Origins of the Uncertainty and Methods

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Deterministic safety analysis (frequently referred to as accident analysis) is an important tool for confirming the adequacy and efficiency of provisions within the defence in depth concept for the safety of nuclear power plants (NPPs). Requirements and guidance pertaining to the scope and content of accident analysis have, in the past, been described in various IAEA documents. To a certain extent, accident analysis is covered in several documents of the revised NUSS Series, mainly in the Safety Requirements on Design and in the Safety Guide on Safety Assessment and Verification for NPPs. More detailed guidance has been included in the IAEA Safety Report on Accident Analysis for NPPs. The Safety Report covers all the steps required for accident analyses, i.e. selection of initiating events and acceptance criteria, selection of computer codes and modelling assumptions, preparation of input data and presentation of the calculation results.

The current technical document complements the Safety Standards and the Safety Report listed above. There is a number of IAEA guidance documents devoted to qualified use of advanced computer codes for safety analysis. The objective of the present document is to provide practical guidance for the evaluation of uncertainty as a necessary component for BE safety analysis, and to contribute to broader use of this approach. Such guidance is considered as very important with the aim to avoid misinterpretation of calculation results and unjustified reduction in safety. The document is based on the up to date experience worldwide. Largest experience has been gained from use of thermal-hydraulic system codes, but major part of the document is applicable to any other aspects of reactor accidents.

In essence, following items are included: (a) the most common methodologies for determining the uncertainties for BE calculations completed to describe NPP behaviour during transients or accident or abnormal transients are described and compared, (b) examples of using the methodologies are given, (c) the techniques used to qualify the uncertainty calculations are summarized, and (d) the present trends in this research area are described. Main use of the BE of safety analysis and consequently of this document is expected in applications for design and licensing purposes, both for new reactor projects as well as for periodic safety reviews, safety upgrading, and lifetime extension of existing nuclear power plants.

1. Introduction

The purpose of the document is to address the use of a best estimate approach in licensing type accident analysis with evaluation of uncertainties. It should outline the current issues and future trends in BE accident analysis and share experience and guidance for performing accident analysis based on present good practice worldwide.

The current efforts to assure stable, safe and competitive operation of nuclear power plants go together with advances made in accident analysis domain where the deterministic safety analyses are an

important instrument for confirming the adequacy and efficiency of provisions for the safety of nuclear power plants.

Recently made advances offer two acceptable options for demonstrating that the safety is ensured with sufficient margin: use of best estimate (BE) computer codes either combined with conservative input data or with realistic input data but associated with evaluation of uncertainty of results. Before current BE computational tools were available only conservative licensing calculations with conservative codes were performed.

The IAEA Safety Standards and the Safety Report recommend as one of the options for demonstration of sufficient safety margins use of best estimate computer codes with realistic input data in combination with evaluation of uncertainties of the calculation results. Since this kind of analysis is very complex issue, sharing of experience and developing guidance for this area is very much needed.

All reactor types are to be considered while the main trust is directed towards the water-cooled reactors. In addition the advanced designs and new generation design should not be excluded. The trend in accident analysis continues to move in the direction of Best Estimate approaches rather than conservative analysis, however the conservative analyses are still used in many cases.

At this time the BE methodologies with uncertainty analysis is most widely used for safety analyses performed in the context of transients and Design Bases Accidents, where the associated uncertainties related to the understanding and modelling of physical phenomena are rather well defined and limited. Majority of BE applications in licensing are related the large break loss of coolant analyses. BDBA analyses are currently not part of the licensing processes, however BE analysis for this type of conditions are envisaged with the main concern related to the core melt prevention. Their application to the Beyond Design Bases Accidents is still rather limited due to large uncertainties associated with the calculated results and predictions.

The motivation to use best estimate systems codes and to calculate the uncertainty of the final results is persuasive. They can be used for licensing purposes in cases of plant systems modifications and improvements, power up-ratings, core optimizations, maintenance and repairs planning, core cycle extension, etc. The plant operators may therefore fully exploit a large number of techniques to maximize plant operational efficiencies, power output, and plant operational cycles. These capabilities in turn, enable a utility to operate a plant with the minimum cost and downtime.

The BE analyses provide a good view of existing margins or limits on nuclear power plant operation consequently offering a basis for possible parameter optimizations. Main use of the BE of safety analysis is expected in applications for design and licensing purposes, both for new reactor projects as well as for periodic safety reviews, safety upgradings, power up ratings and lifetime extension of existing nuclear power plants. The BE tools are already widely used by the regulatory organization, research institutes and in many cases by the industry, the utilities and the vendors.

There are already a number of IAEA guidance documents dealing with qualified use of advanced computational tools for safety analysis.

2. Definitions and Issues

The following definitions and issues are selected because of their strong relation to licensing aspects.

Design basis accidents (DBA)

Design basis accidents are more severe than normal maneuvering transients and anticipated operational occurrences. A design basis accident represents a challenge to one or more safety systems or functions. The safety systems should be designed to cope with the relevant design basis accidents without any radiological hazard in excess of acceptable limits, although need for repairs and re-qualifications of components, and some loss in lifetime or need for replacement of fuel may result.

Because of the large number of postulated initial events in combination with the variety of single failures which must be postulated in most deterministic accident analyses, it is convenient to identify those events which represent the bounding cases for various classes. These bounding cases are chosen to provide the most severe challenges to safety systems and functions. Initially, these bounding cases must be selected on the basis of screening analyses and engineering judgement. The selection must be confirmed periodically. For existing plants, the bounding cases are usually well known, but must be re-evaluated in case of significant design change that impacts the accident sequences.

The design basis analysis was introduced at the first stage of the design of Western plants. With this concept the analysis has to demonstrate that NPPs are able to face a list of identified abnormal or accidental conditions without any consequences or damages to the environment and the population. The design basis accident analysis does not intend to demonstrate that the plant is able to operate again after the transient, the plant may be completely be out of order but again without any consequences or damages to the environment and the population.

Beyond design basis accidents (BDBA)

Beyond design basis accidents mainly relate to operating conditions beyond the scope of conventional design. As example of beyond design basis accident, the case of simultaneous failure of redundant trains of safety-related systems should be studied with a deterministic approach. The choice of crucial sequences is generally oriented by probabilistic studies. Examples of beyond design basis accidents are:

- Anticipated transient without scram (ATWS) in several countries (in others ATWS is considered as DBA),
- Total loss of steam generator feedwater supply
- Total loss of power
- Other occurrences of multiple failures of safety systems which require accident management measures.

Deterministic accident analysis

A major part of the process of design and licensing of a nuclear power plant is the safety analysis. The IAEA requirements for safety analysis are stated in the IAEA Safety Standards Series No. NS-R-1 "Safety of Nuclear Power Plants: Design":

"*A safety analysis of the plant design shall be conducted in which methods of both deterministic and probabilistic analysis shall be applied. On the basis of this analysis, the design basis for items important to safety shall be established and confirmed. It shall also be demonstrated that the plant as designed is capable of meeting any prescribed limits for radioactive releases and acceptable limits for potential radiation doses for each category of plant states, and that defence in depth has been effected*."

The IAEA Safety Guide NS-G-1.2 "Safety Assessment and Verification for Nuclear Power Plants" says that the high level of safety should be demonstrated primarily in a deterministic way. "*The aim of the deterministic approach should be to address plant behaviour under specific predetermined operational states and accident conditions and to apply a specific set of rules in judging design adequacy*."

"*In general, the* ***deterministic analysis for design purposes should be conservative****. The analysis of* ***beyond design basis accidents is generally less conservative than that of design basis accidents***." Conservative is understood as leading to pessimistic results relative to specific acceptance criteria, trying to cover uncertainties. In addition, the Safety Guide says that in the past highly conservative computer code model parameters, initial and boundary conditions including safety system actuation set points and equipment availability has sometimes led to misleading sequences of events, unrealistic time-scales being predicted, and some physical phenomena being missed. Due to these shortcomings and the current maturity of best estimate codes, they should be used in a safety analysis in combination with a "*reasonably conservative selection of input data*" and a sufficient evaluation of the uncertainties of the results. The Regulatory Guide states: "*It may also be acceptable to use a combination of a best estimate computer code and realistic assumptions on initial and boundary conditions. Such an approach should be based on statistically combined uncertainties for plant conditions and code models to establish, with a specified high probability, that the calculated results do not exceed the acceptance criteria.*" The USNRC Regulatory Guide 1.157 specifies a common value of 95% for "high probability". 100%-values cannot be provided due to limitations in the number of calculations that can be performed varying parameter values. More clear guidelines about conservative assumptions and realistic assumptions with combined uncertainties should be fixed for deterministic design basis accidents. For beyond design basis accidents clear guidelines should be established with regard to applying realistic and conservative assumptions.

In many countries best estimate codes are used in safety analysis in the licensing process, i.e. codes which represent the physical phenomena as accurately as possible without deliberate bias. The pessimistic bias is introduced by using conservative initial and boundary conditions. The degree of conservatism in the result is not quantifiable using this approach alone. This is mainly due to uncertainties in the ability of the best estimate codes to predict reactor conditions. Uncertainties in plant conditions when an accident starts may be an additional contribution to be taken into account. Regulators reviewing licensing analysis usually want to know the margins between the results of accident analysis as presented, and any changes as

conditions vary. To quantify these margins, uncertainty analysis is used. Uncertainty analysis is even more important if best estimate codes should be used with best estimate initial and boundary conditions. Such a practice is not yet common for licensing analysis.

The approach of using best estimate codes in combination with conservative initial and boundary conditions is not allowed in the USA according to the Code of Federal Regulation (CFR). The 10 CFR 50.46 allows either a conservative calculation with code models satisfying the Appendix K of the CFR, or if using a best estimate code, it is required to identify and quantify the uncertainties.

Probabilistic accident analysis

Probabilistic safety analysis is defined in IAEA Safety Guide NS-G-1.2 "Safety Assessment and Verification for Nuclear Power Plants" as follows:

"*Probabilistic safety analysis provides a comprehensive, structured approach to identifying accident scenarios and deriving numerical estimates of risks.*"

According to the IAEA Safety Guide says that the deterministic analysis should be complemented by a probabilistic approach due to its ability to provide " *insights into plant performance, defence in depth and risk that are not available in the deterministic approach. The PSA should determine all significant contributors to risk from the plant and should evaluate the extent to which the design of the overall system configuration is well balanced, there are no risk outliers and the design meets basic probabilistic targets. The PSA should preferably use a* ***best estimate approach***."

Risk informed regulation

Introducing PSA results into requirements and assumptions in deterministic analysis is the main issue of risk informed regulation. Based on resulting frequencies of postulated accidents, consequences may be derived with regard to establish inspection intervals, and to move the boundary between design basis accidents and beyond design basis accidents. A proposed change of 10 CFR 50.46 refers to introduce a transition break size. Below that transition break size design basis requirements apply, and above that size beyond design basis requirements will be imposed with lower requirements, e.g. not assuming single failure, etc. This reduces the requirements that are in place up to now for large break accident analysis.

Sensitivity and uncertainty analysis

The IAEA Safety Guide NS-G-1.2 on assessment and verification recommends performing both sensitivity and uncertainty analysis. It is important to stress the difference.

Sensitivity analysis means evaluation of the effect of variation in input or modelling parameters on the overall code results to investigate the influence of one parameter, and to show that there is no abrupt change in the result of the analysis for a realistic variation of inputs ("cliff edge" effects). This can be performed subsequently for different parameters, but gives no indication of combinations of these parameters. Sensitivity studies should be used to identify the important parameters necessary for the analysis.

Uncertainty analysis is a statistical combination of code uncertainties, representation uncertainties and plant data uncertainties. These investigations should be used to confirm that the actual plant parameters will be bounded by the results of calculation plus uncertainty with a specified high confidence.

These two analyses may coincide only under special conditions, when only a weak interdependence between various uncertain input parameters exists. A combination of sensitivity studies, code to data comparisons and expert judgement are typically used to estimate uncertainties.

Bounding approach

A bounding approach intends to provide a bound of the uncertainties in a simplified way. "Reasonable" uncertainty ranges or bounding values are specified that encompass for example available data. The statements of the uncertainty of code results are deterministic, not statistic.

Accuracy

Accuracy is defined as the known deviation (or bias) between a code prediction and the actual transient performance of a real facility. Methods exist which extrapolate the accuracy of calculation results of experiments and their data to reactor conditions.

Best estimate code

A combination of the best estimate models necessary to provide a realistic estimate of the overall response of the plant during an accident. The term "best estimate code" means that the code is free from deliberate pessimism, and contains sufficiently detailed models and correlations to describe the relevant processes for the transients that the code is designed to model.

Conservative code

A combination of all the models necessary to provide a pessimistic bound to the processes relating to specified acceptance criteria.

Conservative input data

Conservative input data are initial, boundary conditions and plant parameters leading to pessimistic results, when used in a safety analysis code, to specified acceptance criteria.

Realistic input data

Realistic input data are initial, boundary conditions and plant parameters chosen to give a realistic (also "nominal", "as designed", "as operated") result.

Conservative assumptions for availability of systems

Assumptions about availability of equipment chosen to give a pessimistic result, when used in a safety analysis code, to specified acceptance criteria.

Realistic assumptions for availability of systems

Assumptions about availability of equipment are chosen to give a realistic (also "as designed", "as built", "as operated") result.

3. Conservative and Best Estimate Approaches

As stated in the IAEA Safety Requirements for Design of NPPs "a safety analysis of the plant design shall be conducted in which methods of both deterministic and probabilistic analysis shall be applied". The main objective of safety analysis is to demonstrate in a robust way that all safety requirements are met, i.e. that sufficient margins exist between real values of important parameters and their threshold values at which damage of the barriers against release of radioactivity would occur. It is required that "the computer programs, analytical methods and plant models used in the safety analysis shall be verified and validated, and adequate consideration shall be given to uncertainties".

Various options exist for combining computer codes types and input data for safety analysis as shown in Table 1. Recent advances in development of best estimate code and introduction of new uncertainty evaluation methods are gradually replacing the conventional conservative evaluation methods. Uncertainties are present in calculations due to the computer codes and due to input data for the code. Two different sets of input data define the calculated plant behaviour during the postulated accident, namely:

- Data related to the availability of NPP systems (normal operation systems, control systems, safety systems, i.e. single failure, loss of power supply)
- Data related to all other NPP initial and boundary conditions.

Table 1. Various options for combination of a computer code and input data

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

All conservative approaches, still widely used, were introduced to cover uncertainties due to limited capability for modeling and understanding of physical phenomena at the early stages of safety analysis. The results obtained by this approach are quite unrealistic and the level of conservatism is not fully known. As to using the BE codes two choices are on hand:

- The use of BE codes in combination with a reasonably conservative selection of input data and a sufficient evaluation of the uncertainties of the results.
- The use of BE codes with realistic initial and boundary conditions. If this approach is selected it should be based on statistically combined uncertainties for plant initial and boundary conditions, assumptions and code models, so that the calculated results do not exceed the acceptance criteria.

Both options are considered as acceptable and suggested by the existing IAEA Safety Standards. The option 2 is still more typically used at present for safety analysis in many countries. The international code validations as well as various evaluations of data uncertainties, and sensitivity studies helped to establish confidence in calculated results.

The current trends are going into direction of the option 3 indicating wider use of full BE analysis. The BE analysis with evaluation of uncertainties offers in addition a way to quantify the existing plant safety margins. Its broader use in the future is therefore envisaged, even though it is not always feasible because of the difficulty of quantifying code uncertainties with sufficiently narrow range for every phenomenon and for each accident sequence.

The option 4 is not yet a part of current licensing practices. This option is connected to future development in risk informed regulations.

Uncertainty quantification methods are available today, and several applications have been and will be performed in reactor safety research as well as in licensing. It is clear that the qualified uncertainty methodologies are mature and can be used with confidence to license any type of NPPs. Best-estimate applications of complex thermal-hydraulic system codes are recommended that are supported by uncertainty evaluation for the relevant output quantities. In this context the development of consistent procedures for the application of BE and uncertainty methods within the licensing process deserves a close assessment and attention. However, it is acknowledged that the foremost factor in the promotion and use of the various uncertainty methodologies is the acceptance and preference of the national licensing authorities.

4. Computational Tools and Applications

The purpose of the chapter is to establish the connection between ‘the use of a best estimate (BE) approach in licensing’ and the computational tools including their development features, the capabilities and limitations, and the application results.

The licensing needs that are of concern within the present framework are connected with the accident analysis in NPP. Postulated accident lists are available and systems and phenomena expected to occur have been characterized. Therefore, the key-issue when pursuing a BE approach in licensing is constituted by the availability of computational tools that are capable of modelling systems and phenomena that are expected to play a role.

While the entire spectrum of accident situations is of interest in relation to the development and the use of computational tools, the attention is focused hereafter to DBA and BDBA conditions that occur before the degradation of core, namely before the loss of geometric integrity.

Accident Management in BDBA situations aimed at preventing core melt are within the present scope, while situations like core-concrete interaction and steam explosion, though not fully excluded from the current scope, are given low priority.

4.1 Issues for Using Computational Tools in Licensing

A wide variety of computational tools have been produced that are of interest to the nuclear technology with special regard to the safety of nuclear power plants. In some situations, e.g. severe accident analysis, the use of computational tools is the only way to answer questions or to address problems.

The computational tools of interest to licensing constitute a sub-ensemble within the entire category. These can be distinguished between best estimate and conservative, depending upon the adopted modelling and assumptions in setting up the equations. Only the best estimate tools are considered here.

4.1.1 Classes and Areas for Computational Tools

Various best estimate computational tools exist. A comprehensive classification requires the consideration of a number of characteristics. Some of these are the structure of the software including the numerical scheme, the goals for the development and application, the geometric and/or the material regions of the concerned NPP. Such a comprehensive classification is beyond the purpose of this document. Nevertheless, the following classes of codes are distinguished:

- System thermal-hydraulic codes.
- Component codes.
- Phenomena or situation oriented codes.
- Computational Fluid Dynamics (CFD) codes.
- Coupled codes.

In relation to each class of codes, the area of application is relevant and is connected with separate steps of the licensing process. The areas of application include but are not limited to thermal-hydraulics, structural mechanics, fission products release and tracking, severe accident, neutron kinetics, control systems simulation, electrical system simulation, chemistry, radiation protection including the tracking of radioactive materials in the environment, fuel performance including burn-up, fluid structure interaction and vibration.

The class “System thermal-hydraulic codes” includes those computational tools that are capable of modelling, even separately, the primary system, the containment (or confinement) and the Balance of Plant (BoP) systems. The class “Component codes” includes computational tools that are specialized in the evaluation of steady-state or transient performance of components like pumps, separators, vessel, turbine. The class “Phenomena or situation oriented codes” includes computational tools that are specialized in the evaluation of individual phenomena like Critical Heat Flux (CHF) in steady state or transient conditions, thermal energy to the fuel following control rod ejection, dynamic loads on components associated with pressure wave propagation or break occurrence. The “CFD” codes are specialized computational tools based on a more detailed thermal-hydraulic modelling than the system thermal-hydraulic codes. The capability to predict, for instance, fluid temperature and velocity profiles in a section is typical for CFD codes. “Coupled codes” include those computational tools that are formed by the combination of codes belonging to two or more classes or areas as defined above. Relevant examples of coupled codes are constituted by 3D Neutron Kinetics / System thermal-hydraulics, Pressurized Thermal Shock (PTS) codes (combination of thermal-hydraulics, stress analysis and fracture mechanics).

All the identified areas and classes of computational tools have the potential to be applied with the aim of addressing issues and providing solutions for the best estimate licensing approach. However, the attention is focused hereafter to the codes that are used in accident analysis mostly covering the areas of thermal-hydraulics, neutron kinetics, fuel behaviour (mainly addressing the phenomena of ballooning and hydrogen production) and structural mechanics.

4.1.2 Items of Interest for the Application of Computational Tools to the Licensing

The use of best estimate computational tools in the (best estimate) licensing process is up to national regulatory bodies that must accept the codes and their applications modalities. Topics that are relevant within the present context that also constitute preconditions for the application of codes include the demonstration of:

- Verification and Validation (V & V) for the computational tool.
- Qualification for the user of the tool
- Qualification for the developed input deck (nodalisation qualification).

The V & V as well as the qualification for user and nodalisation are time consuming processes that have been the objective of different documents in the recent past. Definitive and commonly agreed criteria for defining the acceptability of those processes do not exist. Therefore, the processes identified by the dashed items above, as well as the acceptability thresholds for those processes constitutes elements of interest within the present context if finalized to the best estimate licensing approach.

5. Uncertainty Methods

The purpose of the chapter is to emphasize the role of uncertainty evaluation within a best estimate approach in licensing of NPP. Uncertainty in the results obtained from the application of best estimate

computational tools is a synonymous of ‘lack of precise knowledge’. This applies to both steady state and transient results. The term uncertainty shall be distinguished from terms like accuracy, sensitivity and bounding, as described in section 2.

The evaluation of uncertainty constitutes the necessary supplement of Best Estimate (BE) calculations performed to understand accident scenarios in water cooled nuclear reactors. The need for quantifying uncertainty in predictions comes from the unavoidable approximations embedded in the development and application processes of computational tools including inadequate knowledge of a number of input parameter values.

The first best estimate and uncertainty based methodology was the Code Scaling, Applicability and Uncertainty (CSAU) developed in USA in the early 90’s. Since then a number of other methodologies have been proposed including the GRS method, the UMAE method and the AEA technology method. These methods, although sharing a common goal with CSAU, use different techniques and procedures to obtain the uncertainties on key calculated quantities.

5.1 Key-Issues for Uncertainty Evaluation

A pioneering effort aimed at the quantification of uncertainty was made by the US NRC that in the early 90’s produced a comprehensive documentation under the acronym of CSAU. In parallel, development activities were conducted in a number of other countries, that ended up toward the middle of the same decade in suitable uncertainty methods. In this context, concepts like addressing the scaling issue, phenomena identification table, statistical treatment of code input and output uncertainties, accuracy quantification and extrapolation and deterministic treating of code uncertainties were established.

Key issues for evaluating the uncertainty in code calculation results can be synthesized by two main categories that include the concepts listed above are:

- Establishing the sources of uncertainty.
- Identifying the approaches to calculate the uncertainty.

A detailed discussion concerned with all these topics can be found in a document recently (TECDOC January 2005) issued by IAEA.

5.1.1 Sources of Uncertainty

Sources or origin of uncertainty shall be identified and characterized before implementing uncertainty methods. Three major classes for sources of uncertainty are mentioned in the above referenced IAEA TECDOC:

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

Uncertainties associated with scaling and user-effect are embedded into the three listed classes. A more detailed list of basic sources of uncertainties is available from the mentioned document.

5.1.2 Approaches to Evaluate the Uncertainty

An uncertainty analysis consists of identification and characterization of relevant input parameters (input uncertainty) as well as of the methodology to quantify the global influence of the combination of these uncertainties on selected output parameters (output uncertainty). These two main items are treated in different ways by the various methods.

One approach is to evaluate the ‘propagation of input uncertainties’: uncertainty is derived 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 either by deterministic or by probabilistic methods.

The other approach is the ‘extrapolation of output uncertainty’: uncertainty is derived from the (output) uncertainty based on the comparison between calculation results and significant experimental data.

5.2 Milestones in the Field of Uncertainty Evaluation

The following milestones can be considered in the area of development, qualification and application of uncertainty methods:

a) The issue of documents describing the CSAU (beginning of 90's)
b) The issue of journal papers describing approaches that can be characterized as 'application of the Wilks formula' or 'accuracy extrapolation' (middle of 90's).
c) The UMS study within the OECD/CSNI framework (end of 90's).
d) The introduction and the exploitation of the concept 'internal assessment of uncertainty' (end of 90's)
e) The overview provided by the IAEA TECDOC (beginning of 00's).

6. Acceptance Criteria

Acceptance criteria are limitations for acceptance of the results of safety analysis. They may consist of:

- Numerical limits on values of parameters to be predicted by accident analysis;
- Conditions for plant states during and after an accident (e.g. "keep coolable core geometry" and "assure long term cooling");
- Requirements on the need for actions by the operator, or the ability to credit for these actions.

Acceptance criteria are most commonly applied to licensing calculations, both conservative and best estimate.

Basic, high-level acceptance criteria are aimed at achieving an adequate level of defence in depth. Examples would be doses to the public or prevention of consequential pressure boundary failure in an accident. They are usually defined as limits by a regulatory body.

Specific acceptance criteria are used to set authorised limits to allow for adequate safety margins beyond these limits to allow for uncertainties and to provide defence in depth. They are sufficient but not necessarily meet the basic acceptance criteria. They may be developed by the designer and/ or utility and approved by the regulatory body. The regulatory body itself also may set them. An example is a limit on the cladding temperature in a LOCA in a PWR.

Different criteria apply for different vulnerability of barriers and different probability of occurrence. More strict criteria apply for events with higher probability of occurrence. For example, a no consequential containment damage criterion is appropriate for all DBAs. A no boiling crisis criterion is appropriate for anticipated operational occurrences, whereas a cladding temperature less than 1200 C criterion is used for LOCAs. A no cladding damage criterion is only appropriate for normal operation and anticipated occurrences. For DBAs, the fuel damage must be limited for each type of accident, to ensure coolable core geometry. Using a best estimate approach may have an effect on the subsequent calculation of fission product transport from the failed fuel rods to calculate the radiological consequence under best estimate plus uncertainty conditions.

Other examples of acceptance criteria from different areas are given in IAEA Safety Report Series No. 23: "Accident Analysis for Nuclear Power Plants". The Finnish regulatory Guidelines, for example, apart from other acceptance criteria, also define acceptance criteria for severe accidents.

7. Relevant Topics in Best Estimate Licensing Approach

The purpose of the chapter is to list significant topics within the licensing process that may need further emphasis within a best estimate approach. All the topics that are needed to complete a licensing process and require the application of a computational tool are of interest within the present framework.

7.1 Selected Relevant Topics

Various categories of relevant topics can be distinguished (e.g. related to physical phenomena, to modelling, to initial or boundary values, etc.). The considered, not exhaustive list of relevant topics includes the following items and the classification in chapters 7.2.1 to 7.2.4 reflects the classification of the table in chapter 2 of the present document (namely sections 7.2.1, to 7.2.3 correspond to the headings in columns four, two and three of the table, respectively):

1) Initial power.
2) Decay power.

3) Linear power (i.e. maximum Linear Heat Generation Rate, LHGR).
4) Gap model.
5) Hydrogen production and clad embrittlement.
6) Clad Burst and fission product release, transport and deposition.
7) Ballooning including coolability after ballooning.
8) Radiation heat transfer.
9) Modelling of fuel (i.e. pellet deformation and cracking) including the consideration of high burn-up and MOX fuel.
10) Consideration of Single Failure criterion.
11) Mechanical loads.
12) Long term cooling including the debris effect upon core cooling and cavitation of ECCS pumps.

7.1.1 Initial and Boundary Conditions

Topics 1) to 3) of the previous section fall in this category. The problem with the core power value is connected with the precision of its measurement at a given time. If the nominal power is characterized as 100% the precision of instruments is estimated of the order of 4 – 6% by various experts (this implies at a given moment the power can be as large as 106%). The problem that occurs here is: what is the value to be considered as code input for a best estimate analysis?

Various models and results from measurements are available for decay power. The interpretation of the related information generates some lack of agreements among licensing analysts.

The maximum value of LHGR has a noticeable effect in predicting an important safety parameter that is the peak-clad temperature. Engineering factors are added to get the value currently used in licensing. These factors (at least some of these) are not justified by the results of a three-dimensional and rod-by-rod neutron kinetic modelling. The problem that occurs here is: to what extent best estimate three-dimensional coupled neutron kinetics and thermal-hydraulic calculation can be used to fix the maximum LHGR?

7.1.2 Computer Code

Topics 4) to 9) of the section 7.1 fall in this category. Minor inadequacies including insufficient qualification characterize the modelling of phenomena identified under the items above. A detailed discussion about the reasons of these inadequacies, as well as about the current state of the art in the concerned sectors is beyond the scope of the present document. However, a few additional statements are provided hereafter.

In relation to gap model, the different burn-up along the fuel rod may cause different gap sizes along the rod length. This should be considered in best estimate modelling.

In relation to clad burst, a number of licensing authorities impose the consideration of 100% fuel rod failure (owing to ballooning) for calculating fission produce release, e.g. following the Large Break LOCA. This assumption, part of the licensing process, should be reconsidered according to the results of best estimate analyses and more robust models should be developed to predict clad burst.

In relation to fission product transport in primary system, zero retention in primary system is currently assumed and accepted (or imposed) by licensing authorities. Again, the assumption should be reconsidered according to the results of best estimate analyses and more robust models should be developed to predict fission product interactions in primary loop.

In relation to fuel modelling, peculiarities of MOX and high burn-up fuel should be considered in best estimate analysis distinguishing fuel elements one-by-one. In addition the fuel physical configuration changes during a transient; mostly following reactivity initiated accidents (e.g. pellet cracking, gap thickness variation). These variations might have a significant feedback upon the calculated transient evolution.

7.1.3 Availability of Systems

Topic 10) of the section 7.1 falls in this category. Several examples may be considered that deal with component and sub-systems failure. The risk informed regulatory concepts should be used to identify issues connected with this subject.

The selected topic deals with the single failure of the most ‘critical’ component or sub-system. The consideration of the single failure is mandatory when the conservative approach is used. The issue here is to

evaluate whether the single failure concepts must be extended to the best estimate approach (realistic input data and uncertainty are considered).

7.1.4 Other licensing requirements

Topics 11) and 12) of section 7.2 fall in this category. Among the criteria that are requested to be fulfilled by regulators there is the demonstration of no loss of geometric integrity for the core and the long term core cooling. It is current practice today, to address these criteria with 'other-than-best-estimate' techniques. The issue here is to calculate the related system behaviour by best estimate computational tools.

In the case of loss of core geometric entity, the solution of the problem involves, among the other things, the calculation of the pressure wave originated at the break (function of the break opening time) and the propagation of this wave into the vessel.

For the long term cooling, pump cavitations may occur owing to high liquid temperature in the containment combined with the presence of (even minor) debris. The debris may additionally deposit in the core region or at the core inlet, decreasing, in the long term, the cooling effectiveness of the core.

8. Conclusions

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

A review of existing uncertainty methods has been accomplished in the present document, making reference to the best estimate prediction of NPP accident scenarios. Sources of uncertainties, significant features of the uncertainty methods, as well as significant results from their application have been described.

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

Uncertainty quantification methods are available today, and several applications have been and will be performed in reactor safety research as well as in licensing. Experience of applications show that the difference between predicted upper bound or 95th percentile and 95% confidence level PCT to a calculation using nominal "best estimate" input values and default values for the computer code options and input data for models (reference calculation) is about 200 K for a typical large break loss of coolant accident, see Appendices 1 and 2. These relatively large values are due to the numerous models and correlations that are incorporated into a thermal-hydraulic code, and to the uncertainties associated with those individual models.

Two broad classes of uncertainty methods have been identified dealing with propagation of "input uncertainties" and of "output uncertainties" (note—propagation of "output uncertainties" is also characterized as "extrapolation of output errors"), respectively. In the former class, deterministic and probabilistic approaches have been distinguished.

The main characteristics of the methods based upon the propagation of input uncertainties derive from the need to reduce the number of input uncertain parameters, to assign subjective probability distributions and to propagate the uncertainty throughout codes that by their nature (see above) are approximations of the physical behavior. The main characteristics of the methods based upon the propagation of output uncertainties, derive from the need of having available relevant experimental data and from the process of error extrapolation that is not supported by theoretical formulations.

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

As a main conclusion from the present effort, it is clear that the qualified uncertainty methodologies are mature and can be used with confidence to license any type of NPPs. However, it is acknowledged that the foremost factor in the promotion and use of the various uncertainty methodologies is the acceptance and preference of the governing NPP licensing authority in the various countries. Also, it is acknowledged that the process to gain approval for using any of the uncertainty methodologies by the appropriate licensing bodies can consume considerable resources and require substantial time. Hence it is important that these factors are considered when proposing the use of a given uncertainty methodology.

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

Recommendations

Best-estimate applications of complex thermal-hydraulic system codes are recommended that are supported by uncertainty evaluation for the relevant output quantities.

The IAU is a desirable capability in the area that was already identified by the technical community in 1996: it allows the 'automatic' association of uncertainty bands to code calculations results, where uncertainty is a 'peculiarity' of an assigned code. The influence of code-user upon the predicted uncertainty values should be negligible when a robust method is available. The recommendation here is to explore this area considering the economic benefit of IAU applications.

The development of consistent procedures for the application of uncertainty methods within the licensing process is recommended.

Fundamentally, there do not exist any quantitative or qualitative standards for "qualifying" the uncertainty methodologies in use today. The uncertainty methodologies accepted for use (and thus considered qualified uncertainty methodologies) share the following characteristics:

- The results are reproducible.
- The results are traceable.

Users of the systems codes, in particular for licensing calculations, should be trained and certified.