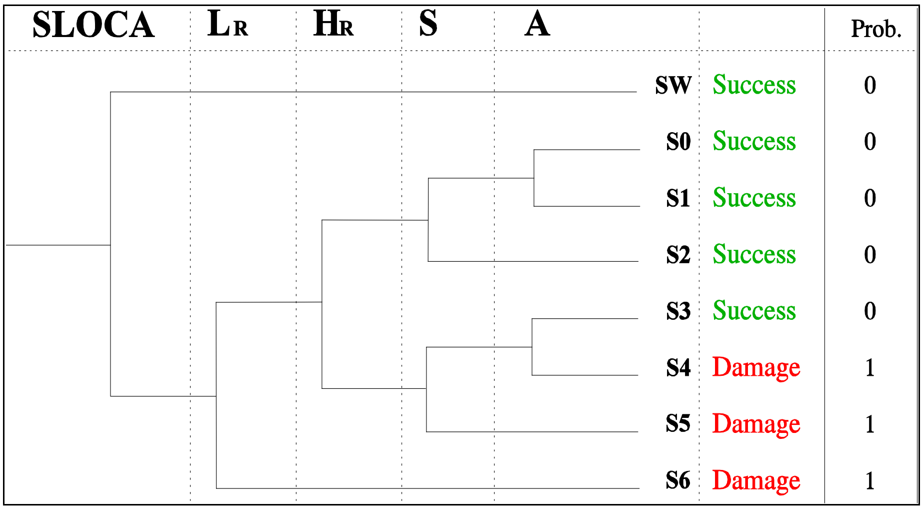
2. Sequence generation module. DET simulation.

The objective of Block A of ISA is to simulate the reference DET stemming from an initiating event. At present, the simulations of DET performed by coupling MAAP and DENDROS are performed in an automatic way[12]. However, in the case of simulations with TRACE code the sequences are still simulated one by one. Coupling TRACE to SCAIS system, in order to be able to an automatic simulation of DET, is currently being developed by Indizen in collaboration with CSN.

2.1. Application to loss of CCWS sequences with SLOCA.

In a first step, several loss of CCWS event trees corresponding to PSA studies of similar nuclear power plants (Westinghouse design with 3 loops) have been analyzed to build a generic loss of CCWS event tree in order to obtain the candidate headers for the DET analysis that is described in the next step of ISA methodology, see Figure 2. These headers could be modified depending on the results obtained from the DET simulations. From the results of this analysis the following sequence headers have been considered: SLOCA (Seal LOCA); H (High Pressure Safety Injection – HPSI with 1/2 trains available); A (Accumulator Safety Injection – ACCUM, 3/4 ACC available); L (Low Pressure Safety Injection – LPSI, 1/2 trains available) which includes the recirculation phase; and S (Primary cooling, at a rate of 55 Kelvin per hour, and depressurization by means of Steam Generators - SG). Additionally, one must be aware that effective intervention of headers H and L depends on the recovery of the CCW system. That is done by adding a recovery condition (R) to the headers H and L, being now H_R and L_R. All the simulations performed in this analysis include the hypothesis of reactor coolant pumps trip coincident with the loss of CCWS event and manual control of auxiliary feedwater system.

Figure 2. Generic event tree for loss of CCWS sequences

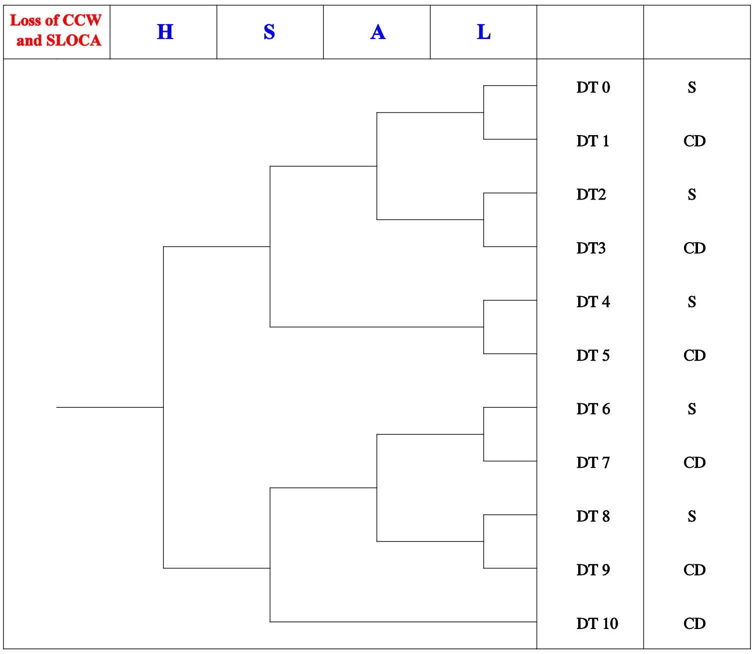


Sequences of the reference DET have been simulated with TRACE assuming that the CCW system is recovered right after the occurrence of the seal LOCA (which occurs at a fixed time, t=2500 s) and

[12] Fernández, I. et al., 2010 “A Code for Simulation of Human Failure Events in Nuclear Power Plants: SIMPROC". Nuclear Engineering and Design (*available online 30 April 2010*).

that all the subsequent actions represented by event tree headers occur (if not failed) without delay from the time they are required. A set of 11 sequences (sequences DT) was generated, see Figure 3. A sample of simulation results is shown in Figure 4 and a summary of significant results for each sequence is given in Table 1**,** where the identification of each sequence is done by the concatenation of header status: a header in upper case means success when demanded and in lower case means failed when demanded. Results shown in Table 1 (TRACE simulations) indicate that headers have been always demanded at the same order. However, in previous simulations performed with MAAP some headers appeared in different orders, depending on the sequence.

Figure 3. Reference DET of Loss of CCWS with SLOCA (TRACE code).



The information obtained from the reference DET (DT sequences, without time uncertainty) allows to identify in which sequences with time/parameter uncertainty (U sequences in Figure 5) the final state is not always success or damage. Introducing time uncertainties an U sequence could be similar to different DT sequences depending on the values of the action delays. For example, delaying the CCWS recovery would make sequence U0 to change from DT0 to DT5 due to the recovery dependence of H and L. Table 2 shows how each U sequence is related with D sequences. From this comparison, it can be concluded that for five U sequences (U0, U1, U2, U3 and U4) the end state is not always success or damage. These sequences are identified in the generic dynamic event tree with uncertainty (GETU), see Figure 5, as sequences with “Damage Domain”. For those sequences, it is necessary to obtain their DD, i.e. the time/parameter region where the paths reach the damage condition. In these sequences the damage/success end state depends on:

- The starting time of the S header, which is a human action with a probability distribution given by the density function $h^S(t)$, t being time from the demand,
- the CCWS-recovery time, which also has a probability density function $h^R(t)$, t being time from the initiating event (which is also the demand for recovery).
- the SLOCA time, which also has a probability density function $h^{SLOCA}(t)$, t being time from the initiating event .

Each probability distribution is described further below. The damage domains are obtained in the Path Analysis Module (Block B).

Figure 4 PCT. Reference DET of Loss of CCWS with SLOCA (TRACE code).

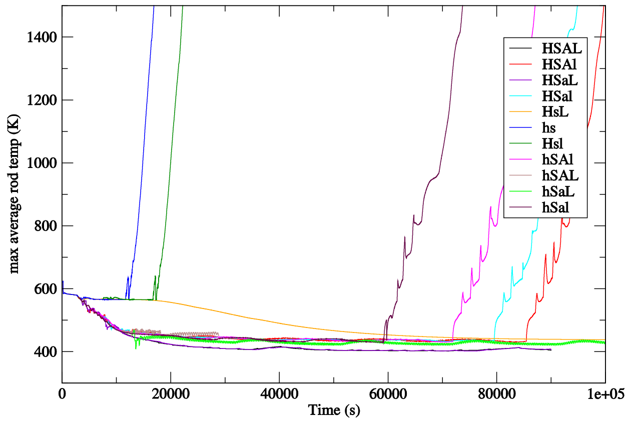


Table 1. Sequence information obtained from the reference DET (TRACE code)

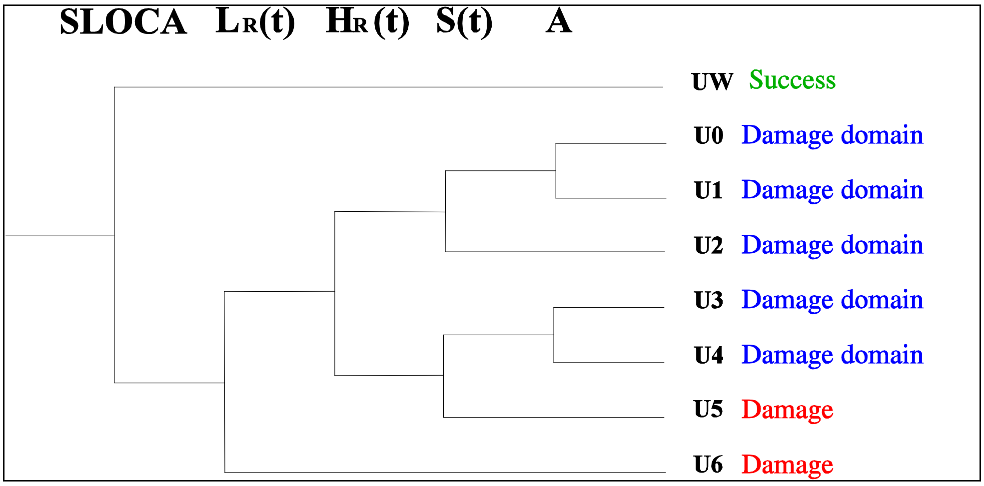
DET Sequence	Time of H header	Time of S header	Time of A header	Time of L header	Time of PCT	PCT
DT0 (HSAL)	2619	3100	5773	7400, R	0	622 K
DT1 (HSAl)	2619	3100	5773	(7400, R)	99386	DAMAGE
DT2 (HSaL)	2619	3100	(5773)	7400, R	0	622 K
DT3 (HSal)	2619	3100	(5773)	(7400, R)	94496	DAMAGE
DT4 (hSAL)	(2619)	3100	5395	11152	0	622 K
DT5 (hSAl)	(2619)	3100	5395	(11152)	86748	DAMAGE
DT6 (hSaL)	(2619)	3100	(5395)	10450	0	622 K
DT7 (hSal)	(2619)	3100	(5395)	(10450)	61787	DAMAGE
DT8 (HsL)	2619	(3100)	-	7400, R	0	622 K
DT9 (Hsl)	2619	(3100)	-	(7400, R)	21934	DAMAGE
DT10 (hs)	(2619)	(3100)	-	-	16631	DAMAGE

R means “demanded in recirculation phase”, time value between brackets means demanded but failed

Table 2. Connection of sequences with time uncertainty and the sequences obtained in the DET

Sequence of GETU (with time uncertainty)	Reference sequences of the DET (without time uncertainty)	Final status of uncertain sequence
U0 (H_RSAL_R)	DT0/DT5/DT8/DT10	S/D/S/D ⇒ DD
U1 (H_RSaL_R)	DT2/DT7/DT8/DT10	S/D/S/D ⇒ DD
U2 (H_RsL_R)	DT8/DT10	S/D ⇒ DD
U3 ($hSAL_R$)	DT4/DT5/DT10	S/D/D ⇒ DD
U4 ($hSaL_R$)	DT6/DT7/DT10	S/D/D ⇒ DD
U5 (hs)	DT10	D ⇒ D
U6 (l)	DT1/DT3/DT5/DT7/DT9	D/D/D/D/D⇒ D

Figure 5. Simplified GETU for loss of CCWS sequences

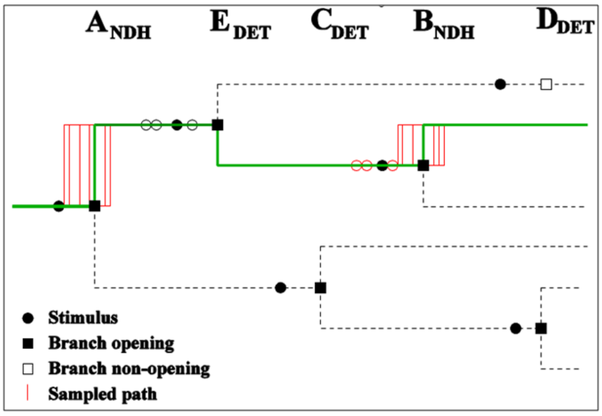


3. Path analysis module

The Path Analysis Module (Block B) receives the sequence and parameter information of all branches of DET from the Sequence Generation Module (Block A) and determines the DD of the candidate sequences. At present, the simulations performed by coupling MAAP and PATH-ANALYSIS are performed in an automatic way, whereas in the case of simulations with TRACE code the sequences are still simulated one by one.

Headers that could occur at uncertain times (mainly operator actions but also events with stochastic phenomenology) are defined as Non Deterministic Headers (NDH). In order to take into account this uncertainty a time sampling between the minimum time when the header event becomes possible and a maximum time (or the mission time, 24 hours) is performed for each NDH, see Figure 6. An example of how to obtain the DD of a sequence is shown in next section. If there are several non-deterministic headers and/or uncertain parameters, a multidimensional time/parameter sampling will be needed. Each sample gives rise to a path belonging to the sequence and the set of paths leading to a damage condition (i.e., limit exceedance) define the DD of the sequence.

Figure 6. Path analysis in a sequence with two NDH (headers A and B).



3.1 Application to loss of CCWS with SLOCA. Simulations performed with TRACE code

The objective of this analysis is to obtain the DD of each sequence. With the DD information it is possible to obtain the DEF of each sequence, which is carried out in the last stage of the ISA methodology (Block D: Risk Assessment) and described in the next section. As an example, the results obtained for sequence U1 (H_RSaL_R) with TRACE code are shown in detail in Figure 7. The calculation process performed in sequence U1 is the following:

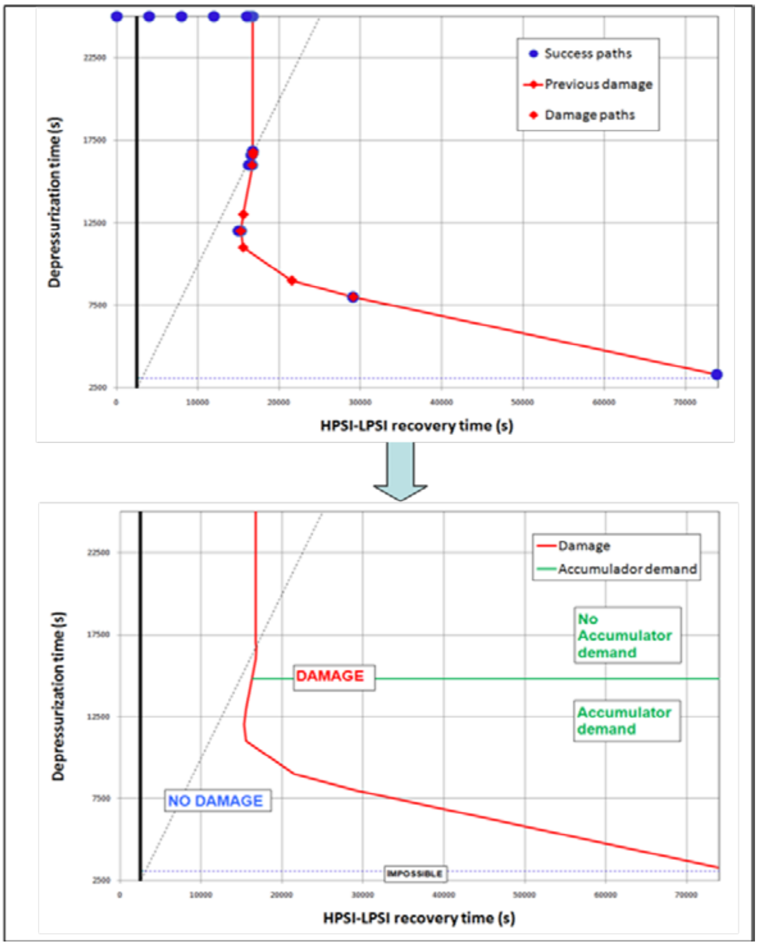
1. Failure of S header is assumed, with the result of no manual depressurization in secondary side. A transient (path) is simulated for each CCWS-recovery time considered. In these sequences, damage will arrive at a certain time t0, which sets the maximum time for the (manual) start of depressurization. Starting depressurization later than t0 time is not useful to avoid damage and no more analysis is required. This region corresponds to time points from the line of Previous Damage (PD) above the diagonal that is shown in Figure 7.

2. Failure of recovery event R is assumed, with the result of no SIS injection due to loss of equipment cooling. A path is simulated for each time of initiation of manual depressurization in secondary side. In these sequences, damage will arrive at certain time t1, which sets the maximum time for the recovery of CCWS. Recovering of CCWS later than time t1 is not useful to avoid damage and no more analysis is required. These time points lead to the line of PD below the diagonal.

3. A set of paths are simulated with different times for the beginning of depressurization and CCWS-recovery times, always below the PD line. Some of the paths exceed the damage condition (red diamond) while other paths do not reach it (blue circle).

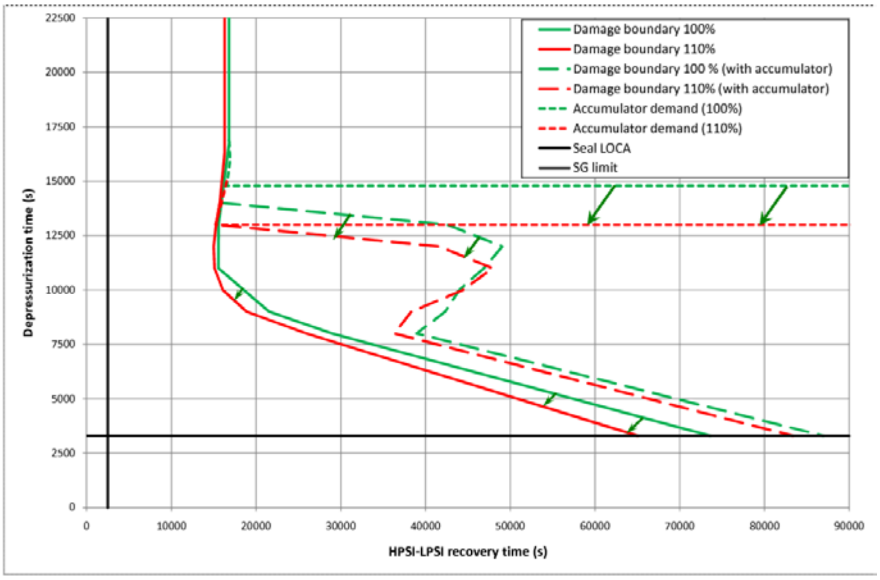
4. The same process has to be done for different times of SLOCA occurrence. For the sake of clarification, this step has not been included in Figure 7.

Figure 7. Path analysis of sequence H_RSaL_R. Loss of CCWS with SLOCA. TRACE code



The analysis has been also applied subsequently to sequence U0 (H_RSAL_R). The comparison of both analysis for U1 and U0 (H_RSAL_R and H_RSaL_R) sequences shows that the availability of accumulators is quite important because it avoids the core damage in a large area, see Figure 8. In addition, Figure 8 also shows the damage domains for sequences U1 and U0 with a 10% of power uprate. It can be seen how the increase of power has modified slightly the shape of the DDs (indicated by the arrows) causing an increase in the area of the DDs.

Figure 8. DD of sequences H_RSAL_R and H_RSaL_R. Loss of CCW (100% and 110%)



These results also illustrate other important difference between classical PSA and ISA methods: as long as several actions with uncertain time are present in a sequence, the available time for each action becomes a function of the previous occurrence times. This effect can be efficiently afforded in ISA while only fixed sequence specific available times are used in PSA. For example, in this analysis the available time for recovery of the CCW system is a function of the sequence (with/without ACC) but also of the starting time of depressurization action and the initial reactor power.

4. Probability calculation and risk assessment

The DEF is obtained by integrating the equations of the Theory of Stimulated Dynamics (TSD) inside the DD of each sequence (further explanation on the equations involved in this module can be found in Izquierdo (2008)[1]. This integration module constitutes the Risk Assessment module (Block D). The equations of the TSD evaluate the frequency density of each path of a sequence and need several probabilistic data that can be obtained from several sources like pre-existing PSA's and stochastic phenomena models (Block C). In the application of the TSD the concept of "stimulus" of a dynamic event plays a fundamental role. The stimulus of an event is a condition that makes the event possible. In the simple case of a protective action the stimulus is the demand of that action. In this analysis the TSD equations the stimuli of all the dynamic events are assumed deterministic, i.e., they can be directly derived from the simulation results. In addition, the probability distributions of NDH do not show mutual dependences and they do not depend on physical variables either. In other words, these probability distributions are known functions of the delay between the activation of the stimulus and the actual occurrence of the event.

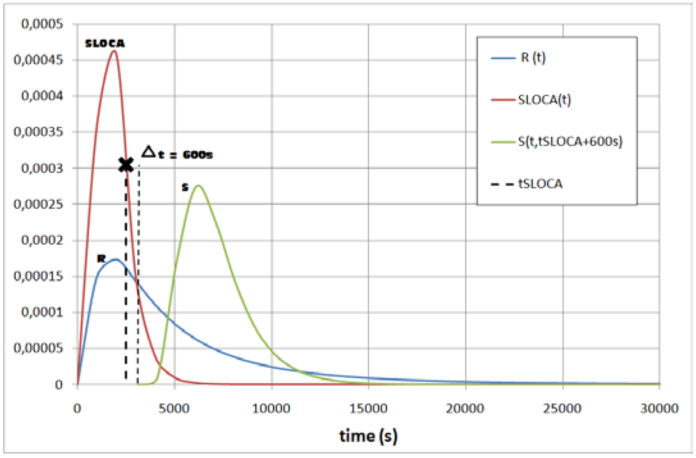
4.1 Application to loss of CCWS with SLOCA. Simulations performed with TRACE code

The data needed are the frequency of the initiating event (Loss of CCWS), the failure probabilities of the headers (H, S, A, L) and the distributions of the delays of NDH, see Table 3 and Figure 10.

Table 3. Initiator frequency and headers failure probabilities

Initiator	**Uncertainty**	**Frequency** (1/year)	**Distribution**
Loss of CCW/W		2.0e-3	
Header	**Type of header**	**Failure probability**	**PDF**
SLOCA	Stochastic	2.1e-1	Lognormal
R	Stochastic	---	Lognormal
H	Deterministic	2.2e-5	---
S	Stochastic	---	Lognormal
A	Deterministic	9,4e-4	---
L	Deterministic	5.6e-5	---

Figure 10. Probability Density Functions (PDF) of S, R and SLOCA.



By integrating the frequency density inside the damage domains the DEF of every sequence is obtained; the numerical integration has been performed by evaluating the frequency density in a uniform grid with integration steps $\Delta\tau_S$, $\Delta\tau_R$ and $\Delta\tau_{SLOCA}$ for depressurization and recovery delays and time of SLOCA, respectively. As an example the integration process for U1 sequence in the two first dimensions is shown in Figure 11. Subsequently, the DDs for different values of time of SLOCA are obtained, as shown in Figure 12. In order to show the relative importance of each region of the damage domain, the product of depressurization and recovery PDFs inside the DDs is showed in Figure 12 for different values of time of SLOCA. Results obtained from the analysis taking into account the three time dimensions for every sequence of the GETU at 100% and 110% of power reactor are shown in Table 4. In this table, only the sequences with blue color needed to be quantified with TSD methods. A frequency threshold of 10^{-7} y^{-1} was established for the SM2A exercise; sequences with lower frequency have not been evaluated.

The TRACE results show that DEF has increased by 30% due to the 10% power uprate, see Table 4.

Figure 11. Example of the integration of PDF inside the DD of sequence S1: H_RSaL_R.

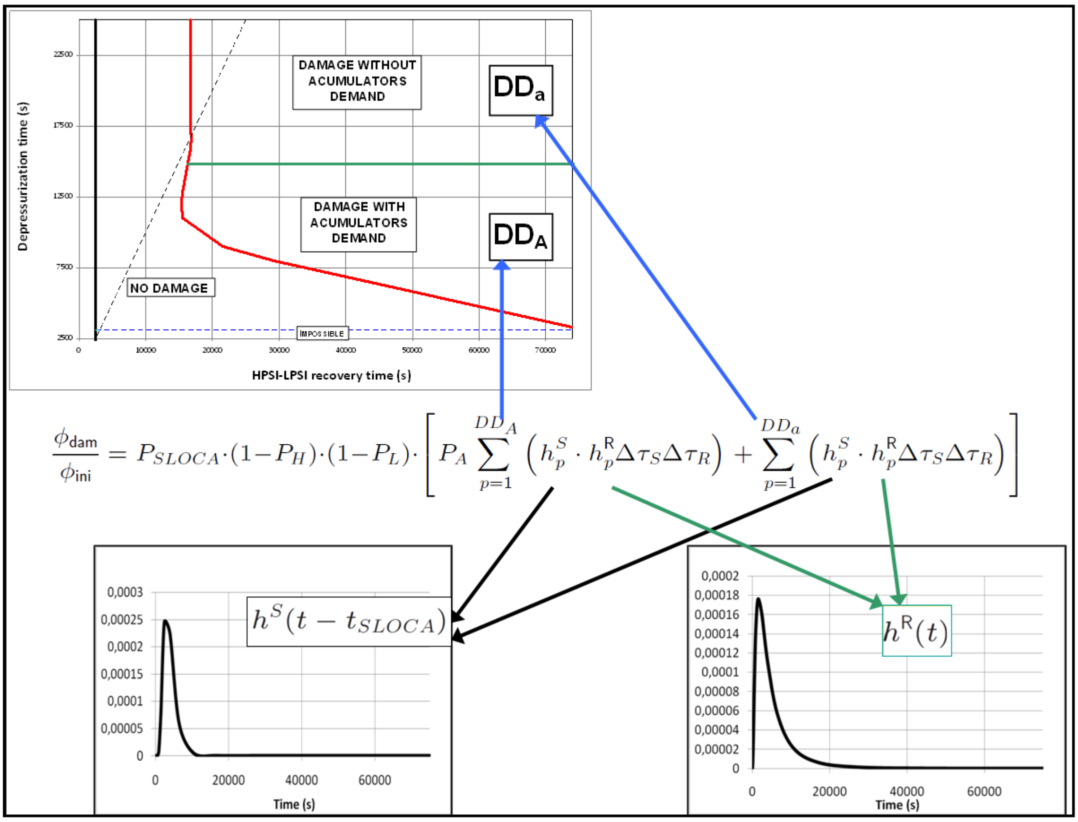


Figure 12. PDF inside the DD of sequences H_RSAL_R and H_RSaL_R for different values of τ_{SLOCA}

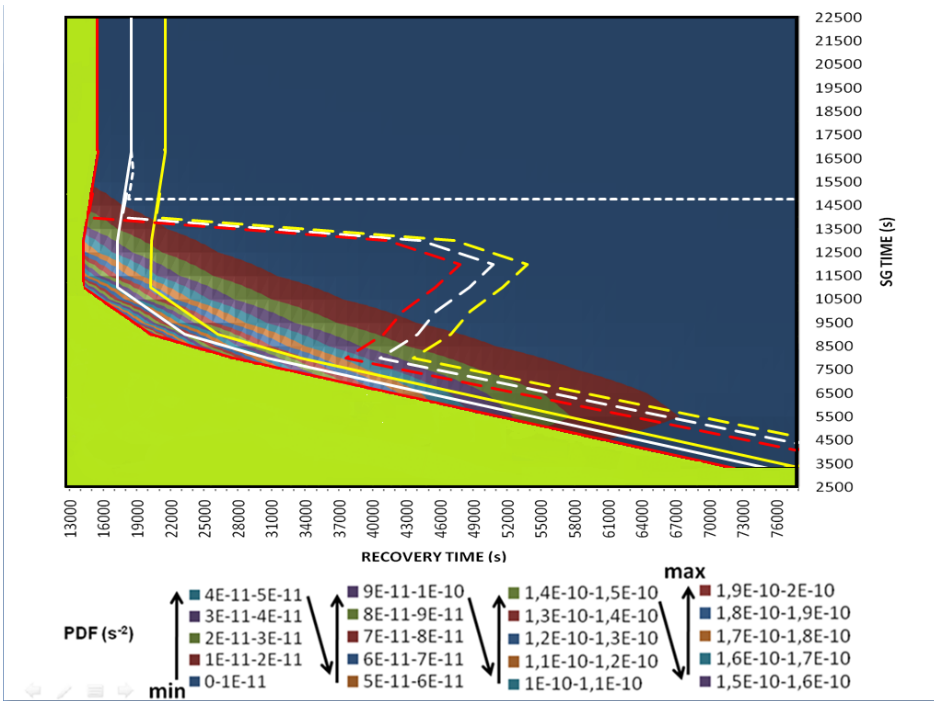


Table 4. DEF of Loss of CCWS event tree (from the simulations performed with TRACE code).

Sequence of GETU	Sequence Frequency (1/year) 100%/110% TRACE	DEF(1/year) 100%/110% TRACE	Conditional exceedance probability 100%/110% TRACE	DEF (1/year) 100%/110% MAAP
UW (R and no SLOCA)	1.58e-03	0	0	0
U0 (H_RSAL_R)	3.94e-04	7.83e-07 /10.03e-07	0.0020/0.0026	3.94e-07 /5.22e-07
U0/U1 (H_RSL_R)	0.89e-07 /0.93e-7	0.93e-07 /0.97e-07	1	15.39e-07 /15.03e-07
U1 (H_RSaL_R)	3.71e-07	0.04e-07 /0.06e-07	0.0108/0.0162	0.08e-07 /0.08e-07
U2 (H_RsL_R)	0	0	N/A	0
U3 ($hSAL_R$)	9.24e-9 < 1.0e-7	--	--	--
U4 ($hSaL_R$)	9.24e-12 < 1.0e-7	--	--	--
U5 (hs)	0	0	N/A	0
U6 (l)	0.22e-7	0.22e-7	1	0.22E-7
Total	**2.0e-3**	**9.02e-7 /11.82e-7**	**0.00045/0.00059**	**19.63e-7 /20.55e-7**
ΔDEF		**2.80e-7 (31%)**		**0.92-7 (5%)**

Results included in table 4 show how the sequence frequency is distributed between success and damage so that almost any sequence can contribute to the DEF. There is also a non-null probability of sequences with non demanded accumulators; this emphasizes the need to distinguish among the lack of actuation of a safety system due to lack of demand or due to system failure. Table 4 introduces also a comparison between results from MAAP and TRACE codes, suggesting that MAAP leads to a more conservative results ($DEF_{MAAP}>DEF_{TRACE}$).

5. Conclusions

The ISA application presented in this paper has showed the importance of Path Analysis and Risk Assessment. In PSA each sequence has a well defined final state, success or damage. However, this analysis has pointed out that it is possible to have for the same sequence a damage probability (PD) and a success probability (PS).

This paper shows an application of the ISA methodology for the analysis of Loss of Component Cooling Water System with SLOCA sequences using two different codes, MAAP and TRACE. Although it cannot be stated as a general conclusion, comparison of both analyses suggests that the results from MAAP code tend to be more conservative than those from TRACE code.

In general, the results have shown the capability and necessity of the ISA-like methodology in order to properly accounting for uncertainties in the time delay of operator response and other stochastic events along with usual parametric uncertainties in the evaluation of the safety in a NPP.

ACKNOWLEDGMENTS
This work is funded by Consejo de Seguridad Nuclear (SM2A project)

Uncertainty and Sensitivity Analyses for CFD Codes: an Attempt of a State of the Art on the Basis of the CEA Experience

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1. Introduction

Uncertainty and sensitivity analyses, associated to best-estimate calculations become paramount for licensing processes and are known as BEPU (Best-Estimate Plus Uncertainties) methods. A recent activity such as the BEMUSE benchmark[1] has shown that the present methods are mature enough for the system thermal-hydraulics codes, even if issues such as the quantification of the uncertainties of the input parameters, and especially, the physical models must be improved[2]. But CFD codes are more and more used for fine 3-D modelings such as, for example, those necessary in dilution or stratification problems. The application of the BEPU methods to CFD codes becomes an issue that must be now addressed. That is precisely the goal of this paper. It consists of two main parts. In the chapter 2, the specificities of CFD codes for BEPU methods are listed, with focuses on the possible difficulties. In the chapter 3, the studies performed at CEA are described. It is important to note that CEA research in this field is only beginning and must not be viewed as a reference approach

[1] BEMUSE Phase VI Report: "Status report on the area, classification of the methods, conclusions and recommendations", OECD/NEA/CSNI/R(2011)4, March 2011.

[2] T. Skorek, A. de Crécy, "PREMIUM – Benchmark on the quantification of the uncertainty of the physical models in the system thermal-hydraulic codes", OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations, Barcelona, Spain, November 2011.

2. Uncertainties in CFD: generalities

2.1. The sources of uncertainties in CFD

Theoretically, the sources of uncertainties are the same for CFD codes as for system codes. But for system codes, the uncertainties of the physical models, the boundary and initial conditions and the material properties are generally the most important ones, and consequently other uncertainties such as those related to discretization of the equations are neglected. That is not the case for CFD codes. A possible list of the uncertainties to be considered is given below (see also chapter 2 of [3]). They are:

- The uncertainties related to discretization, sometimes called "numerical errors". They are the differences between solutions of the exact equations and the discretized equations. In [3], the authors distinguish the "spatial discretization errors", coming from the discretization of the spatial derivatives, the "time discretization errors" coming from the discretization of the derivatives versus time, the "iteration errors" depending on the value of the residuals and the round-off errors. Richardson's method, quoted in [3], makes it possible to know the numerical errors, by using 2 or 3 more or less refined meshings and by knowing the order of the different numerical schemes. Nevertheless some conditions must be fulfilled for its application: the behavior of the output with respect of the size of the cells must be of polynomial type, and the use of different schemes must not lead to bifurcations, in other words the solution must be stable.
- The simplification of the geometry. The geometry of the different parts of a reactor is complex and miscellaneous and consequently its precise description is not always possible. More precisely, one can distinguish:
  - The non-controlled errors;
  - The errors coming from a voluntary simplification in order to avoid numerical problems or too long calculations, for example for the description of the wire-wrapped fuel rods in reactors of SFR type;
  - The use of a porous medium for some volumes.
- The uncertainties related to the physical models, also called "modeling errors". These uncertainties are of two kinds:
  - The uncertainty of the constants of the models. One can quote for example the 5 constants: C_1, C_2, C_μ, Pr_k and Pr_ε of the k-ε model;
  - The choice among different models: for example the different models of turbulence (RANS: k-ε or k-ω, LES, etc.), making or not a hypothesis of non-compressibility, physical amounts such as viscosity or conductivity depending or not on temperature, etc.
- The initial and boundary conditions. Taking them into account can be more difficult than for the system codes. For example, an experiment will provide only flow rates as boundary conditions whereas the code uses velocities: a hypothesis must be made for the density. Another difficulty is that hypotheses must be done for the inlet profiles: flat, parabolic, logarithmic, etc. Besides CFD codes need to give a turbulence rate at the inlet, and its definition is often arbitrary.
- The user's effect. As explained above, it is present for a lot of sources of uncertainties: choice of the numerical schemes, geometrical modelling, choice of the physical models, description of the initial and boundary conditions. Another origin of the user's effect is the analysis of the results: the users can consider a non-converged solution or not completely master the post-processing tools.

[3] European Commission, 5th Euratom Framework Programme 1998-2002, ECORA, Contract n°FIKS-CT-2001-00154, F. Menter et al., "CFD Best Practices Guidelines for CFD Code Validation for Reactor-Safety Applications", February 2002.

– Software errors. They include both “bugs” and mistakes in the documentation. Such errors are generally considered as negligible.

2.2. The difficulties specific to CFD codes

The first difficulty specific to CFD codes is the CPU cost of a code run, generally higher by far than the cost of a run performed with a system code.

Another difficulty comes from the large variety of the modeled objects, which correspond to different parts of a nuclear reactor.

But the main difficulty comes perhaps from the large variety of uncertainty sources, described in §2.1. In particular, it seems compulsory to consider a new type of input variables, the so-called categorical variables (which can be present for system codes, but are rarely taken into account). Discretization schemes are a typical example of such variables: for instance, the user has different choices for the time scheme: explicit Euler, Runge-Kutta order 3, Cranck-Nicholson, etc. These different choices correspond with different “levels” of the variable “time scheme”, which are neither numerical, nor continuous. Consequently this variable requires a specific treatment, especially for sensitivity analysis.

Using an approach based on the propagation of the uncertainty of the input parameters (as for system codes) poses the question of the quantification of these “input” uncertainties. This question is already difficult for system codes, especially for the physical models, and that is why the PREMIUM benchmark[2] is launched. But it is more difficult for the CFD codes for the reasons detailed in both following paragraphs. As explained in §2.1, the modeling errors, i.e. related to the physical models are of two kinds: i) uncertainty of the constants of the models, ii) choice of the models.

For the first kind of uncertainties, expert judgment is rather difficult to apply. Indeed an expert can have an idea of the range of variation of a physical model in a system code, because he has a representation of the associated physics, even if it is approximate. That is more complicated for constants of turbulence models or wall laws. Besides, statistical methods of inverse quantification of uncertainties such as those expected in PREMIUM benchmark[2], can also be theoretically used. But such methods require having experiments with few influential physical parameters. Such experiments exist in CFD but correspond to very specific configurations: plane channel, homogeneous isotropic turbulence for example. There are also more sophisticated experiments (jets, plumes, flows with obstacle, etc.), but this time all the physical parameters are influential together.

The second kind of uncertainties: “choice of the models” is described via the use of categorical variables. The quantification of the uncertainty of the categorical variables (for the choice among different physical models, but also among different discretization schemes) is a difficult issue, not yet solved nowadays. It must be performed by giving a probability to each level of the variable. Without any information, the hypothesis of equiprobability can be made. In the opposite case where the user has more confidence in a level than in another one, this quantification becomes very arbitrary.

Nevertheless, at least for numerical schemes, this issue is less relevant if the calculation is mesh-converged: in this case, all the schemes are equivalent. But the notion of converged calculation is not always obvious and involves an increase of the CPU cost.

In other methods, not based on the propagation of the “input” uncertainties, the user considers several experiments at different scales, devoted to the same transient as that considered for uncertainty analysis (e.g. Pressurized Thermal Shock in the reactor case). The accuracy of the output, i.e. the code-experiment difference, is estimated for these different experiments and extrapolated to the case for which uncertainty analysis has to be done. The extrapolated value of the accuracy represents the desired uncertainty. Even if it is not clearly written, these methods seem close to the UMAE developed by University of Pisa for system codes[4]. The advantage of such methods is that the notion of sources of uncertainties is no more considered and the difficult issue of quantifying their uncertainty is avoided. A drawback is that it requires having at its disposal a lot of experiments at different scales for the same kind of transients. In addition, the method used to extrapolate the accuracy is not clearly defined. Another difficulty comes from the need to take into account the experimental uncertainty of the output considered for the different experiments, because, contrary to the system codes, the accuracy is often of the same magnitude as the experimental uncertainty. Finally, it is important to note that such methods do not provide sensitivity measures, unlike methods based on propagation of "input" uncertainties.

3. The studies performed at CEA

3.1. Introduction

The CEA approach is based on the propagation of the uncertainty of input variables, of real or categorical type. Two studies were performed, both devoted to the treatment of the categorical variables. The considered cases are in 2-D and with monophasic flow. The CPU time per code run is short, no more than two hours. The list of the uncertain input variables was established in order to be plausible, but without being exhaustive. Ranges of variation for real input variables were arbitrarily defined; the levels of the categorical variables were considered as equiprobable. In both cases, experimental data are not considered. These studies are only aimed at defining the methods to take into account the presence of categorical variables.

These 2 studies were carried out with the CFD code Trio_U[5], developed by CEA.

3.2. First study: deterministic approach based on analysis of variance

The first obvious difficulty due to the presence of categorical variables is the sensitivity analysis. Indeed, as categorical variables have not real values, classical measures such as correlation or regression coefficients[6] cannot be used.

The first study[7], performed at CEA and funded by IRSN, tackles only this issue. The studied case is air at 50°C entering a rectangular channel with obstacle as shown in figure 1. The inlet velocity has a linear profile: low at the bottom of the channel, higher at the top. The flow is laminar (Re ≈ 500). The CPU cost is low, around 15 mn per code run, for 125 meshes along the x-axis and 100 meshes along the y-axis. The output is chosen because it is the most sensitive one: it is the standard deviation

[4] F. D'Auria, N. Debrecin, G. M. Galassi, "Outline of the Uncertainty Methodology based on Accuracy Extrapolation (UMAE)", Journal of Nuclear Technology, Vol. 109, No. 1, pg. 21-38, 1995.

[5] C. Calvin, O. Cueto, P. emonot, "An object-oriented approach to the design of fluid mechanics software", Mathematical Modelling and Numerical Analysis, 36 (5), 2002.

[6] A. Saltelli, K. Chan and E.M. Scott, "Sensitivity Analysis", Wiley series in Probability and Statistics, Wiley (2000).

[7] A. de Crécy, "CFD codes : Analysis of variance in the case of a 2-D flow with obstacle", Technical meeting IAEA-OECD (NEA) on Application of deterministic Best Estimate safety analysis, Pisa, Italy, September 21-25, 2009.

of the pressure behind the obstacle, on the symmetry axis, calculated during the time interval [80, t] s, 80s being the time of stabilization of the calculation.

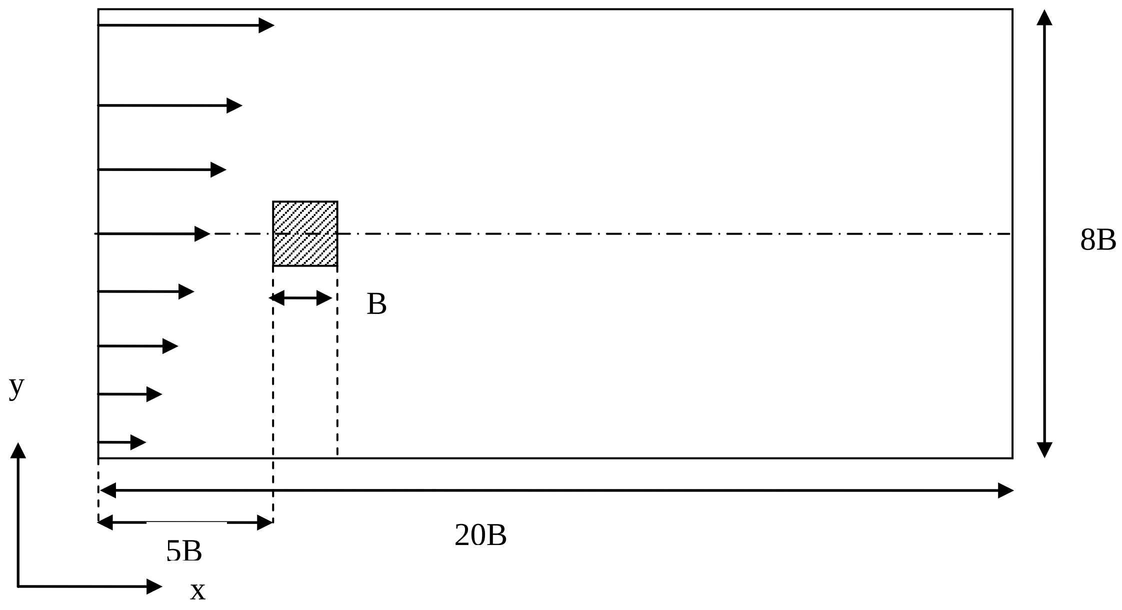

Figure 1: Geometry of the considered case in the first CEA study: channel with obstacle

Eight input variables are considered: real variables such as entering mean velocity or viscosity, but also three categorical variables which are the time scheme, the convection scheme and the pressure solver. The levels of these variables are indicated in table 1. The uncertainty on the meshing is also taken into account with 3 levels: the reference meshing with 6 cells along the obstacle, a coarse meshing with 5 cells along the obstacle and a fine meshing with 7 cells along the obstacle. For each case, all the cells have the same size inside the channel; consequently and for instance, all the cells are smaller in the fine meshing than in the reference meshing.

Table 1: Categorical variables for the case “flow with obstacle”

Description of the variable	Number of levels	Numbering of the levels	Description of the level
Time scheme	4	1	Explicit Euler
		2	Runge-Kutta order 3
		3	Cranck-Nicholson
		4	Implicit diffusion
Convection scheme	3	1	Quick
		2	Centered order 4
		3	Centered order 2
Pressure solver	4	1	PCG (Preconditioned Conjugated Gradient) threshold 10^{-6}
		2	PCG threshold 10^{-5}

		3	PCG threshold 10^{-8}
		4	Cholesky

In this study, classical tools of Analysis of Variance (ANOVA) are used. The interest is that commercial softwares of ANOVA know how to deal with categorical variables, because such variables are usual in industrial problems. The approach is the following:

- One model between the output y and the inputs is supposed. In our case, a second order polynomial model is chosen. It includes quadratic effects for the real variables and all the interactions of order 1, for both real and categorical variables.
- The Design Of Experiments (DOE) corresponding to the model is generated. We have chosen a D-optimal DOE. It is important to note that this DOE is purely deterministic: only the ranges of variation of the real variables are used, with a large majority of the points at the boundaries of these ranges of variation and only few points at their center: no function of density is necessary. The categorical variables are sampled by assuming an equiprobability of their levels. 125 code runs are performed with Trio_U.
- The polynomial regression is built with the 125 points of the DOE. Finally results of ANOVA, i.e. the contribution of each effect (linear, quadratic or interaction) to the total variance of the output is obtained. These results, for example: “The viscosity explains 3.7% of the total variance or the numerical schemes considered together explain roughly 25% of the total variance”, can be viewed as sensitivity analysis results.

The way of doing for obtaining regression coefficients for the categorical variables is as follows. Each categorical variable with n levels is replaced by $(n\text{-}1)$ real coded variables, the values of which are equal to 1, -1 or 0, as explained below for the pressure solver. Let us denote it as X. It has 4 levels (cf. table 1). Three real coded variables, denoted as X_1, X_2 and X_3 are associated to it, according to the table 2.

Table 2: Principle of the coding of the categorical variables

Levels of the variable X : pressure solver	Value of X_1	Value of X_2	Value of X_3
PCG threshold 1.10^{-6}	1	0	0
PCG threshold 1.10^{-5}	0	1	0
PCG threshold 1.10^{-8}	0	0	1
Cholesky	-1	-1	-1

The variable X being replaced by the 3 variables X_1, X_2 and X_3, 3 regression coefficients x_1, x_2 and x_3 are estimated by usual techniques of regression by least squares. When X is at the level 1 (PCG threshold 1.10^{-6}), the effect of X is equal to x_1, when X is at the level 2, its effect is equal to x_2, idem for the 3rd level. Finally, if X is at the forth level (Cholesky), its effect is equal to $x_4 = -x_1 - x_2 - x_3$.

This coding, which is not the only one possible, enables to take into account categorical variables for the determination of the regression and for ANOVA. If the DOE is orthogonal, the formula giving the part of variance explained by a categorical variable X is:

Contribution of X to the total variance = $\sum_{k=1}^{n} p_k x_k^2$, with:

- n: the number of levels of X
- p_k: the probability of the level k
- x_k: the regression coefficient of the k^th coded variable

If the DOE is not orthogonal, as it is the case for a D-optimal design, approximations exist for the estimation of the contribution of each categorical variable X (cf. [8]).

At the end of this study, the issue of sensitivity analysis with categorical variables is considered as being solved. The used approach with the coded variables has nevertheless the drawback to increase the number of regression coefficients to be estimated.

In addition, as written before, the used approach is deterministic. The 125 performed code runs do not enable to have directly uncertainty analysis results because the points of the DOE are not sampled according to the probability density function (pdf) of the input variables. Nevertheless, a probabilistic approach can be used by considering the polynomial regression: the input variables are sampled a very large number of times according to their pdf, the value of the output being at each time calculated by using the regression. Order statistics can then be used to derive percentiles and consequently performing uncertainty analysis. Sobol's indices[8] can also be used, always with the regression, in order to have sensitivity measures. These probabilistic methods have however the drawback to consider the regression instead of the CFD code itself. Metamodels more sophisticated than the polynomial one can be considered, provided that they are able to take into account categorical variables: one can quote, for example, neural networks or kriging.

3.3. Second study: probabilistic approach

This time, a probabilistic method close to that often used for system codes and based on propagation of the uncertainty of input parameters[1] is used. For sensitivity analysis, at least two methods are tested, including a method close to that developed in the first study, but by considering only a first order regression.

The considered physical case cannot be described, for confidentiality reasons, according to a request of IRSN, which funded the study. Nevertheless, the main features of this case can be quoted and are: stationary calculation and turbulence described via a k-ε model. One reminds too that it is a 2-D case, in monophasic, as already written in §3.1.

Anyway, the foremost is the method developed for this study, which is detailed below. Among the considered outputs, there are two scalar quantities: the first one is simple since it is a temperature at a given point and a given time; the second one is more complex since it is a maximum value of the temperature profile at a given time. 24 input variables are of real type: for instance, boundary conditions, physical properties of the fluid, constants in the k-ε model, etc. There are only 3 categorical variables. The first one is related to the initial conditions with 3 possible levels corresponding with 3 different configurations, the second one is the time scheme with also 3 levels and the last one is the type of boundary condition at the outlet with 2 levels: imposed pressure or imposed pressure gradient. Like in the first study presented in §3.2, more attention was paid to the applicability of the method than to the definition of the uncertain input parameters and the quantification of their uncertainty. In particular, only input variables which can be easily modified from the input data deck are considered.

Uncertainty analysis is usually performed by using order statistics. And just to apply them, there is only one hypothesis, which concerns the probability density function (pdf) of the output: it must be continuous, in other words the different values of the output obtained by sampling the input variables must be all different. The presence of categorical variables introduces a certain degree of discontinuity of this pdf. More precisely, if there were only categorical variables, the pdf would be "perfectly discontinuous", with a succession of peaks, each of them corresponding to the value of the output for a given combination of the levels of the categorical variables. With the mixing of

[8] I.M. Sobol, "Sensitivity estimates for non linear mathematical models", Mathematical Modelling and Computational Experiments, 1, 407-414, 1993.

categorical and real variables, the pdf of the output is smoother. Nevertheless, it seems possible to have the case, where, for a given level of a categorical variable, the continuous real input variables have no effect. Consequently, each time that this level is sampled, all the values of the output are strictly equal. Such a theoretical case is illustrated in figure 2, for an output of temperature type: there is a discontinuity when the temperature is equal to 1000°C.

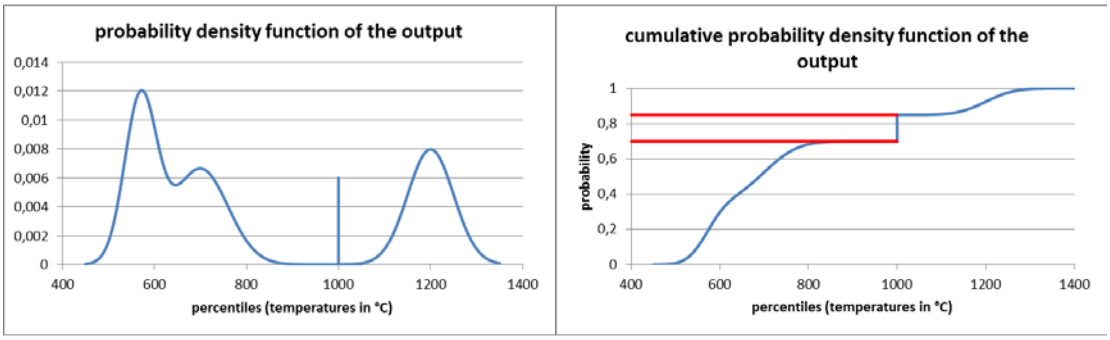

Figure 2: An example of discontinuity of the pdf of the output, resulting from the presence of categorical variables

In the figure 2 (on the right), it is apparent that the value of the temperature equal to 1000°C corresponds to any probability between 70% and 85%, which is a somehow difficult situation for uncertainty analysis.

One out of the goals of the study presented in this §3.3 is to check that this situation does not occur. Consequently, the 24 real and the 3 categorical input variables are sampled according a SRS (Simple Random Sampling) DOE. This sampling takes into account the pdf of the real inputs, whereas a hypothesis of equiprobability is made for the levels of the categorical variables. 100 code runs of Trio_U are performed. One checks that, for each output, two strictly equal values are never obtained. As a consequence, order statistics can be used without problem: quantities such as 2.5% and 97.5% percentiles can be derived. However, this promising result is perhaps due to the low proportion of categorical variables among all the input variables and must be checked in other studies.

The values of the scalar outputs obtained with the 100 code runs are also used for sensitivity analysis. A first method similar to that presented in §3.2 is used, each categorical variable with n levels being replaced by $(n-1)$ coded real variables. But this time, the considered regression is of linear type, the DOE is simply SRS, and takes into account the pdf of the inputs.

A second method based on coefficients of correlation is used. The definition of the contribution of each input to the total variance of the output is different depending on the nature of the input, real or categorical. For the real variables, let ρ be the coefficient of correlation of the X real input parameter with the output y. The contribution of the X parameter to the total variance of y is equal to $\rho^2 \times \hat{var}(y)$, where $\hat{var}(y)$ is the empirical variance of y. The categorical variables are treated with the following method. Let X be a categorical variable, with n levels. Let e_k be the mean value of the output y when X is at its level k, i.e. $E(y/X)$. Let p_k be the frequency of the number of cases where X is at its level k (close to $1/n$ if the levels are isoprobable). Finally, let $\bar{e}$ be the mean value of the e_k, considering all the levels: $\bar{e} = \sum_{k=1}^{n} p_k e_k$. The contribution of X to the variance of y is the empirical variance of the e_k: $\sum_{k=1}^{n} p_k (e_k - \bar{e})^2$, in other words the conditional variance of the expectation value of y knowing X: $var[E(y/X)]$.

Both sensitivity methods give very similar results for the first scalar output (temperature at a given point and a given time): the dominant parameters are the same and their sensitivity is very close in both methods. The coherence between both methods is only qualitative for the second scalar output, of type maximum. The three dominant parameters are the same, but their sensitivity is significantly different, depending on the method. This result can be explained by the complex nature of the output, which is a maximum, and also by the fact that the linear regression is of poor quality with a coefficient of determination equal to 0.42.

4. Conclusion

This paper deals with the relatively new and complex issue of uncertainty and sensitivity analyses for CFD codes.

In the first part of the paper, the difficulties specific to the CFD codes are listed, as well as the different sources of uncertainties, more numerous and various than for the system codes. Two possible approaches are mentioned for uncertainty analysis. The first one lies on the propagation of the uncertainty of input variables, as it is often done for system codes. But, in CFD, some input variables are of categorical type: the variables have different levels which are not numerical. It can be the case of the time scheme, the levels of which being for example explicit Euler, Runge-Kutta order 3, Cranck-Nicholson, etc. Another difficulty, more important for CFD codes than for system codes, is the quantification of the “input” uncertainties: those of the classical real input variables, as for system codes, but also the determination of the probability of each level for the categorical input variables. A second approach lies on the extrapolation of the accuracy of the output, i.e. the code-experiment difference, calculated for several experiments devoted to the study of the same phenomena and at different scales. Finding sufficiently numerous experiments and defining a proper way to extrapolate the accuracy, are difficulties to be solved for this second approach.

In the second part of the paper, the CEA approach is presented. It is based on propagation of “input” uncertainties, i.e. of the first type approach among both possible approaches quoted above. Two simple studies were performed, where solutions are proposed to take into account the presence of categorical variables, both in uncertainty and sensitivity analysis.

CEA work on the issue of BEPU approaches for CFD codes is only beginning and must be completed. To this end, another study is planned where, unlike the two former ones, experimental data are available. This study is aimed at checking uncertainty results with respect to experimental data. As a consequence, a special attention will be paid to the list of input variables. The issue of the quantification of their uncertainty seems too difficult to be considered now. This latter point must be, of course, improved.

Finally, CEA, in collaboration with EDF, intends to tackle another difficulty specific to CFD codes: the high CPU cost of a code run. To this end, the method based on autoregressive models will be tested[9]. For this method, one considers that the CFD code can be run with different levels of sophistication, depending for example on the refinement of the meshing. Efficiency of uncertainty analysis is improved by combining few expensive runs of the most complex level of the code (with a refined meshing) with more numerous and relatively cheap runs performed with the simpler levels of the code (with coarse meshing).

[9] M.C. Kennedy and A. O’Hagan, “Predicting the output from a complex computer code when fast approximations are available”, Biometrika, 87, 1, pp. 1-13, 2000.

Preliminary insights of the impact of steady state uncertainties on fuel rod response to an RIA

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ASTRACT

Nuclear fuel cycles are driven to higher burn-ups under the nowadays more demanding operational conditions. The operational safety limits that ensure the long-term core coolability are set through the analysis of the design basis accidents, RIA scenarios. These analyses are mainly performed through computer codes that encapsulate the present understanding of fuel behavior under such events. They assess safety margins and therefore their accuracy is of utmost interest. Their predictability is extended and validated through separate effect and integral experiments data.

Given the complexity of working phenomena under RIA conditions, neither existing thermo-mechanical models nor rod state characterization at the moment of power pulse reflect exactly reality. So, it is essential to know how these deviations influence code predictions. This study illustrates how uncertainties related to pre-transient rod characterization affect the estimates of nuclear fuel behavior during the transient. In order to do so, the RIA scenario of the CIP0-1 test (CABRI program) has been analyzed with FRAPCON-3 (steady state irradiation) and FRAPTRAN 1.4 (transient) codes. The input uncertainties have been propagated through the codes by following a deterministic approach.

This work is framed within the CSN-CIEMAT agreement on "Thermo-Mechanical Behavior of the Nuclear Fuel at High Burnup".

1. INTRODUCTION

The current nuclear industry trend to increase fuel burnup under more demanding operational conditions, has highlighted the importance of ensuring that, even under those conditions, a fuel rod would be capable of withstanding thermal and mechanical loads resulting from any design basis accident and, particularly Reactivity Initiated Accidents (RIA). Computer codes were created and are being updated consistently. At the same time, they are validated against separate effect and integral tests.

Given the huge complexity of RIA scenarios, models encapsulated in codes can not capture reality with a perfect accuracy. They rely on hypotheses and approximations and they are fed with variable values that are uncertain to some extend. Estimating the impact of such uncertainties is of a high interest in Best Estimate (BE) safety analysis.

Uncertainty analysis (UA) have been profusely used and developed in the area of Thermohydraulics, the OECD BEMUSE project being an example. Some applications have been conducted also in RIA related fields, such as the bundle response to a RIA (Diamond et *al.*, 2000, Le Pallec and Tricot, 2002, Panka and Kereszturi, 2004, Avvakumov et *al.*, 2007, and Anchel et *al.*, 2010) or even when modeling experimental rodlet transients (Laats, 1981, Barrio and Lamare, 2005). However, very few applications are known in the RIA field itself, the most recent one being the work by Sartoris et al. (2010), where uncertainties are considered in the development of a methodology to set safety criteria for RIAs.

This work is aimed to assess the impact of uncertainties in the description of steady state irradiation on the results of a RIA accident. Hence, uncertainties affecting steady and transient models and uncertainties defining the transient scenario are out of the scope of this work. To do so, the CIP0-1 test of the CABRI program was chosen as a case study. Fuel rod thermo-mechanical behavior was simulated by means of FRAPCON-3 (steady state irradiation) and FRAPTRAN (transient) performance codes. After a brief presentation of the computing strategy, a UA methodology is proposed and applied to the quantification of uncertainties when modeling CIP0-1 scenario, with the objective to 1) identify the critical steady state uncertain input parameters and 2) quantify the maximum accuracy that can be reached in a RIA modeling.

2. APPROACH AND TOOLS

2.1. Scenario

The CIP0-1 integral test has been chosen as our RIA reference scenario. The CIP0-1 experiment was performed in 2002 in the framework of OECD CABRI International Program (CIP). Around 99 cal/gUO_2 were injected in 32.4 ms into a short rodlet previously irradiated up to an average burnup around 68.5 GWd/tU.
Several reasons support this test selection: its initial and boundary conditions are prototypical of a Rod Ejection Accident in PWR; it was thoroughly characterized experimentally both in its steady state irradiation (ENUSA, 2009) and transient (Georgenthum, 2011) stages; there is available documentation; and it was previously modeled by other researchers (e.g. Romano et *al.*, 2006).

2.2. Best Estimate Modeling

2.2.1. Analytical tools

The "FRAP" codes family has been used in this study. FRAPCON-3 is the steady state fuel performance code maintained by PNNL on the behalf of US-NRC (Geelhood et *al.*, 2011a). FRAPTRAN is the corresponding transient fuel performance code (Geelhood et *al.*, 2011b). FRAPCON-3 predicts fuel thermo-mechanical performance by modeling the material responses of both fuel and cladding under normal operating conditions, whereas FRAPTRAN include models, correlations and properties for various phenomena of interest during fast transient and accidental conditions. Both codes are considered as BE codes, since they are capable of modeling all the relevant processes in a physically realistic way and were validated against integral test databases (Geelhood et *al.*, 2011c and d).

2.2.2. Global strategy

The modeling of the full CIP0-1 test is the result of a 3-step calculation chain, as shown in **Fig. 1**. The characterization at base irradiation EOL is provided by the use of FRAPCON-3 code version 3.4a: a modeling of the complete mother rod behavior is run first, before limiting it to the span which is submitted to the power pulse. The transient modeling is then carried out using FRAPTRAN code version 1.4.

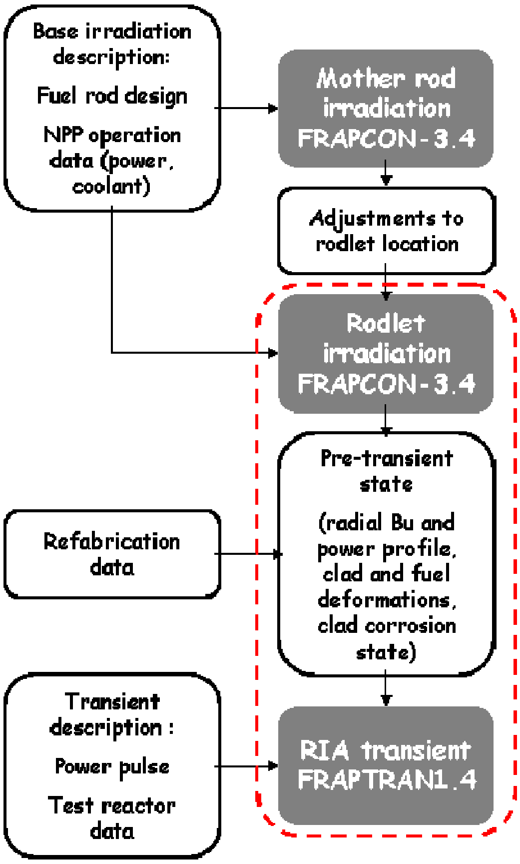


Fig. 1*. CIP0-1 modeling scheme*

2.2.3. Hypotheses and Assumptions

The CIP0-1 mother rod was irradiated during five cycles (eq. 2078 EFPD) in the Spanish reactor Vandellós-II at a core average power of 18 kW/m. The CIP0-1 rodlet was reconditioned from span 5 of the mother rod in Studsvik facilities.

The CIP0-1 test was performed on November 2002 in the CABRI sodium loop facility (Cadarache). 99 cal/gUO_2 were injected at PPN without failure.

Concerning the transient phase, the input deck built can be considered as a BE since a major part of the information is adjusted to the available experimental data: power pulse and axial power profile, coolant temperatures, oxide layer thickness. For simplification sake, no transient FGR is considered (default option).

2.3. Uncertainty analysis setup

2.3.1. Sources of uncertainties

Modeling a RIA involves a number of uncertainties. They may be classified as follows:

- Transient uncertainties. They include both those involved in the scenario description (i.e., affecting variables, such as power pulse width, total injected energy, etc…) and those derived from the own simulation approach (i.e. specification of materials, modeling of specific phenomena, definition of boundary conditions, etc…)
- Steady state uncertainties. These include all uncertainties coming from the steady state irradiation. Their significance lies on their effect on the characterization of the rod at the onset of the transient. As the former type, they entail both uncertainties from the irradiation description and uncertainties from its simulation, which affect input parameters and steady state models.
- In the case of CIP0-1, one should add the uncertainties related to the rodlet refabrication to fit it to the experimental device.

2.3.2. Scope

The UA conducted here entails assessing the impact of uncertainties in FRAPCON-3 input deck on FRAPTRAN predictions of the rodlet response to a power pulse. That is, uncertainties from any other sources discussed about are out of the scope of this study. As a consequence, the resulting band of FRAPTRAN estimates should be seen as a minimum, since addition of any other type of uncertainty introduced above would mean broader bands. It is noteworthy that in this case FRAPCON-3 and FRAPTRAN codes behave just as propagation tools.

It is assumed that only FRAPCON-3 input data are subject to uncertainty and none propagation is sent back to FRAPTRAN input file. The final response behavior is studied as a function of the perturbations made on FRAPCON-3 input deck file, while FRAPTRAN input file remains unchanged. In the case of CIP0-1 FRAPTRAN BE simulation, restart file has to be manually updated at each run, in order to account for refabrication gas and clad corrosion states.

2.3.3. Responses of interest

Regarding the code response, our attention is focused on the following safety-related mechanical variables:

- Maximum clad elongation $\Delta l\big|_{max}$: data are experimentally available, and it reflects in a physical way the clad tube mechanical response to the power insertion.

- Peak residual clad circumferential deformation ε_θ^p : experimental data are also available, foremost ε_θ^p is the critical parameter of the clad ductile failure model implemented in FRAPTRAN1.4.

- Peak Strain Energy Density $SED\big|_{max}$ value is used to estimate the cladding failure damage through the SED-to-CSED ratio (e.g. (Huguet et *al*., 2010)); given the similar behavior between both SED formulations calculated in FRAPTRAN codes (PNNL and EPRI), ^{PNNL}SED has been selected by default.

2.3.4. Selected methods

Given that, on the one hand the analytical tools selected are not equipped with integrated UA package, and on the other hand PDF are difficult to assign to the selected uncertain parameters, it has been decided to explore deterministic propagation methods inspired of Design of Experiment (DOE) theory. The two methods used in the frame of this study are detailed hereafter: OAT and RSM.

i. One-At-a-Time (OAT) method
The OAT approach is a sort of systematic parametric study. The OAT experimental design consists in perturbing only one input factor per computer run: each input is evaluated at its upper or lower value, meanwhile other stay at their nominal value; it implies a total of 2k+1 runs, if k is the factor number. The information from the OAT design can be used to screen the input variables, i.e. to make a preliminary identification of the most sensitive inputs, at a local scale (e.g. Kazeminejad, 2007). Such a simple method is convenient when the number of uncertain factors is high, but not enough robust to give quantitative insights.

ii. Response Surface Method (RSM)
The main idea of RSM is to approximate the original code response by an analytical "function" (called surrogate, proxy or metamodel) from a prescribed database of computations, and then use this replacement model for subsequent UA (e.g. Chun et *al.*, 1996, or Zangeneh et *al.*, 2002). Different types of response surface can be generated, according to response linearity degree: polynomial, thin plate splines, neural networks, generalized linear model, partial least square regression, MARS, boosting…Depending on the response surface form, computer experimental sampling size and design should be carefully optimized: complete or fractional factorial plan, Hadamard matrix, Placket-Burman matrix, and even more elaborate sequential and adaptive plan.

2.3.5. UA sequences

The UA follows a progressive application methodology, as illustrated by Kerrigan and Coleman (1979).

i. Screening analysis
The first step in UA involves identifying the "a priori" relevant FRAPCON-3 input parameters likely to influence the responses of interest defined above. This screening analysis is achieved on the basis of subjective expert judgment.

ii. Importance assessment analysis
In a second step, after having assigned variation range to each factor identified in the previous section, an OAT computing method is applied to highlight the most relevant ones.

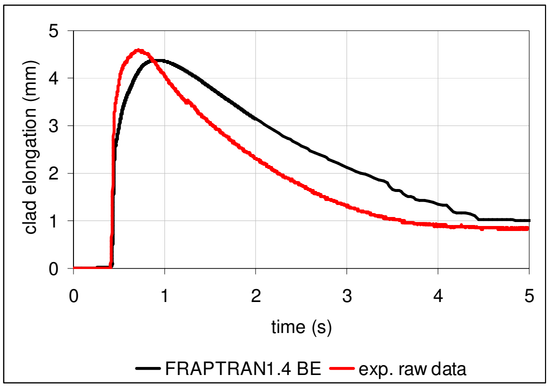
iii. Sensitivity analysis
Sensitivity analysis (SA) is the study of how the uncertainty in the output of a mathematical model can be allocated, qualitatively or quantitatively, to different sources of uncertainty in the model input data.
The RSM computing method is chosen as the basis of SA. Its application proceeds according to the following procedure:

- Design of the experimental matrix: steady state input uncertain factors are perturbed in a prescribed way in order the numerical output response database gets the appropriate properties for subsequent model exploration.
- Determination of the adequate surface response function: given that this study is an exploratory work of UA methods application, the priority would be given to a simple function.
- Estimation of the response uncertainty by means of a second-order error propagation technique: the response "surface" developed replaces the original model in order to quantify response uncertainty.

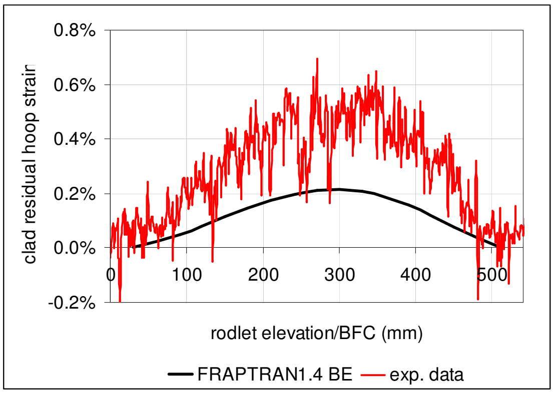
3. RESULTS AND INTERPRETATION

3.1. Best Estimate results

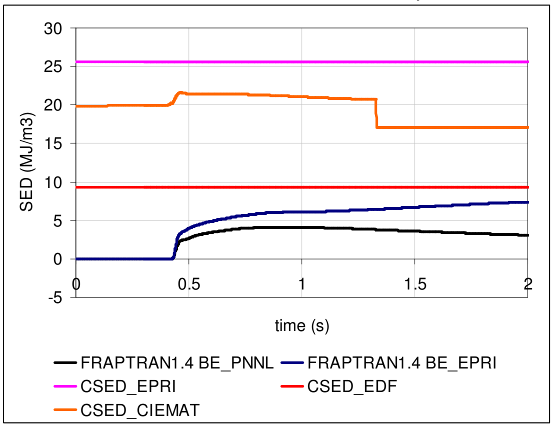
A synthesis of the BE modeling results is encapsulated in **Fig. 2** and **Table 1**.



(a) Clad elongation evolution (normalized) : predicted *vs* experimental



(b) Clad residual hoop strain along rodlet : predicted *vs* experimental



(c) SED *vs* CSED evolutions

Fig. 2. *BE results for the 3 responses of interest defined in section 2.4.3.*

Table 1*. BE numerical results of interest*

Variables		BE numerical result	Experimental data	Relative error (%)
$\Delta l\|_{max}$	Maximum clad elongation (mm)	4.99 4.39 (normalized)	- 4.59	4
ε_{θ}^{p}	Permanent hoop strain at PPN (%)	0.209	0.694	232
$SED\|_{max}$	Peak PNNL-SED value (MJ/m^3)	4.09	-	-

3.2. Screening analysis

A set of FRAPCON-3 input parameters has been selected. Two major criteria were adopted a priori: parameters likely to influence transient behavior and uncertain data present in the input deck. Additionally, if the input value is needed, and no default value is available and/or the assigned value is arbitrary, the parameter is also considered.

A systematic review of the restart file from FRAPCON-3 that feeds FRAPTRAN calculations, allowed identifying the information passed and then choosing 9 parameters considered uncertain (**Table 2**) in the FRAPCON-3 input deck (out of a total of 70). All these parameters are assumed to be independent.

Table 2*. FRAPCON-3 uncertain input parameters*

Code flag	**Definition**	**BE nominal value**	**Unit**
deng	open porosity fraction for pellets	0(1)	%TD
flux	conversion factor between fuel specific power and fast neutron flux	2.21E+16(1)	n/m^{2}s per W/g
fgpav	as-fabricated filling gas pressure (at room temperature)	2.35E+06(2)	Pa
cldwks	clad cold-work (fractional reduction in cross-section area due to manufacturing processing)	0.5(3)	ND

rsntr	expected increase in pellet density during in-reactor operation	52.656(4)	kg/m^3
slim	swelling limit	0.05(1)	volume fraction
cpl	cold plenum length	0.047(5)	m
roughf	surface fuel roughness	4	microns
roughc	surface clad roughness	2	microns

(1) Code default value
(2) Experimental value
(3) Recommended value by code developers
(4) Determined from standard resintering test performed on fresh fuel (NUREG-0085)
(5) Variable adjusted to fit with experimental free volume value

3.3. Importance assessment analysis

3.3.1. Input parameter characterization

For each input parameter defined in **Table 2**, large upper and lower limits have been defined according to available technical data and expert judgment (**Table 3**).

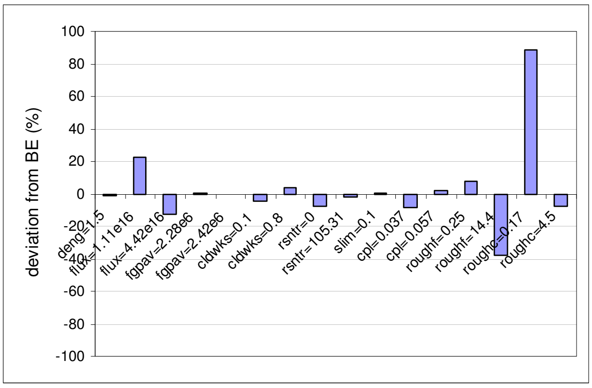
Table 3. *FRAPCON-3 uncertain input parameters and their range of variation*

Code flag * required input	**Unit**	**Lower bound**	**Upper bound**	**Perturbation rationale**
deng	%TD	0 (=BE)	1.5	(i)
flux	n/m^{2}s per W/g	1.11E+16	4.42E+16	/ and * by 2 [-50 %|+100 %]
*fgpav**	Pa	2.28E+06	2.42E+06	+/-3 % (Geelhood et *al.*, 2009)
cldwks	ND	0.1	0.8	Applicability range of FRAPCON-3 mechanical correlations (Geelhood et *al.*, 2008)
*rsntr**	kg/m^3	0	105.31	(ii)
slim	volume fraction	-	0.1	
*cpl**	m	0.037	0.057	+/- 1 fuel pellet height (+/-27 %) (Geelhood et *al.*, 2009)
*roughf**	microns	0.25	14.4	(iii)
*roughc**	microns	0.17	4.5	

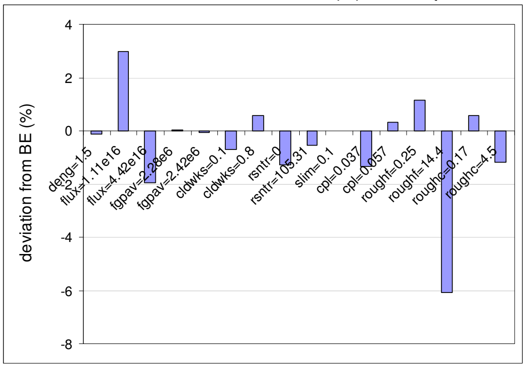
Some of the values set in **Table 3** deserve further explanation:
(i) When *deng* is set to 0, an internal model takes over the calculations of open porosity; and for fuel density higher than 95.25 %TD, the corresponding fraction is set to 0. Commercial fuel rods are not expected to show high open porosity values. However, according to Na et *al.* (2002), for a fuel density of 95.7 %TD (as fabricated density of CIP0-1 fuel), the open porosity ε_0 may reach 1.5 %TD.
(ii) The study of the uncertainty related to *rsntr* and *slim* input variables involves exploring fuel densification and swelling steady state models. Upper limits were fixed arbitrarily to the double of their nominal values; in the case of *rsntr*, this criterion is consistent with the value of 1% (=109.6 kg/m^3) recommended by the code developers.
(iii) Fuel and clad roughnesses are hardly reported in papers and reports. Based on data found (Lassmann and Blank, 1988) it has been decided to set large margins, realistic though.

3.3.2. Importance sampling

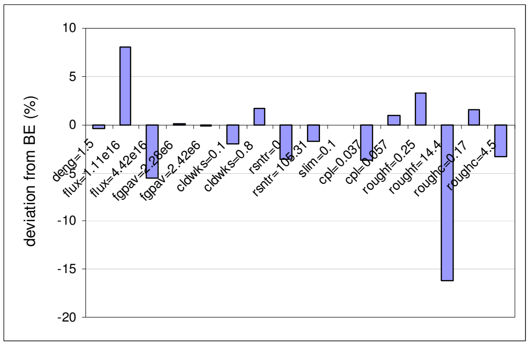
Once the uncertainty ranges assigned, an OAT method was conducted and the results obtained are summarized in **Fig. 3**. The rationale adopted to choose relevant parameters is sketched in **Fig. 4**.



(a) Clad permanent hoop strain at PPN



(b) Maximum clad elongation



(c) Peak PNNL-SED value at PPN

Fig. 3. *Relative sensitivity of responses of interest to steady state input parameters perturbation*

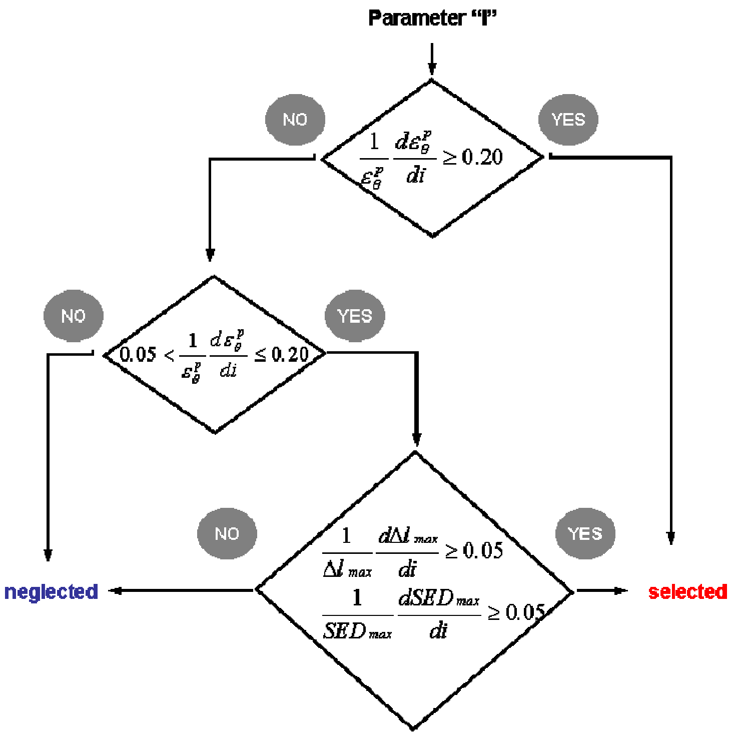


Fig. 4*. Factor discrimination methodology*

As a result, a classification is here proposed:

- Negligible: *slim*, *deng* and *fgpav*
- Moderate: *cpl, cldwks,* and *rstnr*
- Dominant: *roughf*, *flux* and *roughc*

It is worth reminding no quantitative information (such a ranking between *roughf*, *flux* and *roughc*) can be extracted from an OAT analysis.

i. *flux (F)*

The variable *flux* is called at the beginning of each time step to convert the linear heat generation as given in input to the corresponding fast flux, and by summing, to the fast fluence. This cumulative dose measures the material damage due to irradiation and is involved in the calculation of clad mechanical properties for instance. The information on the flux-to-heating conversion factor (also called kerma factor), which is requested as a function of the axial elevation, is rarely given by NPP operators. Its value which depends on neutron cross section data and reactor design can be calculated by means of 3D neutronic codes. The uncertainties associated to individual kerma factors quoted in the literature may reach up to 10 % (Bichsel, 1974). As a consequence, large bounds have been assigned to *flux* parameter, also to account for axial deviations.

ii. *roughf* (ρ_f) and *roughc* (ρ_c)

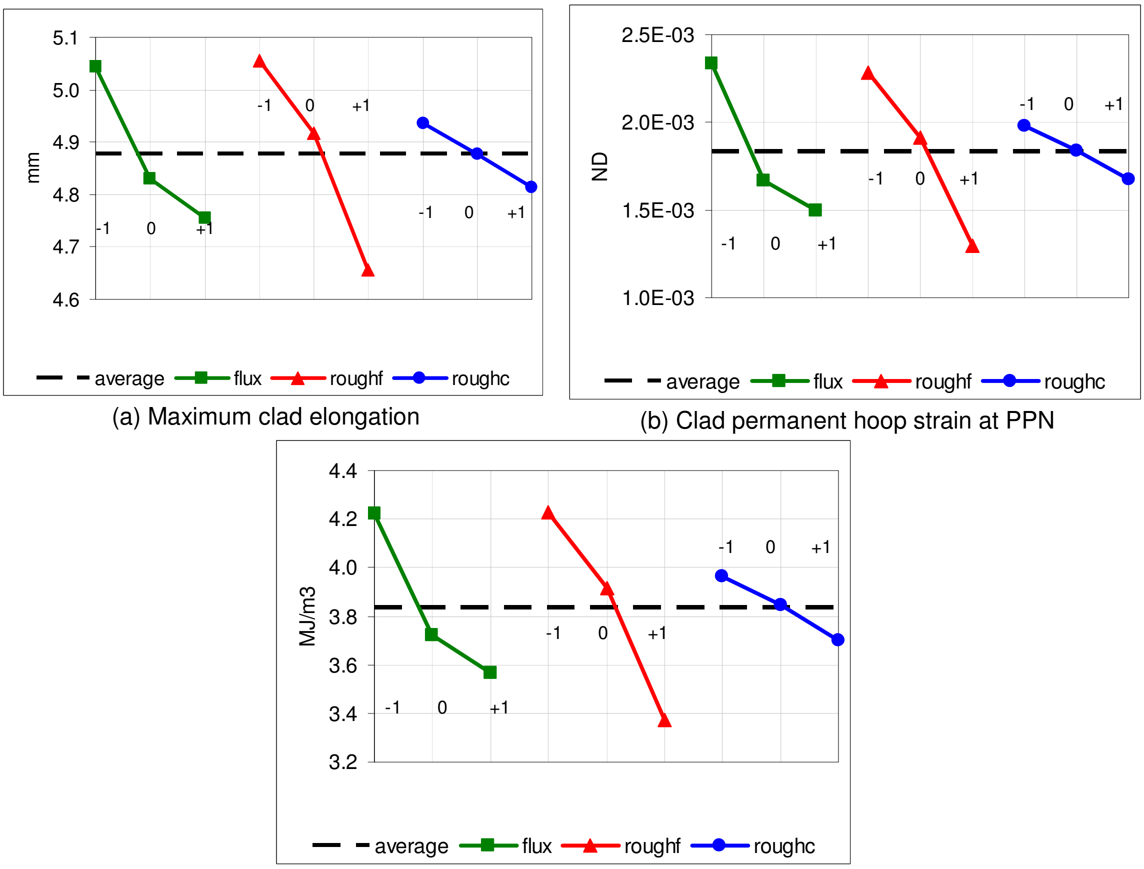
roughf and *roughc* are respectively the fuel pellet and clad tube surface roughnesses (arithmetic mean, peak-to-average). These variables are involved in the calculation of gap heat transfer through both gas and solid contact conductance terms, for which no validity range is reported. Besides, their sum imposes the pellet-to-clad minimal gap width in calculations. Their values are not expected to change along irradiation. In reality, creep effects and chemical reactions affect surface characteristics in a manner which can not be specified accurately.

3.4. Sensitivity analysis

A deeper SA has been conducted on the three parameters highlighted in the previous UA step.

3.4.1. Experimental design

An orthogonal array (OA) design has been chosen as the sampling method to generate the database which is used to build the response surface approximation. To ensure a high degree of resolution and explore interaction effects, a full factorial design made of all the combinations of 3 factors on 3 levels [-1, 0, 1] has been built, leading to a total of 27 “runs” required. Individual factor effects can be visualized in **Fig. 5**.



(a) Maximum clad elongation

(b) Clad permanent hoop strain at PPN

(c) Peak PNNL-SED value

***Fig. 5**. Main effects graphs*

The observation of **Fig. 5** calls the following comments:
- Increasing *flux*, *roughf* and *roughc* values leads to decrease the values of selected responses,
- Sensitivity to *roughc* variations is symmetric while it is not for *flux* and *roughf*,
- For a given input factor, the response to the changes (shape) is the same from one output variable to the other.

3.4.2. Regression analysis

Because FRAPCON-3 is a steady state code, it is reasonable to assume a continuous response on the range of variation of its input factors. As a consequence, simple second-order polynomial surface responses have been explored. Three equations have been derived by a least-squares fitting of each output values:

$$Y_i = a_0 + a_1 \cdot F + a_2 \cdot \rho_f + a_3 \cdot \rho_c + a_{12} \cdot F \cdot \rho_f + a_{23} \cdot \rho_f \cdot \rho_c + a_{13} \cdot F \cdot \rho_c + a_{11} \cdot F^2 + a_{22} \cdot \rho_f^2 + a_{33} \cdot \rho_c^2 \qquad (\textbf{Eq. 1})$$

with

Y_i = estimated output value ($\Delta l\big|_{max}, \varepsilon_\theta^p, SED\big|_{max}$)

a_0, a_i, a_{ij} = response equation coefficients (1 mean effect term, 3 main effects and 6 quadratic terms, all reported in **Table 4**)

F, ρ_f, ρ_c = input variables normalized to [-1,1] range

Table 4. *Regression coefficients of surface response equations*

Regression coefficients	$\Delta l\big|_{max}$	ε_θ^p	$SED\big|_{max}$
a_0	4.87	0.132	3.81
a_1	-0.145	-4.21E-02	-0.327
a_2	-0.200	-4.94E-02	-0.427
a_3	-6.01E-02	-1.50E-02	-0.134
a_{12}	-3.57E-02	-8.58E-03	-8.51E-02
a_{23}	-3.38E-02	-5.90E-03	-6.06E-02
a_{13}	-8.41E-03	-1.34E-03	-1.84E-02
a_{11}	6.86E-02	4.43E-02	0.171
a_{22}	-5.99E-02	7.70E-03	-0.116
a_{33}	-3.33E-03	1.86E-02	-1.31E-02

The accuracy of the regression equations was checked by examining residual plots. The ability of surface response equations to reproduce maximum clad elongation and peak PNNL-SED value at PPN looks suitable: high values of correlation coefficients obtained (0.9967 and 09956, respectively); and the biggest residuals are respectively, 0.7 and 3 %. A not so good but acceptable result is the equation corresponding to the permanent clad hoop strain (R^2=0.9234). A cubic-order polynomial would not increase accuracy notably.

3.4.3. Sensitivity indices

The relative influence of each input variable on the computed outputs is obtained by examining the coefficients (in absolute value) in the response equations (**Table 4**). The conclusions of this analysis are the following:
- Whatever is the response of interest, the order of sensitivity is: *roughf>flux>roughc.*
- *roughf* and *flux* have the same sensitivity, while *roughc* is three times less significant.
- Quadratic effect of *flux* has the same magnitude order than *roughc* linear one (interaction effect are known to be important in such studies).

As a main result of this exploratory UA, the most influential steady state parameter regarding transient modeling appears to be *roughf,* which value did not raise concerns up to now.

3.4.4. Responses uncertainty

Once constructed, the response surfaces can be used as a surrogate in a Monte Carlo simulation to estimate responses uncertainty. 2000 random sampling have been calculated without emitting any hypothesis on uncertainty distribution of the three selected inputs (uniform PDF in the bounds as specified in **Table 3**). Results are presented in **Table 5**.

Table 5. *Sensitivity analysis final results*

Responses	Maximum clad elongation (mm)	Permanent clad hoop strain at PPN (%)	Peak PNNL-SED value (MJ/m^3)
Standard deviation σ_d	0.15	0.040	0.33

95% confidence interval applied to BE response (x_0 +/-·$2\sigma_d$)	[4.68-5.30]	[0.129-0.289]	[3.43-4.75]
Precision on BE response (%)	12	77	32

According to this SA, considering the propagation of large uncertainties related to *roughf*, *roughc* and *flux* steady state variables, clad elongation and SED peak values along subsequent transient are calculated with a (95 % confidence) precision of respectively 12 and 32 %.
Assuming the response surface assigned to clad hoop strain response as valid, the "precision" on this nominal response reaches 77 %.

4. CONCLUSIONS AND PROSPECTS

This study is an exploratory work on uncertainty analysis application to RIA modeling. Despite its complexity, the interest of this work is outstanding: on one side, it can provide insights concerning how accurate one can become when analyzing RIA; on the other side, key variables affecting calculation precision can be identified. Both aspects should assist in defining research priorities.
In particular, this study investigates how much initial steady state uncertainties can affect accuracy in a RIA simulation. Based on the CIP0-1 test of the CABRI program, and by applying the "One-at-A-Time" and Response Surface Methods (OAT and RSM, respectively), the accuracy in the peak value of key transient variables (i.e. clad elongation, clad residual hoop strain and strain energy density) has been assessed.
From the analysis, it has been highlighted that fuel roughness, in-reactor heat-to-flux conversion factor, and clad roughness, are key characteristic values that should be specified as accurately as feasible. Among them, fuel surface roughness is the most important.
The study has allowed noting that just when uncertainties affecting steady state irradiation are considered, the accuracy in RIA modeling is limited, particularly of variables like clad strain. Fortunately, others like SED of extreme relevance characterizing fuel rod integrity do not seem too drastically affected. In other words, their accuracy is acceptable.
Given the intrinsic limitations of this study, it will be extended in the future. In the short term, it is planed to assess the impact of FRAPCON-3 models. In the long term both uncertainties should be combined in order to quantify steady state global inaccuracy.

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