Licensing and Regulatory Requirements for Best Estimate Plus Uncertainty Applications

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Abstract

Best-Estimate methods plus uncertainty (BEPU) analyses and evaluations are increasingly gaining importance in the licensing process in many of the areas such as fuel performance methodology, transient analyses and loss of coolant accident (LOCA) analyses. This paper will elaborate the details of the application of BEPU methods in various areas of design and operation of nuclear plants and how the BEPU licensing applications are reviewed by the U.S. Nuclear Regulatory Commission (NRC) staff. Fuel vendors are increasingly using best-estimate plus uncertainty methods in their fuel performance codes and methodology for analysis of thermal-mechanical behavior of fuel rods during normal and transient conditions. The methodology uses random sampling of various pertinent parameters, such as power, manufacturing, model and operational parameters in the BEPU method as wells as non-parametric statistics to evaluate uncertainties associated with each of the fuel design parameters in order to assure that the fuel is not damaged during normal and anticipated operational occurrences. BEPU method is used for meeting the requirements for realistic calculations of transients and emergency core cooling system (ECCS) performance during a LOCA and for estimating the uncertainty in the results. The paper will present the regulatory requirements for a best-estimate model that includes thermal-hydraulic best estimate (BE) codes, models and correlations, and computational models. Application of code scaling and uncertainty evaluation (CSAU) to quantify the uncertainty as well as the use phenomena identification and ranking table (PIRT) will be discussed.

1.0 INTRODUCTION

Requirements for licensing of U.S. commercial nuclear power plants (NPPs) in early 1950s were based mainly on deterministic safety analysis to ensure core integrity for all events, both abnormal and normal operation. The licensing regulations were formulated on the basis of both safety limits and analysis methods. For example, the safety criteria are peak cladding temperature (PCT), maximum hydrogen generation, coolable geometry, long term core cooling, maximum cladding oxidation for loss of coolant accidents; maximum fuel enthalpy content, stored energy in the fuel, and safety limits related to critical heat flux related to accidents and other events. Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that light-water nuclear reactors fueled with uranium oxide pellets within cylindrical zircaloy cladding be provided with emergency core cooling systems (ECCS) that are designed in such a way that their calculated core cooling performance after a postulated LOCA conforms to certain criteria specified in paragraph 50.46(b). Paragraph 50.46(b)(1) requires that the calculated maximum temperature of fuel element cladding not be greater than 2200° F. In addition, 10 CFR 50 46 (b)(2) through (b)(5) specify limits for calculated maximum cladding oxidation and maximum

hydrogen generation, core remains amenable to cooling and that the core maintains long-term cooling.

The original Section 50.46 of 10 CFR specified the acceptance criteria of ECCS performance. In 1988, the NRC staff amended the requirements of 10 CFR 40.46 and Appendix K "ECCS Evaluation Modes" in order that the regulations reflected an improved understanding of ECCS performance during transients and accidents that was obtained through extensive research. Paragraph 50.46(a)(1) permits licensees to use either Appendix K features or a realistic evaluation model. The 1980's regulations allowed the use of BE models and codes instead of conservative code models. These BE or realistic models must include justification to demonstrate that the analytical techniques employed realistically describe the behavior of the reactor system during postulated LOCA. 10 CFR 50.46 allowed the use of BE codes instead of conservative code models, stipulating, however, that uncertainties be identified and quantified.

One of the first major uncertainty methodologies presented was the CSAU evaluation (Reference 6). The CSAU methodology was developed at the NRC to address, in a unified and systematic manner, issues related to scaling capability of a best-estimate code, to its applicability to scenarios of interest to NPPs safety studies, and to the evaluation of uncertainties in calculating parameters of interest when the code is used to analysis of a specified scenario and NPP design. CSAU is a systematic procedure that leads to a quantified evaluation calculation uncertainty. The CSAU methodology is capable of analyzing complex phenomena because it subdivides the task into a series of manageable steps applied specifically to a single power plant (or generic group of plants) for a single accident scenario. Though the application of this methodology leads to a statement of uncertainty for a specific set of conditions, it can be separately applied to various combinations of plants and scenarios. The complexity in the CSAU procedure is further minimized by combining hierarchical top-down and detailed bottom-up approaches. In the top-down approach, the problem is analyzed from the perspective of the end state. As an example, the variable most affecting the PCT must be defined. The selection of the dominant contributors to uncertainty is based upon experimental evidence, analysis and code calculations, engineering judgment, and appropriate subjective decision-making techniques. The dominant contributors or the most important phenomena are identified and are listed as "highly ranked" phenomena, based on experimental data and code predictions of the scenario under investigation. In the resulting PIRT, the 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.

This paper is organized as follows. Section 2 describes summary of practices that are followed at the NRC for best-estimate licensing calculations. Section 3 describes the CSAU methodology used in the safety analysis of NPPs. Section 4 describes the uncertainty and sensitivity analysis used in the best-estimate safety analysis of NPPs.

2.0 SUMMARY OF PRACTICES IN BEST-ESTIMATE LICENSING CALCULATIONS

Currently there are two licensing approaches for evaluation of LOCA: conservative and best estimate. Conservative approach is referred to that involves conservative analysis using conservative code and conservative assumptions. BE approach includes BE calculations using uncertainty analysis in accordance with the guidance provided in RG 1.157. RG 1.157 describes the general requirements for using BEPU methodology for meeting Section 50.46 of CFR 50, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." In order to demonstrate the use of the RG 1.157, NRC sponsored a study to test the use of best estimate tools NUREG/CR-5249, "Quantifying Reactor Safety Margins,

Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident," which used TRAC-PF1/MOD1 code in demonstrating the LOCA results for peak cladding temperature, cladding oxidation rate, and quantity of hydrogen generated during a Large Break LOCA in a Westinghouse 3 or 4 loop pressurized water reactor (PWR). CSAU is a systematic procedure that leads to a quantified evaluation of code calculation uncertainty. The CSAU method is capable of analyzing complex phenomena by subdividing the task in to a series of manageable steps for a single or group of power plants for a single accident scenario.

The CSAU methodology comprises of several steps (described later) that can be grouped in to three key elements: such as, modeling requirements and code capabilities, assessment and ranging of parameters, and sensitivity and uncertainty analysis.

The NRC staff developed regulatory guidance to provide for the development of more generalized, BE methods for other accidents and transients. This RG (RG 1.203) describes the process that the NRC staff considers acceptable for use in developing and accessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant. Section 15.0.2 of the Standard Review Plan (NUREG-0800) covers the same material and is intended to be complementary and provides guidance to the reviewers providing practices and principles for the benefit of method developers. RG 1.203 defines the model known as evaluation model (EM) and its development of "figure of merit (FOM)" in evaluating the importance phenomena and processes. RG 1.203 has six basic principles identified as important to follow in the development and assessment of EM. They are:

- Determine requirements for the evaluation model to provide focus throughout the evaluation model development and assessment process (EMDAP).
- Develop an assessment base consistent with the determined requirements: Since an EM can only approximate physical behavior for postulated events, it is important to validate the calculational devices, individually and collectively, using an appropriate assessment base.
- Develop the evaluation model with the calculational devices needed to analyze the events in accordance with the requirements determined in the first principle should be selected or developed.
- Based on the application of the first principle, especially the phenomena importance determination, an assessment should be made regarding the inherent capability of the EM to achieve the desired results relative to the figures of merit derived from the General Design Criteria (10 CFR 50 Appendix A).
- Follow an appropriate quality assurance protocol during the EMDAP as required in Appendix B to 10 CFR Part 50.
- Provide a comprehensive, accurate, up-to-date documentation in order to enable a credible NRC review.

Appendix A of RG 1.203 provides additional considerations in the use of this RG for ECCS analysis. It provides regulatory structure in support of the BE plus uncertainty analysis methods. The best-estimate option in 10 CFR 50.46(a)(1)(i) requires that uncertainties in the analysis method and inputs must be identified and assessed so that "uncertainty in the calculated results can be estimated in order make sure that there is a high level of probability

that the ECCS performance criteria would not be exceeded." In support of the revised 1988 ECCS rule, NRC developed an uncertainty evaluation methodology, CSAU and the RG 1.203 is oriented toward the CSAU approach. CSAU uses the PIRT process in which relative importance is established by appropriate experts based on experience, experimental evidence, or computer based sensitivity studies.

Appendix B of RG 1.203 describes the EMDAP that guides the development of an evaluation model from the ground up. This appendix provides examples demonstrating a graded approach to EMDAP for evaluation model changes that are relatively small and have low safety significance. The first elements of the EMDAP consists of establishment of requirements for evaluation model capability which consists of specification of analysis purpose transient class, and power plant class, specification of FOM, identification of systems, components, phases, geometries, fields, and processes that must be modeled, and identification and ranking of key phenomena and processes (PIRT). The second element consists of development of an assessment base that involves specification of objectives for assessment base, performing scaling analysis and identification of similarity criteria, identification of existing data or perform integral effects test (IET) and separate effects test (SET) to complete the database, evaluation of the effects of IET distortion and SET scale up capability, and determination of experimental uncertainties. The third element is the development of the evaluation model which consists of initial planning, software design, coding, software testing and installation and acceptance. The fourth element is assessment evaluation model adequacy which consists of preparation of input and performance of calculations to assess model fidelity or accuracy as intended, determination of field equations capability to represent processes and phenomena and the ability of numeric solutions to the set of equations, assessment of scalability of integral calculations and data for distortions, determination of evaluation model biases and uncertainties and judgement for adequacy of the model.

3. CSAU METHODOLOGY

As indicated in Section 2, the CSAU methodology can be grouped in to three key elements: such as, modeling requirements and code capabilities, assessment and ranging of parameters, and sensitivity and uncertainty analysis.

The modeling requirements and code capabilities consists of various steps; (1) Determination of a code's applicability and uncertainty which is scenario dependent and identification of most important phenomena, (2) Selection of type of NPPs since individual components of NPPs such as fuel rod design, core loading, steam generator tubes, safety injection systems and control systems differ from plant-to-plant, (3) phenomena identification and ranking since plant behavior is not equally influenced by all processes and phenomena that occur during a transient, (4) maintenance of a frozen code version with changes allowed for corrections only, (5) documentation supporting code consistent with the frozen code version, and (6) determination of code applicability.

The code applicability must consider elements such as the capability of field equations to address global processes, code capability to model and scale particular processes, code capability to perform efficient and reliable calculations, and the code capability that address model plant geometry and perform efficient and accurate plant calculations.

Assessment and ranging parameters consist of total uncertainty contribution from code limitations, scaling effects embedded in experimental data and uncertainties associated with the state of the reactor at the initiation of a transient. This process includes establishment of

assessment matrix that should be designed to provide a data base for evaluating the code accuracy to calculate phenomena important to the scenario, the code capability to scale-up the phenomena to NPP conditions, and the influence of nodalization on the calculation. The plant nodalization must be done finely enough to represent both the important phenomena and design characteristic of the NPP but coarsely enough to remain economical. The differences for similar physical processes, at scales up to and including full scale, should also be quantified for bias and deviation to establish a statement of potential scale-up effects.

Sensitivity and uncertainty analysis objective is accomplished when the effect of important individual contributions to uncertainty in the primary safety criteria Figure of Merits are determined. These individual contributions are combined to give the desired uncertainty statement. The first step is in determination of uncertainties in NPP simulations that may result from uncertainties in the plant operating state at the initiation of the transient. For example, "the state of the fuel is a function of burnup history prior to accident initiation and of the original fuel manufacturing tolerances." The second step is sensitivity calculations that are used to determine the code's output sensitivity to various plant operating conditions that arise from uncertainties in the reactor state at the initiation of the transient. The third step is to determine combined bias and uncertainties that result from code limitations, scaling effects, NPP input variations which may contribute to a combined uncertainty to determine its probability density function through Monte Carlo sampling. The final step is to determine the total uncertainty for the code as an error band or statement of probability for the figures of merit. The effect of uncertainty contributors that cannot be quantified as bias and distribution because the data are limited, or because it is not economical, can be quantified as separate biases based on bounding sensitivity calculations with the NPP model.

The CSAU methodology is structured, traceable, and practical as is needed in the regulatory arena. The methodology is systematic and comprehensive as it addresses and integrates the scenario, experiments, code, and plant. The methodology combines a top-down approach to define the dominant phenomena, with a bottom-up approach to quantify uncertainty.

4. SENSITIVITY AND UNCERTAINTY ANALYSIS

One of the most important functional part of the BE methods for NPP safety analyses is the identification and characterization of uncertainties. The sources of uncertainties fall within five general categories:

- Code or model uncertainties associated with code models, material properties, correlations, model options, data libraries
- Representation uncertainties: The discretization of the systems associated with nodalization or mesh cells to obtain the control volumes that are represented by the field equations.
- Scaling uncertainty: scaled experiments and the reliance on scaling laws to apply the data results to full scale systems.
- Plant uncertainty: The uncertainty bands associated with the boundary and initial conditions for the nuclear power plant condition under operating conditions, such as, core power
- User effect: user errors occurred while creating and apply a system analysis code.

4.1 Code Uncertainty

A thermal-hydraulic system code is a computational tool that typically, includes three different sets of balance equations, closure or constitutive equations, material and state properties, special process or component models, and a numerical solution method. The three sets of balance equations deal with the fluids of the system, the solid structures including the fuel rods, and the neutronics. The closure (constitutive) equations deal with the interaction between the fluid and the environment as well as with the interaction of the two phases of the fluid (i.e., the gas and the liquid phase). The interfacial drag coefficient, wall to fluid friction factor and heat transfer coefficient are typically expressed by constitutive equations. The code uncertainty arises from balance equations where not all interactions between phases are included. Also code uncertainty can be from geometry averaging at a cross-section scale. Code uncertainty can also originate from extensive and unavoidable use of empirical correlations which are needed to close the balance equations. Code uncertainty can originate from imperfect knowledge of boundary conditions.

The code assessment process emphasizes differences between predicted and experimental data that cannot be directly or easily assigned to any of the above listed categories. In addition, improvement in the capability of the code to predict a particular experiment does not imply improvement of the capability to predict a different experiment.

4.2 Representation Uncertainty

Representation uncertainty is related to the process of setting up the nodalization which constitutes the connection between the code and the ‘physical reality’ that is the subject of the simulation. The process for setting up the nodalization aims at transferring the information from the real system (e.g. the nuclear power plant), including the related boundary and initial conditions, into a form understandable to the code. The result of the process can affect the response of the code.

4.3 Scaling Uncertainty

Scaling is a broad term used in nuclear reactor technology as well as in basic fluid dynamics and in thermos-hydraulics. Scaling involves the process of transferring information from a model to a scaled prototype. The model and prototype are typically characterized by different geometric dimensions, but parameters such as pressure, temperature and velocities may be proportionally different in the model and prototype.

4.4 Plant Parameter Uncertainty

Uncertainty or limited knowledge of boundary and initial conditions and related values for a particular nuclear power plant are referred to as plant uncertainty. Typical examples are the pressurizer level at the start of the transient, the thickness of the gap of the fuel rod, the conductivity of the UO2, and the gap between the pellets and the cladding. The quantities such as gap conductivity and thickness are relevant for the prediction of safety parameters (e.g., the PCT) and are affected by other parameters such as burnup, knowledge about which is not as detailed as required (e.g., knowledge about each layer of a fuel element that may be part of the nodalization). Thus a source of error of this kind in the class of ‘plant uncertainty’ cannot be avoided and should be accounted for by the uncertainty method.

4.5 User Effect

Complex system codes such as ATHLET, RELAP5 and TRAC have many degrees of freedom that lead to misapplication (e.g., not using the countercurrent flow limiting model at a junction where it is required) and errors by users by inputting an incorrect dimension of a system component. Two competent users will not approach the analysis of a problem in the same way and are therefore likely to take different paths to reach a solution. The cumulative effect of user community members to produce a range of answers for a well-defined problem with rigorously specified boundary and initial conditions is called the user effect. To reduce the user effect, several features are to be implemented:

- Misapplication of the system code should be eliminated
- Errors should be minimized
- The user community should preferably use the same computing platform.
- The system code should preferably be used by a relatively large user community
- The problem to be analyzed should be rigorously specified; that is, all geometrical dimensions are unambiguously defined, the initial conditions and boundary conditions are clearly specified, etc.

The application of realistic best-estimate computer codes to the safety analysis of NPPs involves the evaluation of uncertainties. The above mentioned sources of uncertainty affect the predictions and must be taken in to account. The major sources of uncertainty can be summarized in to three categories: code or model uncertainty, representation or simulation uncertainty, and plant uncertainty.

5.0 CONCLUSION

BEPU analyses and evaluations are increasingly gaining importance in the licensing process in many of the areas such as fuel performance methodology, transient analyses and LOCA analyses. The 1980's NRC effort allowed the use of BE models and codes instead of conservative code models. A pioneering effort in the area of thermal hydraulics was made by NRC with the publication of the CSAU method at the beginning of the 1990s. The improvement in physical knowledge of the associated phenomena coupled with the exponential increase of computer power has allowed license applicants to switch from basic conservative calculations to highly detailed best-estimate ones coupled with uncertainty.

6.0 REFERENCES

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