Objectives and Status of the OECD/NEA sub-group on Uncertainty Analysis in Modelling (UAM) for Design, Operation and Safety Analysis of SFRs (SFR-UAM)

G. Rimpault[1] , L. Buiron[1], N. E. Stauff[2], T. K. Kim[2], T. A. Taiwo[2], Y-K Lee[3], A. Aures[4], F. Bostelmann[4], E. Fridman[5], A. Keresztúri[6], B. Batki[6], I.-A. Kodeli[7], K. Mikityuk[8], R. Lopez[9], A. Gomez[9], F. Puente-Espel[9], E. del Valle[9], A. Peregudov[10], M. Semenov[10], G. Manturov[10], Y. Nakahara[11], J. Dyrda[11], T. Ivanova[11]

[1] CEA, DEN, DER, SPRC, Cadarache, F-13108 St Paul-Lez-Durance, France

[2] Nuclear Engineering Division, Argonne National Laboratory (ANL), Argonne, U.S.A

[3] CEA, DEN, DM2S, SERMA, Saclay, F-91191 Gif-sur-Yvette, France

[4] Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) gGmbH Forschungszentrum Boltzmannstr. 14, D-85748 Garching bei München, Germany

[5] Helmholtz-Zentrum Dresden-Rossendorf (HZDR) Bautzner Landstraße 400 D-01328 Dresden

[6] MTA EK Center for Energy Research, H-1525 Budapest 114, Hungary

[7] Institut Jozef Stefan, Jamova 39, 1000 Ljubljana, Slovenia

[8] Paul Scherrer Institut, CH 5232 Villigen PSI, Switzerland

[9]Instituto Nacional de Investigaciones Nucleares (ININ), Edo. de México, Mexico.

[10] Institute for Physics and Power Engineering (IPPE), Obninsk, Russian Federation.

[11] OECD Nuclear Energy Agency (NEA), 92100 Boulogne-Billancourt, France

E-mail contact of main author: gerald.rimpault@cea.fr

Abstract. An OECD/NEA sub-group on Uncertainty Analysis in Modelling (UAM) for Design, Operation and Safety Analysis of Sodium-cooled Fast Reactors (SFR-UAM) has been formed under the NSC/WPRS/EGUAM and is currently undertaking preliminary studies after having specified a series of benchmarks.

The incentive for launching the SFR-UAM task force comes from the desire to utilize current understanding of important phenomena to define and quantify the main core characteristics affecting safety and performance of SFRs. Best-estimate codes and data together with an evaluation of the uncertainties are required for that purpose, which challenges existing calculation methods. The group benefits from the results of a previous Sodium-cooled Fast Reactor core Feed-back and Transient response (SFR-FT) Task Force work under the NSC/WPRS/EGRPANS.

Two SFR cores have been selected for the SFR-UAM benchmark, a 3600MWth oxide core and a 1000MWth metallic core. Their neutronic feedback coefficients are being calculated for transient analyses. The SFR-UAM sub-group is currently defining the grace period or the margin to melting available in the different accident scenarios and this within uncertainty margins.

Recently, the work of the sub-group has been updated to incorporate new exercises, namely, a depletion benchmark, a control rod withdrawal benchmark, and the SUPER-PHENIX start-up transient. Experimental evidence in support of the studies is also being developed.

Key Words: SFR, uncertainties, OECD benchmark, reactivity coefficients.

1. Introduction

There is a strong incentive to design reactors with improved safety performance while preserving a sustainable source of energy at a rather low cost. The Generation IV

International Forum (GIF) has defined the key research goals for advanced Sodium-cooled Fast Reactors (SFR):

- improved safety performance, specifically a demonstration of favorable transient behavior under accident conditions;
- improved economic competitiveness;
- demonstration of flexible management of nuclear materials, in particular, waste reduction through minor actinide burning.

Sodium-cooled Fast Reactors offer the most promising type of reactors to achieve such Generation IV goals at a reasonable time scale given the experience accumulated over the years. However, it is recognized that new regulations and safety rules as they exist worldwide are requiring improved safety performance. In particular, one of the foremost GIF objectives is to design cores that can passively avoid core damage when the control rods fail to scram in response to postulated accident initiators (e.g., inadvertent reactivity insertion or loss of coolant flow). The analysis of such unprotected transients depends primarily on the physical properties of the fuel and the reactivity feedback coefficients of the core.

Under the auspices of the Working Party on Scientific Issues of Reactor Systems (WPRS), an Expert Group task force was formed to investigate Sodium Fast Reactor core Feed-back and Transient response (SFR-FT) in order to identify recent progress in this field. The work was focused on a shared analysis of the feedback and transient behavior of the next generation SFR concepts [1].

The analysis on the transient behaviors under accident conditions was not completed under the SFR-FT benchmark and will be completed within a work group hosted by the UAM (Uncertainty Analysis in Modelling) working group.

Recently, the International Atomic Energy Agency (IAEA) produced guidance on the use of deterministic safety analysis (DSA) for the design and licensing of nuclear power plants (NPPs): ''Deterministic Safety Analysis for Nuclear Power Plants Specific Safety Guide''. Since the early days of civil nuclear power, the conservative approach has been used and is still widely used today. However, the desire to utilize current understanding of important phenomena and to maximize the economic potential of NPPs without compromising their safety has led many countries to use best-estimate codes and data together with an evaluation of the uncertainties.

The group benefits from the results of a previous Sodium Fast Reactor core Feed-back and Transient response (SFR-FT) Task Force work which demonstrated that for the benchmark cores under study the major source of bias between participants is coming from nuclear data. Nevertheless, the detailed review of modeling effects shows that there is a need to:

- describe heterogeneous subassembly for cross section generation,
- use fine group energy description for self-shielding effects,
- have a specific treatment in order to take into account spatial self-shielding effect for cross sections of control rod materials.

Doppler and Void coefficients were calculated as well as some important dynamic characteristics of the core. Missing in the benchmark were feedback coefficients associated to thermal expansions and hence transient studies were not performed.

The UAM-SFR working group will have to define the grace time or the margin to melting available in the different identified accidental scenarios, have to apply the Best Estimate Plus

Uncertainty (BEPU) methodology and possibly recommend some changes to the design so that it meets some safety concerns.

The work is progressive and has the aim to resolve the remaining inconsistencies of the previous benchmark. Two SFR cores among the 4 being studied in the SFR-FT task force were selected. Those are the large oxide core proposed by CEA and the medium metallic core proposed by ANL [2]. A benchmark has been added for studying burnup on a fuel sub-assembly. This will help understanding the consequences of burnup on core characteristics on a simpler case.

There is the desire to apply the Best Estimate Plus Uncertainty (BEPU) methodology to some unprotected transients. At first two simple Unprotected Transients over Power (UTOP) and Loss of Flow (ULOF) are proposed because they allow useful insights without need for complicated modeling:

- no need to model the secondary loop,
- lower impact of the primary vessel.

Another benchmark on control rod withdrawal has been added recently and will challenge tools on a particularly difficult asymmetrical transient.

Experimental evidence in support to the studies has also been launched with:

- A selection of beta-effective measurements.
- The SEFOR Doppler experiment.
- One of the SUPER PHENIX start-up experiments.

2. Best Estimate Neutronic Results for the SFR 3600 MWth Core and the ABR 1000 MWth Core

The section focuses on the neutronic contributions of the different participants on the two core benchmarks: the SFR 3600MWth Core and the ABR 1000MWth Core, which are presented in more details in [3]. The main core characteristics of the large and medium SFR cores investigated are summarized in Table 1.

Table 1: Comparison of The Main Core Characteristics

SFR Cores	ABR 1000MWth Core	SFR 3600MWth Core
Thermal Power (MW) Type of fuel used	1,000 U-Pu-10Zr	3,600 $(U,Pu)O_2$
Cladding / Duct material	HT-9	ODS/EM10
Number of fuel assemblies in: - inner fuel - outer fuel	 78 102	 225 228
Number of control rods in: - primary system - secondary system	 15 4	 24 9
Inlet sodium temp. (°C) Outlet sodium temp. (°C) Avg. Fuel temperature (°C)	355 510 534	395 545 1,227
Height of fissile zone (cm) Lattice pitch (cm)	85.82 16.25	100.56 21.22
Fuel cycle duration (efpd[1])	328.5	410

[1] Equivalent Full Power Days

The results expected are for the End Of Cycle (EOC) parameters such as steady state reference reactivity/multiplication factor, feedback coefficients as “perturbation” from nominal operating conditions, materials thermal expansion configurations as well as Doppler effect, kinetics parameters (βeff, Λ, …).

2.1. SFR 3600 MWth oxide fuel core

The oxide core description is a large 3600 MWth core that exhibits power densities that result in low reactivity swing during the equilibrium burn cycle. Details of the core are given in a companion paper [3].

Nine participants from different organizations provided the results:

- ANL, Argonne, USA
- CEA Cadarache, France
- CEA Saclay, France
- CER, Budapest, Hungary
- GRS, Garching, Germany
- HZDR, Dresden, Germany
- IKE, Stutgart, Germany
- ININ, Edo. de México, Mexico
- IPPE, Obninsk, Russia

Most of the results are based on ENDF/B-VII cross sections library and were obtained using both deterministic and stochastic calculation methods. Results are shown in the following Table 2.

Table 2: Results of the 3600 MWth SFR core, oxide fuel benchmark

Institute		ANL	CEA Cadarache	CEA Saclay	CER	GRS	HZDR	IKE	ININ	IPPE
Library		ENDF/B-VII.1	ENDF/B-VII.1	JEFF 3.1.1	ENDF/B-VII.1	ENDF/B-VII.1	ENDF/B-VII.1	ENDF/B-VII.1	JEFF-3.1.1	ABBN-RF (ROSFOND)
Code		MC^2/ VARIANT	ERANOS	TRIPOLI4	SERPENT	KENO-IV	SERPENT	MCNP	SERPENT	TRIUM (MMKK)
K-effective		1.0162	1.0102	1.0185	1.0289	1.0164	1.0134	1.0075	1.0164	1.0087
βeff	[pcm]	351	372	361	348	344	361	353	360	361
Control rod worth (fully inserted)	[pcm]	-6360	-6511	-6135	-5556	-6218	-6315	-6439	-6111	-6206
Control rod worth (5cm from top)	[pcm]	-140	-139	-146	-126	-134	-133	-138	-127	-136
Doppler Constant	[pcm]	-857	-929	-875	-758	-848	-778	-800	-791	-787
Na Void Worth	[pcm]	1863	2005	1768	1726	1677	1821	1690	1851	1889
1% Sodium	[pcm/K]	0.420	0.448	0.466	0.446	0.523	0.500	0.366	0.828	0.480
1% Wrapper	[pcm/K]	0.023	0.022	0.025	0.019	0.021	0.017	0.019	0.027	0.027
1% Cladding	[pcm/K]	0.036	0.041	0.038	0.041	0.043	0.047	0.034	0.051	0.039
1% Fuel	[pcm/K]	-0.300	-0.310	-0.304	-0.292	-0.295	-0.306	-0.312	-0.310	-0.318
1% Fuel + Axial	[pcm/K]	-0.127	-0.133	-0.120	-0.144	-0.125	-0.139	-0.128	-0.127	-0.152
1% Grid	[pcm/K]	-0.745	-0.755	-0.758	-0.726	-0.757	-0.761	-0.822	-0.614	-0.811

(*) JEFF3.1 data for ν_d

(**) ROSFOND available at http://www.ippe.ru/podr/abbn/libr/rosfond.php

For the Doppler calculation, new cross sections need to be calculated at an increased temperature chosen to be twice the nominal one (in Kelvin).

For the fuel, sodium, cladding or wrapper calculations, new mediums and cross sections need to be created with the density of the corresponding material being multiplied by a 0.99 constant, simulating a decrease in the concentration of 1%.

Grid calculation:

- Homogeneous cell: the whole volume fractions change: except the sodium, each fractions is divided by a factor of (1.01)2, simulating an increase of the core pitch of 1%. The sodium fraction is calculated to replace the created void by the new fractions.
- Heterogeneous cell: the cell pitch is multiplied by the 1.01 factor.

The results displayed in this table show quite satisfactory results. A statistical analysis is conducted in [3] and most of the results are within 2-σ. A few outliers were identified for the control rod worth, cladding and wrapper coefficients, and under investigation. Although, the comparison as it stands looks quite satisfactory, there are further in-depth analyses required before being able to draw any definitive conclusions [4].

2.2. ABR 1000MWth metallic Core

The 1000 MWth Advanced Burner Reactor (ABR) metallic core is a compact core concept with a transuranics (TRU) conversion ratio of ~0.7 which was developed for a one-year cycle length with 90% capacity factor. Detailed description is presented in a companion paper [3].

Six participants (ANL, CEA of Cadarache, CEA of Saclay, GRS, ININ, and IPPE) provided their preliminary results to the medium-size metallic SFR core problem. Both deterministic and stochastic approaches are used with the ENDF/B-VII.1, JEFF-3.1.1 and ABBN-RF nuclear data libraries. The results are presented in the following Table 3.

Table 3: Results of the 1000 MWth SFR core, metallic fuel benchmark

Institute		ANL	CEA/Cad	CEA/Saclay	GRS	ININ	IPPE
Library		ENDF/B-VII.1	ENDF/B-VII.1	JEFF-3.1.1	ENDF/B-VII.1	JEFF-3.1.1	ABBN-RF (ROSFOND)
Code		MC^2-3/ VARIANT	ERANOS	TRIPOLI4.9®	KENO-IV	SERPENT	MMKK
K-effective		1.0171	1.0128	1.0299	1.0197	1.0284	1.0215
β_{eff}	[pcm]	332	352	342	324	342	343
Control Rod Worth (fully inserted)	[pcm]	-9905	-10029	-9540	-9796	-9640	-9542
Control Rod Worth (5cm from top)	[pcm]	-239	-230	-241	-232	-233	-241
Doppler constant	[pcm]	-383	-407	-394	-378	-384	-351
Na Void Worth	[pcm]	1327	1219	1579	1370	1247	1423
1% Sodium	[pcm/K]	0.383	0.340	0.405	0.261	0.565	0.393
1% Wrapper	[pcm/K]	0.021	0.022	0.022	0.023	0.032	0.023
1% Cladding	[pcm/K]	0.043	0.050	0.050	0.049	0.070	0.040
1% Fuel	[pcm/K]	-0.553	-0.568	-0.538	-0.567	-0.594	-0.570
1% Fuel + Axial	[pcm/K]	-0.257	-0.265	-0.260	-0.277	-0.307	-0.267
1% Grid	[pcm/K]	-1.137	-1.115	-1.074	-1.093	-1.097	-1.162

Good agreement is observed for most of the parameters as all the results are within 2-σ. Remaining variations on the sodium and cladding density coefficients require in-depth investigation.

2.3. Preliminary conclusions

As conclusions on the neutronic benchmark on the reactivity coefficients, these preliminary results are in relatively good agreement. Variations in k-effective, delayed neutron fractions and sodium void worth were explained by previous analyses [1]. Remaining discrepancies are still being investigated. Such variations in the results should not have a significant impact on the transient simulations [4].

3. Best Estimate Neutronic Results for the SFR Sub-Assembly Depletion Benchmark

A burnup benchmark has been defined in order to help understand the consequences of burnup on core characteristics on a simpler case. It focuses on depletion calculation using compositions of the 3600 MWth fresh sub-assembly of the SFR-FT benchmark.
Requested results are:

- K-eff + neutron flux
- Nuclide densities
- Branching ratios for capture reactions, (e.g. Am-241, Am-242g/m);
- Energy release
- Na void reactivity (100% to 1% density change)
- Doppler reactivity (1500 K to 750 K)
- Fission yield values

For completeness, k-inf and nuclide densities are analysed for a $(U,Pu)O_2$ fuel assembly (MOX 3600 core) at BOC (Beginning of Cycle) and EOC. The initial contributors are from:

- GRS – Codes: Serpent 2.1.26, MCNP 6; Data: ENDF/B-VII.0/VII.1, JEFF-3.1.2/3.2
- CEA/Cadarache – Code: ECCO/ERANOS 2.3; Data: JEFF-3.1

Results are shown in Table 4 for k-inf and Table 5 for nuclide densities. As can be seen from the results in Table 4, the choice of the nuclear data library has a significant influence on the initial k-inf with values ranging from 1.12331 to 1.13360 (1129 pcm). The reactivity swing is affected by this initial reactivity but other sources of data have an influence such as the energy release per fission and the branching ratio.

Table 4: Results on k-inf for fuel sub-assembly burn up benchmark

Burn-up step [days] (cum.)	0	102.5	205	307.5	410
Serpent 2.1.26 - ENDF/B-VII.0 (GRS)	1.12528	1.12788	1.12973	1.13169	1.13318
MNCP 6 - ENDF/B-VII.1 (GRS)	1.12748	1.12993	1.13142	1.13325	1.13439
MCNP 6 - JEFF-3.1.2 (GRS)	1.13360	1.13500	1.13636	1.13740	1.13823
MCNP 6 - JEFF-3.2 (GRS)	1.12515	1.12748	1.12891	1.13037	1.13122
MMKK - ABBN (IPPE)	1.12331	1.12642	1.12917	1.13144	1.13328
Serpent - ENDF B-VII.0 (HZDR)	1.12903	1.13169	1.13399	1.13593	1.13742
Serpent - JEFF-3.1 (HZDR)	1.13240	1.13423	1.13577	1.13707	1.13809
ECCO/ERANOS 2.3 - JEFF-3.1 (CEA)	1.13261	1.13502	1.13724	1.13908	1.14058

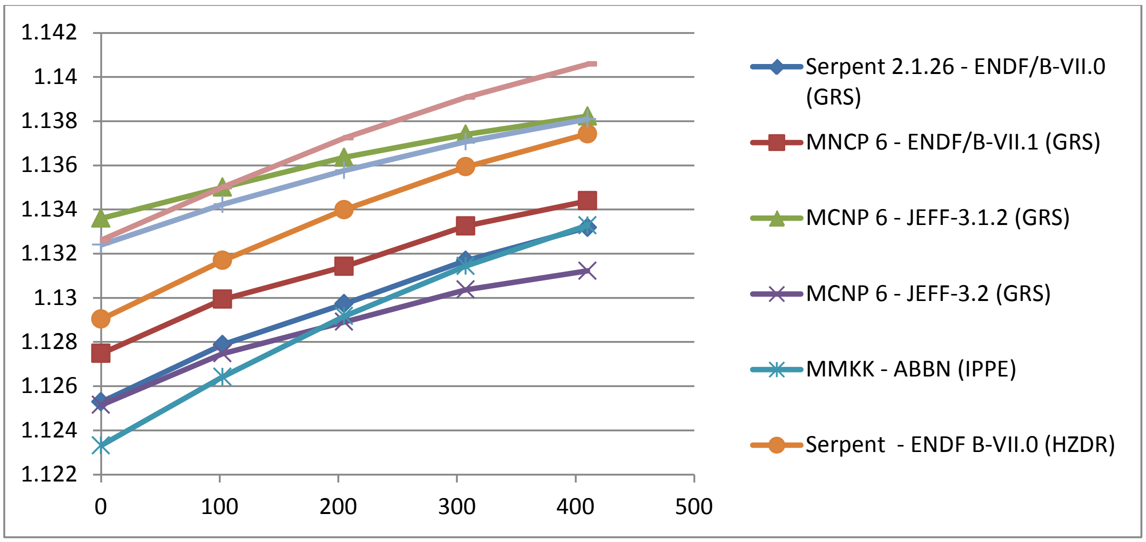

Figure 1: Results on k-inf for fuel sub-assembly burn up benchmark

Table 5: Rel. deviation of nuclide densities obtained by GRS with Serpent 2.1.26 – ENDF/B-VII.0 and by CEA with ECCO/ERANOS 2.3 - JEFF-3.1.

Burn-up step [days] (cum.)	0	102.5	205	307.5	410
U-234	0.0%	-0.1%	-0.2%	-0.3%	-0.4%
U-235	0.0%	-0.1%	-0.1%	-0.2%	-0.2%
Np-237	0.0%	0.5%	-0.5%	-1.2%	-1.8%
Pu-239	0.0%	-0.1%	-0.1%	-0.2%	-0.3%
Pu-241	0.0%	-0.3%	-0.6%	-0.9%	-1.2%
Am-242	-0.4%	-9.2%	-9.1%	-9.1%	-9.1%
Am-242m	-0.4%	8.8%	16.9%	23.9%	30.1%
Cm-242	0.0%	-4.2%	-6.4%	-7.8%	-8.5%
Cm-243	0.0%	6.5%	11.3%	14.8%	17.5%
Cm-244	0.0%	1.3%	2.3%	3.2%	4.0%
Cm-245	0.0%	6.3%	10.8%	14.2%	16.8%
Cm-246	0.0%	1.4%	3.9%	6.7%	9.2%

In summary, a simple fuel subassembly depletion benchmark has been formulated in order to perform an in-depth analysis on one axial section of a fuel assembly of the inner core region of the MOX-3600 core. Beside the analysis of integral quantities, the comparison of the branching fractions, the flux level, and the nuclear data might explain why the results deviate.

4. Identification of the Different Sources of Uncertainties

As feedback coefficients are the main neutronic inputs in the transient analysis, uncertainty estimation have to focus on these parameters. Uncertainties may come from different origins:

- Uncertainties from nuclear data knowledge (cross section, delayed neutron fraction, etc…)
- Uncertainties on isotopic number densities from manufacturing processes such as
  - geometrical tolerances for pellets, cladding and wrapper geometries
  - various material densities (porosity, etc..)
- thermal expansion correlation for sodium as coolant

Other items of interest to be investigated are:

- uncertainties coming from depletion effects,
- bias from core modeling assumptions.

Uncertainty levels can be computed using several methods such as sensitivity studies (from either deterministic methods or direct calculations) or probabilistic propagation. The work has started already with the major source of uncertainties among these, the nuclear data uncertainties.

At first, work was performed on the effective delayed neutron fraction (βeff) and the neutron generation lifetime (lambda) and is presented in companion papers [4,5].

The effective delayed neutron fraction (β_{eff}) is an important characteristic of nuclear reactors since it affects transients significantly. It is therefore important to characterise it correctly. The use of best-estimate codes and data together with an evaluation of the uncertainties is required not only for their use in safety studies but also to assess reactivity effects which are being measured relative to the effective delayed neutron fraction (β eff) in $.

The nuclear data uncertainty propagation has led to a 2.8% uncertainty for U-Pu core and 2.6% for enriched uranium cores (with JEFF-3.2) with main contributors being the delayed neutron fission yield and the fission cross section of U-238.

Also calculations were performed for reactivity, ρ_{Doppler} and $\rho_{\text{Na Void}}$. The uncertainties due to nuclear data are being calculated with different covariance matrices (COMAC, ENDF/B-VII.1, JENDL4.0). The different covariance matrices (COMAC, ENDF BVII.1, JENDL4.0) should in principle reflect the way the nuclear data evaluation has been made. This means that differential measurement uncertainties should have been propagated towards the evaluation itself. Since, there is a rationale in the choice of these differential measurements which depends on the evaluator; the final induced uncertainty might differ.

The nuclear model being used links together the different differential measurements and might add some more correlations (some exists already since differential measurements are conducted in a limited range of energy) in energy and between different cross section types.

Assimilation of integral experiments might have been done (JEZABEL is often used) which might add more correlations and a reduction of the variances.

These findings are similar for K_{eff} (Table 6), $\rho_{Doppler}$ (Table 7) and $\rho_{Na\ Void}$ (Table 8). However, for this last one, there are also the influence of Na-23 cross sections, and ENDF/B-VII Na-23 capture, elastic and inelastic uncertainties are much larger than for the other two.

TABLE 6: Keff uncertainties with different covariance matrices

Library	FISSION	CAPTURE	ELASTIC	INELASTIC	N,XN	NU	SUM
COMAC	0.00565	0.00252	0.00068	0.00403	0.00023	0.00156	0.00758
ENDF-BVII	0.00220	0.00418	0.00150	0.01137	0.00007	0.00175	0.01253
JENDL-4	0.00281	0.00451	0.00076	0.00601	0.00009	0.00156	0.00821
ABBN	0.00771	0.00520	0.00294	0.00266	-	0.00323	0.01062

TABLE 7: $\rho_{Doppler}$ uncertainties with different covariance matrices

Library	FISSION	CAPTURE	ELASTIC	INELASTIC	N,XN	NU	SUM
COMAC	1.47%	1.82%	1.23%	1.35%	0.05%	0.20%	2.97%
ENDF-BVII	0.46%	1.47%	2.07%	2.44%	0.02%	0.21%	3.56%
JENDL-4	0.56%	1.20%	1.76%	1.52%	0.03%	0.20%	2.68%
ABBN	2.20%	2.04%	3.08%	1.14%	-	0.52%	4.48%

TABLE 8: $\rho_{Na\ Void}$ uncertainties with different covariance matrices

Library	FISSION	CAPTURE	ELASTIC	INELASTIC	N,XN	NU	SUM
COMAC	3.16%	2.10%	1.39%	1.90%	0.03%	0.50%	4.50%
ENDF-BVII	1.20%	2.73%	2.70%	4.96%	0.01%	0.48%	6.41%
JENDL-4	1.37%	1.87%	1.39%	3.63%	0.01%	0.50%	4.55%
ABBN	4.88%	3.70%	7.47%	6.96%	-	0.88%	11.93%

ABBN covariance matrices were obtained with taking into account integral experiments which enable to refine nuclear cross sections.

The conclusions are the following:

- The keff uncertainty is predominantly due to the uncertainties in inelastic scattering of ^{238}U and in the fissions of ^{239}Pu and ^{238}U.
- The $\rho_{Doppler}$ uncertainty is predominantly due to the uncertainties in inelastic scattering of ^{238}Uand in the capture of ^{239}Puand in the elastic removal of ^{23}Na and ^{56}Fe.
- The $\rho_{Na\ void}$ uncertainty is predominantly due to the uncertainties in inelastic scattering of ^{238}U and ^{23}Na, in the capture of ^{238}U, in the fission of ^{239}Pu and in the elastic and inelastic removal of ^{23}Na.
- There are significant differences between covariance matrices from COMAC, ENDF/B-VII.1 and JENDL-4.0.
- The differences in the ^{238}U inelastic cross section are presumably due to the optical models being used, differential measurements being scarce.
- The differences in the ^{23}Na cross sections are due to the use of more recent differential measurements performed at IRMM (inelastic) and Oak Ridge (total).

Independently, the OECD/NEA conducted a work on "JEFF-3.3T1 Processed Covariances: Uncertainty Propagation Analysis and Comparison" with the goal to compare nuclear data uncertainties propagated from new COMAC V1 (JEFF3.2^{++}), ENDF/B-VII.1 and JENDL-4.0 covariance data. At first, the work shows the missing covariance data and those available while highlighting the most important differences and the underlying reasons.

Testing of ^{23}Na, ^{56}Fe, ^{235}U, ^{238}U & ^{239}Pu has been done as example of nuclides used in SFRs. The OECD WPEC SG33 representative benchmark cases have been mostly used. Uncertainty has been propagated from covariance files using standard linear 'sandwich' equation. This has been done with the NDaST tool public β-version http://www.oecd-nea.org/ndast/.

The conclusions of this study are the following:

- The covariance matrices for ^{235}U Intermediate & Fast uncertainties (5-9% on fission and up to 25% on capture) appear inconsistent and could be over-estimated:
- The covariance matrix for ^{239}Pu is in better shape, containing more complete information
  - – But intermediate values (2-10%) appear higher over a wider range significantly affecting some benchmarks (e.g. PMI)
  - – Also, the lower 'valley' (<0.5%) on the fast peak may also be questionable as this affects PMF type benchmarks
- Generally: ^{238}U and ^{56}Fe do not show any obvious flaws but more specific tests may be beneficial:
  - – Explore origins of negative correlations that differ between evaluations
  - – Need to look at different benchmarks and / or different integral parameters for testing ^{23}Na.

Given the large differences between covariance sets, there was a proposal to create an NEA Subgroup with the aim to improve/select/recommend covariances for Uncertainty Quantification in reactor physics domain. The establishment of the subgroup under NSC Working Party on Integral Nuclear Data Evaluation Co-operation (NSC/WPEC) is in progress at the moment.

5. Experimental evidence in support to calculations and their associated uncertainties

Again, given the large differences between covariance sets and even if the Subgroup of the NEA/WPEC action is successful, improving the reliability of the nuclear data covariance set will not go without using integral experiments of great confidence. It is the aim of the last component of the SFR-UAM task force.

In order to study in more details the relevance of OECD experimental benchmarks (committed in the ICSBEP and in the IRPhE experimental data bases) to the SFR-UAM cores, it is envisaged to provide sensitivities to the NEA Data Bank to be able to calculate representability factors as well as some means to possibly reduce the final SFR-UAM core characteristics uncertainty.

5.1 β_{eff} experimental validation

This task is for assessing the calculation of β_{eff}. There are a number of available experiments in the ICSBEP and in the IRPhE experimental databases among which are JEZEBEL, SNEAK7A and SNEAK 7B. To these experiments, one can add the BERENICE experiments performed in MASURCA. An upgrade of the modelling was however necessary for the BERENICE experiments. The use of the Iterated Fission Probability method in the Monte Carlo code TRIPOLI4® confirms results obtained with deterministic codes such as ERANOS for calculating β_{eff}. The asset of TRIPOLI4® lies in the possibility to get a better representation of experimental cores. Its use for evaluating the calculated components of the β_{eff} of the BERENICE experimental programme of the MASURCA facility [5] has led to significant improvements of the C/E ratios, especially the R2 experimental core. However, there are still discrepancies between different types of measurements such as ^{252}Cf source or noise ones. Significantly is the fact that the comparison is quite excellent with the noise technique results which is more reliable. However, the quoted uncertainties do not allow a reduction of predicted uncertainty (2.8%). In order to get a reduced uncertainty it is hence recommended to measure β_{eff} within the future GENESIS experimental programme in the refurbished zero power reactor (ZPR) MASURCA with an improved noise measurement technique.

Based on the interpretation calculations of α_{Rossi} and β_{eff} measurements, a series of C/E comparisons is being done with modern tools such as MCNP IFP method, TRIPOLI4 IFP method, SUSD3D, SERPENT IFP method and the latest evaluated nuclear data ENDF/B-VII.1, JEFF3.2, JENDL4.4. The importance of a neutron is needed to calculate β_{eff}, the Iterated Fission Probability method (IFP) [6] is the most accurate method to obtain it with Monte Carlo and has been implemented in various codes quite recently. Uncertainty assessments due to nuclear data (including those for delayed neutron constant values) have been done using the SUSD3D and ERANOS tools. Uncertainties on delayed neutron constant values are only available in the JENDL4.0 library.

The calculations of uncertainties were carried out by JSI, CEA and GRS for a series of experimental benchmarks: SNEAK 7A, SNEAK 7B, JEZEBEL, POPSY, BERENICE ZONA2, and the SFR 3600MWth. These calculations of uncertainties have been done with various sets of covariance matrices including JENDL4.0 on which one can compare the calculations done at JSI, CEA and at GRS [9,10].

Since uncertainties for delayed neutron constant values are available only in the JENDL4.0 library, a series of actions (differential measurements, models) are studied at CEA, ILL and Subatech-Nantes in order to provide in the future, new recommended values.

On the basis of the computation-experimental evaluation and experimental uncertainties due to the nuclear data, it is already necessary to consider new experiments of kinetics constants with significantly reduced uncertainties in order to improve the prediction of the tools. The test of the new techniques could be done within the JSI TRIGA reactor.

5.2 Doppler measurements

Doppler coefficient is an important dynamic characteristic of the core. A review of relevant experiments in the IRPhE database has identified a lack of experiments on Doppler. The SEFOR reactor has been built for the purpose of measuring the Doppler coefficient [7]. SEFOR documentation is not in the IRPhE standard but has been used in the past [8] and is worth being investigated. Uncertainties on Doppler coefficient lie in the 100eV-1keV energy domain and are mainly due to the knowledge of the flux level at the bottom edge of the fast reactor flux.

The SEFOR static tests were performed at power levels up to 20 MW while maintaining the average core coolant temperature constant at 678K. The reactivity effects due to power changes were measured by the reflector positions, adjusted to compensate the reactivity feedback. The Doppler coefficients were then evaluated by subtracting the contributions from the fuel axial expansion. Since SEFOR was particularly designed, using segmented fuel rods and dished fuel pellets, the reactivity change due to the axial expansion is as small as 5% of the total feedback and its uncertainty little affects the Doppler reactivity evaluation.

Hence, the SEFOR static tests are recommended as an experimental evidence of the validity of Doppler calculations.

Careful attention should be given to the different sources of experimental uncertainties (fuel thermal conductivity, temperature increase, etc…). Uncertainties on Doppler coefficient calculations lie in the 100eV-1keV energy domain and are mainly due to the nuclear data uncertainties. Calc. & Exp. uncertainties should be compared together with C/E values to give an estimation of recommended uncertainty.

5.3 Super-Phénix start up measurements

A proposal for a benchmark based on the selected Super-Phénix (SPX) start up test has been done and aims at supporting transient calculations performed within the SFR-UAM working group i.e. ULOF (Unprotected Loss Of Flow), UTOP (Unprotected Transient Over Power) and CRW (Control Rod Withdrawal).

SPX is a SFR reactor which operated within the 1986 – 1996 period. The design power was set to 3000 MWth/1200 MWe with an inlet/outlet coolant temperature of 395ºC / 545ºC and a coolant flowrate 16.4 t/s. The sub-assembly uses MOX fuel and SS cladding. The available data on the start-up tests is documented in [9].

The proposal is to select one test for the benchmark: the 3-step negative reactivity insertion.

The power is initially at 51% nominal power (1540 MWth) with a 63% nominal flowrate (10.4 t/s). The perturbations are achieved through inlet coolant temperature reduction and control rod insertion in three steps (–25 pcm $\times$ 3). The proposal is set up with Excel template based on the input requirements of the TRACE code used at PSI. Reactivity coefficients are provided and the aim of the work is to verify the ability of participants to reproduce experimental results. The UAM-SFR Benchmark will take advantage of these data as an experimental evidence to support its activities.

6. Conclusions

The OECD/NEA sub-group on Uncertainty Analysis in Modelling (UAM) for Design, Operation and Safety Analysis of Sodium-cooled Fast Reactors (SFR-UAM) has started its work under the NSC/WPRS/EGUAM two years ago and has been meeting every year.

The participants to the sub-group have been launching a series of benchmarks to support current understanding of important phenomena to define and quantify the main core characteristics affecting safety and performance of SFRs. Different codes and data have been used to support the evaluation of the uncertainties which challenges existing calculation methods.

Two SFR cores have been selected for the SFR-UAM benchmark, a 3600 MWth oxide core and a 1000 MWth metallic core. Their neutronic feedback coefficients are being calculated for transient analyses. The SFR-UAM sub-group is currently defining the grace period or the margin to melting available in the different accident scenarios and this within uncertainty margins.

Recently, the work of the sub-group has been updated to incorporate new exercises, namely, the depletion benchmark, the control rod withdrawal benchmark, and the Super-Phénix start-up transient. Experimental evidence in support of the studies is also being developed.

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