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The Role of Uncertainty Quantification for Reactor Physics

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The quantification of uncertainties is a crucial step in design. The comparison of a-priori uncertainties with the target accuracies, allows to define needs and priorities for uncertainty reduction. In view of their impact, the uncertainty analysis requires a reliability assessment of the uncertainty data used. The choice of the appropriate approach and the consistency of different approaches are discussed.

I. INTRODUCTION

The role of uncertainty quantification has been stressed and has been the object of several assessments in the past (see e.g. Refs. [1] and [2] among many others), in particular in relation to design requirements for safety assessments, design margins definition and optimization, both for the reactor core and for the associated fuel cycles. The use of integral experiments has been advocated since many years, and recently re-assessed [3] in order to reduce uncertainties and to define new reduced "a-posteriori" uncertainties. While uncertainty quantification in the case of existing power plants benefits from a large data base of operating reactor experimental results, innovative reactor systems (reactor and associated fuel cycles) should rely on limited power reactor experiment data bases and on a number of past integral experiments that should be shown to be representative enough. Moreover, in some cases, in particular related to innovative fuel cycle performance and feasibility assessment, nuclear data uncertainties are the only available information. Uncertainty quantification in that case becomes a tool for detecting potential show stoppers associated to specific fuel cycle strategies, besides the challenges related to fuel properties, fuel processing chemistry and material performance issues.

II. THE DESIGNER DILEMMA

The quantification of uncertainties is a crucial step in different phases of a nuclear system design. In a preliminary (conceptual) design phase, the comparison of calculation scheme (nuclear data and modeling) a-priori uncertainties with the target accuracies for the most important design parameters, allows to define needs and priorities for calculation scheme improvement and uncertainty reduction. The designer analysis establishes the quantified penalties due to uncertainties beyond the target accuracy range and their impact on the design (*e.g.* extra margins on fuel performances, choice of alternative or back-up solutions *etc.*). Successively, and in parallel with preliminary design, the choice of the most adapted approach to uncertainty reduction could be done according to time-frame, project schedule etc., but also according to safety requirements (*e.g.* demonstration of validated uncertainties). In view of their impact, the uncertainty analysis requires a reliability assessment of the uncertainty data that have been used. The choice of the appropriate approach can be a dilemma for the designer. In practically all case, the uncertainty quantification for design will imply the use of experiments (past experiments or ad-hoc experiments still to be performed):

- Performance of a full series of design oriented experiments (critical mass, reaction rate distributions, reactivity coefficients, control rod worth etc.) in a representative reactor mock-up. This is the most ambitious (in terms of resources deployment), but not necessarily the most effective or even feasible approach (facility availability, cost, difficulty to achieve representativity etc.). If available, the uncertainty reduction by integral parameter R is a function of the a-priori covariance data D [4]

$$\Delta R_0'^2 = \Delta R_0^2 \cdot \left(1 - r_{RE}^2\right), \tag{1}$$

$$r_{RE} = \frac{\left(S_R^+ D S_E\right)}{\left[\left(S_R^+ D S_R\right)\left(S_E^+ D S_E\right)\right]}, \tag{2}$$

where the S_E and the S_R are the sensitivity vectors

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of the experiments and of the design parameters, respectively.

- A more flexible approach is to use a large set of "representative" integral experiments and to perform a global assimilation or adjustment that allows to obtain an "adjusted" nuclear data set and an "a-posteriori" covariance matrix D′. This new covariance matrix can be used to assess the new (reduced) uncertainty for each integral design parameter R of interest [4]

$$\Delta R_0'^2 = S_R^+ D' S_R. \quad (3)$$

Both approaches 1 and 2 rely a) on the reliability and completeness of the covariance data; b) on the reliability of the integral experimental uncertainties; c) on the capability to detect possible systematic errors in the experiments and their overall consistency and d) on the drastic reduction of modeling errors.

- A third approach can be (and has been) envisaged that relies on the existence or performance of selected integral experiments that provide information on "elemental" phenomena or on separated individual physics effects. This approach can provide practical "uncertainties", derived from the calculation/experiment analysis, the observed C/E dispersion and consistent with the experimental uncertainties, on each "elemental" phenomena and/or on separated physics effects. It could alternatively also provide "bias factors" extracted from the residual C/E′s that can be combined appropriately.

This last method (that relies on criteria b), c) and d) indicated for methods 1 and 2) is of particular interest when applied to design parameters that result from the compensation of several separated (or elemental) effects, potentially of different sign and potentially of comparable magnitude. This is the case of most reactivity coefficients and of the core reactivity evolution with time. An interesting feature of this approach is represented by the possibility to compare the overall uncertainties obtained both using the a-priori covariance data (method 2) and the "uncertainty" derived from the analysis of a selected set of integral experiments (method 3). Consistency of the two approaches will be a strong argument in support of the robustness of the a-priori covariance data. In the following paragraph we will discuss two typical cases: the sodium void reactivity coefficient in an innovative FR and the general case of the reactivity loss/cycle. In both cases it is possible to express the integral parameter as a sum of physical components that can in principle be measured separately. In the examples we will indicate how old experiments can be used or how some simple experiments can be designed to meet the needs of this approach.

III. INTEGRAL PARAMETERS AND THEIR SEPARATED COMPONENTS

A. The Na Void Coefficient in an Axial Heterogeneous Innovative Fast Reactor Core

Low void reactivity coefficient is an innovative feature of the French ASTRID design [5]. Studies performed on a representative configuration [6] confirm a global, full core and upper structure void reactivity coefficient that is fairly small and even negative. However, the sodium void reactivity is the result of compensation of large components with different sign, according to the formulation, in diffusion theory, given below

$$\begin{aligned}\frac{\Delta k}{k} = \frac{1}{F}\{N_{Na}\sum_j \sigma^c_{Na,j}\Phi_j\Phi_j^+ \\ -N_{Na}\sum_{j,k}\sigma_{Na}\left(j\to k\right)\Phi_j\left(\Phi_k^+ - \Phi_j^+\right) \\ -\delta D_j\int \Delta\Phi_j\Delta\Phi_j^+ dV - \sum_i N_i\sum_j \delta\sigma^a_{i,j}\Phi_j\Phi_j^+\} \\ = A_{\mathrm{Na}} - S_{\mathrm{Na}} - L - A_{\mathrm{selfSh}}. \quad (4)\end{aligned}$$

For instance in the case of the reactivity change associated to voiding the sodium in the fissile + internal fertile + upper plenum and calculated in R-Z S_4P_1 the total value of -1024 pcm is the result of the difference between -3587 pcm (leakage component, L), and +2563 pcm (non-leakage component, of which +2222 pcm of spectral component, $A_{\mathrm{Na}} - S_{\mathrm{Na}}$ and +341 pcm of self-shielding variation, A_{SelfSh}). In effect, in order to be more conservative, one should calculate the uncertainties by component and combine them with some degree of correlations, with the completely uncorrelated hypothesis being the most conservative. In order to evaluate the uncertainty per component the previously mentioned case of the fissile + internal fertile + upper plenum region sodium void was considered. This is the situation that has been considered the most relevant to further safety calculations. Table I shows the uncertainty results (in pcm) for the total (all components) sodium void reactivity, obtained using the COMMARA-2 covariance data [7]. The breakdown by isotope and reaction is provided. From the total value, considering a 95% confidence interval (2σ), an uncertainty of 1$ [8] should be associated to this reactivity coefficient for the successive safety calculations. Tables II and III show the uncertainties for, respectively, the leakage and non-leakage components. In terms of contributions one can notice that for the non-leakage component the uncertainty is largely dominated by the anisotropic elastic cross sections of sodium. Therefore, one has to be very careful with the related covariance evaluation. The non-leakage uncertainty is also larger, by itself, than that of the total effect. In combining the non-leakage and leakage component uncertainties the safest assumption is a conservative assumption of simple sum of the uncertainties. With that kind of assumption, and, again