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Nuclear Data Uncertainty Propagation in Depletion Calculations Using Cross Section Uncertainties in One-group or Multi-group

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Several approaches have been developed in last decades to tackle nuclear data uncertainty propagation problems of burn-up calculations. One approach proposed was the Hybrid Method, where uncertainties in nuclear data are propagated only on the depletion part of a burn-up problem. Because only depletion is addressed, only one-group cross sections are necessary, and hence, their collapsed one-group uncertainties. This approach has been applied successfully in several advanced reactor systems like EFIT (ADS-like reactor) or ESFR (Sodium fast reactor) to assess uncertainties on the isotopic composition. However, a comparison with using multi-group energy structures was not carried out, and has to be performed in order to analyse the limitations of using one-group uncertainties.

I. INTRODUCTION

Nuclear data uncertainties are currently on the spotlight, due to the assessment of their impact on criticality safety analysis, and also in burn-up/depletion/activation analysis. Through Uncertainty Quantification (UQ) methodologies, the uncertainties on nuclear data are propagated to final response functions of interest, such as the effective neutron multiplication factor (k_{eff}) or concentrations of burn-up trackers [1].

Monte Carlo sampling (forward approach) is the simpler method to be adopted because it is easy to implement without any major code modification. Moreover, in principle, it could also take into account non-linear effects. Examples of implementations are Total Monte Carlo (TMC) [2], XSUSA [3] or NUDUNA [4]. However, its main drawback is that it requires a very large amount of direct calculations, while a backward approach, e.g. the TSUNAMI sequence of SCALE-6.1 [5], needs only a number of importance calculations equal to the number of response functions addressed.

A Monte Carlo sampling approach has been developed by UPM (Universidad Politécnica de Madrid), referred as "Hybrid Method (HM)" [6]. It only performs propagation of nuclear data uncertainties on the depletion part of a coupled transport-depletion (burn-up) problem, and on decay problems. For solving depletion equations, it makes use of the ACAB depletion/activation code [7]. It has been improved to tackle burn-up problems with neutron spectrum variations [8], when using collapsed one-group cross section uncertainties. But checks with a multi-group cross section approach were not performed in order to validate them.

Therefore, the aim of this paper is to recall the improvement done for HM when using collapsed one-group cross section uncertainties, and to address its limitations by comparing with the results of using multi-group cross section uncertainties. With two cases, the comparison is carried out: for the ESFR (European Sodium Fast Reactor) fuel cycle [8], and hypothetical case of large neutron spectrum variations. Two sources of uncertainties are used here: EAF-2010 [9] and SCALE-6.1 [5].

II. THE HYBRID METHOD

A. Description of the Hybrid Method

The HM [6] is aimed to propagate nuclear data uncertainties on depletion/activation/transmutation calculations, decoupling the depletion part from the transport calculation, by means of Monte Carlo sampling of the nuclear data. That means the uncertainties are only propagated to response functions that come from the depletion problem, mainly isotopic composition and its derivatives such as decay heat. With this method, neither uncertainties on the neutron spectrum nor the feedback of uncertainties in isotopic composition to neutron spectrum are taken into account, because they are kept constant in every burn-up step. However, when such feedbacks are smaller compared with the explicit effect of nuclear data uncertainties on isotopic compositions, or the problem is addressed with the assumption of constant flux, there is

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no need to include the transport part in these calculations, reducing the amount of time required to perform such an uncertainty study.

The scheme of work is presented in Fig. 1, and explained below, when multi-group cross section uncertainties are used:

1. A single complete coupled transport-depletion problem is performed, from which the spectrum in every depletion step is retrieved.
2. Random multi-group cross sections are drawn by sampling appropriate probability density functions (PDFs), Normal or Lognormal, accordingly to the covariance data used.
3. With one sample of the step 2, a complete depletion calculation is performed, obtaining one history. In every burn-up step, random multi-group cross sections are collapsed with neutron spectrum of the burn-up step to one-group.
4. After a large enough set of histories are carried out, the cross section uncertainties are propagated, and their effects on response functions are assessed with statistical analyses.

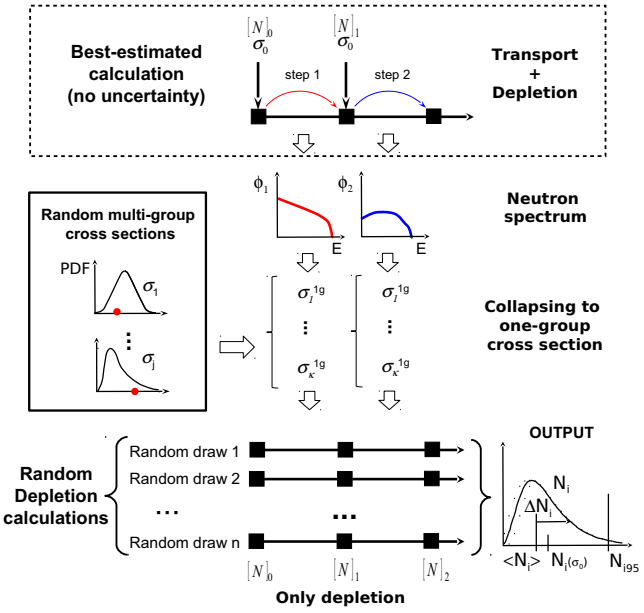

FIG. 1. (Color online) Scheme of the Hybrid Method using multi-group cross section uncertainties.

B. Hybrid Method with One-group Uncertainties

For depletion equations, only collapsed one-group cross sections are required as input. So, uncertainties collapsed in one-group can be used instead, reducing the amount of variables to sample.

In the same way as the multi-group cross sections (σ_i, where i refers to the energy group of a total o n) are collapsed to one-group (σ_{1g}) with Eq. (1) using the multi-group neutron spectrum normalized (ϕ_i), their uncertainties can be collapsed as well with Eq. (2) (derived from the Taylor series and propagation of moments), where V is the covariance matrix of multi-group cross sections. Then, the one-group cross sections can be treated as random variables, and sampled using these values. Under the assumption of keeping the neutron spectrum invariant in every burn-up step, and considering that only depletion equation are being solved, Eq. (2) conserves reaction rate uncertainties,

$$\sigma_{1g} = (\phi_1, \cdots, \phi_n)\,(\sigma_1, \cdots, \sigma_n)^T = \omega^T \sigma \qquad (1)$$

$$\mathrm{var}(\sigma_{1g}) = \omega^T V \omega \quad ; \quad \omega = (\phi_1, \cdots, \phi_n)^T . \qquad (2)$$

The scheme to follow when using one-group cross section uncertainties is presented in Fig. 2, and explained below:

1. *Idem* as multi-group approach.
2. Collapse the multi-groups cross section library and their uncertainties to one-group for every burn-up step, using the neutron spectrum obtained in the best-estimated calculation.
3. Sample the one-group cross sections accordingly to their collapsed covariance matrix.
4. *Idem* as multi-group approach.

As shown in Ref. [8], sampling one-group cross sections of different burn-up steps cannot be done independently. Statistically, the random one-group cross section of two different burn-up steps (e.g j and k) should be correlated, as given in Eq. (3), because of the underlaying multi-group cross sections

$$mathrmVar(\sigma_{1g}^j, \sigma_{1g}^k) = (\omega^j)^T V \omega^k . \qquad (3)$$

If spectrum variations between burn-up steps are small, correlations between the same reaction cross section of two different burn-up steps are close to 1. But also, if relevant group cross sections of different burn-up steps (because high values of their spectrum group) are high correlated through V, correlations close to 1 could be obtained.

In such cases, and in order to keep a Monte Carlo scheme and to avoid the introduction of spectrum variation terms into Eq.(2), correlated sampling is implemented. As represented in Fig. 3, it uses the same random vector drawn from the Gaussian PDF N(0,1) for calculating the random one-group cross sections in every burn-up steps (for the same history). In this way, one-group cross sections are determined with such a vector for this draw/history, and the correlation between cross sections of different burn-up steps is kept to 1.