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Uncertainty analysis of a large break loss of coolant accident in a pressurized water reactor using non-parametric methods

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ABSTRACT

The safety analysis of nuclear power plant is moving toward a realistic approach in which the simulations performed using best estimate computer codes must be accompanied by an uncertainty analysis, known as the Best Estimate Plus Uncertainties approach. The most popular statistical method used in these analyses is the Wilks' method, which is based on the principle of order statistics for determining a certain coverage of the Figures-of-Merit with an appropriate degree of confidence. However, there exist other statistical techniques that could provide similar or even better results. This paper explores the performance of alternative non-parametric methods as compared to the Wilks' method of obtaining such Figure-of-Merits tolerance intervals. Three methods are investigated, i.e. Hutson and Beran–Hall methods and a bootstrap method. All the techniques have been used to perform the uncertainty analysis of a Large-Break Loss of Coolant Accident. The Figure-of-Merit of interest in this application is the maximum value reached by the Peaking Clad Temperature. In order to analyze the results obtained by the different methods, four performance metrics are proposed to measure the coverage, dispersion, conservativeness, and robustness of the tolerance intervals.

1. Introduction

The International Atomic Energy Agency's guidance (IAEA) on the use of deterministic safety analysis (DSA) for the design and licensing of nuclear power plants (NPPs) "Deterministic Safety Analysis for Nuclear Power Plants Specific Safety Guide, Specific Safety Guide No. SSG-2′ [1] (hereinafter referred to as SSG-2) addresses four options for DSA applications. Due to the importance of taking the current understanding of physical phenomena into account, and thanks to the availability of reliable tools for more realistic safety analyses without compromising plant safety, many countries have chosen Option 3.

Option 3 involves the use of best-estimate codes and data together with an evaluation of the uncertainties, the so-called Best Estimated Plus Uncertainty (BEPU) methodologies. Table 1 shows the different options addressed in the SSG-2 guide.

The IAEA Safety Report Series N°.23 "Accident Analysis for Nuclear Power Plants" [2] recommends a sensitivity and uncertainty analysis if Best Estimate (BE) codes are used in the licensing analysis. A comprehensive overview of uncertainty methods can be found in the IAEA Safety Report Series N°.52 "Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation", issued in 2008 [3]. References [4,5,6,7] deal with the evolution of BEPU analysis and describe some of the most frequently used techniques. Some of these techniques have been developed by International programs which have discussed the BEPU approaches in order to address the issue of the capabilities of best-estimate computational tools and uncertainty analysis.

This is the case, for example, of the BEMUSE, promoted by the Working Group on Accident Management and Analysis (GAMA) of the OECD. These discussions have led to the development of BEPU approaches insofar as they have been accepted for performing deterministic safety analysis by the regulatory authorities. The scope of BEMUSE Phase V, in which fourteen participants from twelve organizations and ten countries participated, is the uncertainty analysis of a Large Break Loss-Of-Coolant-Accident (LBLOCA) in a Pressurized Water Reactor. The results and the main lessons learned from this BEMUSE program are presented in reference [8].

In a BEPU design-basis accident it is normally assumed that the uncertainty in the safety outputs (i.e., the figures of merit (FOMs) involved in the acceptance criteria of the analysis) derives from the uncertainties in the input parameters (initial and boundary conditions) and those arising from the computational model [4].

These FOMs are usually extreme values (minima, maxima) of safety variables during the transient, such as Peak Clad Temperature (PCT), Critical Heat Flux (CHF), etc. Current BEPU methodologies mainly rely on a probabilistic description of the uncertainty and on the use of sta-

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Notation

α	acceleration parameter in BCα method
$B(n,p)$	binomial distribution of parameters n and p
C_j	coverage of the sample j
CC	conservativeness
$F(\bullet)$	cumulative distribution function (cdf)
$\hat{F}^{-1}$	inverse of the empirical cdf
$f(\bullet)$	probability density function (pdf)
$I[c]$	indicator function. This function is equal to 1 if c is true and 0 if c is false
$I_x(a,b)$	incomplete beta function of parameters x, a and b
Z	reference distribution sample size
n	sample size for estimating TL
N	number of samples (repetitions)
p	coverage of the tolerance interval
$U_{n'p:n}$	Dirichlet process
$X_{i:n}$	ith order statistics from a sample of size n of independent and identically distributed random variables
z_0	bias correction parameter in BC$_\alpha[\gamma]$ method
z_γ	z score from the standard normal distribution
γ	confidence level of the tolerance interval
ξ_p	p percentile
$\Phi(\bullet)$	standard normal cdf

Acronyms

BC$_\alpha[\gamma]$	bias corrected accelerated bootstrap method
BE	best estimate
BEPU	best estimate plus uncertainty
CD	coverage standard deviation
CHF	critical heat flux
CM	coverage mean
CV	coverage coefficient of variation
DSA	deterministic safety analysis
ECCS	emergency core cooling systems
FOM	figure of merit
FOS	first order statistic
GAMA	working group on accident management and analysis
IAEA	international atomic energy agency
LBLOCA	large-break loss of coolant accident
LPIS	low-pressure injection system
NPP	nuclear power plant
OS	order statistic
PCT	peaking clad temperature
PIRT	process identification and ranking tables
PWR	pressurized water reactor
SV$_{ref}$	reference safety value
TH	thermal hydraulic
TL	tolerance limit
UA	uncertainty analysis
TRACE	TRAC-RELAP advanced computational engine
SNAP	symbolic nuclear analysis package

tistical techniques to estimate it [9,10,11,12,13]. In this framework, the uncertainty of a FOM can be identified with its probability distribution.

Most BEPU approaches accepted by the regulatory authorities are based on the propagation of input uncertainties and make use of methods based on Wilks' formula, which is based on the principle of order statistics for determining a certain coverage of the figures-of-merit (FOM) with a certain degree of confidence. The German Technical Safety Organisation (TSO) Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) was the first to introduce Wilks' tolerance limits in uncertainty analyses with TH codes, so that this type of analysis is renowned as the GRS method [9]. This method determines the number of code runs needed to obtain a sample of outputs, i.e. FOMs, which are required to verify compliance with acceptance criteria. In accordance with current regulatory practice a 95% coverage with a 95% confidence level is required. So, if a one-side FOM tolerance interval is applied based on the use of the First Order Statistics (FOS) with a 95/95 coverage/confidence level a sample size of $n = 59$ runs is required.

This paper focuses on the deterministic safety analysis of a Large-Break Loss of Coolant Accident (LBLOCA) scenario in a PWR NPP based on a BEPU approach and the use of order statistics according to the current practice for the formulation, propagation, and analysis of uncertainties. In addition, the paper introduces alternative non-parametric methods to the traditional first order statistics based on Wilks' formulae. The results of the alternative methods are compared with those of the traditional method based on appropriate performance metrics also proposed in this paper. The study specifically focuses on the analysis of the uncertainty associated with the maximum of the PCT (Peak Cladding Temperature) as the FOM.

2. Overview of the BEPU approach

Fig. 1 outlines a typical procedure used in BEPU approaches [14], which consist of the following twelve steps:

1. Selection of the accident scenario. Reactor system and transient selection.
2. Selection of the safety criteria linked to the accident scenario under study and the FOM involved in the acceptance criteria.
3. Identification and ranking of relevant physical phenomena based on the safety criteria.
4. Selection of the appropriate uncertain TH (Thermal Hydraulic) parameters to represent those phenomena.
5. Identification of relevant safety-related functions and systems involved in the accident scenario.
6. Identify relevant trains and components of the safety-related functions and systems developing their possible redundancies.
7. Development of the TH computer model of the accident scenario, e.g. developing an input for the TRACE integrated into the SNAP platform [15,16,17].
8. Allocation of PDF (Probability Density Functions) for each selected uncertain TH parameter.
9. Establishing conservative assumptions on the availability of trains/components of safety systems.
10. Random sampling of the selected uncertain TH parameters according to PDF. Sample size (n) will depend on the particular statistical

Table 1
SSG-2 DSA options.

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 input data plus uncertainty; partly most unfavorable conditions[a]
4. Risk-informed	Best estimate	Derived from probabilistic safety analysis	Best 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.