BFS-115: An experimental physics program to measure sodium void and control rod worth in an axially-heterogeneous fast reactor core

J. Tommasi[1], R. Jacqmin[1], F. Mellier[1], M. Yu. Semenov[2], D.A. Klinov[2], G.M. Mikhailov[2], A.M. Tsibulya[2]

[1]CEA, DEN, DER, SPRC, Cadarache, F-13108 Saint-Paul-lez-Durance, France

[2]JSC "SSC RF-IPPE", Obninsk, Russia

E-mail contact of main author: jean.tommasi@cea.fr

Abstract. As part of a bilateral agreement on the study of large axially-heterogeneous oxide-fueled SFR cores, CEA and IPPE have recently performed neutron physics experiments in the BFS facility. The configurations of interest are pancake-shape cores with a split fissile column and a sodium plenum, designed to favor a high inner plutonium conversion ratio and a low sodium void worth. Separate effect tests, including local and global sodium void situations as well as various rodded cases, have been done. The measurements included reactivity effects, spectral indices, detailed reaction rate traverses, neutron importance, etc. The analysis of the experiments with Monte Carlo codes and recent nuclear data files shows the following trends: core reactivity is predicted within 1.5 $, depending on the nuclear data file used; sodium voiding in the 91 central tubes is predicted within 0.25 $; the calculated axial reaction rate traverses match the experimental ones; the weight of the simulated control rod is predicted within 10%.

Key Words: ASTRID, BFS, validation, sodium void

1. Introduction

Under the French Act of 2006 on the sustainable management of radioactive materials and waste, stipulating the commissioning of a Generation-IV reactor in the 2020 decade, CEA is working on the conceptual design of a pool-type sodium-cooled fast reactor (SFR) with $(U,Pu)O_2$ fuel, in cooperation with industrial partners. This reactor is called ASTRID (for Advanced Sodium Technological Reactor for Industrial Demonstration).The specified total thermal power output is 1500 MW and the electrical output close to 600 MW.

An innovative core design was adopted to reduce the sodium density and void reactivity effects: the inner core was made heterogeneous in the axial direction by introducing an off-centered fertile zone, the fissile zone was topped by a thick sodium plenum, with a neutron-absorbing layer above it (see the schematic cross-cut of *FIG. 1*). In addition, the outer fuel column was made somewhat higher than the inner one. This heterogeneous axial arrangement results in a net overall (slightly) negative sodium void reactivity effect (SVRE).

For a recent status review of the ASTRID project, see [1]; for an overview of the core design and expected performance, see [2].

Compared with conventional homogeneous cores, such heterogeneous cores represent a challenge for neutron transport calculation codes, especially for sodium void and control rod reactivity worth predictions, as a consequence of a complex interplay of physics phenomena.

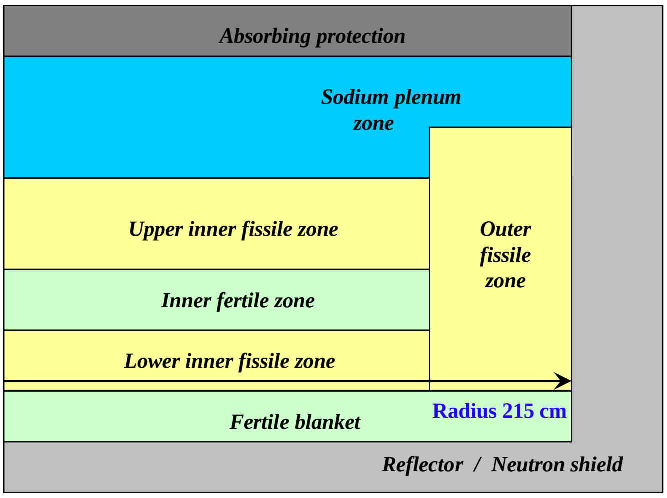

FIG. 1. Schematic R-Z cross-cut of the ASTRID core

This situation calls for specific validation data, which are lacking. CEA and IPPE have therefore decided to perform a joint experimental physics program at the zero-power BFS facilities in Obninsk, with the objective of getting such validation data. This program will be later complemented by experiments at MASURCA, once this facility resumes operation after completion of the current refurbishment program.

The experimental program was divided into four phases. The two first phases were performed at the BFS-2 facility in 2013 and 2014; the third phase was carried out at the BFS-1 facility, mainly in 2015; the fourth phase, also at BFS-1, is scheduled for 2017. The experiments were carried out by the BFS experimentalists. This paper presents the analysis of the third phase at BFS-1. In Section 2, a brief description of the facility and the experiments is given. A short description of the calculation schemes used by CEA and IPPE, followed by the code and data validation results is presented in Section 3.

2. The BFS-115-1 experiments

The BFS complex at IPPE (Obninsk) includes two fully compatible zero-power reactors: a large one (BFS-2) and a small one (BFS-1). *FIG. 2* is a photograph taken at BFS-1. The core is a hexagonal array of cylindrical tubes with a pitch of 5.1 cm. Each tube is loaded with pellets of various materials, e.g. Pu, UO_2, Na (or empty), steel, B_4C. A visual for such a loading can be seen in *FIG. 3*. The basic repetitive patterns (cells) for the fuel and fertile plate are shown in *FIG. 4*. The lower axial blanket, upper shield and radial blanket are made from depleted UO_2 pellets only, the plenum from Na pellets only, and the radial reflector from steel cylinders.

The loading map of the reference core is shown in *FIG. 5*, while the axial layouts of the tubes in the inner and outer fuel zones are displayed in *FIG. 6*.

The sodium voiding and filling experiments are performed over the 91 central tubes (i.e. center tube + 5 rings). These experiments involve a full voiding (Na pellets replaced by empty cans) and a full voiding followed by a progressive filling from bottom to top (see *FIG. 7*).

FIG. 2. A photograph of the BFS-1 facility (taken from [3])

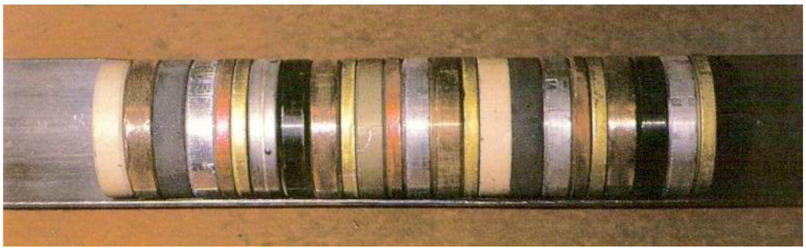

FIG. 3. Example of a pellet loading within a BFS tube (taken from [3])

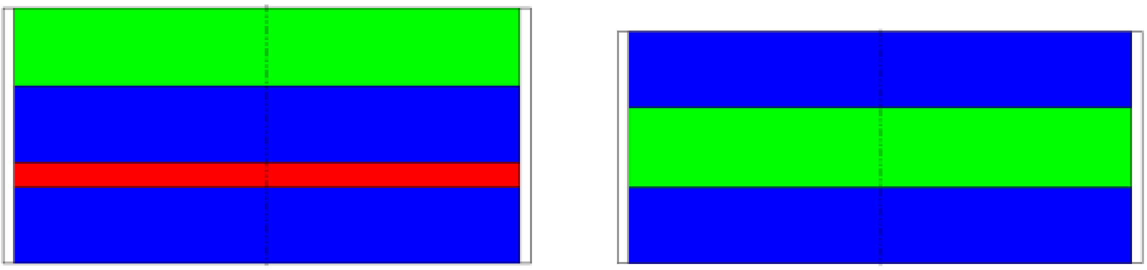

FIG. 4. Basic patterns (cells) of the pellet stacks for the fuel (left) and fertile plate (right) zones
Red = Pu (metal) – Blue = depleted UO_2 – Green = Na

The sodium void reactivity effects (SVRE) were determined using two methods. The first method measures differences of reactivity margins for two critical states of the assembly: (i) tubes loaded with sodium pellets, and (ii) tubes loaded with empty cans in replacement of sodium pellets in the sub-zone under investigation. The integral SVRE effect is then inferred by summing the partial effects measured for each sub-zone. The measured reactivity margins normally range from 0.02 β_{eff} to 0.2 β_{eff}. The BFS operational procedure requires the core periphery to be loaded (or unloaded) with a certain number of fuel rods to maintain a required reactivity margin. Therefore, in the course of these stepwise core sodium void measurements, the number of fuel rods in the core and the core radius vary. Reactivity values are obtained by the classical Inverse Kinetics Equation Solution (IKES) method.

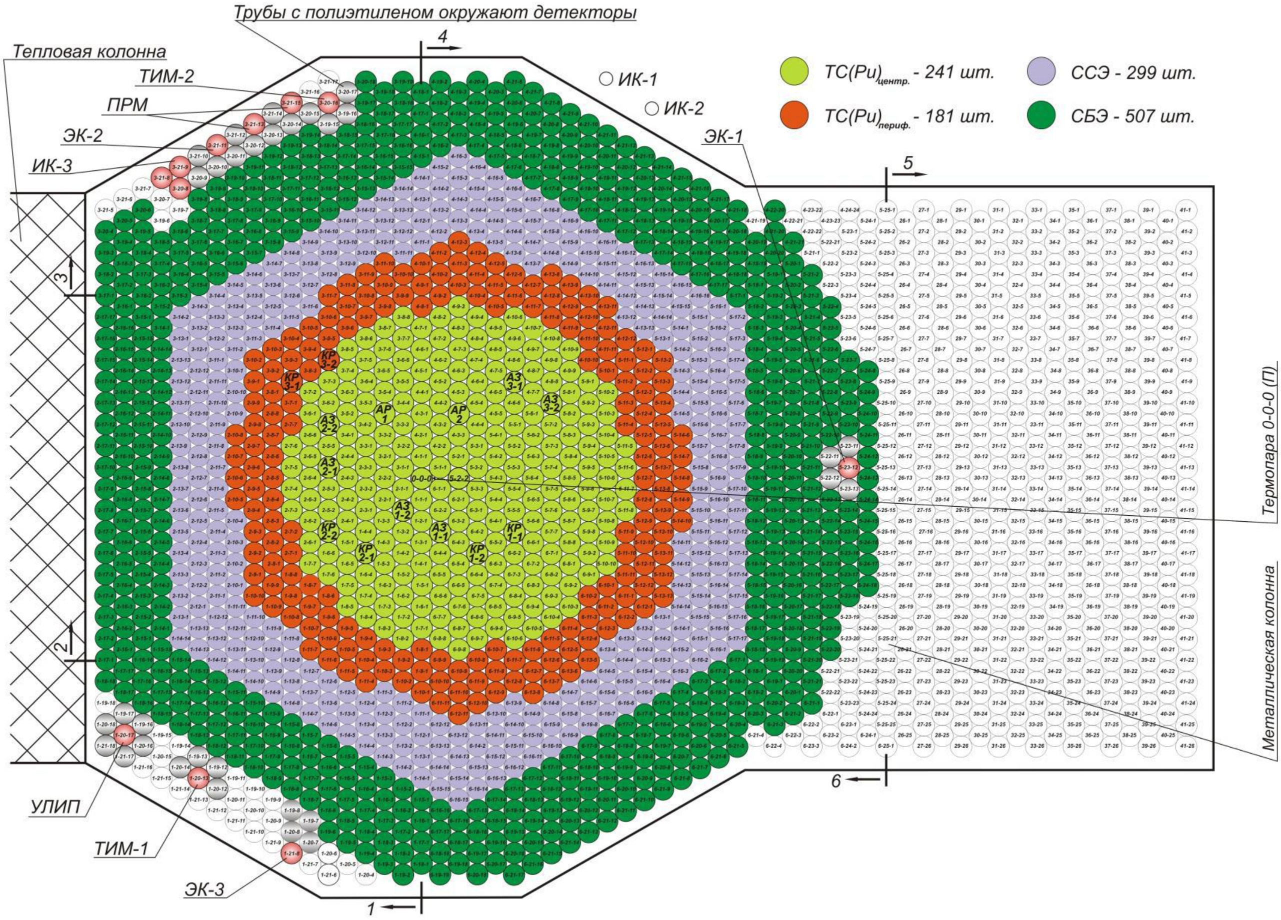

FIG. 5. Loading map of the BFS-115-1 reference core
Inner core = 241 tubes – Outer core = 181 tubes – Radial reflector = 299 tubes

The second SVRE measurement method is the ASM (approached source multiplication) method, which requires subcritical conditions. In the ASM, a reference neutron count rate measurement is first made for the initial configuration of the core. Then, neutron count rates are measured at each stage of core voiding. The value of the reference subcriticality is measured using a “runaway-rod drop” method of one of the control rods or a group of fuel rods (FRs). This is done in such a way that the rod perturbation has a minimum effect on the SVRE under measurement. The ASM allows the direct measurement of the integral SVRE effect over the entire voided volume. This method has therefore the advantage of not requiring loading (unloading) of FRs to complete the core.

The results obtained by these two measurement methods are generally in agreement, i.e. the differences remain within the experimental uncertainties. The total voiding of the 91 central tubes results in a negative reactivity effect with a magnitude of approximately -1.2 β_{eff}, and each of the partial filling steps lies in the range of 0.2 to 0.4 β_{eff} (positive or negative).

Axial reaction rate distributions (traverses) were measured in an inter-tube adjacent to the central tube, by inserting small chambers at different heights, with an axial pitch of 15 mm. The reactions used are: ^{238}U(n,f), ^{237}Np(n,f), ^{235}U(n,f), ^{239}Pu(n,f) and ^{10}B(n,α). These measurements were performed both for the reference core and for the configuration with the 91 central tubes fully voided (Na pellets replaced by empty cans).

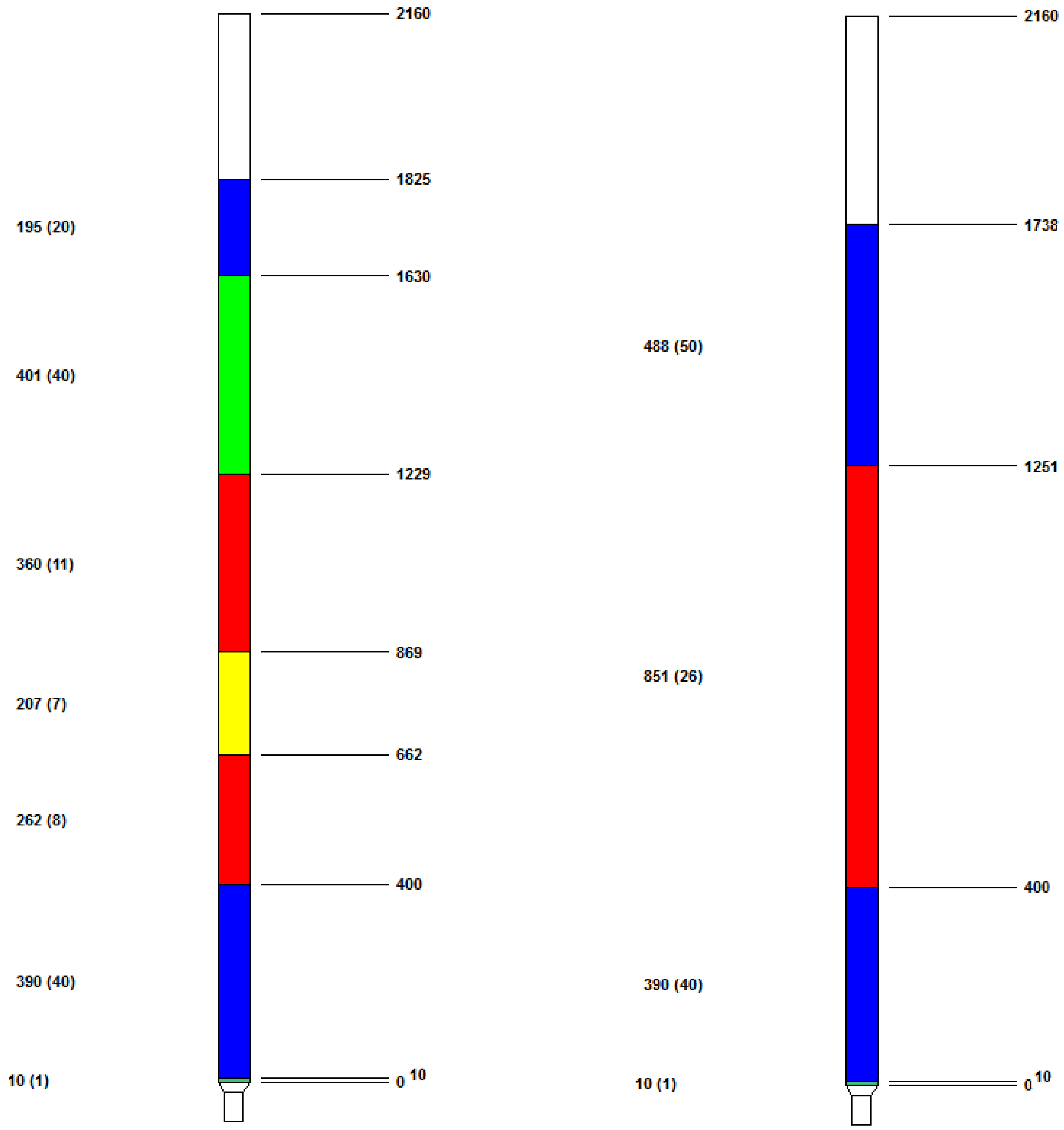

FIG. 6.Axial structure of the inner and outer fuel tubes
Red = fuel – Blue = depleted UO_2 – Yellow = fertile plate – Green = Na

A control rod was simulated by replacing the 7 central fuel tubes by 7 tubes containing absorber cells (made of $^{nat}B_4C$, steel and Na pellets) and follower cells (made of steel and sodium pellets). See the schematic diagram in *FIG. 8*. These 7 tubes were positioned at different heights successively. The control rod total worth and S-curve were inferred using the two measurement techniques, IKES and ASM, for the reference core state and the configurations with the 91 central tubes (including the control rod tubes) voided. Again, the two types of measurement results were found to agree within the experimental uncertainties of the order of 5%.

The full control rod worth is on the order of 2.5 dollars in the reference core and 2 dollars in the core with the 91 central tubes fully voided.

Several other measurements were performed in the BFS-115-1 core as part of the CEA-IPPE bilateral agreement, but are not reported here.

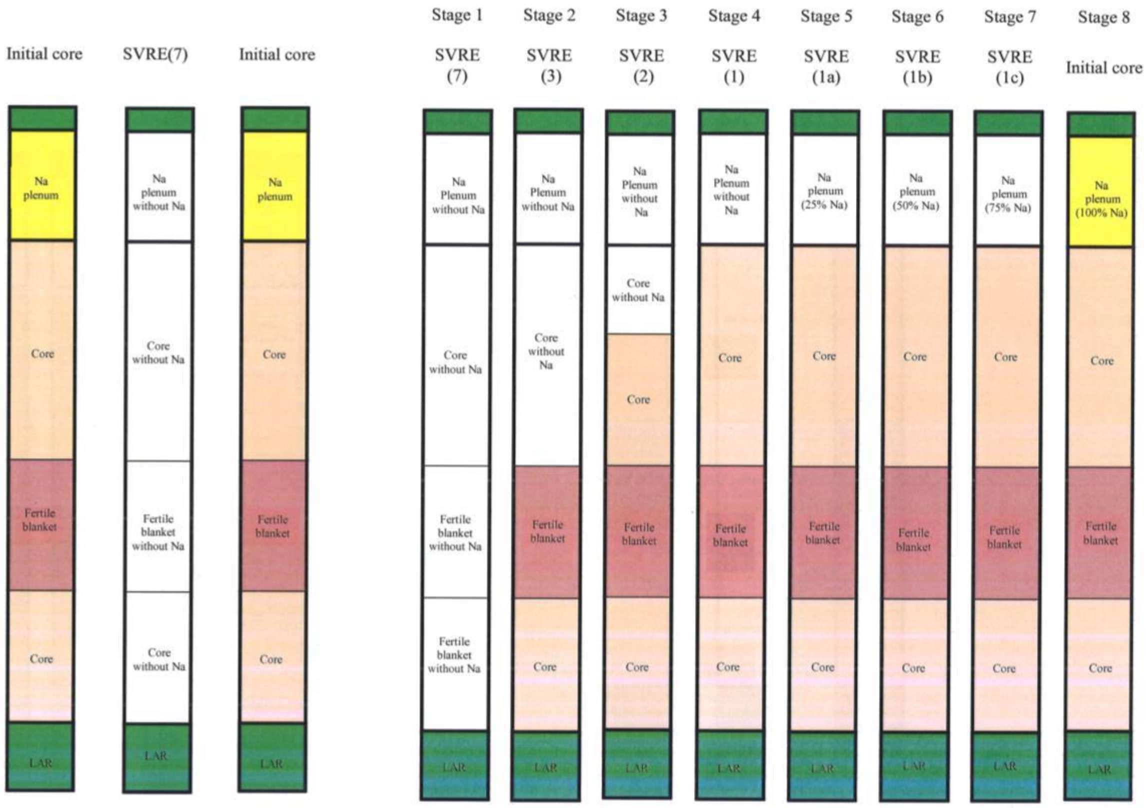

FIG. 7.Schematic view of the sodium voiding and filling sequences (white = voided)
Note the progressive filling of the plenum: 100%→75%→50%→25%→0% void

3. Analysis of the experiments

3.1.Codes and data used

For most of their calculations, IPPE used the MMKKENO Monte Carlo code [4] with the ABBN-93 cross-section data [5] processed in 299 energy group structure with subgroups for ^{235}U, ^{238}U, ^{239}Pu and Fe. Some deterministic calculations (e.g. for the calculation of the delayed neutron fraction) were performed using the TRIGEX diffusion code [6] and the ABBN-93 data, using a 28 energy groups structure. Broad group data are processed from the ABBN-93 data using the CONSYST system [7]. IPPE also used the MMKC Monte Carlo code [8] with continuous energy cross-section data produced from the ROSFOND [9] evaluated nuclear data files with a single nuclide temperature equal to 300 K. Probability tables were then used to account for self-shielding in the unresolved resonance range of the nuclides.

CEA used the TRIPOLI-4 Monte Carlo code [10] with continuous-energy cross-section data produced either from the JEFF-3.1.1 [11] or JEFF-3.2 [12] libraries with a single nuclide temperature equal to 20°C. Probability tables were used to account for self-shielding in the unresolved resonance range of the nuclides.

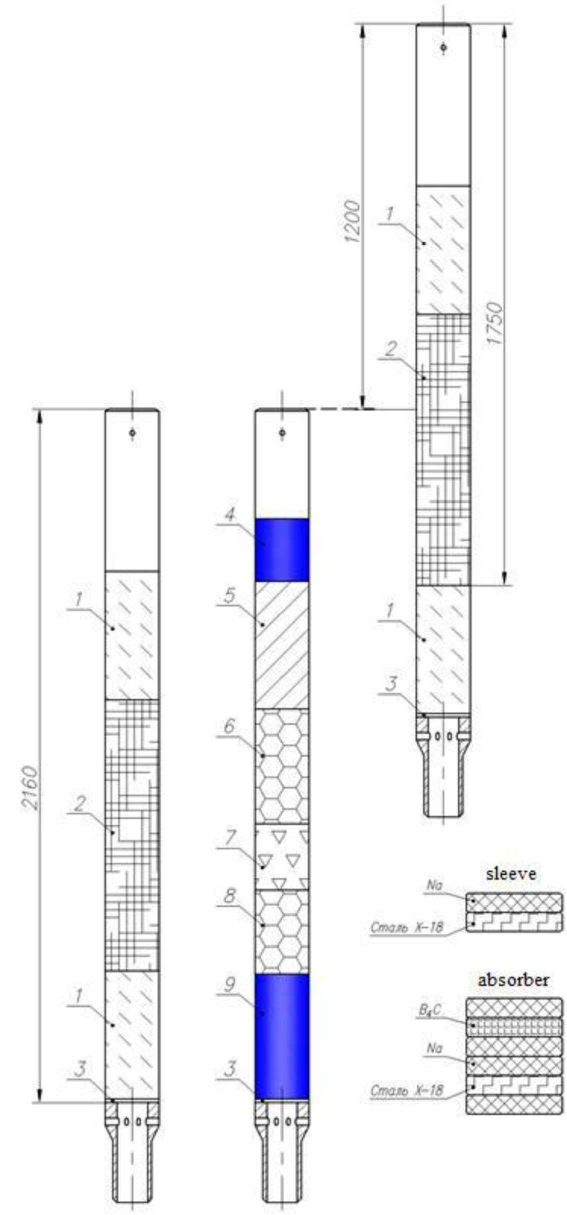

FIG. 8.Schematic view of a control rod mock-up tube fully inserted (left) and lifted (right) compared to a standard fuel tube (center)

In the Monte Carlo calculations, a detailed geometry (pellet by pellet) was modelled, corresponding to an explicit representation of the loaded pellets and tubes. Detailed geometric and composition data were provided by IPPE.

The statistical uncertainties for Monte Carlo calculations are given as 1 standard deviation (1σ). For uncertainty combinations, the experimental uncertainties used are also 1σ values.

3.2.Reference core reactivity

The calculated values of the effective delayed neutron fraction β_{eff} are presented in Table I below. IPPE calculated this parameter using the TRIGEX diffusion code with the ABBN-93 data and delayed neutron data in 6-group format. CEA used the Iterated Fission Probability method recently implemented in TRIPOLI-4 [13] to perform the adjoint weighting necessary to compute β_{eff}. The basic delayed neutron data was in an 8 time-group representation.

As shown in Table I, the calculated results are in very close agreement, within 1%. The value of β_{eff} sets the reactivity scale used for the measurement results, according to the usual definition: β_{eff} = 1 dollar = 100 cents.

The core temperature coefficient (CTC) was measured as -1.0 ± 0.1 cent/K. Due to the forced cooling of the core and appropriate waiting times for temperature stabilization, all measurements were performed in a narrow temperature interval, i.e. between 17 and 27°C (measured in specific inter-tubes by 4 thermocouples), so as to make reactivity effects induced by temperature differences negligible. For comparisons to calculations, the measurement results were scaled to 20°C using the measured CTC value.

Reactivity margins for the BFS-115-1 reference core (loading as shown in *FIG. 5*) were measured in the range from 15 to 24 cents, with the average at 19 cents. The computed reactivity margins (Table II) are within 1.5 dollar from the experimental value. The major discrepancy observed, between the CEA results using JEFF-3.1.1 and JEFF-3.2 data, is mainly due to differences in ^{23}Na, ^{239}Pu and ^{238}U cross-section data.

3.3. Sodium void reactivity effect

For the full voiding of the 91 central tubes (Na pellets replaced by empty boxes, see left part of *FIG. 7*), the comparison of calculated values to experiment is given in Table III. The order of magnitude of the experimental value is -120 cents.

For the seven steps of progressive sodium filling (empty boxes replaced by Na pellets, see right part of *FIG. 7*) the differences between calculation and experiment results are more scattered due to lower reactivity effects with similar absolute experimental uncertainties. We only give here a synthetic dimensionless indicator of the agreement between calculation and experiment, the average relative quadratic discrepancy defined as (N=7 here):

$$\Delta = \sqrt{\frac{1}{N}\sum_{i=1}^{N}\frac{(C_i - E_i)^2}{\delta C_i^2 + \delta E_i^2}} \tag{1}$$

where C_i are the calculated values and δC_i their statistical uncertainties, E_i the experimental values and δE_i their uncertainties. The values for this indicator are given in Table IV.

For the CEA calculations, going from JEFF-3.1.1 to JEFF-3.2 data globally improves the sodium void/fill reactivity effect prediction. This is probably due in part to the new ^{23}Na cross-section data evaluation in the JEFF-3.2 library [14], but a thorough sensitivity analysis still remains to be done.

TABLE I: RESULTS FOR THE DELAYED NEUTRON FRACTION CALCULATION.

	Code	**Data**	**β_{eff} (pcm)**
CEA	TRIPOLI-4	JEFF-3.1.1	366.1 ± 1.4
		JEFF-3.2	370.3 ± 1.4
IPPE	TRIGEX	ABBN-93	367.2

TABLE II: COMPUTED REACTIVITY MARGINS FOR THE REFERENCE CORE.

	Code	**Data**	**reactivity (cents)**
CEA	TRIPOLI-4	JEFF-3.1.1	8.6 ± 1.1
		JEFF-3.2	-146.3 ± 1.3
IPPE	MMKKENO	ABBN-93	56.7 ± 0.5
	MMKC	ROSFOND	-104.5 ± 2.4

TABLE III: FULL Na VOIDING IN THE 91 CENTRAL TUBES.

	Code	**Data**	**C/E**
CEA	TRIPOLI-4	JEFF-3.1.1	1.22 ± 0.03
		JEFF-3.2	1.12 ± 0.03
IPPE	MMKKENO	ABBN-93	1.11 ± 0.03

TABLE IV: PROGRESSIVE Na FILLING IN THE 91 CENTRAL TUBES.

	Code	**Data**	**Δ from Eq.(1)**
CEA	TRIPOLI-4	JEFF-3.1.1	5.3
		JEFF-3.2	3.4
IPPE	MMKKENO	ABBN-93	3.0

3.4.Spectral indices and reaction rate traverses

Three spectral indices (reaction rate ratios) were measured in an inter-tube close to the central tube, at mid-height of the upper fuel column. The comparison of Monte Carlo results to experiment is given in Table V. With well-converged Monte Carlo runs, the major contributor to the quoted uncertainty is the experimental uncertainty. The effects of cross-section changes between libraries (and possibly from multigroup to continuous energy data) are apparent.

Excellent agreement is found between the calculated and experimental shapes of the axial traverses of ^{238}U(n,f), ^{237}Np(n,f), ^{235}U(n,f), ^{239}Pu(n,f). Two visual examples are provided in *FIG. 9*. Larger discrepancies are found for the ^{10}B(n,α) traverses and spectral indices. They are attributed to calibration and discrimination threshold problems of the ^{10}B chambers.

TABLE V: SPECTRAL INDICES (C/E RATIOS).

	IPPE		**CEA**	
Code	**MMKKENO**	**MMKC**	**TRIPOLI-4**	
Data	**ABBN-93**	**ROSFOND**	**JEFF-3.1.1**	**JEFF-3.2**
^{238}U(n,f)/^{235}U(n,f)	1.014 ± 0.028	0.981 ± 0.028	0.967 ± 0.027	0.967 ± 0.027
^{239}Pu(n,f)/^{235}U(n,f)	1.024 ± 0.015	0.977 ± 0.015	0.990 ± 0.015	0.975 ± 0.015
^{237}Np(n,f)/^{239}Pu(n,f)	0.976 ± 0.020	1.000 ± 0.020	0.945 ± 0.019	0.941 ± 0.019

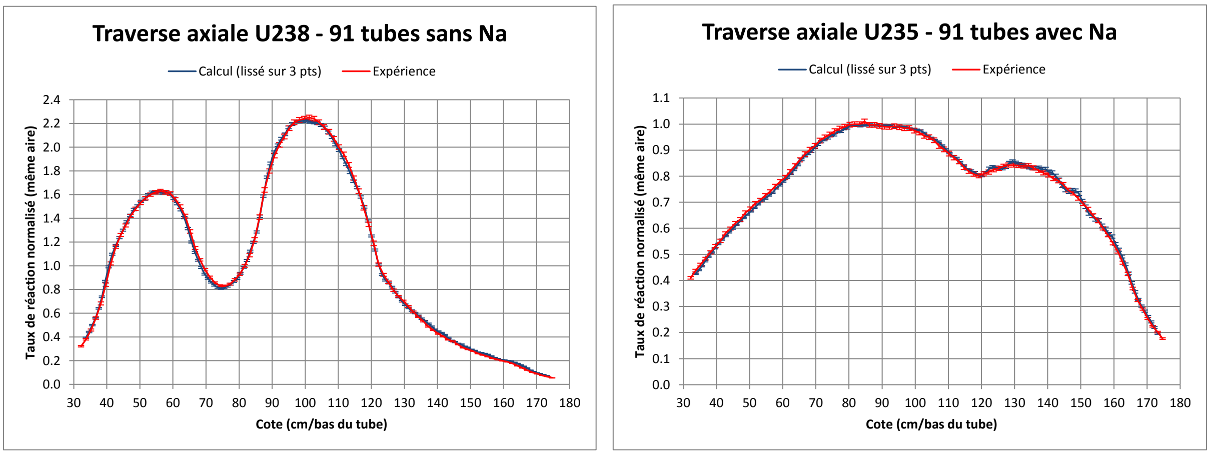

FIG. 9.Examples of axial traverses – Left: $^{238}U(n,f)$ in the core with the 91 central tubes voided – Right: $^{235}U(n,f)$ in the reference core. Blue = calculation (TRIPOLI-4+JEFF-3.1.1), Red = experiment Left (^{238}U): bumps in the lower and upper fuel regions, depression in the fertile plate – Right (^{235}U): smooth behavior in the fuel + plate region, increase in the sodium plenum

3.5.Control rod worth

Comparison between the calculated and experimental results for the full worth of the simulated control rod are given in Table VI. A good agreement is generally found, but the larger discrepancy observed for the CEA calculations in the case of the voided 91 central tubes has to be investigated further.

4. Conclusion and perspectives

CEA and IPPE have recently performed a joint experimental physics program in BFS, in which sodium void and control rod reactivity effects have been measured in axially-heterogeneous SFR core configurations. A subset of the measurements has been analyzed by both organizations. The results of this first analysis have been presented. A global good agreement between calculated and experimental values was found. Core reactivity is predicted within 1.5 $, depending on the nuclear data file used. Sodium voiding in the 91 central tubes is predicted within 0.25 $. The calculated axial reaction rate traverses match the experimental ones. The reactivity worth of the simulated control rod is predicted within 10%. A more thorough analysis of these (and other additional experimental) results has yet to be done, in order to explain some of the reported discrepancies.

TABLE VI: FULL CONTROL ROD MODEL WORTH (C/E)

	Data	Ref. core	91 tubes voided
CEA	JEFF-3.1.1	1.04 ± 0.05	1.11 ± 0.06
	JEFF-3.2	1.03 ± 0.05	—
IPPE	ABBN-93	1.02 ± 0.05	1.03 ± 0.06

5. References

[1] ROUAULT, J., et al., "ASTRID, The SFR GEN-IV technology demonstrator project: where are we, where do we stand for?", Proc. Int. Conf. ICAPP 2015, May 03-06, 2015, Nice, France – Paper 15439.

[2] VENARD, C., et al., "The ASTRID core at the midterm of the conceptual design phase (AVP2)", Proc. Int. Conf. ICAPP 2015, May 03-06, 2015, Nice, France – Paper 15275.

[3] IRPhE (International Reactor Physics Experiments database), "BFS-73-1 assembly – Experimental model of sodium-cooled fast reactor with core of metal uranium fuel of 18.5% enrichment and depleted uranium dioxide blanket" , Id. Number : BFS1-LMFR-EXP-001 CRIT-SPEC-COEF-KIN-RRATE, NEA/NSC/DOE(2006)1

[4] BLYSKAVKA A.A., MANTUROV G.N., NIKOLAEV M.N., TSIBULYA A.M., "The CONSYST / MMKKENO program package for nuclear reactor calculations using the Monte Carlo method in a multigroup approximation with scattering indicatrices in a Pn approximation", Preprint IPPE-2887, Institute for Physics and Power Engineering, Obninsk (2001)

[5] MANTUROV G.N., NIKOLAEV M.N., TSIBULYA A.M., "The system of group constants ABBN-93 Verification Report", Interdepartmental Data Qualification Commission, Moscow (1995)

[6] SEREGIN A.S., KISLITSINA T.S., "Notes on the TRIGEX-CONSYST-BNAB-90 Program Package", Preprint IPPE-2655, Institute for Physics and Power Engineering, Obninsk (1997)

[7] MANTUROV G.N., NIKOLAEV M.N., TSIBULYA A.M., "CONSYST cross-section preparation system", Preprint IPPE-2828, Institute for Physics and Power Engineering, Obninsk (2000)

[8] BLYSKAVKA A.A., ZHEMCHUGOV YE.V., RASKACH K.F. Pilot version of the MMKC code with continuous monitoring of the neutron energy. XXIII-th All-Russian Seminar "Neutron-physical problems of nuclear energy with a closed fuel cycle" (NEUTRONIKA), Obninsk, Russia (2012).

[9] S.V. ZABRODSKAYA, A.V. IGNATYUK , V.N. KOSCHEEV et al., VANT, Nuclear Constants, v. 1-2, p.3 (2007). ROSFOND-2010 Library, Institute of Physics and Power Engineering, Obninsk, Russia (2010).

[10] BRUN, E., et al., "TRIPOLI-4® , CEA, EDF and AREVA reference Monte Carlo code", Annals of Nuclear Energy **82** (2015) 151-160

[11] SANTAMARINA, A. et. al., « *The JEFF-3.1.1 nuclear data library* », JEFF Report 22, NEA No.6807, OECD (2009)

[12] JEFF-3.2 Evaluated Data Library – Neutron Data, March 5 (2014), https://www.oecd-new.org/dbforms/data/eva/evatapes/jeff_32

[13] TRUCHET, G., et al., "Computing adjoint-weighted kinetics parameters in Tripoli-4® by the Iterated Fission Probability method", Annals of Nuclear Energy **85** (2015) 17-26.

[14] ARCHIER, P., NOGUERE, G., DE SAINT JEAN, C., PLOMPEN, A.J.M. and ROUKI, C., "New JEFF-3.2 sodium neutron induced cross-sections evaluation for neutron fast reactors applications: from 0 to 20 MeV", Nuclear Data Sheets, **118** (2014) 140-143