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

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G. Rimpault, Laurent Buiron, Ne Stauff, Tk Kim, Ta. Taiwo, et al.. Current Status and Perspectives of the OECD/NEA sub-group on Uncertainty Analysis in Modelling (UAM) for Design, Operation and Safety Analysis of SFRs (SFR-UAM). ANS Best Estimate Plus Uncertainty International Conference (BEPU 2018), May 2018, Lucca, Italy. cea-02339103

HAL Id: cea-02339103
https://cea.hal.science/cea-02339103

Submitted on 13 Dec 2019

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

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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 to check the use of best-estimate codes and data. This work comes from the desire to design reactors with improved safety performance while preserving a sustainable source of energy at a rather low cost.

Two SFR cores are being studied: a large 3600MWth oxide core and a medium 1000MWth metallic core. In order to assess tools being used for studying these cores, various sub-exercises have been set up for what concerns neutronics with cell, sub-assembly, super-cell and core benchmarks under steady state conditions either at BOL conditions or at EOEC. A sub-assembly depletion benchmark is being set up before going into full core calculations with depletion.

Since the objective is to define the grace period or the margin to melting available in the different accident scenarios and this within uncertainty margins, uncertainties of different origins (methods, neutronics, thermal-hydraulic, fuel behavior) once identified and evaluated will be propagated through.

In order to ensure validity to these exercises, the sub-group incorporates some experimental validations on neutronics, thermal hydraulics, fuels and systems. This will be done with experiments from IRPhE & ICSBEP, SEFOR, THORS and the SUPER-PHENIX start-up transient programme.

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 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 to check the use of best-estimate codes and data.

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 [1] which demonstrated that for the benchmark cores under study the major source of bias between participants is coming from nuclear data.

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 so as to avoid possible compensating errors. Two SFR cores are being studied: a large 3600MWth oxide core and a medium 1000MWth metallic core [2]. In order to assess tools being used for studying these cores, various sub-exercises have been set up for what concerns neutronics with cell, sub-assembly, super-cell and core benchmarks under steady state conditions either at BOL conditions or at EOEC depending on the benchmark. A sub-assembly depletion benchmark is being set up before going into full core calculations with depletion.

Since the objective is to define the grace period or the margin to melting available in the different accident scenarios and this within uncertainty margins, uncertainties of different origins (methods, neutronics, thermal-hydraulic, fuel behavior) once identified and evaluated will be propagated through. At first two simple Unprotected Transients over Power (UTOP) and Loss of Flow (ULOF) are proposed because they allow useful insights without need to model the secondary loop and the primary vessel (negligible impact).

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

In order to ensure validity to these exercises, the sub-group incorporates some experimental validations on neutronics, thermal hydraulics, fuels and systems. This will be done with neutronic experiments from IRPhE & ICSBEP, SEFOR, thermal hydraulic experiments from THORS and system experiments with the SUPER-PHENIX start-up transient programme.

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, …).

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 [3].

Currently 9 participants provided their results for the large-size oxide SFR core problem. Among those, only 6 provided their results for the medium-size metallic SFR core. 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^2-3/ VARIANT	ERANOS	TRIPOLI4	SERPENT	KENO-IV	SERPENT	MCNP	SERPENT	TRIUM (MMKK)
K-effective		1.0162	1.0102	1.0185	1.0289	1.0194	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	-6243	-6315	-6439	-6111	-6206
Control rod worth (5cm from top)	[pcm]	-140	-139	-146	-126	-140	-133	-138	-127	-136
Doppler Constant	[pcm]	-857	-929	-875	-758	-886	-778	-800	-791	-787
Na Void Worth	[pcm]	1863	2005	1768	1726	1650	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	0.018	0.017	0.019	0.027	0.027
1% Cladding	[pcm/K]	0.036	0.041	0.038	0.041	0.044	0.047	0.034	0.051	0.039
1% Fuel	[pcm/K]	-0.300	-0.310	-0.304	-0.292	-0.298	-0.306	-0.312	-0.310	-0.318
1% Fuel + Axial	[pcm/K]	-0.127	-0.133	-0.120	-0.144	-0.119	-0.139	-0.128	-0.127	-0.152
1% Grid	[pcm/K]	-0.745	-0.755	-0.758	-0.726	-0.744	-0.761	-0.822	-0.614	-0.811

The results displayed in this table 2 show quite satisfactory agreement. A statistical analysis is conducted in [3] and most of the results are within 2-σ.

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 [3]. Both deterministic and stochastic approaches are used with different nuclear data libraries. The results for the 1000 MWth ABR metallic core 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	1.0223	1.0284	1.0215
β_{eff}	[pcm]	332	352	342	324	342	343
Control Rod Worth (fully inserted)	[pcm]	-9905	-10029	-9540	-9801	-9640	-9542
Control Rod Worth (5cm from top)	[pcm]	-239	-230	-241	-233	-233	-241
Doppler constant	[pcm]	-383	-407	-394	-397	-384	-351
Na Void Worth	[pcm]	1327	1219	1579	1464	1247	1423
1% Sodium	[pcm/K]	0.383	0.340	0.405	0.362	0.565	0.393
1% Wrapper	[pcm/K]	0.021	0.022	0.022	0.022	0.032	0.023
1% Cladding	[pcm/K]	0.043	0.050	0.050	0.048	0.070	0.040
1% Fuel	[pcm/K]	-0.553	-0.568	-0.538	-0.553	-0.594	-0.570
1% Fuel + Axial	[pcm/K]	-0.257	-0.265	-0.260	-0.271	-0.307	-0.267
1% Grid	[pcm/K]	-1.137	-1.115	-1.074	-1.078	-1.097	-1.162

Good agreement is observed for most of the parameters as all the results are within 2-σ.

However, in order to allow the identification of the cause of the observed differences, several sub-exercises have been set up. The models defined in the following exercises are derived from the medium-sized metallic core (MET1000) and the large oxide core (MOX3600) of the UAM-SFR specifications.

Some transient studies have been done by 4 different organizations (CEA, ANL, CER and IKE) on simple unprotected transients. Comparisons between results are still to be done but conclusions are the following ones for the 2 UTOP (fast and mild) and 1 ULOF.

For the ULOF, sodium boiling occurs at 873^{o}C within 17s in the large Oxide core.

- Max fuel temp: center of the hot pin in the third (middle) node
- Max cladding temp: inner wall of the cladding in the hot pin in the fifth (top) node

For the UTOPs in the large oxide core, fuel temperatures increase up to 2700^{o}C around.

Those modelled transients and core models used are meant to be simple and not especially representative of the real behavior of SFRs during transients. However, modifying the specifications in order to avoid fuel melting or sodium boiling are recommended to avoid misinterpretation of results and to allow meaningful uncertainty propagations.

3. UNCERTAINTIES

3.1 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 [14],
- bias from core modeling assumptions.

3.2 Nuclear Data Uncertainties

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.

Calculations were performed for reactivity, $\rho_{Doppler}$, $\rho_{Na\ Void}$ and β_{eff}. The uncertainties due to nuclear data are being calculated with different covariance matrices (COMAC, ENDF/B-VII.1, COMMARA-2.0, JENDL4.0). The different covariance matrices 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 and, in practice there are significant differences.

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

The conclusions [4,5] 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}U and in the capture of ^{239}Pu and 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.
- 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. 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.

3.3 Uncertainty propagation

Three different studies have been conducted for propagating different sources of uncertainties through UAM SFR ULOF and UTOP transients:

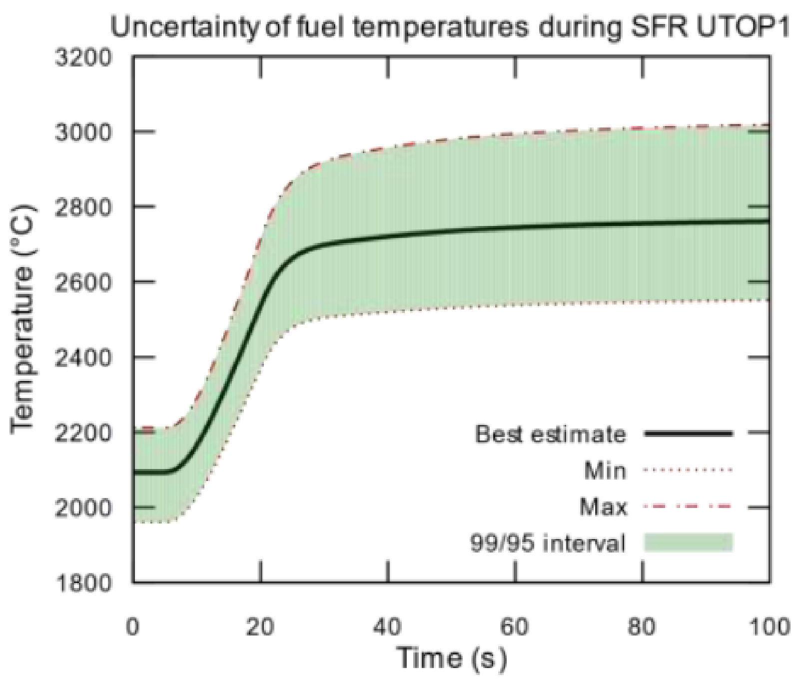
- the work conducted by CER uses the standard deviations between benchmark results of the different participants,
- the work conducted by IKE uses different pellet clad gaps to account for the heat transfer coefficient,
- the works conducted by ANL and CEA use nuclear data uncertainties.

Simulated transients are the ones specified in the specifications with 2 UTOP (fast and mild) and 1 ULOF.

3.3.1 Standard Deviation between participants

Hot channel results with their uncertainties (Figure 1) from CER are performed with the ATHLET3.1A code. Uncertainties come from the standard deviations between benchmark results of the different participants which come from both nuclear data and methods.

Figure 1: Hot Channel results with their uncertainties



The maximum temperatures including the uncertainties lead to sodium boiling in case of ULOF and to fuel melting both in the UTOP1 and the UTOP2 cases.

3.3.2 Pin mechanical models

Results of the UTOP and ULOF transient Benchmark for the large oxide core were calculated by IKE using DYN3D/ASTEC-Na. ASTEC-Na includes, in particular, an advanced pin mechanical models, a fission gas model and an in-pin fuel motion.

Since ASTEC-Na embark no neutronic module in opposition to SIMMER, a coupling of ASTEC-Na with nodal diffusion code DYN3D (HZDR) for more accurate neutronics description has been done. Macroscopic, group-wise XS data for DYN3D are generated with SERPENT2.

Since the calculations could not use the heat transfer coefficient specified, various gap sizes were used as a parametric study. As a result, different gap sizes induce different fuel temperatures with a smaller gap size leading to a lower fuel temperature and a later boiling onset from 30 to 40 s for the ULOF.

3.3.3 Nuclear data

ANL and CEA results conducted ULOF & UTOP transients with SAS4A and MAT4DYN respectively. ANL uses DAKOTA to propagate uncertainties on SAS4A calculations while CEA uses URANIE to propagate uncertainties on MAT4DYN calculations (Table 4).

Table 4: Hot Channel results with their standard deviations

	CEA OXIDE 3600 MWth		ANL METAL 1000 MWth	
Transient	ULOF	"Mild" UTOP	ULOF	"Mild" UTOP
Number of transients simulated	1000	1000	500	500
Peak outlet coolant temperature during transient				
Reference value [deg C]	1448	592	866	696
Average value [deg C]	1449	592	867	696
Standard deviation [deg C]	22	1	10	8
Upper 95 percentile [deg C]	1486	595	884	709
Peak fuel temperature during transient				
Reference value [deg C]	2574	3043	913	1066
Average value [deg C]	2575	3044	914	1066
Standard deviation [deg C]	45	20	10	7
Upper 95 percentile [deg C]	2654	3078	932	1072

As a conclusion, the impact of nuclear data uncertainties is generally small on transient behaviours.

Sensitivity analysis performed show:

- LHS provides quicker convergence (~100 simulations) but random sampling can be used as well
- Inter-reaction correlations can be evaluated, but effect is small
- Largest impact comes from uncertainties on Doppler and sodium density effects

4. EXPERIMENTAL EVIDENCE IN SUPPORT TO CALCULATIONS

For a reliable prediction of the characteristics of the core, it is necessary to use 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 [6, 7] (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.

4.1 Validation Exercises

The methods that were applied for the analysis of the specified theoretical exercises shall be used for the calculation of experiments. The calculated output quantities can be compared to the experimental values while considering the obtained uncertainties.

The validation experiments were chosen based on a similarity assessment between various experiments from the International Criticality Safety Benchmark Experiment Handbook (ICSBEP) handbook [6] and the MET1000 and MOX3600 fuel assemblies, respectively. First the experiments and the fuel assemblies were calculated using the TSUNAMI code of the SCALE 6.2 code package[8] to obtain the energy-dependent sensitivities of the eigenvalue to the cross sections. Then TSUNAMI-IP was used to determine the correlation coefficient index c_k between the systems using on ENDF/B-VII.1 covariance data. This index describes an estimate of the correlated uncertainty between systems. Systems with the same materials and similar spectra are correlated, while systems with different materials or spectra are not correlated.

The TSUNAMI-IP calculations resulted in negligible correlation factors between the fuel assemblies and experiments with the identifiers PU-MET-FAST and MIX-MET-FAST. Only for the MIX-COMP-FAST experiments, significant correlation factors were obtained (Table 5). Based on this assessment and the description of the MIX-COMP-FAST experiments, the ZPR-6 Assembly 7 and the ZPPR-2 experiment were chosen as validation exercises in this benchmark.

Table 5: Correlation factor between the experiment and the MET1000 and MOX3600 fuel assemblies, respectively, determined with TSUNAMI-IP.

	MET1000	**MOX3600**
MIX-COMP-FAST-001-001	0.8269	0.9117
MIX-COMP-FAST-002-001	0.8264	0.9124
MIX-COMP-FAST-003-001	0.8421	0.9197
MIX-COMP-FAST-003-002	0.8550	0.9352
MIX-COMP-FAST-004-001	0.6438	0.7346
MIX-COMP-FAST-005-001	0.8849	0.9546
MIX-COMP-FAST-006-001	0.8143	0.8882

4.1.1 ZPR-6 Assembly 7

The first validation exercise is the ZPR-6 Assembly 7, a fast reactor core with mixed (Pu,U)-oxide fuel and sodium with a thick depleted-uranium reflector. A description of this experiment including the detailed specifications is given in the International Handbook of Evaluated Reactor Physics Benchmark Experiments[8] under the acronym ZPR-LMFR-EXP-001. The parameters for which the uncertainties due to nuclear data shall be compared are the following:

- Multiplication factor,
- Sodium void worth for loading 46.

4.1.2 ZPPR-2

The second validation exercise is the ZPPR Assembly 2, cylindrical assembly with mixed (Pu,U)-oxide fuel and sodium reflected by depleted uranium, sodium, and steel. A description of this experiment including the detailed specifications has recently been added to the IRPhEP handbook under the acronym ZPPR-LMFR-EXP-011. The parameters for which the nominal value and the corresponding uncertainty caused by uncertainties of the nuclear data shall be compared are the following:

- Multiplication factor (case 1),
- Sodium void worth for case 9.

4.2 β_{eff} experimental validation

There are a number of available experiments for assessing the calculation of β_{eff} 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 [5]. 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) [9] 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. 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.

4.3 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 [10]. SEFOR documentation is not in the IRPhE standard but has been used in the past [11] 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.

4.4 Super-Phénix start up measurements

A benchmark based on the selected Super-Phénix (SPX) start up test has been developed [12, 13] 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 of 16.4 t/s. The sub-assembly uses MOX fuel and SS cladding.

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 × 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.

5. 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.

Experimental evidence in support of the studies is also being developed with neutronic, thermal hydraulic and system experiments.

6. ACKNOWLEDGEMENT

Argonne National Laboratory's work was supported by U.S. Department of Energy (DOE) under Contract number DE-AC02-06CH11357. The direction and support of the U.S. DOE Office of Nuclear Energy is greatly appreciated. TRIPOLI-4® is a registered trade mark of CEA. In case of MTA EK, this work has been carried out in the frame of VKSZ_14-1-2015-0021 Hungarian project supported by the National Research, Development and Innovation Fund. The GRS contribution to this work is supported by the German Federal Ministry for Economic Affairs and Energy.

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