

Transposition studies with a hybrid experimental database combining ZPR and PWR measurements

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Abstract. Transposition offers the possibility of extending the conclusions of the validation of a scientific calculation tool to a wider field of use by integrating the experimental information in the calculation process. It generally results in bias adjustment and posterior uncertainty reduction. The aim of this paper is to apply a transposition method on main neutronics parameters to a Gen-III PWR benchmark core, based on a combination of measurements from both critical mock-ups and industrial reactors, thus distinguishing ourselves from most of the studies carried out on the subject. The studies carried out focus mainly on highlighting the main nuclear data requirements for reducing the uncertainties propagated in the EDF and Framatome industrial core code COCAGNE. Two observables are analyzed in this context, one is global: the critical boron concentration, rarely studied in the literature. The second is local: the center/periphery fission rate ratio. It is an indicative measure of the center/periphery power bulge inside a reactor core and has received particular attention. A general overview of the advantages of using transposition for integral quantities requiring significant improvements in nuclear data is also given. As a result, it is shown that the hybrid experimental database enables a wide range of sensitivity profiles to be covered and thus a very large number of nuclear data to be constrained, which ultimately leads to significant reductions in the uncertainties after assimilation (about 70% in the case of the fission rate ratio and 80% in the case of the critical boron concentration). The main nuclear data contributors are identified in each case and some recommendations are given to improve them.

1 Introduction

For nuclear power plant operators, uncertainties in neutronics are generally estimated by comparison of calculations and measurements. However, the lack of experimental data in certain cases leads to the exploration of new techniques. The transposition approach, in addition to offering the possibility of extending the conclusions of the validation of a scientific calculation tool to a wider field of use, are of great interest for this purpose. As discussed in the French Nuclear Safety Authority (ASN) “Guide 28” reference document [1], this procedure can be justified either by expert judgement, or by means of a much more rigorous sensitivity analysis between the validation cases and the application case. In this case, the nuclear data are adjusted by data assimilation and the deviations transposed to the application case to improve the knowledge on integral quantities both global and local.

The use of transposition in neutronics goes back a long way, as shown by Gandini’s work [2,3] in the 80s on trans-

posing experimental data to design reference programs. It has been mainly used for fast reactors in a context of the validation of a calculational system for a specific application, with its library of adjusted data (typically ERANOS/ERALIB1 for SFR [4], or more recently the American studies on advanced burners [5]), or for the design of experimental programs (CIRANO [6], designed in support of the Superphénix program). Studies for PWRs are more recent, and were initially dedicated to the design of experimental programs (BASALA [7] or EGERIE [8] in France for example).

The aim of this paper is to apply and analyse a rigorous transposition approach based on Bayesian inference on an industrial context using the core calculation code COCAGNE [9] with experimental database made up of a combination of measurements from both critical mock-ups and industrial reactors, applied to the UAM benchmark [10]. The UAM Gen-III cores were initially designed for neutron characterization and study at zero burnup (zero and full power). However, our study extends the validation scope of the benchmark to include burnup analysis,

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and will also determine the range of representativity of a critical mock-up at zero power, with no fuel burnup indicator, in relation to fuel evolution over a cycle of a Gen-III core. On the other hand, critical experiments have been designed to study the effective multiplication factor (k_{eff}), reactivity effects and local power distributions on a rod-by-rod basis, and are therefore partly used outside their original domain, i.e. extension to large cores. The present study is therefore an attempt to test the limits of using mock-ups in transposition studies. Two observables are analyzed: the critical boron concentration and the ratio of total fission rate between the central assembly and a peripheral assembly, representative of core/peripheral power bulge. They will be studied by using standard and generalized perturbation theory respectively, both implemented in the COCAGNE code.

We utilize Bayesian inference to introduce the equations applied for transposition, focusing particularly on the notion of representativity (Sect. 2). Then, after a brief description of the different configurations of the study (Sect. 3), these concepts are then taken up and applied to the UAM benchmark and an experimental database made up of a combination of measurements from four different critical experiments and four PWRs (Sect. 4). The benefits of the combined use of the two sources of experimental information during the transposition process, which constitutes the original character of this work, are analyzed.

2 Mathematical background

2.1 Data assimilation by Bayesian inference: hypothesis of linear model and gaussian distributions

Let Y be a continuous random variable describing the state of a system, i.e. the interest data. We assume that each realisation of Y depends on $\mathbf{X}$, a vector of continuous random variables representing parameters. In this article, the quantities of interest Y are the effective multiplication factor and the fission rate ratio, calculated by a neutron deterministic calculation scheme. The input data $\mathbf{X}$ are the nuclear data. The measured experimental and calculated values associated with the true unknown value Y are respectively denoted by Y_E and Y_C , such as $Y_E = Y + \xi_E$ and $Y_C = Y + \xi_C$, where ξ_E and ξ_C represent the measurement and calculation errors.

The *prior distribution* represents our initial knowledge of the parameters $\mathbf{X}$ before considering the experimental observations. Let's assume a normal prior distribution for $\mathbf{X}$, $\mathbf{X} \sim \mathcal{N}(\mathbf{x}_0, \mathbf{B}_{\mathbf{x}_0})$ with $\mathbf{x}_0$ the prior mean of the parameters and $\mathbf{B}_{\mathbf{x}_0}$ is the prior covariance matrix of the parameters, representing the initial uncertainties¹.

¹ Note that the Gaussian prior hypothesis for the nuclear data vector $\mathbf{X}$ is a classical and practical choice [11]. It leverages the Central Limit Theorem, the principle of maximum entropy, and the mathematical properties of the Gaussian distribution to provide a robust and tractable framework for Bayesian inference. This assumption is well-supported by empirical data and simplifies the computational procedures, making it a widely accepted approach in nuclear data analysis.

The *likelihood model* describes how the observations y_E of Y_E are related to the parameters $\mathbf{X}$. To build this model, we use knowledge from the calculation.

Letting $y_C = f(\mathbf{x})$ be the deterministic function associated with the random variable Y_C , such that $y_0 = f(\mathbf{x}_0)$, we assume that Y_C can be approximated by a *first-order Taylor expansion* around the nominal value y_0 :

$$Y_C = y_0 + \mathbf{S}(\mathbf{X} - \mathbf{x}_0) + \eta_{Taylor}, \quad (1)$$

where

$$\mathbf{S} = \left(\left[\frac{\partial f}{\partial x_i} \right]_{x_i=x_{i0}} \right)_{i=1..d} \in \mathbb{R}^d$$

is the gradient or sensitivity vector of y_C with respect to the d parameters $\mathbf{X}$ evaluated at $\mathbf{x}_0$. η_{Taylor} represents the error due to the Taylor approximation, which is typically small for small perturbations and can often be neglected under the *hypothesis of linearity*. Under this hypothesis, $\mathbf{S}$ is determined with the first order perturbation theory. Depending on the integral outputs, different perturbation theories are applied to compute the nuclear data sensitivities: the Standard Perturbation Theory (SPT) [12] for the effective multiplication factor (k) and the Generalized Perturbation Theory (GPT) [13] for the fission rate ratio.

The uncertainty associated to y_C due to the parameters is denoted ε_C^2 and is given by the *sandwich rule* [14]:

$$\varepsilon_C^2 = \mathbf{S} \mathbf{B}_{\mathbf{x}_0} \mathbf{S}^t \quad (2)$$

Finally, applying Bayes' theorem [15] with the conjugate relationship between the normal prior and the normal likelihood allows deriving the *posterior distribution* analytically. The *posterior distribution* of $\mathbf{X}$ given the experimental observations $Y_E = y_E$ is a normal distribution: $\mathbf{X}|Y_E = y_E \sim \mathcal{N}(\mathbf{x}_p, \mathbf{B}_{\mathbf{x}_p})$, with the following parameters:

– the posterior mean,

$$\mathbf{x}_p = \mathbf{x}_0 + \mathbf{B}_{\mathbf{x}_0} \mathbf{S}^t (\mathbf{S} \mathbf{B}_{\mathbf{x}_0} \mathbf{S}^t + \varepsilon_\xi^2)^{-1} (y_E - y_0), \quad (3)$$

– and the posterior covariance,

$$\mathbf{B}_{\mathbf{x}_p} = \mathbf{B}_{\mathbf{x}_0} - \mathbf{B}_{\mathbf{x}_0} \mathbf{S}^t (\mathbf{S} \mathbf{B}_{\mathbf{x}_0} \mathbf{S}^t + \varepsilon_\xi^2)^{-1} \mathbf{S} \mathbf{B}_{\mathbf{x}_0} \quad (4)$$

where ε_ξ^2 is the experimental variance.

2.2 Transposition to a case without measurement

Let's consider experimental data from N measured quantities, from one or more configurations ($E_{i,i=1,\dots,N}$) and M other integral quantities from one or more application configurations ($A_{j,j=1,\dots,M}$). The transposition method involves using the known measurements from (E_i), their uncertainties and their deviation from the calculated values to improve our understanding of (A_j) and their associated uncertainties. The application of the data assimilation presented in the previous section allows determining the posterior mean and covariance $\mathbf{x}_p$ and $\mathbf{B}_{\mathbf{x}_p}$.

A key parameter that strongly determines the transferability of information from (E_i) to (A_j) is the *representativity factor* between the two cases. It is expressed by the Pearson correlation, using the Taylor approximations of $y^{(A_j)}$ and $y^{(E_i)}$ (under the linearity hypothesis). It was introduced by Orlov in the 80's [16] and expresses the degree of shared information between the two cases. Its value ranges between -1 and $+1$:

$$r_{E_i, A_j} = \frac{\mathbf{S}^{(E_i)} \mathbf{B}_{\mathbf{x}_0} \mathbf{S}^{(A_j)^t}}{\varepsilon_C^{(E_i)} \varepsilon_C^{(A_j)}}, \quad (5)$$

where $\varepsilon_C^{(E_i)}$ and $\varepsilon_C^{(A_j)}$ are given by the sandwich rule (Eq. (2)) applied on (E_i) and (A_j) .

The application of the data assimilation presented in the previous section allows determining the posterior mean and covariances $\mathbf{x}_p$ and $\mathbf{B}_{\mathbf{x}_p}$ and therefore, the application integral quantities $Y_p^{(A)}$ and their covariances $\mathbf{B}_{Y_p}^{(A)}$ such that [17]:

$$\begin{cases} Y_p^{(A)} \simeq Y_0^{(A)} \\ + \text{diag}(\varepsilon_C^{(A_j)}) \cdot \mathbf{R} \cdot \mathbf{B}_E^{-1} \cdot \text{diag}(\varepsilon_C^{(E_i)})^{-1} \cdot \begin{bmatrix} \dots \\ y_E^{(E_i)} - y_0^{(E_i)} \\ \dots \end{bmatrix} \\ \mathbf{B}_{Y_p}^{(A)} \simeq \mathbf{B}_{Y_0}^{(A)} - \text{diag}(\varepsilon_C^{(A_j)}) \cdot \mathbf{R} \cdot \mathbf{B}_E^{-1} \cdot \mathbf{R}^t \cdot \text{diag}(\varepsilon_C^{(A_j)}) \end{cases} \quad (6)$$

with $i = 1, \dots, N$, $j = 1, \dots, M$. $\text{diag}(\varepsilon_C^{(*)})$ is the diagonal matrix composed of the uncertainties due to the nuclear data of the various quantities. $\mathbf{R}$ is the matrix of experiment-application representativities such that:

$$\mathbf{R} = \begin{bmatrix} r_{E_1, A_1} & \dots & r_{E_N, A_1} \\ \vdots & r_{E_i, A_j} & \vdots \\ r_{E_N, A_1} & \dots & r_{E_N, A_M} \end{bmatrix}$$

and $\mathbf{B}_E$ the so-called *extended correlations* matrix, made up of correlations between the quantities of the experimental configurations, which take into account both the correlations due to nuclear data and the experimental correlations (due to measurements and technological data). It is defined as:

$$\mathbf{B}_E = \begin{bmatrix} \frac{\varepsilon_{\xi_{1,1}}^2}{\varepsilon_C^{(E_1)} \varepsilon_C^{(E_1)}} + 1 & \dots & \frac{\varepsilon_{\xi_{1,N}}^2}{\varepsilon_C^{(E_1)} \varepsilon_C^{(E_N)}} + r_{E_1, E_N} \\ \vdots & \frac{\varepsilon_{\xi_{i,i}}^2}{\varepsilon_C^{(E_i)} \varepsilon_C^{(E_i)}} + 1 & \vdots \\ \frac{\varepsilon_{\xi_{N,1}}^2}{\varepsilon_C^{(E_N)} \varepsilon_C^{(E_1)}} + r_{E_N, E_1} & \dots & \frac{\varepsilon_{\xi_{N,N}}^2}{\varepsilon_C^{(E_N)} \varepsilon_C^{(E_N)}} + 1 \end{bmatrix}$$

$\varepsilon_{\xi_{i,j}}^2$ is the experimental covariance between experiments E_i and E_j . Determining such a quantity is far from trivial, given the different sources involved. We'll come back to this in Section 4.2 dedicated to the determination of experimental uncertainties and correlations.

3 Description of the study

3.1 Constituting the hybrid experimental database for data assimilation

For this study of the transposition process, the experimental information is based on one hand on measurements in critical mock-ups and in the other hand, on measurements carried out on Nuclear Power Plants (NPPs), representing EDF's feedback from over 40 years of operation. The experimental configurations are presented in the following paragraphs.

3.1.1 Description of the selected critical mock-ups

The critical configurations studied in this article were realized in the EOLE Zero Power Reactor (ZPR) [18], within the framework of experimental programs carried out at CEA (French Atomic Energy Commission) in the 90s with the aim of qualifying simulation tools for PWR-type cores. They were selected on an expert judgement, on the basis of the diversity in terms of fuel type, consumable poisons, but also on the availability of the measurements to be assimilated and their uncertainty. Four critical configurations have been considered: UH1.4, UH1.4-ABS, UMZONE and CAMELEON. They are respectively referred to as UH, UH-ABS, UMZ and CAM in the followings. The three first ones are extracted from the EPICURE experimental program devoted to the validation of partial MOX loading in PWRs [19], as the fourth one is from the CAMELEON program [20], devoted to the measurement of absorbers. In each of these experiments, the fuel pins are clad in zircaloy-4 and surrounded by an aluminum overcladding (AG3) to simulate the moderation ratio of a hot PWR, operating at ~ 155 bar, $\sim 300^\circ\text{C}$ while the mock-ups operate at ambient temperature and pressure ($\sim 22^\circ\text{C}$, 1 bar). They are all shown in Figure 1.

3.1.2 The Nuclear Power Plants

In addition to highlighting the valuable database built up during physical tests on industrial reactors, the integration of these measurements into the transposition process aims to study their contribution compared with critical mock-ups, which are generally used in this type of study. The PWR configurations consist of four cores of different nominal power (900, 1300 and 1450 MW_e) comprising 157 (referred to as BU and DA in the following), 193 (CA) and 205 (CZ) fuel assemblies respectively. They are all considered at their first start-up configuration. Some of these assemblies contain Pyrex-poisoned rods, whose effect is to provide enough anti-reactivity at the beginning of life to remain critical throughout the cycle. Some measurements at the fifth cycle of the CA reactor were also considered which is distinguished from the start-up configuration by the introduction of assemblies containing gadolinium rods in addition to irradiated assemblies (containing plutonium isotopes) from previous cycles.

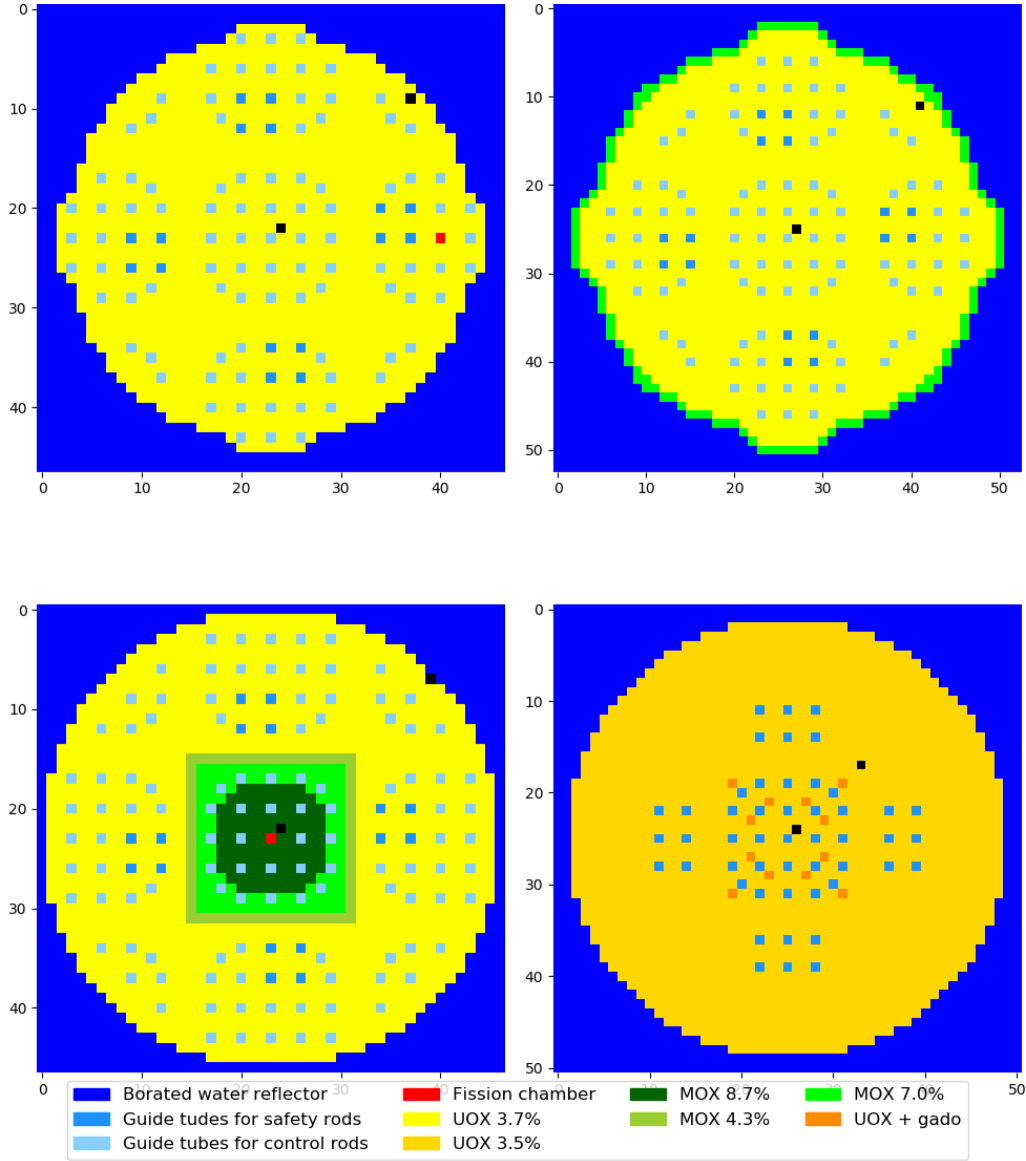


Fig. 1. 2D radial cross-section in the mid-plane of the mock-ups. UH1.4 at the top left, UH1.4-ABS at the top right, UMZONE at the bottom left and CAMELEON at the bottom right. The black dots corresponds to the position of local fission rate measurements used in the present work.

3.2 The Gen-III MOX core of the UAM benchmark as application case

The application configuration is the Gen-III MOX core of the UAM benchmark [10]. It comprises 241 fresh fuel assemblies of different natures: LEU assemblies enriched in ^{235}U to 2.1% and 3.2%, surrounded by trizoned MOX assemblies (consisting in fuel rods of 9.8%, 6.5% and 3.7% of plutonium) on the last ring of the active core. The 3.2% enriched assemblies contain 20 gadolinium-poisoned rods (8% gadolinium oxide on a 1.9% ^{235}U -enriched support), the purpose of which is to provide enough anti-reactivity to compensate for the high reactivity of the fuel at the start-up, in the same way as the Pyrex rods in the previous industrial cores. The active core is surrounded by a homogeneous heavy stainless steel reflector,

the size of an assembly, and itself bathed in borated water. Figure 2 shows a 2D cross-section of the UAM Gen-III MOX.

4 Results and analysis

All calculations are made at critical conditions using a simplified spherical harmonics transport calculation scheme truncated to order 3 (SP3) in a 8-group energy mesh and anisotropy of scattering cross sections of order 3 (P3) for PWRs and order 0 with transport correction (P0_c) for mock-ups. The nuclear data library used is the CEA modified V520 library, based on the JEFF3.1.1 evaluation.

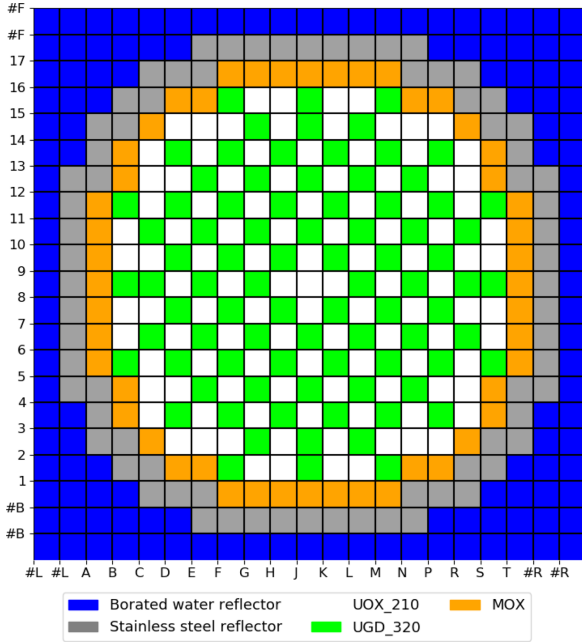


Fig. 2. 2D radial cross-section in the mid-plane of the UAM Gen-III MOX.

4.1 Sensitivity and uncertainty analysis

4.1.1 Generating few group nuclear data covariance for uncertainty propagation

The covariance data needed for this purpose have been pre-processed from the 36 energy group COMAC [21] to obtain 8 energy group new covariance data, which is the energy mesh in which the sensitivities are calculated. The multigroup quantities considered in this paper are essentially cross sections, fissile nuclei multiplicities and fission spectra. The condensation principle is based on the reaction rates conservation and can be resumed by the following expression [22], given for the relative covariance as in the case of COMAC:

$$rcov(\sigma_G^r, \sigma_{G'}^j) = \sum_{\substack{g \in G \\ g' \in G'}} \frac{\tau_g^r \cdot \tau_{g'}^j}{\tau_G^r \cdot \tau_{G'}^j} \cdot rcov(\sigma_g^r, \sigma_{g'}^j) \quad (7)$$

with $\tau_g^r = \phi_g \cdot \sigma_g^r$ the rate of the reaction r in group g and so on.

In the particular case of covariance data involving the total scattering ($scat$), they are obtained by summing the condensed covariance data involving the elastic (el) and inelastic (in) scattering cross sections according to the relation:

$$rcov(\sigma_G^{scat}, \sigma_{G'}^j) = \sum_{\substack{g \in G \\ g' \in G'}} \frac{\tau_g^{el} \cdot \tau_{g'}^j}{\tau_G^{el} \cdot \tau_{G'}^j} \cdot rcov(\sigma_g^{el}, \sigma_{g'}^j) + \sum_{\substack{g \in G \\ g' \in G'}} \frac{\tau_g^{in} \cdot \tau_{g'}^j}{\tau_G^{in} \cdot \tau_{G'}^j} \cdot rcov(\sigma_g^{in}, \sigma_{g'}^j) \quad (8)$$

where j is any other reaction, including scattering reactions.

This treatment is due to the fact that the COMAC covariance data are given separately for the elastic and inelastic scattering cross sections while the sensitivities are only available for the total scattering cross section of each isotope (issued from the spectral code CARABAS, based on APOLLO2 [23]).

However, as the elastic and inelastic scattering rates are not directly computable by the code, they are deduced from the total scattering rate such that

$$\tau_g^{scat} = \tau_g^{el} + \tau_g^{in} = \gamma_g^{el} \cdot \tau_g^{scat} + (1 - \gamma_g^{el}) \cdot \tau_g^{scat}$$

where γ_g^{el} stands for the fraction of the total scattering rate due to elastic scattering in group g and is to be estimated. We used continuous microscopic cross sections with an assumption of constant flux per energy interval corresponding to the different energy meshes involved in the condensation, and expressed it by the relation:

$$\gamma_g^{el} = \frac{\overline{\sigma_g^{el}(E)}}{\overline{\sigma_g^{el}(E)} + \overline{\sigma_g^{in}(E)}} \quad (9)$$

with $\overline{\sigma_g(E)}$ the mean microscopic cross section in group g , and so on for g' , G and G' .

4.1.2 Nuclear data sensitivities and uncertainties

The critical boron concentration sensitivities are deduced from the effective multiplication factor's ones, $\tilde{S}(k, x)$, according to the relation [24]:

$$\tilde{S}(C_b, x) = - \frac{\tilde{S}(k, x)}{\alpha \cdot C_b^{(0)} \cdot k^{(0)}} \quad (10)$$

where $C_b^{(0)}$ and $k^{(0)}$ are the boron concentration and the effective multiplication factor of the nominal (unperturbed) state. α is the boron differential efficiency² and x the nuclear data. The nuclear data uncertainties are then propagated via the sandwich rule (Eq. (2)), using the previous calculated sensitivity vectors.

4.2 Experimental uncertainties and correlations calculation

Experimental uncertainties (δ_E) result from the combination of measurement uncertainties (δ_{mes}) and technological uncertainties (δ_{techno}) such that:

$$\delta_E = \sqrt{\delta_{mes}^2 + \delta_{techno}^2} \quad (11)$$

The technological uncertainties account for the slight technological differences (geometric tolerance, isotopic

² Since they add nothing to this article, and for the sake of conciseness, sensitivity values have not been presented here.

composition, etc.) between actual experimental conditions and the ideal modeled state. While the measurement uncertainties are usually provided by the experimenters, the technological uncertainties are to be calculated from the various sources that contribute to it. In this case, the previous formalism applied to nuclear data can be used [25], resulting to the following relation:

$$\delta_{\text{techno}} = \sqrt{\mathbf{S}_{\text{techno}} \mathbf{B}_{\text{techno}} \mathbf{S}_{\text{techno}}^t} \quad (12)$$

where $\mathbf{S}_{\text{techno}}$ and $\mathbf{B}_{\text{techno}}$ are respectively the sensitivity vector to technological data and their covariance matrix. All the difficulty here is to correctly estimate the technological covariance matrix as the information is frequently unavailable. For this study, we have assumed that only isotopic concentrations contribute to technological uncertainty and are uncorrelated with each other. Moreover, without any indication on the isotopic concentration uncertainties of the PWRs, they were taken to be equal to those of the critical mock-ups, taken from internal reference. A sensitivity study to those uncertainty values will nevertheless be carried out in the following section, in order to deduce their impact on the transposition results.

We have taken as measurement uncertainties, **1% for the boron concentration in both cases and respectively 0.5% and 1% for the assembly/pin fission rate of the mock-ups and PWRs**, which are based on expert judgement since the latter are generally confidential and not readily available in the literature.

The experimental uncertainties are then calculated and are shown in Table 1 below, for the different experimental configurations and for the different variables.

In all cases, it should be noted that the experimental uncertainties are much smaller than the uncertainties due to the nuclear data in the case of the critical boron concentration for all configurations, which is a positive point in the ability to extract information by transposition. The same applies to the center/periphery fission rate ratio of PWRs, but not for the mock-ups, for which the technological uncertainty is at least of the same order of magnitude as the uncertainty due to nuclear data.

Regarding experimental correlations, if we understand correlation as the result of an information exchange between two variables or points in a system (two measurements for us) which leads to a measure of the dependence of their respective fluctuations, 3 main causes of correlations can be identified between experiments (mock-ups or PWRs):

- common components (especially fuel) or experimental conditions.
- Measurement techniques (fission chamber, γ -scanning, titrimetry, ...).
- The measurement system itself: instrumentation, signal acquisition and processing chain.

The first relates to technological data and the last two relates to the measurement itself. Under these assumptions, the experimental correlation between two configurations

can be expressed by the relation:

$$\rho_{E_1, E_2} = \frac{\mathbf{S}_{\text{techno}} \mathbf{B}_{\text{techno}} \mathbf{S}_{\text{techno}}^t + \delta_{\text{MeasTech}}^2 + \delta_{\text{instru}}^2}{\delta_{E_1} \delta_{E_2}} \quad (13)$$

with $\delta_{\text{MeasTech}}^2$ and δ_{instru}^2 the variance associated with the measurement technique and instrumentation respectively, and are defined by:

$$\delta_{\text{measure}}^2 = \delta_{\text{technique}}^2 + \delta_{\text{instru}}^2 \quad (14)$$

At this stage, we assume that $\delta_{\text{technique}} \simeq \delta_{\text{instru}}$ and it is worth noting that $\delta_{\text{MeasTech}} = 0$ if the measurement technique is not the same and also, $\delta_{\text{instru}} = 0$ if the instrumentation is not the same. However, as in the case of the variance due to technological data, identifying the sources contributing to these different terms could enable a more rigorous estimation of the latter, as in equation (12).

Under these assumptions, two measurements of the same variable made using the same technique in the same core (common components) will have a maximum correlation, while two measurements of two different variables (hence different techniques) made in different cores will have a minimum correlation.

The experimental correlations computed in this way are shown in Table 2 for the configurations of the experimental database.

From these results, we can observe that:

- mock-ups are highly correlated with each other when the same type of variable is involved, and uncorrelated when the variables differ. This behavior can be explained by the fact that they all share practically the same fuel, which has little impact on the uncertainty of the fission rate ratio, unlike boron concentration.
- Conversely, the PWRs are almost uncorrelated with each other and with mock-ups, whatever the variable of interest. However, when it comes to inter-cycle measurements for the same reactor, correlations can become significant due to common assemblies or common measuring equipment (CA and CA⁽⁵⁾, for example).

4.3 Representativity and transposition: the nuclear data adjustment

Now that the experimental database has been characterized, the information can be assimilated and transposed to the application case. The results are presented in the following paragraphs.

4.3.1 Study of the representativity between the configurations of the experimental database and the Gen-III UAM-MOX core

The representativities of the configurations in the experimental database compared with UAM-MOX are shown in Table 3 for the zero-power situation. For the sake of conciseness, the first start-up configurations (BU, DA, CZ and CA) are represented in the table by the CA configuration, whose response is identical.

Table 1. Experimental uncertainties for the different observables and for the different configurations at zero power. The critical boron concentration (C_b^{crit}) uncertainties are expressed in ppm. π_f denotes the fission rate ratio.

Cores	Observables	meas. uncert.	techno. uncert.	Total
CA		11	11	15
CA4 ⁽⁵⁾	C_b^{crit}	16	15	22
	π_f	1.4%	1.23%	1.9%
UH	C_b^{crit}	6	10	11
	π_f	0.7%	0.17%	0.7%
UH-MOX	C_b^{crit}	9	11	14
	π_f	0.7%	0.13%	0.7%
UMZ	C_b^{crit}	5	11	12
	π_f	0.7%	0.46%	0.8%
CAM	C_b^{crit}	3	7	7
	π_f	0.7%	0.04%	0.7%

Table 2. Some experimental correlations of the different observables and for the different configurations at zero power.

		CA		CA ⁽⁵⁾		UH		UH-ABS		UMZ		CAM	
		C_b^{crit}	C_b^{crit}	π_f	C_b^{crit}	π_f	C_b^{crit}	π_f	C_b^{crit}	π_f	C_b^{crit}	π_f	
CA	C_b^{crit}	1.00	0.51	0.00	0.17	0.00	0.22	0.00	0.16	0.00	0.15	0.00	
CA ⁽⁵⁾	C_b^{crit}		1.00	0.10	0.18	0.00	0.22	0.00	0.16	0.00	0.16	0.00	
	π_f			1.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
UH	C_b^{crit}				1.00	0.07	0.99	0.05	0.93	−0.13	0.99	0.00	
	π_f					1.00	0.06	1.00	0.08	0.86	0.05	0.97	
UH-ABS	C_b^{crit}						1.00	0.04	0.92	−0.11	0.97	0.00	
	π_f							1.00	0.05	0.85	0.04	0.98	
UMZ	C_b^{crit}								1.00	0.06	0.90	−0.00	
	π_f									1.00	−0.17	0.82	
CAM	C_b^{crit}										1.00	0.01	
	π_f											1.00	

Table 3. Representivity factors of the UAM-MOX observables compared with those of the experimental database configurations: all at zero power.

	CA ¹			CA ⁵			UH		UH-MOX		UMZ		CAM	
	C_b^{crit}	C_b^{crit}	π_f	C_b^{crit}	π_f	C_b^{crit}	π_f	C_b^{crit}	π_f	C_b^{crit}	π_f	C_b^{crit}	π_f	
C_b^{crit}	0,93	0,96	-0,25	0,59	-0,02	0,64	0,04	0,66	-0,24	0,55	0,06			
π_f	0,51	0,24	-0,43	0,22	-0,10	0,22	0,05	-0,04	-0,57	0,22	0,17			

Apart from these zero-power configurations, power measurements performed on the CA core at different irradiation levels (270, 2062, 5937 and 12724 MWd/t³) have also been included in the experimental database. The evolution of the representivity of the UAM-MOX observables during the cycle compared to some of those

³ Corresponding to the irradiations at which the measurements have actually been made.

configurations and to the mock-ups are shown in Table 4⁴. In these tables, the columns represent the different variables of the experimental database configurations, the rows represent the variables of the application configuration at different irradiation levels and the representivities are computed according to equation (5). It can be

⁴ For conciseness, all the different configurations can not be presented in the table.

Table 4. Evolution of the representivity factors of the UAM-MOX observables compared with those of some experimental database configurations. PWRs are at hot Full Power and mock-ups at Cold Zero Power.

Bu (MWd/t)		CA4 ⁽⁵⁾				UH		UH-ABS		UMZ		CAM	
		270		12724		0		0		0		0	
		C_b^{crit}	π_f	C_b^{crit}	π_f	C_b^{crit}	π_f	C_b^{crit}	π_f	C_b^{crit}	π_f	C_b^{crit}	π_f
0	C_b^{crit}	0.95	-0.13	0.83	-0.06	0.55	-0.03	0.59	-0.01	0.58	-0.09	0.52	0.07
	π_f	0.25	-0.52	-0.04	-0.44	0.24	-0.07	0.25	0.07	-0.01	-0.55	0.24	0.16
500	C_b^{crit}	0.96	-0.11	0.85	-0.04	0.54	-0.03	0.59	-0.02	0.58	-0.08	0.51	0.07
	π_f	0.26	-0.51	-0.02	-0.43	0.24	-0.07	0.25	0.07	-0.01	-0.55	0.24	0.15
2000	C_b^{crit}	0.98	-0.03	0.89	0.02	0.52	-0.03	0.57	-0.02	0.58	-0.05	0.49	0.07
	π_f	0.29	-0.47	0.02	-0.41	0.24	-0.07	0.24	0.07	-0.01	-0.54	0.23	0.15
4000	C_b^{crit}	0.98	0.05	0.93	0.09	0.49	-0.03	0.54	-0.03	0.58	0	0.47	0.06
	π_f	0.33	-0.42	0.08	-0.36	0.23	-0.07	0.24	0.06	-0.01	-0.53	0.22	0.14
6000	C_b^{crit}	0.97	0.12	0.96	0.14	0.47	-0.03	0.51	-0.04	0.57	0.04	0.44	0.05
	π_f	0.37	-0.35	0.15	-0.31	0.22	-0.07	0.23	0.05	0	-0.51	0.22	0.13
8000	C_b^{crit}	0.96	0.17	0.97	0.19	0.45	-0.03	0.49	-0.05	0.57	0.07	0.42	0.03
	π_f	0.41	-0.29	0.22	-0.26	0.22	-0.07	0.23	0.05	0.01	-0.49	0.21	0.08
10 000	C_b^{crit}	0.95	0.21	0.98	0.22	0.43	-0.02	0.47	-0.05	0.57	0.09	0.41	0.02
	π_f	0.44	-0.24	0.26	-0.21	0.21	-0.08	0.22	0.04	0.01	-0.48	0.21	0.05

seen how the representativity between two variables evolves with the irradiation.

Interpretation For the critical boron concentration, the PWR has a very high representativity compared to UAM-MOX, whatever the burnup. Moreover, it increases over the cycle, with plutonium production and spectrum hardening. Mock-ups, on the other hand, are only moderately representative.

With regard to the fission rate ratio, at fixed PWR burnup, while representativity is moderately satisfactory at the beginning of the cycle, it decreases over time until it is almost insignificant. Compared with the mock-ups, representativity depends on each of them and is only moderately satisfactory at best. While the low values for sensitivities can be explained by compensation effects between center and periphery due to their small size, the low representativity is due to the difference in the main influencing data between the two types of core. These trends are interpreted by taking two distinct elements into account: on the one hand, the assemblies themselves, counting for the direct effect on sensitivities and therefore representativity, and on the other, the immediate environment of each of them, counting for the indirect effect.

4.3.2 Transposition: nuclear data adjustment and uncertainties reduction

The transposition from the experimental database, comprising all the previous configurations (at zero power and the four irradiation points of CA⁽⁵⁾), gives the results shown in Table 5.

The posterior uncertainties obtained by transposing the hybrid experimental database to the application case

shows a significant reduction compared with the prior uncertainties, the result of adjusting the nuclear data by integrating measurements from both critical mock-ups and large-scale reactors. This diversification of experimental sources enables us to cover a wide range of operating situations, with different characteristics from the point of view of nuclear data, and thus to better constrain them.

With regard to adjusted nuclear data, in the case of the UAM-MOX fission rate ratio at zero power, the adjustment results for the main nuclear data are given in Table 6.

The main adjustments are made to nuclear data which are both the most influential in terms of sensitivities, and for which the transposition process has enabled a significant reduction in uncertainty. This is the case, for example, with the capture cross section and fission spectrum of ²³⁵U, or the capture and fission cross sections of ²³⁹Pu. However, it is important to precise that the adjustments obtained in this study are case-specific and just provide valuable indication of the main nuclear data requirements for reducing uncertainties.

5 Conclusions

This work was dedicated to the application of the transposition method based on Bayesian inference, using a hybrid experimental database made up of critical mock-ups and industrial reactors. The Gen-III UAM-MOX core of the UAM benchmark was modeled for the purposes of this study. It was chosen for its particular design, which was conceived to exacerbate neutron effects due to the presence of MOX fuel with a high plutonium content at the core periphery (which also exacerbates the effect of the steel reflector).

Table 5. Results of the transposition from the experimental database to UAM-MOX configuration.

(% PN)	bu (MWd/t)	prior (ppm)	uncert. (ppm)	C_b^{crit}		prior	uncert. (%)	π_f	
				posterior (ppm)	uncert. (ppm)			posterior	uncert.(%)
0%	0	1467	56	1443	11	0.85	19.2%	1.03	5.7%
	0	1069	64	1047	11	1.45	15.8%	1.70	4.9%
	500	1002	54	984	9	1.50	14.9%	1.76	4.7%
	2000	977	59	971	10	1.66	12.3%	1.98	4.1%
100%	4000	885	60	896	11	1.73	10.2%	2.10	3.7%
	6000	795	63	816	12	1.80	8.7%	2.18	3.3%
	8000	730	65	747	12	1.87	7.4%	2.11	2.9%
	10 000	644	60	639	11	1.83	6.6%	1.89	2.7%

Table 6. Adjustment results for the main nuclear data and their uncertainties. All results are given in relative value. Groups 1 to 3: fast, 4 to 6: intermediate, 7 and 8: thermal.

Isotope	data	group	adjustment	prior uncertainty	posterior uncertainty	uncertainty variation
^{235}U	χ_g	1	9.9%	5.1%	2.9%	-43%
		3	-5.9%	3.2%	1.8%	-44%
		4	-15.9%	6.9%	4.1%	-41%
		2	0.7%	1.5%	1.5%	0%
	ν_g	7	0.8%	0.3%	0.3%	0%
		8	0.8%	0.3%	0.3%	0%
		σ_g^c	-16.9%	8.6%	5.8%	-33%
		σ_g^f	2.0%	1.5%	1.2%	-20%
^{239}Pu	σ_g^c	5	0.8%	2.0%	1.8%	-10%
		7	8.8%	2.5%	1.8%	-28%
	σ_g^c	5	9.6%	4.5%	4.2%	-7%
		6	-16.3%	2.4%	1.4%	-42%
^{240}Pu	σ_g^c	5	-9.0%	1.9%	1.4%	-26%
^{241}Pu	χ_g	1	6.0%	13.7%	10.0%	-27%
^{242}Pu	σ_g^c	5	28.3%	11.6%	9.6%	-17%

The combined use of measurements from industrial reactors and mock-ups finally results in significant uncertainty reductions. For example, the uncertainty in the fission rate ratio at zero power for the UAM-MOX configuration has been reduced from 19% to 6%. This reduction is especially significant since, in addition to its diversity in terms of sensitivity profiles, the database is mostly made up of quasi-independent measurements. However, caution must be taken with extensive experimental databases, since although they are likely to contribute to a greater reduction in uncertainties, all the modeling errors (C/E deviations) of all the configurations in the database will be reflected in the a posteriori deviation of the application configuration. It may therefore be necessary to optimize the experimental database so as to minimize the risk of over-adjusting the nuclear data or introducing excessive modeling errors into the a posteriori value of the variable of interest. This means paying particular attention to the estimation of experimental uncertainties and correlations, and to the nominal modeling of experimental database configurations.

The study also shows the importance of critical mock-ups in a transposition process. Although a priori not very

representative of PWRs for local quantities, they are still of great interest for transposition, provided they are chosen wisely and different types of measurements are combined to constrain as many nuclear data as possible.

These results underline the value of transposition in improving the predictability of calculation codes, and can provide valuable guidance to the nuclear data evaluation community in focusing efforts on the most important data in reactor physics (fission spectrum and capture cross-section of ^{235}U , diffusion cross-sections of ^{238}U and $^1\text{H}_{\text{H}_2\text{O}}$ or capture and fission cross sections of plutonium isotopes for example.). This can be achieved through the development of new integral and/or differential experiments specifically designed for this purpose.

As an outlook, a sensitivity study of the transposition results to uncertainties in isotopic concentrations, as well as a sensitivity study to the type of configurations in the experimental database (mock-ups vs PWRs) should be realized in order to evaluate their impact on the results. Similarly, the impact of modelling bias should also be taken into account, given that this study assumes an unbiased model. These issues could be the subject of future work.

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Conflicts of interest

The authors declare that they have no competing interests to report.

Data availability statement

The authors confirm that the associated data of this review are available within the article and online ressources in the references section.

Author contribution statement

All the authors gave important contributions to the paper. The individual contributions to the article can be summarized as follow: **E. Njayer**: Data elaboration, simulations, formal analysis, writing – original draft. **P. Blaise**, **D. Couyras**, **J.-P. Argaud**, **L. Clouvel**, **N. Dos Santos**: formal analysis, supervision, writing – review & editing.

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