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Nuclear data uncertainty propagation on a sodium fast reactor

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ABSTRACT

Uncertainties on criticality parameters and reactivity effects in a SFR (Sodium Fast Reactor) were calculated and characterized. An SFR model loaded with ≈ 10% MAs (Minor Actinides) was chosen as test case of a burner reactor, its core geometry was implemented in MCNP-5 according to benchmarking specifications. Monte Carlo random sampling on the ENDF/B-VII.1 neutron-induced nuclear data was performed using the SANDY code developed at the Belgian nuclear research center SCK•CEN. SANDY was proved to be a successful tool for uncertainty quantification in the model case of a MA burner SFR. The results show that despite the high content in MAs, the contribution of U-238 data is still dominant in most instances, especially the inelastic scattering cross section data. Minor actinides together with Pu make up to 33% of the total calculated k_{eff} uncertainty, which amounts to $\pm$ 1345 pcm of reactivity. Relative uncertainty greater than $\pm$ 3.47% was estimated for β_{eff} with significant contribution from the prompt multiplicity data of U-238 and Pu-239. Relative uncertainty on the Doppler reactivity worth was found to be at least $\pm$ 12.6%. Coolant voiding in different core zones yields reactivity worth with relative uncertainties whose lower bounds range from $\pm$ 6.37% to $\pm$ 22.47% depending on the voiding conditions, with important contributions due to the Na-23 nuclear data uncertainty.

1. Introduction

The calculation of uncertainty on the neutronic parameters is a crucial step of a reactor core design, because it provides information on how confidently these parameters comply with safety margins. In recent years, two factors enabled some major improvements of uncertainty quantification tools for neutronic calculations: an increase in computation power and the addition of covariance data in the nuclear data libraries. The contributions of each specific nuclear parameter on the total propagated uncertainty can be now accurately assessed, and they can give insights to the nuclear data expert groups about where future efforts should be spent to improve weaknesses in the existing data.

A class of uncertainty quantification methods relies on perturbation theory to calculate sensitivity coefficients of each output parameter with respect to input perturbations, GPT (Generalized Perturbation Theory) is being used in reactor safety calculations in combination to Monte Carlo neutronics solvers such as SERPENT (Fratoni and Aufiero, 2016; Leppänen et al., 2013) and SCALE (Rearden et al., 2015) with its TSUNAMI module (Williams and Rearden, 2008). As an alternative to perturbation theory, Monte Carlo uncertainty quantification methods rely on the random sampling of the input data and the examination of the output distribution, obtained by feeding the sampled inputs to a solver code of choice. Some of the existing codes include NUDUNA (Buss et al., 2012), NUSS (Zhu et al., 2015), TMC (Total Monte Carlo), SCALE modules SAMPLER and XSUSA (Williams et al., 2013; Zwermann et al., 2013) and SANDY (Fiorito et al., 2017). Optimized methods such as fast-TMC and GRS have also been developed with the goal to reduce the computational cost required to target a certain accuracy (Zwermann et al., Jul 2012; Rochman et al., 2014). Monte Carlo methods are often referred to as "model-free" as they do not require alterations on the problem model, any solver can be used to process the sampled input on any kind of system. Unlike in finite order perturbation theory, there is no need to introduce approximations on the physics of the system, and input nuclear data uncertainties are consistently propagated whatever their relative magnitude is.

In the framework of Gen-IV (World Nuclear Association, 2017) research, special attention is being addressed to these uncertainty quantification tools. Several benchmarks are being carried out to cross check the reliability of these methods on Gen-IV concepts for reactor safety calculations. When compared to traditional LWRs (Light Water Reactors), fast reactors loaded with large amounts of Pu and MAs exhibit a deterioration of some important safety parameters such as the effective delayed neutron fraction β_{eff} and the reactivity feedback effects. These shortcomings are due to fundamental design choices, for example the presence of a fast neutron flux or the large amounts of MAs (Minor Actinides) in the fuel of a fast burner reactor. Therefore, accurate calculations of the margins of uncertainty become more stringent in these

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types of system.

Relative uncertainties associated to nuclear data at fast energies are often larger than at thermal energies. This difference arises mainly because cross section measurements at fast neutron energy are scarcely available to date. As a consequence, larger propagated uncertainties on the safety parameters are expected for calculations with a fast neutron spectrum. Nuclear data expert groups are interested in identifying the isotopes and reaction data that should be improved for the benefit of fast reactors safety.

In the present work, nuclear data covariances available in the library ENDF/B-VII.1 (Chadwick et al., 2011) were propagated on the neutronics parameters of a SFR (Sodium Fast Reactor) fueled with a large amount of MAs. The core is modeled and simulated by means of a general purpose radiation transport code: MCNP-5 (Los Alamos National Laboratory Monte Carlo Code Group, 2000; 5 Monte Carlo Team Los Alamos National Laboratory, 2003). The geometry specifications are described in "Benchmark on Low Void SFR Burner Core: Impact of Minor Actinides Nuclear Data Uncertainty on Integral Parameters" (Gabrielli and Rineiski, 2015). The selected method is the Monte Carlo random sampling of nuclear data carried out by the SANDY (SAmpler of Nuclear Data and uncertaintY) code developed at SCK•CEN. The perturbed input uncertainties include cross section data (σ^{XS}), prompt neutron multiplicities ($\overline{\nu}$), angular scattering distribution ($\overline{\mu}$) and fission neutron energy distribution data (χ). The propagated output uncertainty is calculated and characterized on the following parameters: effective neutron multiplication factor k_{eff}, effective delayed neutron fraction β_{eff}, void reactivity worth $\Delta\rho_v$ and Doppler effect quantified as a reactivity worth $\Delta\rho_d$.

2. Reactor Core model and fuel for transmutation

A fully detailed 3D model of the reactor core was built by following the benchmark specifications rigorously. The model is inspired on the French design ASTRID CFV-V1 (Cœur à Faible effet de Vide – Low Void effect Core), developed by CEA and EDF (Gabrielli et al., 2015CEA Nuclear Energy Division, 2012). The model is scaled from the original power of 1500 MWth down to 1200 MWth (Gabrielli and Rineiski, 2015). The core contains two concentric fueling zones, assemblies are placed in an hexagonal lattice as well as the fuel pins within each assembly. Fig. 1 shows a top view of the core. In order to improve the Monte Carlo statistics, axial symmetries were exploited by eliminating 5/6 of redundant volume and reflective boundary conditions were applied along symmetry planes. The equivalence between the full volume and the 1/6 symmetric model was verified for major neutronic parameters.

The choice of fuel envisaged for this work serves the purpose of modeling a MA burner core. The fuel contains about 70 wt% depleted uranium, 20 wt%–22 wt% Pu in the inner and outer fuel respectively, and about 10 wt% MAs. While the choices of depleted uranium and plutonium weight fraction both reflect the composition of a typical MOX (Mixed OXide) fuel, the vector of TRUs (TRans-Uranic elements) reproduces the reprocessing product of a PWR (Pressurized Water Reactor) spent fuel after 30 years from irradiation (Gabrielli and Rineiski, 2015). Table 1 shows the uranium, plutonium and MA vectors in the fuel inventory.

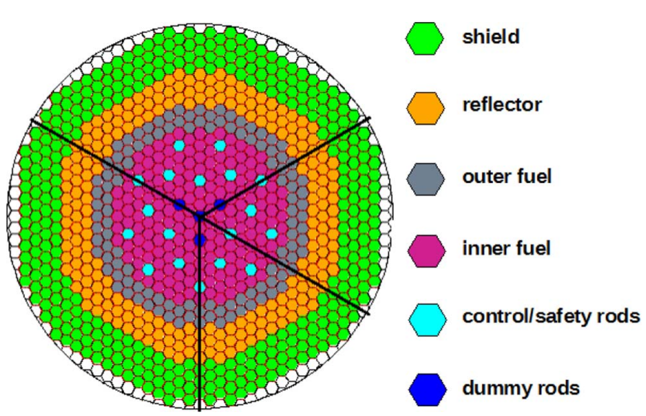

Fig. 1. Top view of the core (xy).

Table 1
Element-wise and region-wise weight percent ratios of all actinides in the fuel inventory, The region-wise ratios are defined over a total weight that includes oxygen and other unknown impurities.

Isotope	[wt%] within group	Inner core	Outer core	Blanket
Uranium isotopic composition		*Region-wise U* [wt%]		
U-235	0.2	57.27	54.583	78.876
U-238	99.8			
Plutonium isotopic composition		*Region-wise Pu* [wt%]		
Pu-238	3.737	20.63	22.426	0
Pu-239	46.446			
Pu-240	34.121			
Pu-241	3.845			
Pu-242	11.85			
Pu-244	0.001			
MAs isotopic composition		*Region-wise MAs* [wt%]		
Np-237	3.884	9.981	10.849	8.994
Am-241	75.51			
Am-242m	0.254			
Am-243	16.054			
Cm-243	0.066			
Cm-244	3.001			
Cm-245	1.139			
Cm-246	0.088			
Cm-247	0.002			

The method chosen to quantify the uncertainty implies that a MCNP-5 criticality calculation of the reactor core model has to be run for each of the numerous samples, which would be computationally expensive. Hence, for the sake of reducing the computational time, simplifications were introduced in the reactor core geometry. A homogenized model of the reactor was built, in which the materials for fuel, gas gap, cladding, structures and coolant were homogenized into an evenly blended mixture. This homogenization resulted in doubling the calculation speed. By altering the geometry and rescaling the volumetric densities, some systematic differences in the behavior of the core with respect to the heterogeneous model were introduced.

However, equivalence in terms of response to perturbations can be proved if a high linear correlation is found between the response of the two models to the same perturbation. If such a condition holds, the relative uncertainties calculated on the homogenized model can be considered valid on the fully detailed model. This case was tested by running the two models over a set of 100 samples of perturbed input data obtained with SANDY (the cross section data block was perturbed at once for each sample). The linear correlation was then measured over the resulting multiplication factors, the one-group fluence tally $\int_{V,E}\Phi$ and the one-group reaction rate tallies for fission $\int_{V,E}\Phi\sigma(n,F)$ and capture $\int_{V,E}\Phi\sigma(n,\gamma)$ of Pu-239, chosen as test case because of its dominant contribution to fissions. Results are summarized in Table 2 and show a high degree of correlation between the two models.

Table 2
Correlation of selected output parameters of the fully detailed and homogenized models as a response to the same perturbation on the input nuclear data.

Parameter	Statistical uncertainty	Correlation
k_{eff}	0.01%	97.11%
$\int_{V,E}\Phi$	0.04%	96.52%
$\int_{V,E}\Phi\sigma(n,F)$ Pu-239	0.04%	96.77%
$\int_{V,E}\Phi\sigma(n,\gamma)$ Pu-239	0.09%	96.86%

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