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Use of forward sensitivity analysis method to improve code scaling, applicability, and uncertainty (CSAU) methodology

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ABSTRACT

Since the code scaling, applicability, and uncertainty (CSAU) methodology was proposed about two decades ago, it has been widely used for new reactor designs and existing LWRs power uprates. In spite of these huge successes, CSAU has been criticized for the need of further improvement, focusing on two main issues – lack of objectiveness and high cost. With the effort to develop next generation safety analysis codes, new opportunities appear to take advantage of new numerical methods, better physical models, and modern uncertainty qualification methods. Forward sensitivity (FS) analysis directly solves the partial differential equations for parameter sensitivities. Moreover, our work shows that time and space steps can be treated as special sensitivity parameters so that numerical errors can be directly compared with physical uncertainties. It should be noted that FS analysis is an intrusive uncertainty quantification method that requires the user of the method to be familiar with the simulation code structure including numerical spatial and temporal integration techniques. When the FS analysis is implemented in a new advanced system analysis code, CSAU could be significantly improved by quantifying numerical errors and allowing a quantitative PIRT (Q-PIRT) to reduce subjective judgment and improve efficiency. This paper will review the issues related to the current CSAU implementations, introduce FS analysis, show a simple example to perform FS analysis, and discuss potential improvements on CSAU with FS analysis. Finally, the general research direction and requirements to use FS analysis in an advanced system analysis code will be discussed.

1. Introduction

The code scaling, applicability, and uncertainty (CSAU) methodology was developed in late 1980s by USNRC (Nuclear Regulatory Commission) to systematically quantify reactor simulation uncertainty. This method was developed in response to the USNRC rule change to allow the use of realistic physical models to analyze the loss of coolant accident (LOCA) in a light water reactor (Boyack et al., 1990). Prior to this time, the evaluation of this accident was subject to a prescriptive set of rules set by Appendix K of the regulations which require conservative models and assumptions to be applied simultaneously, leading to very pessimistic estimates of the impact of this accident on the reactor safety. The rule change therefore promised to provide significant benefits by allowing nuclear power reactor to increase output without major plant modifications. CSAU was developed to apply realistic methods, while properly taking into account uncertainty in data, physical modeling and plant variability.

The method was first demonstrated in 1996 for licensed application by Westinghouse to be structured, traceable, and practical (Young et al., 1998). Since then, best estimate plus uncertainty (BEPU) methods have been extensively used by the nuclear power industry around the world for power upratings, license renewals, and new design certifications. One example is AREVA's realistic large break LOCA (LB-LOCA) analysis methodology which received approval by USNRC in April 2003 (Martin and O'Dell, 2005). It incorporates the nonparametric statistical approach originally incorporated in the Gesellschaft fur Anlagen und Reaktorsicherheit (GRS) methodology for LOCA analysis. Another example is the Westinghouse automated statistical treatment of uncertainty method (ASTRUM) approved at the end of 2004 (Muftuoglu et al., 2004). The ASTRUM uses the same code and uncertainty distributions as the 1996 BELOCA method but uses nonparametric order statistics and more explicit treatment of more uncertainty parameters. For the same PWR plants, the calculated 95th percentile PCT (peak clad temperature) value is reduced by 126 K with the ASTRUM method, comparing with the value obtained from the 1996 BELOCA method. Almost all of demonstration applications of BEPU methods so far are for LOCA including both LB-LOCA and SB-LOCA (small break LOCA) and most of licensing applications of

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Nomenclature

BE	Backward Euler scheme
BEPU	Best estimate plus uncertainty
C-N	Crank–Nicolson method
CSAU	Code scaling, applicability, and uncertainty
FS	Forward sensitivity
IET	Integral effect test
JFNK	Jacobian-free Newton–Krylov method
LBLOCA	Large break LOCA
LOCA	Loss of coolant accident
LTE	Local truncation error
NPP	Nuclear power plant
PCT	Peak clad temperature
PDE	Partial differential equation
PDF	Probability distribution function
PIRT	Phenomena identification, ranking table
Q-PIRT	Quantitative PIRT
RS	Response surface
SBLOCA	Small break LOCA
TL	Statistical tolerance limits

BEPU methods are for LBLOCA (Prosek and Mavko, 2007). Although CSAU methodology has been traditionally employed with nuclear reactor safety analysis codes like TRACE, TRAC, and RELAP5, the authors believe that it can have a more general role that applies to any simulation code employed for nuclear reactor analysis.

1.1. Major steps of CSAU

In developing CSAU, the emphasis was placed on providing a practical engineering approach that could be used to quantify code uncertainties (Boyack et al., 1990). Consequently, for a specified plant and a given scenario, the CSAU method focuses only on important processes and/or phenomena, assesses the code capability to scale them up, and evaluates the accuracy with which the code calculates them. The CSAU evaluation methodology consists of three primary elements.

The first element is requirements and code capabilities. In this element, scenario modeling requirements are identified and compared against code capabilities to determine the code's applicability to the particular scenario in a given plant design. In addition, an effort is made to identify potential code limitations. The modeling requirements are established by identifying and ranking processes and phenomena important to the particular scenarios (phenomena identification, ranking table, PIRT). The PIRT process provides a cost-effective approach to rank process and phenomena by evaluating their importance and modeling uncertainty relative to the primary safety criteria so that only the significant contributors need to be evaluated (Wilson and Boyack, 1998). Code deficiencies and/or limitations are also identified and evaluated as to their potential effects on uncertainties to calculate primary safety criteria. The second element is assessment and ranging of parameters. In this element there are activities to assess the capability of the code to calculate processes important to the scenario by comparing calculations against experimental data to determine code accuracy, to determine scale-up capability, and to specify ranges of parameter variations needed for sensitivity studies. In addition, bounding analyses can be performed and, in such cases, code calculations may not be required. The third element is sensitivity and uncertainty analyses. The total uncertainty in a safety analysis includes contributors that arise from code limitations, scaling inaccuracies embedded in the experimental data (and therefore the code), and uncertainties associated with the state of the reactors at the initiation of a transient. This element contains the activities to calculate, collect, and combine individual contributors to uncertainty into the required total mean and 95% probability statements including separately identified and quantified biases.

Within the third element, different techniques for the uncertainty propagation can be used, including Monte Carlo analysis (MC), response surface (RS) methods, and statistical tolerance limits (TL). (Prosek and Mavko, 2007) Because of demanding calculation requirements, the MC method is currently not applicable to complex thermal-hydraulic codes. In the RS methods the RS replaces the code calculation in the MC analysis. The TL is approached using a random sampling of input parameters N times, and then using the computer code directly to generate N outputs that are used to estimate the actual uncertainty. Both RS and TL methods have been widely used to obtain the 95/95 (95% probability with 95% confidence level) PCTs and other design limit variables. The first CSAU demonstration by USNRC used the RS method while TL methods have gained popularity recently. The consideration of nonparametric tolerance limits was originally presented by Wilks (1941). Guba et al. (2003), Nutt and Wallis (2004), and others studied and extended Wilks' method's applications in BEPU. Depending on conservativeness of selecting tolerance limit, the number of total random sample runs increases, with the most conservative method starts at 59. If more random runs are affordable, less conservative results can be obtained. For example, if a 93 run case is used, the second highest value is used to establish the 95/95 PCT instead of the highest value in the 59 run case.

1.2. Limitations of current CSAU practices

Since CSAU methodology was first proposed about two decades ago, it has been widely used. In spite of these huge successes, several aspects of CSAU have been criticized for the need of further improvement (Wilson et al., 1992). The critiques focus on two main issues – lack of objectiveness and high cost:

- Subjective judgement in PIRT process – PIRT process heavily relies on expert opinions.
- High cost – due to heavily relying on large experimental database, needing many experts man-years work, and very high computational overhead. The first CSAU demonstration project spent about 13 man-years equivalent resource (Boyack et al., 1990). AREVA spent 40 staff-years to obtain NRC license for its BEPU large break LOCA analysis method (Martin and O'Dell, 2005).
- Mixing numerical errors with other uncertainties – due to limitations of two-phase models and numerics in the legacy system analysis codes currently used, it is impossible to separate numerical uncertainty from other uncertainties.
- Grid dependence and same numerical grids for both scaled experiments and real plants applications – since numerical errors cannot be quantified, one of ways to indirectly quantify them is to use same numerical grids for both scaled experiments and real plants applications. However, this is impossible or have large scaling distortion due to different scaling methods.
- User effects – different users can obtain different answers with same codes due to different discretization methods and different values for many users defined parameters.
- Limits on uncertainty propagation – the current uncertainty quantification in CSAU is based on a "black box" approach with frozen codes. The simulation tool is treated as an unknown signal generator, a distribution of inputs is sent in and the distribution of the output is measured and correlated back to the original input distribution. Not all the important uncertainty in the codes can be accessed by users, i.e., heat transfer correlations in RELAP5. If the experts know the potential uncertainty but cannot explicitly perturb it through the code input, a bias is used and superposed