Preliminary SCALE/TSUNAMI Results for the Sub-exercises of the OECD/NEA Benchmark for Uncertainty Analysis in Modeling of Sodium-Cooled Fast Reactors*

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INTRODUCTION

Sodium-cooled fast reactor (SFR) systems have been identified as a promising advanced reactor technology option for securing the energy supply in the future. Accurately simulating the reactor physics and quantifying the associated uncertainties in the computed quantities are essential to forming the basis for safe reactor operation. The use of different materials in these reactor types, the fast neutron spectrum, and the long neutron mean free path and high neutron leakage requires extensive testing of modeling and simulation tools and their corresponding nuclear data libraries that have been developed with focus on thermal reactor systems.

The SCALE Code System [1] is a widely used tool for reactor physics, criticality safety, and sensitivity and uncertainty calculations. Recent activities relating to advanced reactor systems involve the generation of multigroup cross section and covariance libraries for the analysis of SFR systems with SCALE [2, 3]. In this study, these libraries are used in uncertainty and sensitivity analyses of SFR systems specified within the sub-exercise definitions of the OECD/NEA Benchmark for Uncertainty Analysis in Modelling (UAM) for Design, Operation, and Safety Analysis of SFRs [4]. The analysis of models ranging from a pin cell up to a full core is to be performed to systematically assess the influence of nuclear data uncertainties on fast reactor simulations including eigenvalues, reactivity feedback, and the generation of few-group cross sections.

MODELS AND TOOLS

The present study involves the analysis of two-dimensional hexagonal fuel pin cells, assemblies (Fig. 1) and super-cells defined within the OECD/NEA UAM SFR benchmark [4]. The super-cell consists of a central absorber assembly surrounded by a number of fuel assemblies to represent a realistic absorber-to-fuel ratio (Fig. 2). For each model, both the metallic fuel design (MET1000) and oxide fuel design (MOX3600), are studied. The MOX3600 fuel pins contain annular uranium-transuranic (U-TRU) oxide fuel at a temperature of 1500 K, and the MET1000 pins contain cylindrical U-TRU-Zr metal fuel at 807 K. The fuel in both assemblies is beginning-of-equilibrium-cycle fuel. The fission products are replaced by one element (Mo) with a density that accounts for representative absorption.

The impact of uncertainties of nuclear data on several output quantities including the eigenvalue, cross sections, and reactivity effects is studied using the TSUNAMI tools from the SCALE 6.2.3 [1]. TSUNAMI uses linear perturbation theory to estimate the change in the eigenvalue due to changes in the energy-dependent cross section data via sensitivity coefficients. The data-induced uncertainty in the system eigenvalue is then estimated by multiplying the sensitivity coefficients with corresponding covariance matrices (using the *sandwich formula* [1]). The sensitivities of system responses other than the eigenvalue, such as reaction rates and energy integrated cross sections, are calculated using generalized perturbation theory (GPT) [5, 6]. In case of responses of reactivity difference such as Doppler feedback and control rod worth, the SCALE module TSAR is used to combine sensitivity coefficients at two states to calculate the uncertainty of the reactivity difference between these two states [7].

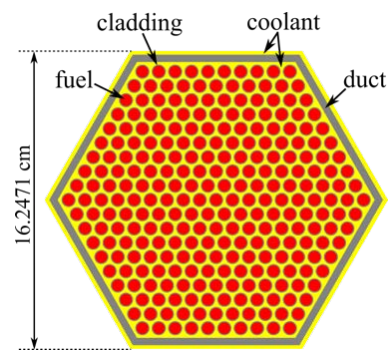

Fig. 1. MET1000 fuel assembly model.

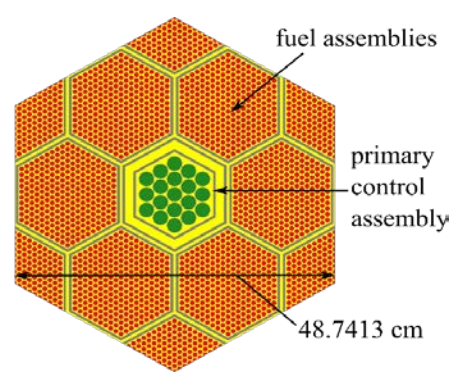

Fig. 2. MET1000 super-cell model.

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For the pin cell and fuel assembly models, both TSUNAMI-2D and CE TSUNAMI-3D are applied. The super-cells are only analyzed with CE TSUNAMI-3D. TSUNAMI-2D uses the two-dimensional deterministic solver NEWT as neutron transport code in combination with a 302-group cross section library that is optimized for these systems in terms of the energy group structure and the weighting spectrum [8]. In addition to the forward calculation, an adjoint transport calculation is performed for each requested output response. CE TSUNAMI-3D uses the three-dimensional Monte Carlo code KENO-VI with continuous-energy (CE) data and requires only one forward calculation [9]. The CLUTCH method [9] is used to determine the eigenvalue uncertainties, and the GEAR method [6] is applied for the GPT responses

The 302-group and CE cross section libraries applied here are based on ENDF/B-VII.1 data. The 302-group library was generated with the AMPX tools [10] distributed with SCALE 6.2. Similarly, the 302-group covariance library was generated with AMPX from the same data sources as the SCALE 6.2 covariance libraries, but with a SFR weighting spectrum.

RESULTS

The nominal eigenvalues and the eigenvalue uncertainties due to nuclear data uncertainties of the pin cell models are consistent between TSUNAMI-2D and CE TSUNAMI-3D considering the statistical uncertainty of the Monte Carlo results, demonstrating the good performance of the 302-group libraries. Table I compares the eigenvalue estimates and the sandwich equation–propagated eigenvalue uncertainty for the pin cell models. For both designs, the eigenvalue uncertainty is larger than 1.5% and therefore about three times as large as that for typical light water reactor (LWR) designs. The most important contributor to this uncertainty is inelastic scattering with U-238. The covariance of this reaction alone causes about 1.35% and 1.56% uncertainty in the eigenvalue for the metallic and oxide pin, respectively. The uncertainties of several relevant microscopic 1-group cross sections and the macroscopic 1-group absorption cross section in fuel range from less than a percent to almost 18% (Fig. 3).

Since the TSUNAMI-2D and CE TSUNAMI-3D results are in very good agreement for the fuel assemblies, only CE TSUNAMI-3D results are presented for these models. For the super-cells, only CE TSUNAMI-3D calculations are performed. The eigenvalue uncertainties for the fuel assemblies and super-cells are similar to the ones observed for the pin cells (Table III and IV). The main contributor to this uncertainty is again inelastic scattering with U-238 for all cases.

For the fuel assemblies, the Doppler and sodium-void reactivities were analyzed using TSAR. For the Doppler reactivity the fuel temperature was doubled, and for the sodium reactivity the sodium was completely removed from the model. The uncertainties obtained for both coefficients are above 5%. The main contributor to the Doppler uncertainty is inelastic scattering on U-238. For the sodium void coefficient, elastic and inelastic scattering of Na-23 contribute most to the overall uncertainty. The uncertainties of the investigated macroscopic 4-group cross sections of the fuel assembly are between 0.5 and 2.5% (Table II, Fig. 4). The scattering uncertainties are higher than those for fission because of the large uncertainties of these cross sections in the fast energy range. The uncertainties of the metallic fuel assembly are always slightly larger than the ones for the oxide fuel assembly, possibly caused by the different fuel composition and harder spectrum.

In the case of the super-cell, the control rod worth is studied by removing the control rods from the model such that the assembly duct is completely filled with sodium. The control rod worth of both super-cells shows uncertainties larger than 2.5%. The main contributor is again inelastic scattering of U-238. Other important contributors are neutron scattering of Fe-56 and Na-23 and the fission spectrum of U-238 and Pu-239. The uncertainties of the macroscopic 4-group cross sections of the central control assembly are in the same range as those of the fuel assemblies. The uncertainties of the scattering cross sections are significantly larger than the uncertainties of the absorption cross sections (Fig. 5).

TABLE I. Pin cell eigenvalue results. The 1σ statistical uncertainties are given in parentheses.

		k_{inf}	Δk/k
MET1000	TSU-2D	1.3585	1.52%
	CE TSU	1.3588(2)	1.52(2)%
MOX3600	TSU-2D	1.1924	1.69%
	CE TSU	1.1920(3)	1.71(1)%

TABLE II. 4-group structure.

Group	1	2	3	4
Upper en.	20 MeV	820 keV	110 keV	15 keV

TABLE III. Fuel assembly CE TSUNAMI-3D results. The 1σ statistical uncertainties are given in parentheses.

		Nominal Val.	Δk/k
MET1000	k_{inf}	1.2800(1)	1.41(1)%
	Doppler	-338(13) pcm	6.3(5)%
	Na-void	5895(9) pcm	5.47(1)%
MOX3600	k_{inf}	1.1467(1)	1.52(1)%
	Doppler	-781(21) pcm	5.2(3)%
	Na-void	2945(10) pcm	5.57(1)%

TABLE IV. Super-cell CE TSUNAMI-3D results. The 1σ statistical uncertainties are given in parentheses.

		k_{inf}	Δk/k
MET1000	k_{inf}	1.0841(1)	1.49(1)%
	CR worth	12081(11) pcm	2.81(1)%
MOX3600	k_{inf}	1.0771(1)	1.52(1)%
	CR worth	4973(11) pcm	2.67(1)%

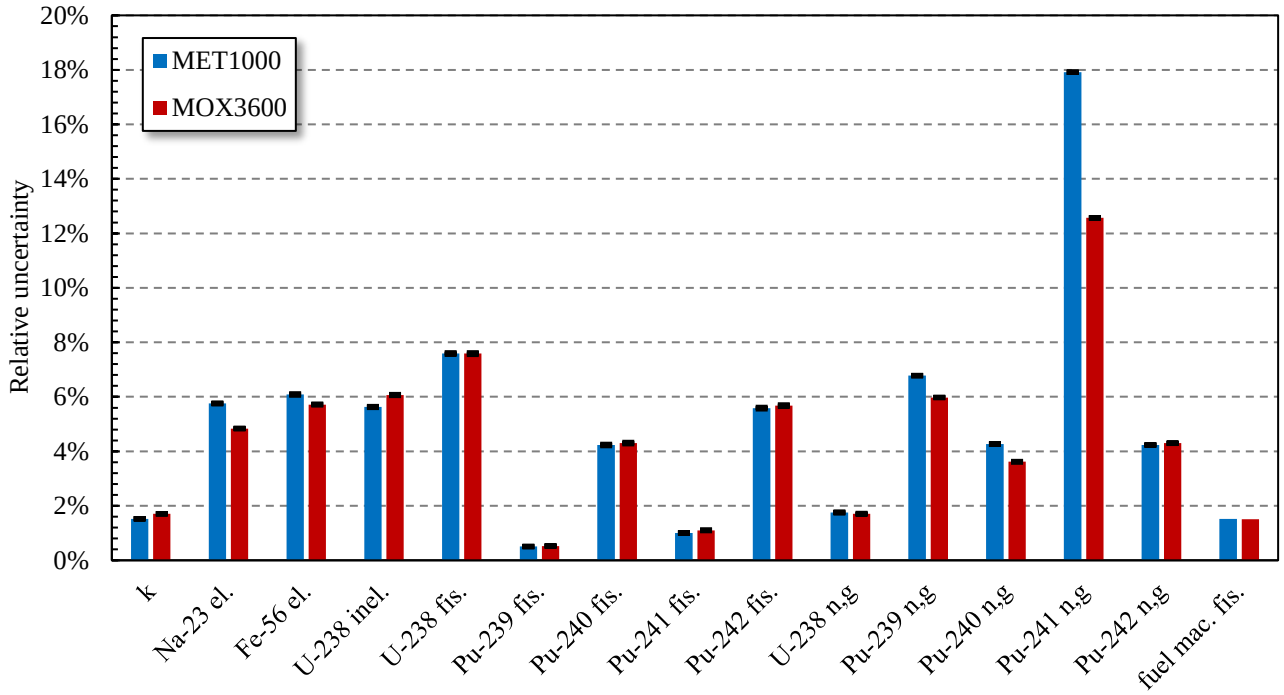

Fig. 3. CE TSUNAMI-3D pin cell eigenvalue and 1-group cross section uncertainties (k: eigenvalue; mac: macroscopic; el: elastic scattering; inel: inelastic scattering; fis: fission; n,g: radiative capture). The 1σ statistical uncertainties from the Monte Carlo calculation are indicated as error bars.

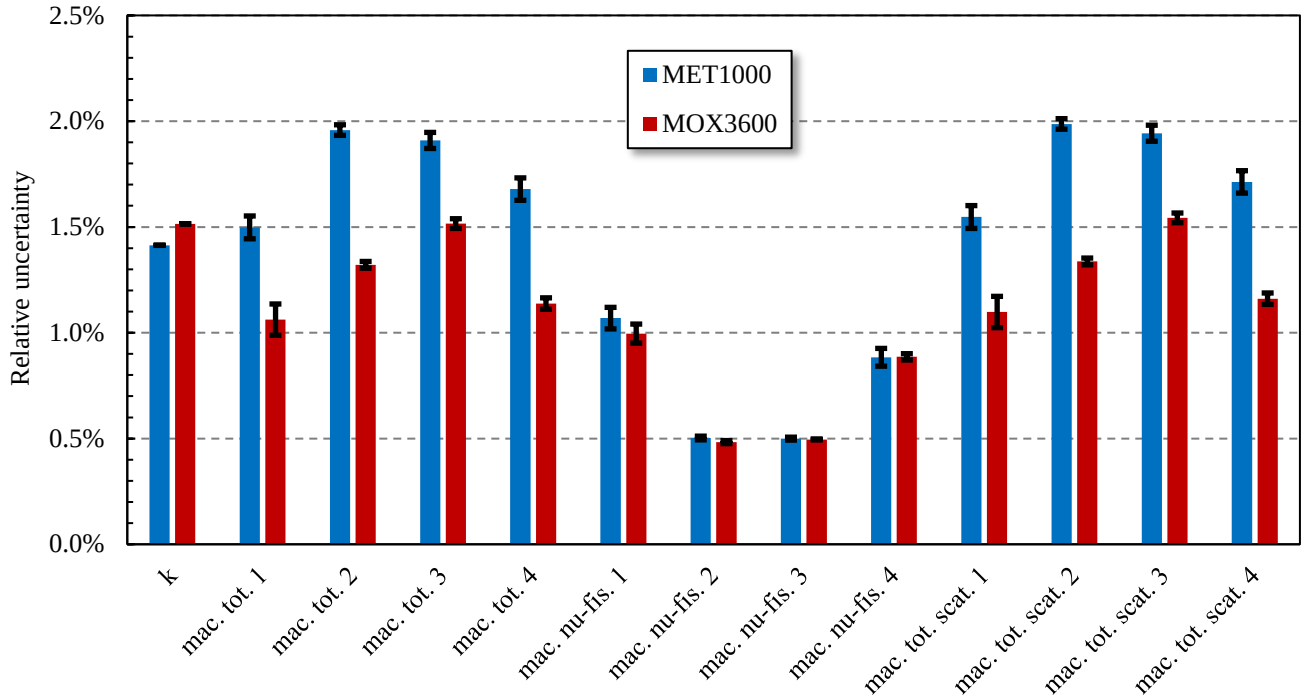

Fig. 4. CE TSUNAMI-3D fuel assembly eigenvalue and macroscopic 4-group cross section uncertainties (k: eigenvalue; mac: macroscopic; tot.: total; nu-fis.: nubar times fission cross section; tot. scat.: total scattering from an individual energy group to all other groups). The 1σ statistical uncertainties from the Monte Carlo calculation are indicated as error bars.

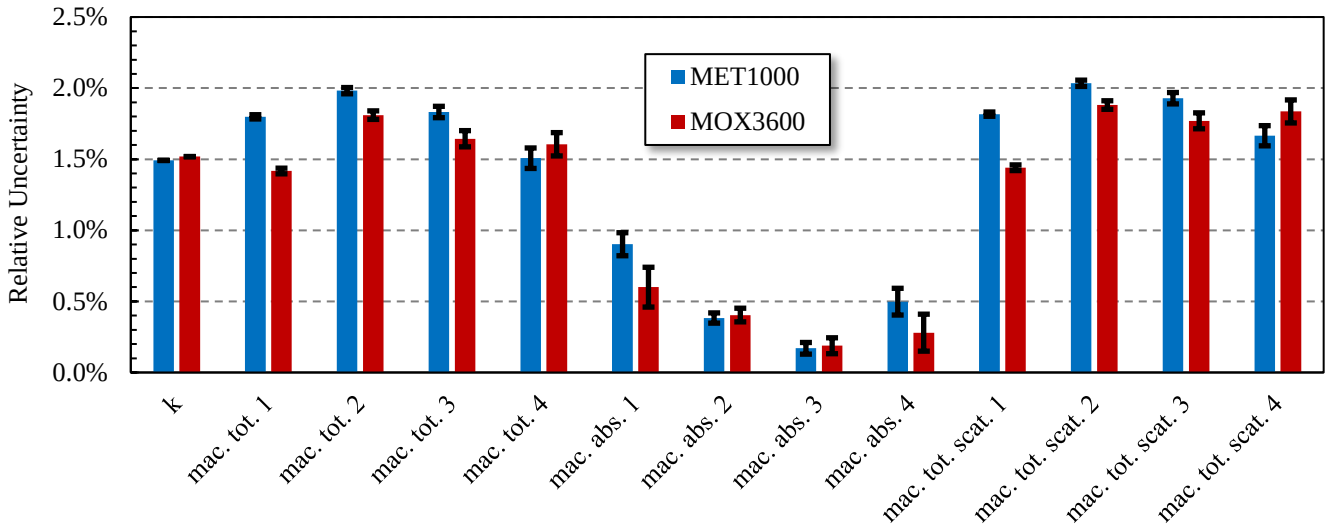

Fig. 5. CE TSUNAMI-3D super-cell eigenvalue uncertainty and macroscopic 4-group cross section uncertainties of the central absorber assembly (k: eigenvalue; mac: macroscopic; tot.: total; tot. scat.: total scattering from an individual energy group to all other groups). The 1σ statistical uncertainties from the Monte Carlo calculation are indicated as error bars.

CONCLUSIONS AND OUTLOOK

The preliminary results of the sub-exercises OECD/NEA Benchmark for Uncertainty Analysis in Modelling for Design, Operation and Safety Analysis of SFRs obtained with SCALE/TSUNAMI were presented. The eigenvalue uncertainties of the pin cells, fuel assemblies, and super-cells were found to be about 1.5% and hence about three times as large as those for corresponding LWR models. The major contributor to these uncertainties is inelastic scattering with U-238. If the uncertainties of this cross section could be reduced by additional measurements or evaluations, the output uncertainties would be significantly reduced. The uncertainties of the Doppler coefficient and sodium void reactivity are above 5%, and the control rod worth uncertainty of the super-cells is more than 2.5%. The investigated macroscopic 4-group cross sections show uncertainties up to about 2%, whereas the microscopic 1-group cross sections show a range of uncertainties from <1% to over 18%.

The next step of the analysis chain is the uncertainty analysis of the full core systems for which several reactivity effects and the fuel assembly power distribution will be studied. Furthermore, two validation experiments will be investigated to assess the performance of these methods by comparing calculated eigenvalues and reactivity effects with measured data while considering the uncertainties due to nuclear data.

ACKNOWLEDGMENTS

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