Evaluation of the OECD/NEA/SFR-UAM Neutronics Reactivity Feedback and Uncertainty Benchmarks

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Abstract. One of the tasks of the OECD/NEA sub-group on Uncertainty Analysis in Modeling (UAM) of Sodium-cooled Fast Reactors (SFR-UAM) under the NSC/WPRS/EGUAM is to perform a code-to-code comparison on neutronic feedback coefficients and associated uncertainties calculated for transient analyses. This benchmark exercise benefits from the results of a previous Sodium-cooled Fast Reactor core Feedback and Transient response (SFR-FT) Task Force work under the NSC/WPRS/EGRPANS. Two SFR cores have been selected for the SFR-UAM benchmark, the 3600MWth oxide and the 1000MWth metallic SFR cores.

Results from six and nine participating international institutes were received for respectively, the metallic and oxide SFR cores, using a wide range of calculation methodologies. The preliminary results display good agreement in the reactivity coefficients estimated, with remaining discrepancies explained by different nuclear data libraries, modeling approximations for deterministic solutions, and statistical convergence for stochastic evaluations on small perturbations. Nuclear data uncertainty evaluations for the reactivity coefficients from two institutions are compared and show consistent results.

Key Words: OECD Benchmark, SFR, Metallic fuel, Oxide fuel, feedback coefficient, uncertainty.

1. Introduction

Within the activities of the Working Party on Scientific Issues of Reactor Systems (WPRS) of the Nuclear Energy Agency (NEA) of the OECD, the Sodium-cooled Fast Reactor Task Force (SFR-FT) conducted a series of benchmarks to evaluate core performance characteristics and reactivity feedback coefficients of the large and medium SFR core concepts. This work is summarized in the final report [1] and confirmed the ability of participants and their neutronic codes to provide generally consistent results when analyzing SFR cores. The initial objectives of the SFR-FT benchmark also included calculating the feedback coefficients and simulating unprotected transients, but those could not be attained within the initial framework. Hence a follow-on benchmark exercise hosted within the Uncertainty Analysis in Modeling (UAM) of Sodium-cooled Fast Reactors (SFR-UAM) sub-group under the NSC/WPRS/ EGUAM, was proposed to conduct a code-to-code comparison on neutronic feedback coefficients and associated uncertainties calculated for transient analyses [2]. Recently, the work of the sub-group has been updated to incorporate new exercises, namely, a depletion benchmark, a control rod withdrawal benchmark, and a SUPER-PHENIX start-up transient. The current status of the various activities conducted within the UAM-SFR benchmark is described in Reference [3].

The objective of the present paper is to describe the current status of the code-to-code evaluation of the neutronic feedback effects on two SFR cores. The neutronic feedback coefficients and their uncertainties at the end of cycle are conducted using a common calculation methodology summarized in Section 2 of this document. Based on the results obtained in the previous step, transient calculations will be performed on a few selected cases for the principal unprotected transients such as the unprotected transient overpower (UTOP) and loss of flow (ULOF) to evaluate the grace period or the margin to melting available within uncertainty margins. Results from four and seven participating international institutes were received for respectively, the metallic and oxide SFR cores, using a wide range of calculation methodologies compared in Section 3. The results obtained by the various organizations are summarized and differences are analyzed in Section 4.

2. Benchmark Description

Two SFR cores among the 4 being studied in the SFR-FT group were selected for the UAM-SFR benchmark and are described in [2]. Those are the large oxide core proposed by CEA and the medium metallic core proposed by ANL. Their main design characteristics are summarized in the following section together with the calculation methodology and the results expected.

2.1. SFR Cores Description

The main core characteristics of the large and medium SFR cores investigated are summarized in Table 1.

For simplification purposes, a representative isotope, Molybdenum, replaced all fission products as both cores are modelled in End of equilibrium Cycle configuration (EOC) state. Each fuel sub-assembly is divided into five axial zones and the core is divided into inner, and outer cores with different fuel compositions provided in [2].

The oxide-fueled SFR core proposed by CEA is a large 3600 MWth core that exhibits a low reactivity swing during the equilibrium burn cycle [4]. It uses Oxide Strengthened Steel (ODS) cladding with helium bond and is based on the “fat pin with small wire” concept that enables to reach self-breeding without the use of fertile blanket. The performance of the ODS cladding allows an average burnup around 100 GWd/t_{HM} for a corresponding cycle length of 410 equivalent full power days with one fifth reloading scheme.

The oxide core layout is presented in Figure 1. It consists of 453 fuel, 330 radial reflector and 33 control subassemblies. The core is divided into inner and outer core zones, which are composed of 225 and 228 fuel assemblies, respectively. Two independent safety-grade reactivity control sub-systems are used. The primary control system consists of 6 control subassemblies in the inner core and 18 control subassemblies at the interface between the inner and the outer zones. The secondary system contains 9 control subassemblies located in the 7th row.

TABLE I: COMPARISON OF THE MAIN CORE CHARACTERISTICS

SFR Cores	Medium Metallic Core	Large Oxide 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

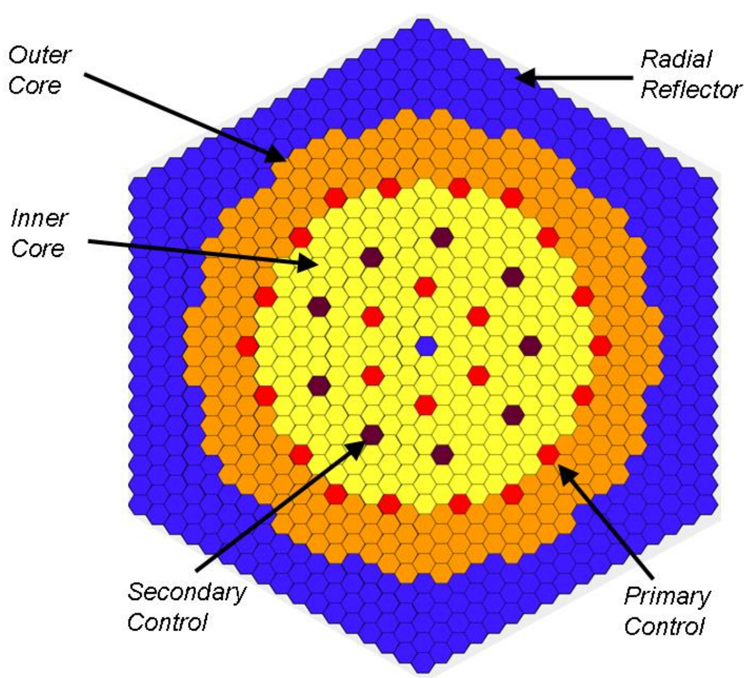

FIG. 1: Radial Core Layout of the Large Oxide SFR Core

The metallic version of the Advanced Burner Reactor (ABR) SFR core proposed by ANL is a medium size 1000 MWth core. The ABR core concepts were developed for the study of fast reactor design options under the Global Nuclear Energy Partnership (GNEP) program [5]. Compact core concepts with a transuranics (TRU) conversion ratio of ~0.7 were developed for a one-year cycle length with 90% capacity factor. Conventional or reasonably proven materials were utilized in the ABR core concepts so that the core stays within current fast reactor technology knowledge base. The detailed descriptions of the cores are available in Reference [2].

Figure 2 shows the radial core layout of the 1000 MWth ABR metallic benchmark core. The core consists of 180 drivers, 114 radial reflectors, 66 radial shields, and 19 control subassemblies. The core is divided into inner and outer core zones, which are composed of 78 and 102 driver assemblies, respectively. Two independent safety-grade reactivity control sub-systems are used. The primary control system consists of three control subassemblies in the fourth row and 12 control subassemblies in the seventh row. The secondary system contains four control subassemblies located at the core center and in the fourth row.

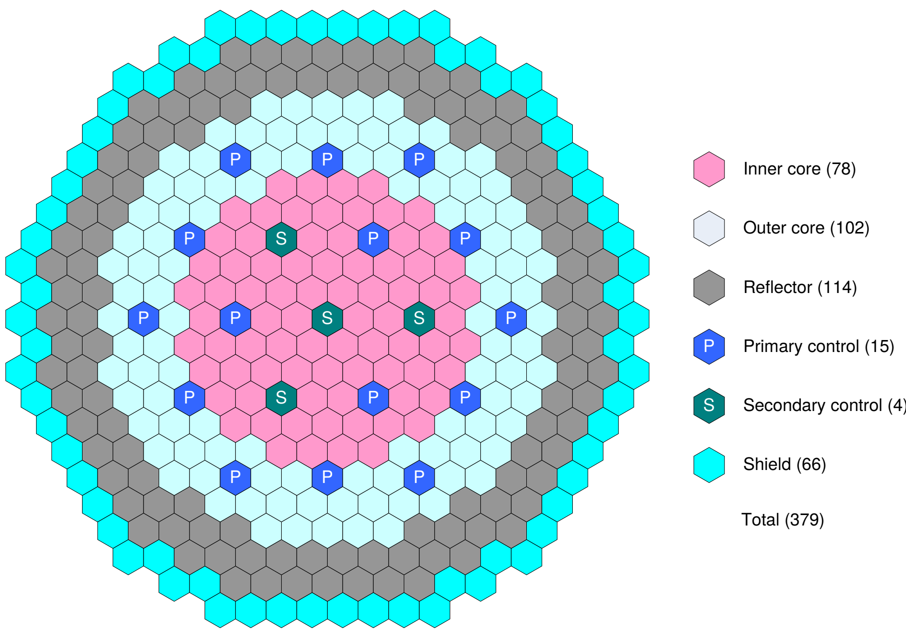

FIG. 2: Radial Core Layout of the Medium Metallic SFR Core

2.2. Calculations Requested

The UAM-SFR benchmark targets the code-to-code comparison of end-of-cycle neutronic parameters. The k-effective and kinetics parameters (β_{eff}, Λ, …) are evaluated together with the Doppler and sodium void worth coefficients. For the Doppler constant, the temperatures (in Kelvin) of the elements in the fuel (U, TRU, O, Zr, Mo) are multiplied by two in the active core region. The sodium void worth is computed by voiding 100% of the sodium in all the driver fuel assemblies. The control rod worths are evaluated by inserting all the rods 5 cm to the top of the core or by fully inserting them in the core.

The main addition to this benchmark (compared with the previous SFR-FT benchmark) consists in the thermal expansion feedback evaluation as those are paramount for transient simulations. For simplicity purposes, the density coefficients are directly compared and all the thermal expansion coefficients can be calculated using the formula detailed in [2]. The fuel, cladding, wrapper and sodium density coefficients are obtained by reducing the density of the materials by 1%. The core axial and radial thermal expansion coefficients are also compared in this study. The axial expansion is obtained by reducing the fuel density by 1% and increasing the length of the driver fuel column by 1%, while moving the control and safety rods so that they remain at the top of the fuel column. The radial thermal expansion coefficient is obtained by increasing the pitch of the assembly by 1% while conserving the masses of fuel and structures and increasing the mass of sodium (sodium is added in the volume gained).

As feedback coefficients are the main neutronic inputs in the transient analysis, uncertainty evaluations are conducted for these parameters. Uncertainties may come from nuclear data knowledge (cross section, delayed neutron fraction, etc…) or can be associated with uncertainties in the isotopic number densities (from manufacturing processes or material properties knowledge).

3. Calculation Methodologies

Nine participating institutions provided preliminary results for at least one of the cores investigated and additional participants might contribute in the future. All of the participants provided results for the large oxide core evaluation and six of them (ANL, CEA of Cadarache, CEA of Saclay, GRS, ININ, and IPPE) provided results for the medium-size metallic SFR core problem. Following is the list of participants that provided preliminary results to the UAM-SFR neutronic feedback coefficients evaluation:

- 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

The participants used a wide variety of calculation methods as summarized in Table 2. Most of the participants used the ENDF/B-VII.1 nuclear data library [6], while CEA/Saclay and ININ provided results with the JEFF3.1.1 library [7] and IPPE used the ABBN-RF library [8]. Among the nine methodologies employed, six used a stochastic (Monte Carlo) approach and three used a deterministic approach. The stochastic solutions model the core explicitly (with detailed description of the sub-assembly) while deterministic codes employ a homogenized model for the flux calculation. However, the deterministic solutions developed at ANL and CEA (Cadarache) use a heterogeneous treatment for the cross-section in the driver fuel and control rods as this was observed to be critical in the previous SFR-FT benchmark [1].

TABLE 2: SUMMARY OF THE CALCULATION METHODOLOGIES EMPLOYED BY THE PARTICIPANTS

Institute	ANL	CEA Cadarache	CEA Saclay	CER	GRS	HZDR	IKE	ININ	IPPE
Method employed	Det.	Det.	Stoch.	Stoch.	Stoch	Stoch.	Stoch.	Stoch.	Det./ Stoch.
Nuclear data library	ENDF/B7.1	ENDF/B7.1	JEFF 3.1.1	ENDF/B7.1	ENDF/B7.1	ENDF/B7.1	ENDF/B7.1	JEFF 3.1.1	ABBN-RF (ROSFOND)
Cross-section processing code	MC^2-3	ERANOS	n/a	n/a	n/a	n/a	n/a	n/a	TRIUM
Geometrical assumption for cell calculations	1D- Het. [2]	2D- Het. [2]	n/a	n/a	n/a	n/a	n/a	n/a	Hom.
Number of broad energy group	33	33	n/a	n/a	n/a	n/a	n/a	n/a	28/299
Core calculation code	DIF3D/ VARIANT	VARIANT	TRIPOLI4	SERPENT	KENO-VI (CE)	SERPENT	MCNP	SERPENT	TRIUM
Geometrical assumption for core calculation [3]	Hom.	Hom.	Het.	Het.	Het.	Het.	Het.	Het.	Hom./ Het.
Diffusion or Transport ?	Transport	Transport	Transport	Transport	Transport	Transport	Transport	Transport	Diffusion/ Transport
Perturbation method	GPT[1]	GPT[1]	Direct	Direct	Direct	Direct	Direct	Direct	GPT/ Direct
Perturbation code name	PERSENT	ERANOS							TRIUM
Covariance matrix	COMMARA2.0	COMAC	-	-	-	-	-	-	ABBN
References	6,11,12,13,17	6,9,20	7,10	6,14	6, 16	6,14	6,15	7,14	8,18,19

[1] Generalized Perturbation Theory

[2] Cross-sections in driver fuel and control regions are calculated using 1D or 2D cell geometries

[3] Core calculation are performed with 3D homogenized compositions (for deterministic calculations) or with explicit geometry (for stochastic calculations)

TABLE 3: NEUTRONIC RESULTS FOR THE LARGE OXIDE CORE

Institute		ANL	CEA Cad	CEA Saclay	CER	GRS	HZDR	IKE	ININ	IPPE	*Average*[1]	*σ*[1]
K-effective		1.0162	1.0102	1.0185	1.0289	1.0164	1.0134	1.0075	1.0164	1.0087	*1.0130*	*0.0041*
β_{eff}	[pcm]	351	372	361	348	344	361	353	360	361	*357*	*9*
Control Rod Worth (fully inserted)	[pcm]	-6360	-6511	-6135	-5556	-6218	-6315	-6439	-6111	-6206	*-6287*	*144*
Control Rod Worth (5cm from top)	[pcm]	-140	-139	-146	-126	-134	-133	-138	-127	-136	*-137*	*6*
Doppler constant	[pcm]	-857	-929	-875	-758	-848	-778	-800	-791	-787	*-833*	*53*
Na Void Worth	[pcm]	1863	2005	1768	1726	1677	1821	1690	1851	1889	*1810*	*106*
1% Sodium	[pcm/K]	0.420	0.448	0.466	0.446	0.523	0.500	0.366	0.828	0.480	*0.469*	*0.035*
1% Wrapper	[pcm/K]	0.023	0.022	0.025	0.019	0.021	0.017	0.019	0.027	0.027	*0.022*	*0.004*
1% Cladding	[pcm/K]	0.036	0.041	0.038	0.041	0.043	0.047	0.034	0.051	0.039	*0.039*	*0.003*
1% Fuel	[pcm/K]	-0.300	-0.310	-0.304	-0.292	-0.295	-0.306	-0.312	-0.310	-0.318	*-0.305*	*0.008*
1% Fuel + Axial	[pcm/K]	-0.127	-0.133	-0.120	-0.144	-0.125	-0.139	-0.128	-0.127	-0.152	*-0.133*	*0.010*
1% Grid	[pcm/K]	-0.745	-0.755	-0.758	-0.726	-0.757	-0.761	-0.822	-0.614	-0.811	*-0.767*	*0.033*

[1] The average and standard deviation were calculated by removing the outliers highlighted in this table.

TABLE 4: NEUTRONIC RESULTS FOR THE MEDIUM METALLIC CORE

Institute		ANL	CEA/Cad	CEA/Saclay	GRS	ININ	IPPE	*Average*[1]	*σ*[1]
K-effective		1.0171	1.0128	1.0299	1.0197	1.0284	1.0215	*1.0216*	*0.0066*
β_{eff}	[pcm]	332	352	342	324	342	343	*339*	*10*
Control Rod Worth (fully inserted)	[pcm]	-9905	-10029	-9540	-9796	-9640	-9542	*-9742*	*202*
Control Rod Worth (5cm from top)	[pcm]	-239	-230	-241	-232	-233	-241	*-236*	*5*
Doppler constant	[pcm]	-383	-407	-394	-378	-384	-351	*-383*	*19*
Na Void Worth	[pcm]	1327	1219	1579	1370	1247	1423	*1361*	*131*
1% Sodium	[pcm/K]	0.383	0.340	0.405	0.261	0.565	0.393	*0.380*	*0.028*
1% Wrapper	[pcm/K]	0.021	0.022	0.022	0.023	0.032	0.023	*0.022*	*0.001*
1% Cladding	[pcm/K]	0.043	0.050	0.050	0.049	0.070	0.040	*0.046*	*0.004*
1% Fuel	[pcm/K]	-0.553	-0.568	-0.538	-0.567	-0.594	-0.570	*-0.565*	*0.019*
1% Fuel + Axial	[pcm/K]	-0.257	-0.265	-0.260	-0.277	-0.307	-0.267	*-0.272*	*0.018*
1% Grid	[pcm/K]	-1.137	-1.115	-1.074	-1.093	-1.097	-1.162	*-1.113*	*0.032*

[1] The average and standard deviation were calculated by removing the outliers highlighted in this table.

The statistical convergence is an important parameter of stochastic calculations that might affect the results obtained for reactivity coefficients for small perturbations. Direct perturbations are being performed at CER with SERPENT with a convergence on k-eff of 13 pcm, which leads to relatively large uncertainty on small reactivity effects, which is why larger perturbations of 5% or 10% were observed. For CEA (Saclay) calculations, the standard deviation on k-effective is lower than 10 pcm and the uncertainty of the associated perturbation calculations for the feedback coefficients ranged from 1% to 5%. The results provided by GRS are converged to a 1-sigma standard deviation of 5 pcm.

IPPE performs “Best Estimate” Monte Carlo simulations using its TRIUM (MMKK) solver for sodium void and control rod worth calculations and deterministic calculations based on diffusion approximation with TRIUM (TRIGEX) solver for 1% perturbation calculations.

4. Preliminary Results

The preliminary results for the neutronic feedback coefficients and their associated nuclear data uncertainties provided by the participating institutes are summarized in this section.

4.1.Preliminary Comparison of Feedback Coefficients Evaluations

The neutronic results on the feedback coefficients are summarized in Table 3 for the large oxide core and in Table 4 for the medium metallic core. As complimentary information, the averaged value and the standard deviation (σ) of the results are displayed on these tables. It should be noted that these statistical parameters were calculated on all the values except on the outliers, those are highlighted in these tables when outside the “Average$\pm$2σ” range. Outliers are only observed on both cores and these larger than expected discrepancies are still under investigations.

The standard deviations in the results obtained are relatively large for the k-effective (413 pcm and 657 pcm for the oxide and metallic cores), the sodium void worth (10% for the medium core), wrapper (16% for large core), and the cladding (10% for the medium core). Otherwise the standard deviations are less than 10%. It should be specifically noted that the results obtained by CER, HZDR, and IKE for the large oxide core display relatively large discrepancies for some parameters (k-effective, control rod worth, delayed neutron fraction) considering they employ the same nuclear data libraries with a stochastic code (SERPENT or MCNP). Those discrepancies should be further investigated in the next steps of this benchmark.

Most of the remaining variations in core multiplication factor, delayed neutron fraction and sodium void worth were already observed and explained in the WPRS/SFR-FT benchmark [1] and are due to differences in nuclear data libraries, delayed neutron fractions, and stochastic/deterministic approach. For instance, 1000 pcm and 500 pcm discrepancies between ENDF/B-VII and JEFF-3.1 were previously observed for the medium metallic and large oxide SFR cores [1]. About 500 pcm underestimation of the k-effective with deterministic codes with regards to stochastic solutions was also observed in [1]. The differences observed between ANL and CEA (Cadarache) mostly on the k-effective were also previously investigated [1] and explained by the difference in cross-section condensation for the reflector regions. A main cause of the observed discrepancy on the delayed neutron fraction comes from different isotopic values of the delayed neutron yields ν_D that were used by the participants.

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

4.2.Preliminary Uncertainty Results

ANL, CEA of Cadarache, and IPPE computed the uncertainties associated with the neutronic feedback parameters above presented and the results are shown in Table 5. These evaluations are based on different nuclear data libraries and different covariance matrices: ANL used the COMMARA-2.0 [17] covariance matrix based on ENDF/B-VII.1, and IPPE used the ABBN covariance matrix [18], and CEA used COMAC covariance matrix [20].

The uncertainty analysis performed by IPPE employs two different methodologies. The first one noted as IPPE-GRS uses the GRS sampled method which implies the TRIUM code [19] and temporary prepared ABBN group data sampled by using the ABBN covariance matrices (a special included code treats sampled results). The second one noted as IPPE-S/U uses the INDECS code system, which includes LEMEX library of descriptions of experimental results, LSENS library of sensitivity coefficients, and LUND28 (or LUND30) uncertainty covariance matrices of the ABBN grouped nuclear data. The advantage of the GRS method is that it allows obtaining the uncertainty estimations for all important neutronics characteristics in one cycle of calculations. But advantage of the first (S/U) method is that it gives a possibility to investigate all the important sources of the total calculation uncertainty. The method used by ANL for uncertainty calculation is fully detailed in Reference [21].

TABLE 5: UNCERTAINTY RESULTS ON FEEDBACK COEFFICIENTS FROM NUCLEAR DATA

Results in [%]	Medium Metallic SFR			Large Oxide SFR		
	ANL	IPPE-GRS	IPPE-S/U	CEA	IPPE-GRS	IPPE-S/U
k-effective	1.3	1.5	1.5	0.8	1.5	1.6
βeff	1.1	1.1	-	3.5	1.2	-
Control Rod Worth	2.8	2.7	2.5	-	2.8	3.6
$K_{Doppler}$	5.8	7.5	6.2	3	5.9	5.3
Na Void Worth	19.2	26.3	28.2	4.5	14.5	13
1% Sodium	13.8	-	18.1	-	-	11.2
1% Structure	7.4	-	7.6	-	-	6.7
1% Fuel + Axial	2.4	-	2.7	-	-	3.6
1% Grid	1.9	2.8	2.7	-	4	3.8

The results obtained by ANL and IPPE display consistent levels of uncertainties for the medium metallic core. Relatively small uncertainties due to nuclear data are observed for the control rod worth, Doppler coefficient, and for the axial and radial coefficients. However, larger uncertainty is observed for the structure density coefficients. For the delayed neutron fraction, larger uncertainties are obtained by CEA since their results include the contribution from delayed neutron constants, which are found to be dominant [22]. Other participants omitted the uncertainties from the delayed neutron constants uncertainties. It should be noted that especially large difference in the level of uncertainty is observed on the sodium density and sodium void worth parameters. This comes from differences in the covariance matrices of sodium, which needs to be understood. A dedicated OECD/NEA working group is being launched to tackle this issue. Future work should focus on analyzing the reasons of such discrepancies, and of propagating those through accidental transients.

5. Summary and Conclusions

Two SFR cores are being analyzed in the framework of the SFR-UAM benchmark to evaluate their neutronic feedback coefficients and the associated uncertainties. Results from six and nine participating international institutes were received for respectively, the metallic and oxide SFR cores, using a wide range of calculation methodologies.

The preliminary results display good agreement in the reactivity coefficients estimated, with most discrepancies explained by different nuclear data libraries, modeling approximations for deterministic solutions, and statistical convergence for stochastic evaluations on small perturbations. Additional evaluations will be required to explain some remaining inconsistencies between participants. Nuclear data uncertainty evaluations for the reactivity coefficients obtained by IPPE, CEA and ANL are compared and show generally consistent results. Future work will focus on the transient simulations and uncertainty propagation.

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