Uncertainty Analysis of Kinetic Parameters for Design, Operation and Safety Analysis of SFRs

I.-A. Kodeli[1], G. Rimpault[2], P. Dufay[2], Y. Peneliau[2], J. Tommasi[2], E. Fridman[3], W. Zwermann[4], A. Aures[4], E. Ivanov[5], K. Ivanov[6], Y. Nakahara[7], T. Ivanova[7], J. Gulliford[7]

[1]Institut Jožef Stefan, Jamova 39, 1000 Ljubljana, Slovenia

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

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

[4]Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS), Garching, Germany

[5]IRSN, 92262 Fontenay-aux-Roses, France

[6]North Carolina State University, Raleigh, NC 27695, USA

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

E-mail contact of main author: ivan.kodeli@ijs.si

Abstract. An OECD/NEA sub-group on Uncertainty Analysis in Best-Estimate Modelling (UAM) for Design, Operation and Safety Analysis of Sodium-cooled Fast Reactors (SFR-UAM) has been initiated in 2015 with the objective to study the uncertainties in different stages of Sodium Fast Reactors.

Best-estimate codes and data together with an evaluation of the uncertainties are required for that purpose, which challenges existing calculation methods. Neutronic status and reactivity feedback coefficients as well as the kinetic parameters are being calculated for transient analyses. Experimental evidence in support of the studies is also being developed.

The use of the Iterated Fission Probability method in the Monte Carlo codes such as Tripoli4® SERPENT-2 and MCNP-6 gives reference values for calculating βeff as well as Λeff and their uncertainties. Deterministic codes like ERANOS and PARTISN/SUSD3D are also used for nuclear data sensitivity analysis and uncertainty propagation. The computational approaches are tested using available integral experiments and the uncertainties of the measurements. A vast series of experiments has been selected and analysed leading to recommendations on the tools, procedures and data to be used for βeff and/or transition functions calculating of the benchmarks including uncertainties.

Key Words: SFR, Beta-effective, Uncertainties

1. Introduction

Advanced Sodium-cooled Fast Reactors (SFR) are among the most promising reactor types studied in the scope of the Generation IV International Forum (GIF) and benefit from the experience accumulated over the past years. Generation IV reactors shall use fuel more efficiently, reduce waste production, be economically competitive, and meet stringent standards of safety and proliferation resistance. Under the auspices of the Working Party on Scientific Issues of Reactor Systems (WPRS) an OECD/Nuclear Energy Agency (NEA) Expert Group Task Force on Uncertainty Analysis in Best-Estimate Modelling (UAM) for

Design, Operation and Safety Analysis of Sodium-cooled Fast Reactors (SFR-UAM) [1] has been initiated in 2015 with the objective to study the uncertainties in different stages of the next generation Sodium Fast Reactors.

Improved safety performance is one of the key objectives of the design of sodium fast reactors and comprises in particular a demonstration of the favourable transient behaviour under accident conditions, such as passively avoiding core damage in case of reactivity excursion following a control rods malfunction. The global neutronic parameters such as the k-effective, beta effective, Doppler coefficient, sodium void worth, control rods worth, the power map and the isotopic content are of interest. This paper focuses on the kinetic parameter studies. Indeed, among the basic fission reactor parameters the kinetic parameters such as the effective delayed neutron fraction (beta-eff, β_{eff}) and the neutron generation lifetime (lambda) play a major role in the reactor safety from the point of view of controlling the reactor. Beta-eff is used to define a unit of reactivity known as the dollar and as such it plays an important role in reactivity accident analysis. Its accuracy should be therefore well understood and evaluated. The values of beta-eff vary from one isotope to the other (from ~200 pcm for ^{239}Pu to ~650 pcm for ^{235}U), therefore the reactor systems containing actinide isotopes in their fuel have to face the problem of lower values of beta-eff due to the presence of plutonium isotopes, making the reactor control of MOX fuelled cores more challenging.

A good understanding and reliable estimation of kinetic parameters is therefore essential. Kinetic parameters were already studied in the scope of the Uncertainty Analysis in Modelling (UAM) project [2] since 2011 and resulted in the in-depth study of the deterministic and Monte Carlo methods for the beta-eff evaluations, including the development of new methods and codes for the corresponding sensitivities and uncertainties.

2. Identification of suitable kinetics benchmark experiments

Several benchmark experiments available in the International Reactor Physics Benchmark Experiments (IRPhE) [3] and International Handbook of Evaluated Criticality Safety Benchmark Experiments (ICSBEP) [4] databases, as well as some found in the literature [5,6] were selected to be studied using deterministic and Monte Carlo codes. The benchmarks considered in these studies are listed in Table 1 together with their main characteristics. Different experimental techniques were used to measure the effective delayed neutron fraction, such as the reactor noise method, ^{252}Cf source method, pile oscillation method, Rossy-α, Nelson, Covariance to mean, Bennet etc. The stated measurement uncertainties are in general around 3 – 5%, but some estimations seem surprisingly low, possibly requiring further verification and/or re-evaluation.

TABLE I: LIST OF BENCHMARK EXPERIMENTS CONSIDERED IN THIS STUDY.

Name	**ICSBEP/IRPhE ref.**	**Description**
Jezebel	PU-MET-FAST-001	bare sphere of 95% ^{239}Pu metal, 4.5% ^{240}Pu, r=6.385cm
Skidoo	U233-MET-FAST-001	bare ~98.1% ^{233}U sphere, r=5.983cm
Popsy (Flattop-Pu	PU-MET-FAST-006	20-cm natural U reflected 94% ^{239}Pu sphere, r=4.533cm
Topsy (Flattop-25)	HEU-MET-FAST-028	~20-cm natural U reflected 93% ^{235}U sphere, r=6.116cm
Flat-top 23	U233-MET-FAST-006	~20-cm natural U reflected 98 at% ^{233}U sphere, r=4.2cm
BigTen	IEU-MET-FAST-007	Cylinder 10% enriched U with depleted U-reflector, r=41.91cm, h=96.428-cm
SNEAK-7A & -7B	SNEAK-LMFR-EXP-001	MOX fuel reflected by metallic depleted U
SNEAK-7A & -7B	SNEAK-LMFR-EXP-001	MOX fuel reflected by metallic depleted U
SNEAK-9C2	Ref. [5]	MOX fuel with Na, reflected by metallic depleted U
Berenice R2	Ref. [6]	30% ^{235}U
Berenice Zona2	Ref. [6]	MOX fuel, UO_2-Na blanket & steel shielding
BFS-73-1	BFS1-LMFR-EXP-001	Na-cooled fast reactor with metal enriched U fuel and depleted UO_2 blanket
BFS-61	BFS1-LMFR-EXP-002	Pb-cooled fast reactor with metal Pu-depleted U fuel and different reflectors
FCA-XIX-1	Ref. [6]	U core with depleted UO_2/Na and depleted U blankets
FCA-XIX-2	Ref. [6]	U/Pu core with depleted UO_2/Na and depleted U blankets
FCA-XIX-3	Ref. [6]	Pu core with depleted UO_2/Na and depleted U blankets

TABLE II: EFFECTIVE DELAYED NEUTRON FRACTIONS MEASURED IN DIFFERENT BENCHMARK EXPERIMENTS.

Name	**β_{eff} measurements (pcm)**	**β_{eff} benchmark data (pcm)**
Jezebel	Rossi-α:	194 ±10(ref.7)
Skidoo	Rossi-α:	290 ±10(ref.7)
Popsy (Flattop-Pu)	Rossi-α:	276 ±7(ref.7)
Topsy (Flattop-25)	Rossi-α:	665 ±13(ref.7)
Flat-top 23	Rossi-α:	360 ±9(ref.7)
BigTen	Rossi-α:	720 ±7(ref.7)
SNEAK-7A	^{252}Cf: 395± 20 Noise: 413± 25	395 ± 20(ref.3)
SNEAK-7B	^{252}Cf: 429± 22 Noise: 450± 27	429 ± 22(ref.3)
SNEAK-9C2	^{252}Cf: 426± 19	426 ± 19(ref.5)
BERENICE R2	^{252}Cf: 697± 20 (JAERI) ^{252}Cf: 739± 23 (CEA) Noise: 728± 15 (CEA) Rossi-α: 739± 23 (IPPE)	721 ± 11(ref.6)
BERENICE ZONA2	^{252}Cf: 346± 9 (JAERI) ^{252}Cf: 346± 9 (CEA) Noise: 350± 7 (CEA)	349 ± 6(ref.6)
BFS-73-1	Rossi-α: 740± 15 ^{252}Cf: 720± 27	735 ± 13(ref.4)
BFS-61		371 ± 60(ref.4)
FCA-XIX-1	Noise: 743± 19 (CEA) Rossi-α: 771± 25 (IPPE) ^{252}Cf: 706± 30 (IPPE) ^{252}Cf: 735± 20 (JAERI-KAERI) Cov.to mean: 724± 13 (JAERI)) Nelson #: 737± 20 (LANL) Bennet: 782± 16 (Nagaya)	742 ± 24(ref.6)
FCA-XIX-2	^{252}Cf: 351± 10 (IPPE) ^{252}Cf: 358± 10 (JAERI-KAERI) Bennet: 368± 6 (Nagaya)	364 ± 9(ref.6)
FCA-XIX-3	Noise: 250± 6 (CEA) ^{252}Cf: 244± 7 (IPPE) ^{252}Cf: 249± 7 (JAERI-KAERI) Cov.to mean: 252± 5 (JAERI)) Bennet: 256± 4 (Nagaya)	251 ± 4(ref.6)

3. Methodologies used for beta-eff computations and corresponding sensitivities

Interest in the β_{eff} sensitivity and uncertainty was expressed in the scope of the Uncertainty in Modelling (UAM) project [2] in 2010. Several perturbation methods were studied and are presented in refs. [8-11]. As demonstrated in these papers (e.g. [10]), the standard Keepin definition of the effective delayed neutron fraction [12] is equivalent to the 1st order derivative (sensitivity) of k_{eff} with respect to the delayed neutron yields. β_{eff} (and the contributions of each individual fissile isotope) can be therefore easily calculated using the standard perturbation codes, be it deterministic or (more recently) Monte Carlo, rather than using approximations such as the Bretscher's prompt k-ratio method [13]. Furthermore, β_{eff} being the 1st derivative of k_{eff}, the sensitivity of β_{eff} can be calculated as the 2nd derivative of k_{eff}, of course if the codes allow such calculations (e.g. Monte Carlo methods available in SERPENT [14] and MCNP6 [15]). Note that no additional code modifications are required in this case.

Alternatively, for methods limited to linear (1st order) perturbation theory, another approach was proposed in [8-11] based on the derivation of the Bretscher's prompt k-ratio method:

$$S_{\beta}^{\sigma} = \frac{\sigma}{\beta_{eff}} \frac{\partial \beta_{eff}}{\partial \sigma} = \frac{k_p}{k - k_p}\left(S_k - S_{kp}\right) = \frac{k_p}{k\beta_{eff}}\left(S_k - S_{kp}\right) \tag{1}$$

where k_p is the k_{eff} taking into account only prompt neutrons and k is the total (prompt and delayed neutron) k_{eff}. The two terms S_k and S_{kp} correspond to the sensitivities of the k and k_p which can be obtained using the standard linear perturbation theory. This approach was used in the SUSD3D [16-17] first-order sensitivity and uncertainty code.

Generalised perturbation method can also be used to obtain the $\boldsymbol{\beta_{eff}}$ sensitivity coefficients and the uncertainties [18]. ERANOS sensitivity calculations have been done with a special procedure using the Generalized Perturbation Theory (GPT).

2.1. Delayed neutron fraction calculations

The values of delayed neutron fraction β_{eff} calculated using different codes and methods are presented in Table II, together with the measured benchmark values. The following methods were used:

- SUSD3D [16-17]: beta effective was calculated as the sensitivity of k_{eff} to the delayed nu-bar [10].
- PARTISN [20] and MCNP5 [15] k-ratio: beta effective was calculated using the Bretscher's approximation [13], sometimes called also the prompt k-ratio method:

$$\beta_{eff} = 1 - \frac{k_p}{k} \tag{2}$$

- ERANOS [19]: beta effective is calculated using the standard perturbation theory
- TRIPOLI [21]: these are various ways to calculate the beta-effective but the most recent development has been used here: the Iterated Fission Probability (IFP)
- SERPENT [14]: the values of βeff were calculated using the iterated fission probability (IFP) method.
- MCNP6.1 [15]: here again the IFP method has been used

Reasonably good agreement between different codes and the measured values of delayed neutron faction was observed (Table III), with maximal difference reasonably within 1σ of the experimental uncertainty.

TABLE III: β_{eff} VALUES CALCULATED USING DIFFERENT METHODS AND NUCLEAR DATA.

Benchmark experiment	**Exp.**	**SUSD3D**	**PARTISN k-ratio**	**MCNP5 k-ratio**	**ERANOS**	**TRIPOLI IFP**	**MCNP6.1 IFP**	**SERPENT IFP**
		ENDF7.0/-7.1*			JEFF3.2	ENDF7.1 /JEF3.2	ENDF7.1 /JEFF3.2	ENDF7.0 /JEFF31
Jezebel	194 ±10	185	186	186				187±1 /188±1
Skidoo	290 ±10	296	297					295±1 /294±1
Popsy	276 ±7	277	278	284				277±2 /287±2
Topsy	665 ±13	688	690					691±2 /691±2
Flat-top 23	360 ±9	374	375					374±2 /381±2
BigTen	720 ±7	720	734					720±2 /736±2
SNEAK-7A	395 ± 20	373	379	369	383	370 ± 3/ 391± 3	363 ± 3/ 391± 8	371±2 /385±2
SNEAK-7B	429 ± 22	419	429	415	426	417 ± 3/ 441± 3	427 ± 8/ 425 ± 8	417±2 /433±2
SNEAK-9C2	426 ± 19				384			383±2 /398±2
Berenice R2	721 ± 11				749	732 ± 3/ 742± 3		
ZONA2	349 ± 6	344	351		362	335 ± 1/ 350 ± 1		
BFS-73-1	735 ± 13					- / 727 ± 1		
BFS-61	371 ± 60				383	370 ± 3/ 391± 3	363 ± 3/ 391± 8	
FCA-XIX-1	742 ± 24							757±2 /761±2
FCA-XIX-2	364 ± 9							368±2 /383±2
FCA-XIX-3	251 ± 4							250±1 /256±1

* ENDF7.1 stands for ENDF/B-VII.1

2.2.Uncertainty in β_{eff} due to nuclear data

To estimate the uncertainty in β_{eff} by the error propagation ("sandwich") formula, both the sensitivity of β_{eff} to the basic nuclear data as well as the corresponding nuclear data covariance matrices are needed. Several sources of covariance matrices needed were considered. However, among the available covariances only the JENDL-4.0u [22] evaluation includes covariance data relative to delayed fission neutron yields, which are a major source of uncertainties in β_{eff}. JENDL-4.0 also covers most of the relevant neutron reaction data and includes both cross-section as well as prompt neutron fission spectra covariance matrices and the P_1 angular distribution uncertainties (MF33, MF34 and MF35 format data, respectively) for the main fissile isotopes. Recently, the SCALE 6.2 [23] package has become available which also contains ENDF/B-VII.1 based covariance data including individual values for prompt and delayed nu-bar. Comparison of JENDL-4.0u covariances with other evaluations (such as ENDF/B-VII [24], SCALE-6 revealed several differences in uncertainty estimations for reactions such as inelastic (^{238}U), elastic, fission, total neutron yield etc.

β_{eff} uncertainties, calculated using different codes are listed in Table IV. Some differences can be observed which are mostly due to different covariance data used.

For the use with the SUSD3D code the JENDL-4.0u, ENDF/B-VII.1 and SCALE-6.0m covariance matrices (MF33, MF34 and MF35 data) were processed by the NJOY/ERRORR [25] and ANGELO [26] code systems. Results using the JENDL-4.0u data are given in Table IV. In the absence of more reliable data, an approximate "two-block" covariance matrices were constructed [10] based on a simple common sense assumption of an energy-uniform standard deviation of 15% and a complete anti-correlation between the energies above and below the mean delayed neutron energy for each of the six delayed groups. Conservative assumption of the complete correlation between the six individual groups was adopted. The neutron spectra of the delayed neutron were however found to play only a relatively small part in the β_{eff} uncertainty.

According to the JENDL-4.0u covariance data the total uncertainty in β_{eff} was found to be in general around 3% (up to 7% for the ^{233}U reactor systems) [10]. The β_{eff} uncertainty is in most cases predominantly due to the uncertainties in delayed neutron yields. In some cases the inelastic and elastic scattering, fission cross sections and prompt neutron yields, as well as the prompt and delayed fission spectra play an important role.

TABLE IV: LIST OF BENCHMARK EXPERIMENTS CONSIDERED IN THIS STUDY.

Experiment	**β_{eff} uncertainty (%)**		
	SUSD3D (JENDL4.0u)	**ERANOS** (JEFF3.2 +JENDL4.0)	**XSUSA** (SCALE-6.1 + JENDL-4.0)
Jezebel	2.5		2.9
Skidoo	7.1		
Popsy	2.6		4.9
Topsy	2.7		2.9
Flat-top 23	5.5		
Bigten	2.5		
SNEAK-7A	2.7	2.9	
SNEAK-7B	2.9	3.3	
SNEAK-9C2		2.9	
Berenice R2	2.6	2.8	
Berenice Zona2		3.6	

ERANOS results were obtained using the Generalized Perturbation Theory with the COMAC covariance data file associated with JEFF3.2 (values for delayed neutron fractions come from JENDL4.0).

The XSUSA [27] results were determined using the SCALE 6.1 covariance data for the cross section uncertainties and the JENDL-4.0u covariance data for the uncertainties of the delayed neutron yields.

The delayed neutron fraction and its uncertainty as a function of operation time were also studied for single light water reactor and sodium-cooled fast reactor fuel assemblies (see [28]). The XSUSA methodology was used to propagate cross section uncertainties (SCALE 6.1) and delayed nu-bar uncertainties (JENDL-4.0) in neutron transport and depletion calculations.

4. Conclusions

A series of studies were performed within the Expert Group of the SFR section of the Uncertainty Analysis in Modeling (UAM) activities of the WPRS cover kinetic parameter studies for fast reactor systems. The suitable experimental measurements of the effective delayed neutron fraction available in the international benchmark experiment databases such as IRPhE and ICESBEP, as well as in the literature were identified. The state-of-the-art Monte Carlo and deterministic methods for the calculation of the beta-eff values, as well as the corresponding sensitivities to nuclear data are presented. Significant improvements in the modelling and computational performances of the computer codes were obtained in the recent years, in particular for the Monte Carlo codes. The differences among the results obtained using different methodologies, neutron data evaluation and library files, computation codes, and applied approximations were studied and compared to the measured values.

The uncertainties of the whole computational process were evaluated for a selected set of benchmarks using the available nuclear covariance data, and the missing data or areas of weakness were identified.

The analyses of the effective delayed neutron fractions calculated using different deterministic and Monte Carlo codes yielded satisfactory agreement for the investigated benchmark experiments. Some differences between the results using different covariance data were observed and need to be further investigated.

The typical computational uncertainty of beta-eff due to nuclear data is around 3 – 4 %, and is of the similar order as the typically reported experimental uncertainty. In order to contribute to further improvement and validation of nuclear data and methods it is therefore recommended to revisit the available kinetic benchmark experiments and to perform new beta-effective measurements using reactor noise or Cf techniques with carefully evaluated experimental uncertainties, if possible with improved accuracy.

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