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JEFF-3.2 LIBRARY VALIDATION

IMPACT OF THE NEW LIBRARY FOR FAST NEUTRONS SFR BENCHMARKS

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Abstract – The JEFF-3.2 library was released by OECD/NEA in March, 2014. The work presented here deals with the validation of this library for Sodium Fast Reactors and attempts to highlight the improvements obtained for this class of reactors. As CEA is involved in the design of a new SFR prototype ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) that should be built around 2020, foreseen reactor safety analyses will be performed using both new neutronics computational codes and new nuclear data libraries. So, there is a strong interest in JEFF-3.2 benchmarking for fast neutron spectrum applications. In addition, some thermal neutron spectrum configurations are also tested to check whether JEFF-3.2 does not degrade the performances shown by JEFF-3.1.1. A huge effort has been made in the JEFF-3.2 library to improve nuclear data for many heavy isotopes either in the high energy domain (^{235}U, ^{238}U) or in the whole energy domain (^{239}Pu, ^{240}Pu , ^{241}Am). Concerning moderator materials, the ^{23}Na evaluated data has also been revisited in the whole domain. Some structure materials like Cr or some fission products have also been modified, but their impact on the class of benchmarks of interest remains low. Across this paper, we report the results of a CEA benchmarking study of the library. Benchmarks originate from the criticality database ICSBEP (about 100 cases are used), from the reactor physics database IRPHE (SNEAK7A, SNEAK7B and ZPPR-10A), from the CEA EOLE Light Water Reactor mock-up (EPICURE and MISTRAL experimental programs) or from the MASURCA Sodium Fast Reactor mock-up (MASURCA1B, PRE-RACINE, RACINE and CIRANO experimental programs), from Superphénix Commissioning Tests or finally from the radiation shielding database SINBAD (ASPIS, REPLICA and JANUS-8). As the basis of JEFF-3.2 library is the JEFF-3.1.1 library, comparisons are made between the two libraries on one hand, and with experimental values on the other hand. Main physical parameters are the effective multiplication factor and the sodium void effect for criticality and reactor physics benchmarks, and the dosimetry reaction rates for radiation shielding benchmarks. All the benchmarks are simulated with the CEA Monte Carlo code TRIPOLI-4®. Some perturbation analyses are performed using the CEA ERANOS/SNATCH code package for some particularly interesting cases. Reactivity effect breakdowns are shown to highlight the origin of the main differences between JEFF-3.1.1 and JEFF-3.2 calculations for some SFR-type benchmarks. Results clearly show that new ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Am and ^{23}Na evaluations have a huge impact on k_{eff} values for SFR type configurations and generally improve the calculated values compared to experimental ones. Moreover, the revisited ^{241}Am data improve results for EOLE thermal cases. Uranium, plutonium or mixed uranium and plutonium thermal configurations are still accurately predicted with the new JEFF-3.2 library.

I. INTRODUCTION

For next generation (GEN-IV) nuclear power plants, France and more particularly CEA have chosen SFR reactors. In 2020, a new SFR prototype should be built to demonstrate the feasibility of innovative concepts. This prototype is called ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) [1]. Within the next years, some different safety reports will be requested that will have to rely on modern and improved calculation schemes on one side and on brand new validated nuclear data on the other side. A crucial issue for these data will be their covariance data to master the uncertainties relevant to each safety parameter (k_{eff}, β_{eff}, Λ_{eff}, $\Delta\rho_{Na}$, $\Delta\rho_{Doppler}$ for example as well as 3D neutronics feedback parameters).

In this framework, CEA is involved in improving current nuclear data evaluation in the JEFF project [2]. Its participation to the new JEFF-3.2 evaluated nuclear data file has focused on the main actinides U, Pu and Am, sodium, structural materials and some fission products. This work has generally been made in collaboration with different divisions of the CEA, ORNL (Oak Ridge National Laboratory) and NRG (Nuclear Research and Consultancy Group).

The benchmarking phase is very important to ensure that new nuclear data improve the results of calculations compared to experimental values for the SFR class of experiments. Our goal is also to check that results obtained for thermal cases after huge efforts to improve evaluations in the thermal region [3] are not degraded. The present benchmarking is made using the CEA computational codes TRIPOLI-4® [4] and SNATCH [5] for the effective multiplication factors. Some reactivity breakdown analyses are also performed using Standard Perturbation Theory to get the exact breakdown of reactivity changes generated by the switch from one evaluation to the other.

Benchmarks analyzed in this paper have different origins. Some of them are extracted from:

- the ICSBEP international database (HEU-MET-FAST, PU-MET-FAST, PU-SOL-THERM, HEU-SOL-THERM, …) [6],
- the IRPHE international database (SNEAK7, ZPPR) [7],
- the SINBAD international database [8] for radiation shielding,
- the CEA facilities, EOLE thermal spectrum mock-up [9] or MASURCA fast spectrum mock-up [10],
- the industrial SFR power plant Superphénix start-up tests [11].

In section II, revisited nuclear data are presented. Then, section III concerns the codes and methods used to analyze the impact of the changes in the libraries. More details on the experiments are supplied in section IV and their choice is briefly justified. Finally, results are listed followed by their analyses based on Standard Perturbation Theory in section V. The impact of the new ^{238}U, ^{239}Pu, ^{240}Pu and ^{23}Na data are highlighted for fast neutron spectrum and the non-regression testing for thermal neutron spectrum is verified.

II. MAIN CHANGES IN JEFF-3.2

This chapter deals with the main changes in new library JEFF-3.2 for fast neutrons configurations and focuses on the changes for which CEA has been involved.

II.A. Uranium, Plutonium and Americium actinides

Uranium isotopes have been reevaluated in the continuum energy range. Two plutonium isotopes, ^{239}Pu and ^{240}Pu, and ^{241}Am have also been changed.

The main changes for ^{235}U are the cross sections and other nuclear data like average prompt fission neutron multiplicity in the continuum region.

For ^{238}U, a small change is made concerning the spin assignment of two resonances (the 551 eV and the 2919 eV resonances) in the Resolved Resonance Range (RRR). The upper limit of Unresolved Resonance Range (URR) has been moved down from 300 keV to 150 keV in order to include only the first inelastic level in the URR. Actually, the code used to process probability tables is better suited for one inelastic level processing. Between 20 keV and 30 MeV, a new evaluation is proposed by CEA Bruyères-le-Chatel. It is based on TALYS calculations [12]. Moreover, new covariance data in MF=33 for MT=1,2,18,102 are provided in 33 groups energy meshing. These data are produced by CONRAD [13] using the marginalization technique [14].

Concerning ^{239}Pu, a collaborative work has been done between ORNL and CEA to analyze the Resolved Resonance Range. The Unresolved Resonance Range data have not been changed compared to JEFF-3.1.1 and the continuum part has also been reevaluated through a collaboration between NRG and CEA.

The analysis of the RRR was motivated by the presence in JEFF-3.1.1 of three disjoint sets of data to cover the energy range (10^{-5} eV to 1 keV, 1 keV to 2 keV, and 2 keV to 2.5 keV). The consequences were mismatches between cross sections and no correlation between the sets. The neutron resonance shape analysis has been performed with the codes SAMMY [15] and CONRAD.

The resonance parameters covariance matrix has been generated using the marginalization technique.

For ^{240}Pu, the Resolved Resonance Range and the Unresolved Resonance Range have been taken from JEFF-3.1.1 evaluation. Some efforts have been done by two

different divisions of CEA and NRG to improve the continuum part. Moreover, some particular data like prompt fission multiplicities have been modified and new high energy reactions have been introduced with new TALYS calculations: (n, nα), (n, 2nα), (n, 3nα), (n, np), (n, n2p), (n, n3p), (n, p) and (n, α).

In ^{241}Am, the Resolved Resonance Range (from 10^{-5} eV to 150 eV) has been analyzed through different works. New IRMM (Institute for Reference Materials and Measurements) measurements performed by C. Sage during its PhD thesis, the resonance shape analysis performed once again with the nuclear data codes SAMMY and CONRAD and the resonance parameters covariance matrix generation have contributed to improve the evaluation. The thermal capture cross section has been increased by about 15% (from 647 b in JEFF-3.1.1 to 748 b +/- 35 b). The capture resonance integral has also grown of almost 20% (from 1526 b in JEFF-3.1.1 to 1826 b +/- 55 b). The fission resonance integral on the contrary decreased from 17.3 b in JEFF-3.1.1 to 15.2 b +/- 0.5 b. Integral Data Adjustment technique in CONRAD was used to extract the capture resonance integral from ICARE-S [16] experiment and direct perturbation analysis was used to fix the thermal capture cross section from SHERWOOD experiment. Both experiments were performed in the CEA research reactor called MELUSINE.

A complete work has been done to re-evaluate the isomeric ratio for capture [17] from 0 eV to 20 MeV. The variances were also evaluated. The isomeric ratio is given pointwise in the evaluation in the form of resonant partial capture cross section.

II.B. Sodium

^{23}Na has been totally re-evaluated [18]. The initial work was motivated on one side because of the CEA ASTRID prototype and the need to calculate accurately the sodium void effect and on the other side because JEFF-3.1.1 cross section data were not consistent with microscopic measurements. Moreover, covariance data were missing in previous JEFF-3.1.1 evaluation file and uncertainty quantification is a crucial issue in current safety analyses.

Two energy ranges have been analyzed: the Resolved Resonance Range and the continuum. The Resolved Resonance Range has been extended up to 2 MeV (against 350 keV in JEFF-3.1.1) and converted in the Reich-Moore R-matrix formalism. Data assimilation was performed for total and inelastic cross sections. As shown in figure 1, the JEFF-3.2 inelastic cross section (red curve) is much higher than in JEFF-3.1.1 above 2 MeV and in good agreement with the experimental data from IRMM [18].

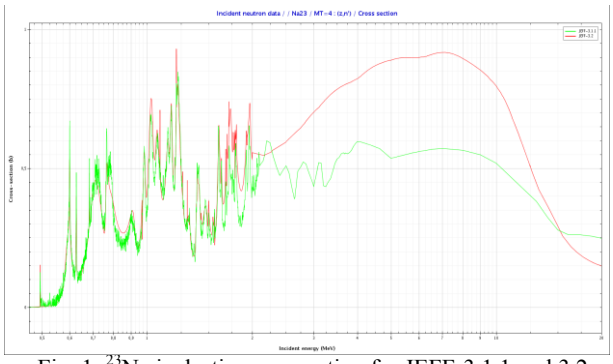

Fig. 1. ^{23}Na inelastic cross section for JEFF-3.1.1 and 3.2

II.C. Light elements and thermal data

Thermal S(α, β) data have not changed from JEFF-3.1.1. At least, no new data were proposed with the new version 3.2.

^{1}H evaluation has been taken from ENDFB-VII.1 [19] evaluation. Deuterium, ^{2}H, is a new evaluation. Cross sections have been performed by ab-initio nuclear physics calculations [20].

^{4}He, ^{9}Be, ^{16}O and ^{19}F are JEFF-3.1.1 evaluation. For ^{19}F, some light modifications concerning energy states and decay states for first and second inelastic levels have been corrected.

For carbon, the main modification concerns the thermal cross section. It has been modified from 3.36 b to 3.81 b according to the work of C.J. Diez, based on new measurements and BR1 reactor experiments analysis [21].

For nitrogen isotopes, ^{14}N and ^{15}N, ENDFB-VII.1 evaluation files have been chosen.

II.D. Absorbers

Absorbers considered here are the B, Ag, Cd, In, Gd and Hf elements.

B isotopes, ^{10}B and ^{11}B come from the ENDFB-VII.1 library.

Ag isotopes, ^{107}Ag and ^{109}Ag are new JEFF-3.2 evaluation files.

All Cd isotopes have also been re-evaluated for the new version : ^{106}Cd, ^{108}Cd, ^{110}Cd, ^{111}Cd, ^{112}Cd, ^{114}Cd and ^{116}Cd. New measurements at the IRMM facility have led to new resonance parameters.

^{113}In and ^{115}In are the same evaluations as the JEFF-3.1.1 ones. The difference comes from adding (n, γ) production data in MF=6. These new data have been calculated by TALYS code using Composite Gilbert Cameron model (CGCM) as level density model and Enhanced Generalized Lorentzian model (EGLO) as γ-strength function model below E_n=1 keV for the continuum part and EGAF data for peaks data. Above E_n=1 keV, both peaks and continuum data are calculated by TALYS with CGCM and EGLO models.

Concerning Gd element, the isotopes have different origins. ^{152}Gd and ^{154}Gd are new evaluations based on ENDFB-VII evaluation files whereas ^{156}Gd, ^{158}Gd and ^{160}Gd are JEFF-3.1.1 evaluation files. Two isotopes have been re-evaluated by the JEFF group: ^{155}Gd and ^{157}Gd.

Finally, for Hf isotopes (^{174}Hf, ^{177}Hf, ^{178}Hf, ^{179}Hf and ^{180}Hf), new measurements performed at IRMM [22] have been taken into account. ^{176}Hf has been kept from JEFF-3.1.1.

II.E. Structural elements

Structural elements modifications in JEFF-3.1.1 library are listed in Table I.

TABLE I

Structural elements modification from JEFF-3.1.1 to JEFF-3.2 libraries

Isotope	Origin	Isotope	Origin
^{24}Mg	JEFF-3.1.1+	^{58}Ni	JEFF-3.1.1
^{25}Mg	JEFF-3.1.1	^{60}Ni	JEFF-3.1.1+
^{26}Mg	JEFF-3.1.1	^{61}Ni	JEFF-3.1.1
^{27}Al	JEFF-3.1.1	^{62}Ni	JEFF-3.1.1
^{28}Si	JEFF-3.1.1	^{64}Ni	JEFF-3.1.1
^{29}Si	ENDFB-VII.1	^{63}Cu	JEFF-3.2
^{30}Si	ENDFB-VII.1	^{65}Cu	JEFF-3.2
^{31}P	TENDL-2012	^{64}Zn	TENDL-2012
^{32}S	TENDL-2012	^{66}Zn	TENDL-2012
^{35}Cl	JEFF-3.1.1	^{67}Zn	TENDL-2012
^{37}Cl	ENDFB-VII.1	^{68}Zn	TENDL-2012
^{39}K	TENDL-2012	^{70}Zn	TENDL-2012
^{40}K	TENDL-2012	^{69}Ga	TENDL-2012
^{41}K	TENDL-2012	^{71}Ga	TENDL-2012
^{40}Ca	JEFF-3.1.1	^{89}Y	JEFF-3.1.2
^{42}Ca	JEFF-3.1.1	^{90}Zr	TENDL-2012
^{43}Ca	JEFF-3.1.1	^{91}Zr	JEFF-3.1.2
^{44}Ca	JEFF-3.1.1	^{92}Zr	TENDL-2012
^{46}Ca	JEFF-3.1.1	^{94}Zr	TENDL-2012
^{48}Ca	JEFF-3.1.1	^{96}Zr	JEFF-3.1.1+
^{46}Ti	JEFF-3.1.1	^{93}Nb	ENDFB-VII.1
^{47}Ti	JEFF-3.1.1	^{92}Mo	JEFF-3.1.1
^{48}Ti	JEFF-3.1.1	^{94}Mo	JEFF-3.1.1
^{49}Ti	JEFF-3.1.1	^{95}Mo	JEFF-3.1.1
^{50}Ti	JEFF-3.1.1	^{96}Mo	JEFF-3.1.1
V	JEFF-3.1.1+	^{97}Mo	JEFF-3.1.1
^{50}Cr	JEFF-3.2	^{98}Mo	JEFF-3.1.1
^{52}Cr	JEFF-3.2	^{100}Mo	JEFF-3.1.1
^{53}Cr	JEFF-3.2	^{180}W	IAEA
^{54}Cr	JEFF-3.2	^{182}W	JEFF-3.2
^{55}Mn	JEFF-3.2	^{183}W	JEFF-3.2
^{54}Fe	JEFF-3.2	^{204}Pb	JEFF-3.1+
^{56}Fe	JEFF-3.2	^{206}Pb	JEFF-3.1+
^{57}Fe	JEFF-3.2	^{207}Pb	JEFF-3.1+
^{58}Fe	JEFF-3.2	^{208}Pb	JEFF-3.1+
^{59}Co	JEFF-3.1.1		

Many isotopes come from the JEFF-3.1.1 library and no major improvement is expected from structural elements.

When the symbol (+) appears, it means that minor changes have been done on isotope's nuclear data. Generally, it concerns corrections in the evaluation file like multiplicities, energy states, …

^{50}Cr, ^{52}Cr, ^{53}Cr and ^{54}Cr are new evaluations called KIT-2010 provided by a collaboration between KIT (Karlsuher Institut für Technoligie) and ORNL. New evaluation data rely on TALYS calculations for cross sections and on Unified Monte Carlo method for covariance data [23].

For iron, ^{54}Fe, ^{56}Fe, ^{57}Fe and ^{58}Fe have all revised gamma production data for (n, γ) reaction. The calculations are based on the same method as the one used for indium isotopes. Moreover, for ^{56}Fe minor changes have been introduced to ensure consistency in the evaluation file. New covariance data in MF=33 (groupwise cross sections covariance) replace the previous ones for reactions (n, n), (n, n') and (n, γ) and are based on integral experiment feedback (CEA mockups EOLE and MASURCA experiments).

^{55}Mn new evaluation comes from a collaborative work between IAEA, IRMM, ORNL and JSI (Jozef Stefan Institute in Slovenia). While thermal capture cross section is very well known (13.27 b), the resonance integral varies up to 20 % from one evaluation to the other. An experimental program led by IRMM in the GELINA facility has been performed to measure first resonances at 338 eV, 1098 eV and 1370 eV.

II.G. Fission products

Some efforts have been done to improve fission product cross sections. As they are not involved in this study, no particular information is given here.

III. COMPUTER CODES AND STANDARD PERTURBATION THEORY

The analysis is performed using two main neutronic codes. The first one is the Monte Carlo code TRIPOLI-4® and the second one is the deterministic S_N code SNATCH.

III.A. Monte Carlo code

TRIPOLI-4® is the French Monte Carlo code developed by CEA and written in C++. It uses continuous energy cross sections with probability tables in the Unresolved Resonance Range produced by the CALENDF code [24]. Some parallelization features are available since 1999 and the code is widely used in the fields of reactor physics, criticality and radiation shielding. It benefits also

since a few years of a depletion calculation capability. It has been validated for 20 years.

For perturbation analysis, one can use the correlated samples method implemented in TRIPOLI-4®. Recently, a PhD work [25] has introduced in the code the capability to calculate an exact continuous energy adjoint flux with the IFP method (Iterated Fission Probability). In this study, only the first method is used.

We generally use the Monte Carlo code to accurately calculate the impact on k_{eff} while using JEFF-3.2 library instead of JEFF-3.1.1.

III.B. Deterministic S_N code

SNATCH is a recent 3D S_N code developed at CEA. It is used in nuclear data validation exclusively. Its capability to calculate an accurate angular flux allows the user to perform uncertainty and sensitivity analyses

Cross sections have to be self-shielded, collapsed and homogenized first by a cell code. In the case of SNATCH, the cell code is the legacy ECCO code [26] used in the ERANOS SFR system [27].

The SNATCH code is used to calculate reactivity effect breakdown with the standard perturbation theory.

III.C. Cross sections processing

At first, it should be noted that the two copper evaluations (^{63}Cu and ^{65}Cu) cannot be used because they cannot be processed with NJOY99 [28]. Some ENDF format errors lead to NJOY failure. The last test files from JEFF-3.2T3 version are taken instead.

For all other evaluations, the cross sections have to be processed to be used in transport codes, either Monte Carlo codes or deterministic codes.

The processing consists for both code types in using NJOY to produce pointwise cross sections. First, they should be produced at 0 K from the resonance parameters with the RECONR module. Then, the cross sections have to be broadened to the desired temperatures with the BROADR module. For deterministic codes, another step is necessary to produce groupwise cross sections. The energetic meshing currently used in calculation schemes is 1968 groups for the cell code and 33 groups for the core code. This task is ensured by the GROUPR module. Another common step in processing is the generation of the probability tables with the CALENDF code.

For TRIPOLI-4® Monte Carlo code, the pointwise cross sections are directly used in PENDF format. The version of NJOY code used is 99-364. The version for CALENDF is 2005-69 revision February 2012 and the probability tables are produced using a 11277 energetic group meshing.

For cell code ECCO, the version of NJOY is also 99-364 but the CALENDF version is 2008-75. Some tests have been done to ensure that the impact of the latter is very low (a few dozen of pcm). When groupwise cross sections and probability tables are done, two other codes are necessary to produce the final ECCO library called ECCOLIB. The first code is called MERGE (version 3.8) to mix groupwise cross sections and probability tables in a same library and the last code is GECCO (version 1.5) to finalize the processing work and produce the same data in the right format for ECCO, while adding weight, atomic number, mass number, mass and energies for fission and capture, and finally decay constant.

III.D. Reactivity effect breakdown

The reactivity effect passing from operator (1) to operator (2) in the Boltzmann equation (by changing the nuclear data) can be expressed as:

$$\delta k_{1\to 2} = -\frac{\left\langle \Phi_1^+, (\Delta A - \frac{\Delta F}{k_2})\Phi_2 \right\rangle}{\left\langle \Phi_1^+, F_1\Phi_2 \right\rangle}$$

With the classical operators A and F of the Boltzmann equation:

$$(A - \frac{F}{k})\Phi = 0$$

$$A\Phi(\vec{r},\hat{\Omega},E) = \hat{\Omega}\cdot\nabla\Phi(\vec{r},\hat{\Omega},E) + \Sigma_t(\vec{r},E)\times\Phi(\vec{r},\hat{\Omega},E) - \iint \Sigma_s(\vec{r},E'\to E,\hat{\Omega}'\to\hat{\Omega})\times\Phi(\vec{r},\hat{\Omega}',E') \times dE'\times d^2\Omega'$$

$$F\Phi(\vec{r},\hat{\Omega},E) = \frac{1}{4\pi}\times\iint \chi(E'\to E)\times\nu(E')\times\Sigma_f(\vec{r},E')\times\Phi(\vec{r},\hat{\Omega}',E') \times dE'\times d^2\Omega'$$

and the scalar product definition:

$$\left\langle f,g \right\rangle = \iiint d^3r\cdot dE\cdot d^2\Omega\cdot f(\vec{r},E,\hat{\Omega})\cdot g(\vec{r},E,\hat{\Omega})$$

Φ_1^+ and Φ_2 are respectively the adjoint flux solution of adjoint Boltzmann equation in situation (1),

here JEFF-3.1.1 library, and the direct flux solution of Boltzmann equation in situation (2), here JEFF-3.2 library.

k_2 is the effective multiplication factor of situation (2), F_1 is the fission source operator in situation 1, and ΔA and ΔF are A and F operators variations passing from JEFF-3.1.1 library to JEFF-3.2 library.

III.E. Spectra characterization

Spectra are characterized in the ICSBEP database by the EALF parameter standing for Energy corresponding to Average Lethargy causing Fission. It is mathematically defined by:

$$EALF = \frac{E_0}{e^{\bar{u}}}$$

With $\bar{u}$ defined by:

$$\bar{u} = \frac{\int_u du \cdot u \cdot \Phi(u) \cdot \Sigma_f(u)}{\int_u du \cdot \Phi(u) \cdot \Sigma_f(u)}$$

IV. BENCHMARKS

The benchmarks used to evaluate the impact of the new library JEFF-3.2 are either criticality benchmarks taken from the ICSBEP database, or reactor physics benchmarks taken from the IRPHE database. Some other experimental configurations from the fast spectrum CEA mockup MASURCA, from the thermal spectrum CEA mockup EOLE are also simulated. Additionally, the Superphénix start-up tests are used. Finally, some radiation shielding experiments from the SINBAD database are considered.

IV.A. ICSBEP benchmarks

Most of the benchmarks used in this study come from the ICSBEP database. Their main advantages are that they are generally easy to model and that they do not mix too many isotopes in the same experiment.

The name of the series of experiment is defined as *<FUEL TYPE>-<CHEMICAL FORM>-<SPECTRUM>*. <FUEL TYPE> are PU for Plutonium, HEU for High Enriched Uranium, LEU for Low Enriched Uranium or MIX for mixtures. <CHEMICAL FORM> can be SOL for Solution, MET for Metal, COMP for Compound. Finally <SPECTRUM> can be FAST for Fast Spectrum (defined as more than 50% of flux has energy higher than 100 keV), INTER for Intermediate Spectrum (defined as more than 50% of flux has energy between 0.625 eV and 100 keV), THERM for thermal flux (defined as more than 50% of flux has energy lower than 0.625 eV) or MIXED when none of the previous rules can be used.

For thermal spectra, Pu cases are taken from PU-SOL-THERM series (001, 004 and 018) and U cases are taken either from the HEU-SOL-THERM series (004 for heavy water and 009 for light water) or from LEU-COMP-THERM series (006 and 007). This leads the EALF to vary from 0.05 eV to 0.37 eV for Pu cases and from 0.07 eV to 13.3 eV for U cases, in different leakage configurations. The PU-SOL-THERM 018 is a quite important series because ^{240}Pu enrichment is a little bit more than 40%. Then, the calculations for the cases of the series are very sensitive to the isotope data.

For intermediate spectra, different series are used only for U cases: HEU-MET-INTER series (001 and 006) and HEU-COMP-INTER series (003, 004 and 006 with case 6 only). These experiments cover EALF values in general from 2 keV to 81 keV, except for HEU-COMP-INTER 004 for which its value is 130 eV. Anyway, they are very useful to test the end of RRR of ^{235}U (from 10^{-5} eV to 2.25 keV) and its URR also (from 2.25 keV to 25 keV) because HEU generally means that ^{235}U content is higher than 95%.

For fast spectra, the Metal Fuel series are used. For Pu, the PU-MET-FAST experiments are modelled and calculated through the series 001, 002, 006, 010, 019, 020, 022, 023, 024, 028, 029, 037 and 041. The EALF values varies from 699 keV to 1.33 MeV for all series, except for series 041 for which spectra are softer and EALF varies from 22 keV to 180 keV. U cases are taken from series 001, 002, 004, 008, 011, 013, 014, 015, 018, 028 and 032. Their EALF values vary from 806 keV to 894 keV, except for series 004 for which it is 34 keV and 011 for which it is 30 keV. The intermediate ^{235}U enrichment or mixed enrichment experiments are the INTER-MET-FAST series (001 also called Jemima, 007 also called Big Ten, 010 and 012). Their EALF values are between 368 keV and 774 keV.

Except for the two very simple cases called Jezebel and Godiva (PU-MET-FAST 001 and HEU-MET-FAST 001 respectively) for which some deterministic calculations are performed also, TRIPOLI-4® Monte Carlo code is used for ICSBEP benchmarks.

IV.B. IRPHE benchmarks

IRPHE database benchmarks are Reactor Physics benchmarks. The experiments are more complex than the ones found in the ICSBEP database.

Two benchmarks are considered in this study: SNEAK-7 and ZPPR-10.

The first one is the SNEAK-7 experiment for which two configurations exist: A, in which fuel plates are surrounded by graphite plates and B, in which fuel plates are surrounded by fertile plates. In the two configurations, core is surrounded by a fertile blanket in all directions. One

inconvenience of this experiment is that sodium is not used.

The second one is the ZPPR-10A (figure 2 shows a figure of the ZPPR reactor). This experiment is a realistic power SFR reactor benchmark in which MOX fuel, sodium and fertile blanket coexist.

Fig. 2. ZPPR reactor view

IV.C. CEA mock-ups

Experimental programs from two CEA mock-ups are considered: EOLE and MASURCA. They are both located at Cadarache and have been working since the 1960s.

The first one is the thermal neutrons EOLE facility. The experiments that have been modelled come from the EPICURE and the MISTRAL programs:

- EPICURE UH1.2 and MH1.2 are UOX reference calculation and MOX configuration respectively of the EPICURE program. The aim of the program is to validate neutronics codes for 30% MOX core loading.

- UMZONE -REF, -AIC, -B4C and -GREY are mixed UOX and MOX fuels core with reference calculation and different absorbers configurations. UMZONE configurations are part of the EPICURE program. The particularity in these configurations is that the central zone is made of MOX fuel surrounded by a UOX zone. Figures 3 and 4 show the TRIPOLI-4® model of the UMZONE core.

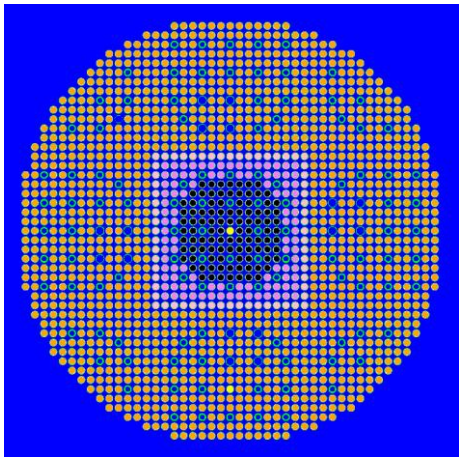

Fig. 3. EOLE UMZONE X-Y model for TRIPOLI-4®

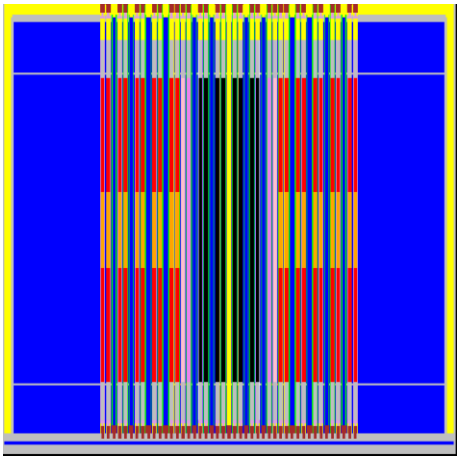

Fig. 4. EOLE UMZONE X-Z model for TRIPOLI-4®

- MISTRAL2 and MISTRAL3 are 100% MOX core with moderation ratio higher than the one of French PWRs (close to 2 considering volume ratio). It was dedicated to Japanese and French studies for 100% MOX cores. Figure 5 and 6 show a picture of the MISTRAL core and the TRIPOLI-4® model.

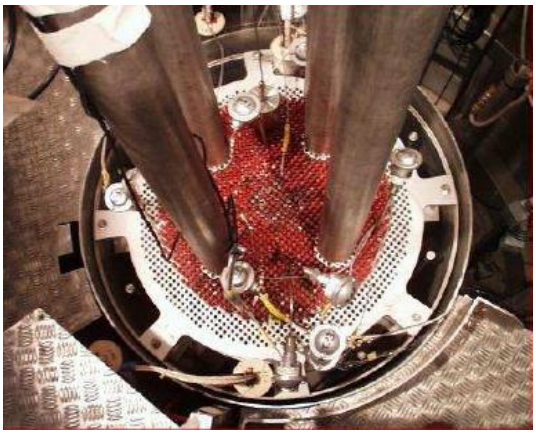

Fig. 5. EOLE MISTRAL-3 experiment view

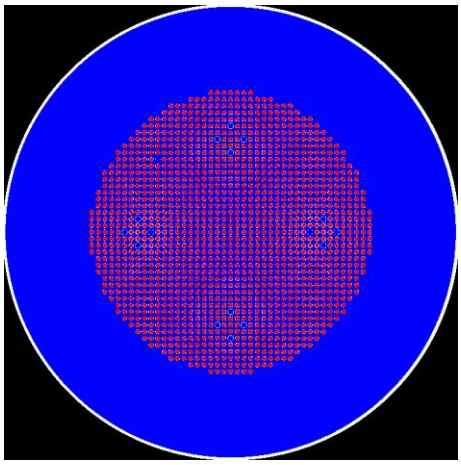

Fig. 6. EOLE MISTRAL-3 model for TRIPOLI-4®

The second one is the fast neutron mock-up MASURCA. This mock-up is used since 1960 to prepare the SFR French program, especially for Phenix [29] prototype and Superphénix industrial reactors. The experiments come from different programs led in the facility:

- MASURCA-1B is a very first experiment of the RZ program with U metal (^{235}U enrichment is 30%) and graphite as moderator with Fe_2O_3 to simulate fuel oxide.
- PRERACINE-2B is an experiment of the PRE-RACINE program dedicated to heterogeneous cores with control rods. Fuel is in some part Pu (so called ZONA1 fuel) and in other part U (so called R1 fuel with 30% ^{235}U enrichment). Sodium void effects were measured in this program.
- RACINE-1A is an experiment of the RACINE program also dedicated to heterogeneous cores with control rods. Fuel is made of Pu only (so called ZONA1 fuel).
- ZONA-2B is an experiment of the CIRANO program dedicated to the study of Pu burner cores. This configuration is a particular case with stainless steel reflector, where fertile blankets have been removed. It is particularly difficult to calculate with deterministic codes because of severe current gradients at the core/reflector interface.

IV.D. Superphénix Power reactor

Superphénix was the French industrial nuclear power plant. The first criticality occurred in September 1985. Its thermal power reached 3000 MW_{th} and its electrical power was 1200 MW_e. The Superphénix core was loaded with 364 fuel subassemblies subdivided in two different enrichment zones to flatten the power density distribution, 233 radial fertile subassemblies, 197 steel reflector subassemblies and 1076 steel radial shielding subassemblies. The fuel subassemblies were made of 271 pins in an hexagonal wrapper tube.

The zero power tests were performed from July to December 1985 and power raising tests between December 1985 and December 1986. The configuration tested here is a critical state of the core obtained during start-up tests in October 1985 and called CMP. It is made of 358 fuel subassemblies (190 inner fuel and 168 outer fuel), 225 radial blanket subassemblies, 3 safety rods and 21 control rods, 18 steel diluent subassemblies and 1326 radial shielding subassemblies. A part of the radial cross section of the TRIPOLI-4® model is shown in figure 7.

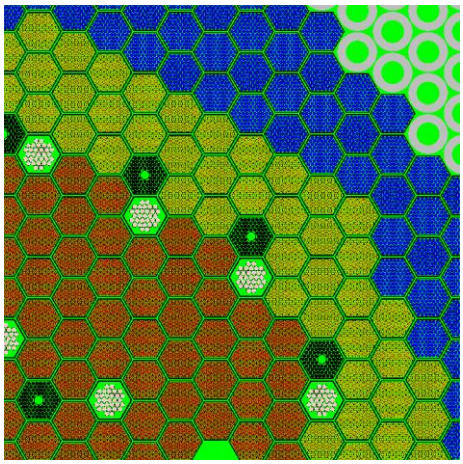

Fig. 7. TRIPOLI-4® model of Superphénix core

IV.E. SINBAD radiation shielding benchmarks

Two benchmarks have been selected in the SINBAD database. They are both calculated with the Monte Carlo code TRIPOLI-4®.

The first one is dedicated to PWRs shielding and is called REPLICA. In REPLICA, one is interested in simulating neutron transport through different blocks of iron and water. The experiment took place in the NESTOR reactor facility. Fission neutrons coming from the NESTOR reactor were thermalized through a graphite plate. Then, hitting a ^{235}U plate they were producing new fission neutrons close to the experimental disposal. The iron and water blocks were to simulate the water between core and baffle, the baffle, water between baffle and vessel, the vessel and void behind vessel.

Detectors used in the experiment are based on ^{103}Rh(n, n'), ^{115}In(n, n') and ^{32}S(n, p) reactions. The cross sections used are IRDF-2002 cross sections. All detectors are fast neutrons detectors. Fast neutrons are a crucial issue for vessel damages and its life expectancy.

A schematic view of the disposal is given in figure 8 as it is modelled with TRIPOLI-4®.

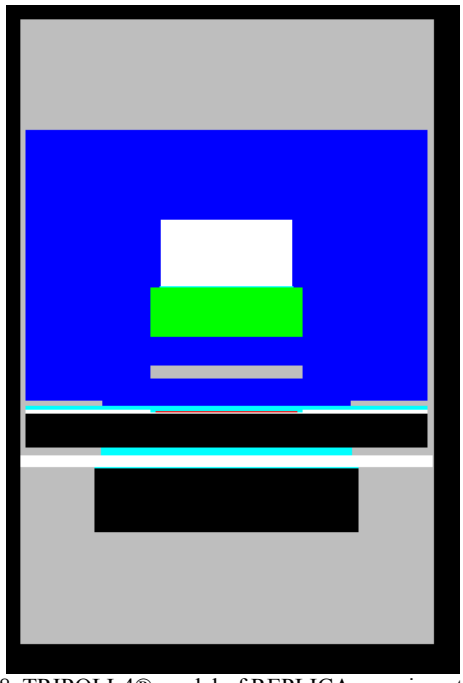

Fig. 8. TRIPOLI-4® model of REPLICA experiment

The second one is more dedicated to SFRs shielding and is called JANUS-8 experiment (see TRIPOLI-4® model in figure 9). It also took place in the NESTOR reactor facility. Instead of crossing water and steel, neutrons travel through tanks of sodium. The total length is slightly higher than 3 meters. This kind of experiment is very useful to validate calculations for secondary sodium activation in intermediate heat exchangers.

Detectors used in this experiment are ^{55}Mn(n, γ), ^{197}Au(n, γ), ^{103}Rh(n, n') and ^{32}S(n, p). The first detectors are thermal or epithermal detectors (high capture resonance around 5 eV for ^{197}Au) and the two other ones are more sensitive to fast neutrons (about 100 keV and 2 MeV respectively).

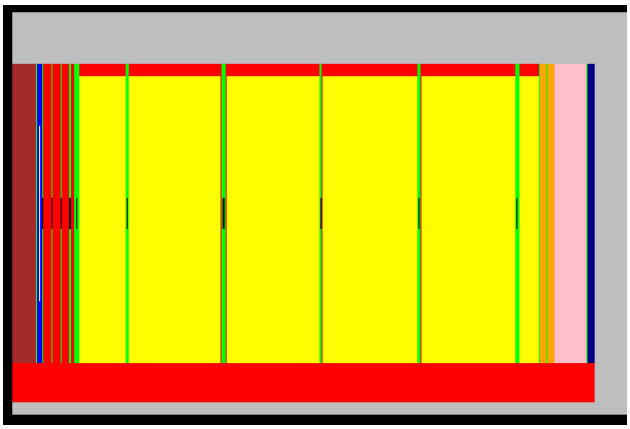

Fig. 9. TRIPOLI-4® model of JANUS-8 experiment

Figure 10 shows the different cross sections for the high energy dosimeters and for IRDFF-1.0 evaluation.

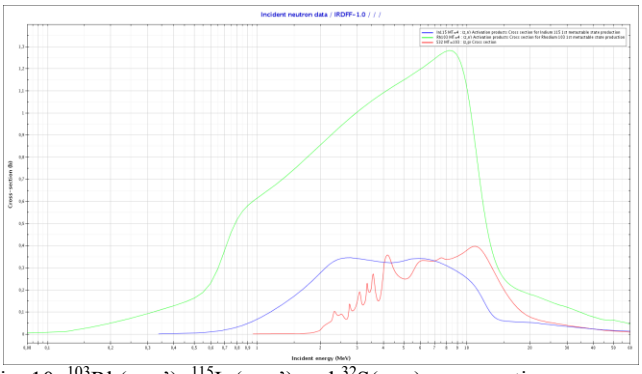

Fig. 10. ^{103}Rh(n, n'), ^{115}In(n, n') and ^{32}S(n, p) cross sections

V. RESULTS AND ANALYSIS

V. A. Thermal spectra

Thermal spectra experiments are used in this study to guarantee that the impact of the new evaluations do not degrade the good results obtained with JEFF-3.1.1 library.

Considering U criticality experiments, the results show large discrepancies passing from JEFF-3.1.1 to JEFF-3.2, for heavy water benchmarks (HEU-SOL-THERM 004) and only tens pcm negative impact for light water configuration (either HEU-SOL-THERM or LEU-SOL-THERM). Table I shows the results obtained with TRIPOLI-4®. k_{eff} uncertainties and k_{eff} discrepancies are given in pcm (10^{-5}) in all tables presenting k_{eff} values.

TABLE I

HEU-SOL-THERM results with TRIPOLI-4®

Series-Case	JEFF-3.1.1	σ	JEFF-3.2	σ	Δk_{eff}	σ
004-1	0.98496	15	0.99403	15	+907	21
004-2	0.98066	15	0.98896	15	+830	21
004-3	0.98785	16	0.99508	15	+723	22
004-4	0.99021	16	0.99721	15	+720	22
004-5	0.98912	16	0.99530	16	+618	23
004-6	0.98575	16	0.99201	16	+626	23
009-1	1.00071	16	1.00059	16	-12	23
009-2	1.00140	16	1.00118	16	-22	23
009-3	1.00119	16	1.00068	16	-51	23
009-4	0.99555	16	0.99493	16	-62	23

The heavy water configurations HEU-SOL-THERM 004 with the new ^{2}H evaluation file show about +800 pcm impact with the use of the new version JEFF-3.2. A breakdown analysis has been performed using on one hand the correlated samples method in TRIPOLI-4® for the (n, n), (n, 2n) and (n, γ) cross sections. And on the other hand the anisotropy file used by TRIPOLI-4® to sample angular

deviation for the (n, n) reaction has been replaced to evaluate anisotropy effect. The results are shown in Table II for case 1.

TABLE II

Perturbation analysis for the new ^{2}H JEFF-3.2 evaluation

Perturbation	Δk_{eff}	σ
(n, n) cross section	-110	6
(n, 2n) cross section	+30	6
(n, g) cross section	+0	0
(n, n) and (n, 2n) anisotropy	+864	17

The results show that the main effect comes from the anisotropy of the elastic scattering. In comparison, cross sections themselves have low effect.

TABLE III

LEU-COMP-THERM results with TRIPOLI-4®

Series-Case	JEFF-3.1.1	σ	JEFF-3.2	σ	Δk_{eff}	σ
006-1	1.00039	3	0.99985	3	-54	4
006-9	1.00043	3	0.99976	3	-67	4
007-1	0.99795	3	0.99756	3	-39	4
007-5	0.99770	3	0.99722	3	-48	4

The following figure shows the results for HEU-SOL-THERM and LEU-COMP-THERM series. They are plotted for both libraries and for experimental values as k_{eff} minus one values and expressed in pcm. All the figures showing k_{eff} results in the article will follow the same rules.

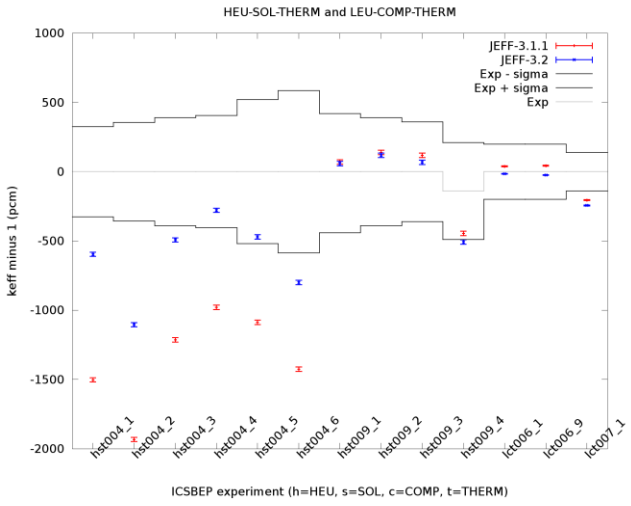

Fig. 11. ICSBEP Uranium benchmarks in thermal spectrum TRIPOLI-4® results

For Pu series, the JEFF-3.2 library has generally low impact on the k_{eff} values. For all series, except PU-SOL-THERM 018, only tens of pcm variations are observed.

TABLE IV

PU-SOL-THERM results with TRIPOLI-4®

Series-Case	JEFF-3.1.1	σ	JEFF-3.2	σ	Δk_{eff}	σ
001-1	1.00116	15	1.00045	15	-71	21
001-2	1.00320	15	1.00304	15	-16	21
001-3	1.00613	15	1.00534	15	-79	21
001-4	1.00014	15	0.99988	15	-26	21
001-5	1.00449	15	1.00387	15	-62	21
001-6	1.00609	15	1.00610	16	+1	21
004-1	1.00001	13	1.00036	13	+35	18
004-2	0.99503	13	0.99491	13	-12	18
004-3	0.99707	13	0.99734	13	+27	18
004-4	0.99502	13	0.99484	13	-18	18
004-5	0.99577	13	0.99655	13	+78	18
004-6	0.99782	13	0.99804	13	+22	18
004-7	1.00167	13	1.00192	13	+25	18
004-8	0.99757	13	0.99771	13	+14	18
004-9	0.99667	14	0.99692	13	+25	19
004-10	0.99838	14	0.99822	14	-16	20
004-11	0.99724	14	0.99664	14	-60	20
004-12	0.99906	13	0.99959	13	+53	18
004-13	0.99665	13	0.99673	13	+8	18
018-1	1.01102	14	1.00858	13	-244	19
018-2	1.01393	13	1.01138	14	-255	19
018-3	1.01112	13	1.00939	13	-173	18
018-4	1.00897	13	1.00676	13	-221	18
018-5	1.00720	13	1.00595	13	-125	18
018-6	1.00515	13	1.00390	13	-125	18
018-7	1.00420	13	1.00288	13	-132	18
018-8	1.00342	12	1.00230	12	-112	17
018-9	1.00132	12	1.00060	12	-72	17

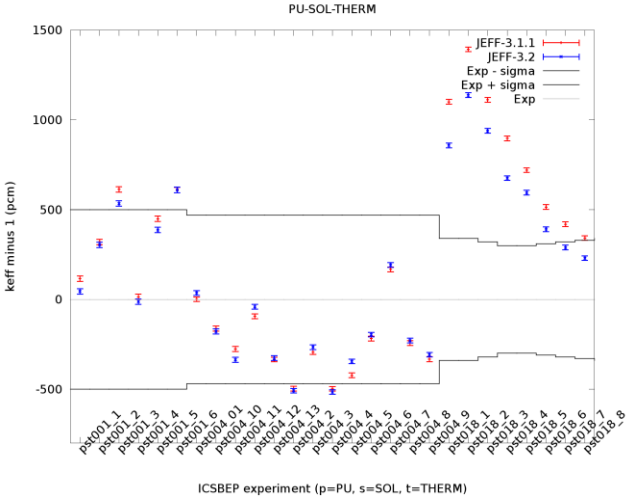

Fig. 12. ICSBEP Plutonium benchmarks in thermal spectrum TRIPOLI-4® results

Particular attention should be paid to series 018. It can be seen that the new ^{240}Pu evaluation improves the

results of the series but a general tendency of overestimation remains.

While considering the EOLE experiments, some important discrepancies appear for high Am content fuels. The results are given in table V.

TABLE V

EOLE benchmarks results with TRIPOLI-4®

Experiment	JEFF-3.1.1	σ	JEFF-3.2	σ	Δk_{eff}	σ
EPICURE program						
UH-1.2	1.00396	6	1.00481	6	+85	8
MH-1.2	1.00101	6	0.99910	6	-191	8
UMZONE-Ref	1.00523	3	1.00497	3	-26	4
UMZONE-AIC	1.00395	3	1.00394	4	-1	5
UMZONE-B_4C	1.00362	4	1.00365	4	+3	6
UMZONE-Grey	1.00358	3	1.00343	3	-15	4
MISTRAL program						
MISTRAL2	1.00616	6	1.00153	6	-463	8
MISTRAL3	1.00728	6	1.00229	6	-499	8

To note that the pilot rod has not been modelled in TRIPOLI-4® and its weight is about tens of pcm.

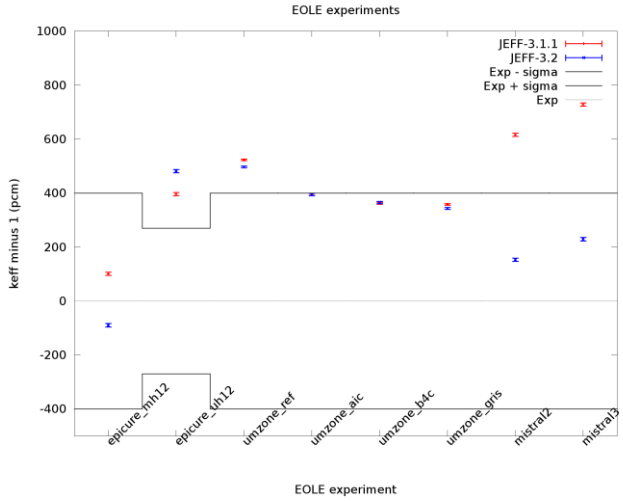

Fig. 13. EOLE experiments in thermal spectrum TRIPOLI-4® results

Some Monte Carlo calculation have been performed with correlated samples perturbation method to highlight the impact of ^{241}Am alone. Calculations are performed without Probability Tables, and then just take into account modification of Resolved Resonance Range and High Energy Range. The results are shown in Table VI.

TABLE VI

Perturbation calculations with correlated samples method for ^{241}Am for MISTRAL-2 and MISTRAL-3 experiments

Cross section modification JEFF-3.1.1→JEFF-3.2	Δρ	σ
MISTRAL-2 (n, γ)	-559	1
MISTRAL-3 (n, γ)	-589	1

It can be seen from the previous table that the effect for ^{241}Am is due to radiative capture cross section. In order to better understand this effect, the energetic impact on the 172 groups XMAS meshing is plotted in figure 14.

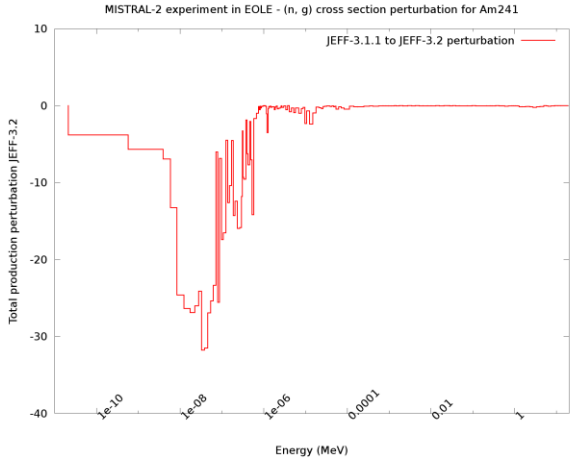

Fig. 14. ^{241}Am (n, γ) perturbation effect for MISTRAL-2

As it has been stated in the paragraph II.A, the results are consistent with the increase of about 15% of the thermal capture cross section. The effect is the most important in the thermal energy region (around 0.025 eV).

V. B. Intermediate spectra

Benchmarks in intermediate spectra are very useful to evaluate the accuracy of nuclear data in the Unresolved Resonance Range. Actually, according to ICSBEP classification, the intermediate spectra benchmarks are the ones for which the part of neutron flux comprised between 0.625 eV and 100 keV is higher than 50%.

One experiment is particularly interesting because it tests the ^{235}U Unresolved Resonance Range. This experiment is identified HEU-MET-INTER 006, or Zeus, experiment. Four cases are available in this series and the EALF of these cases are 2 keV, 9 keV, 22 keV and 81 keV, and the experimental uncertainty is very low (80 pcm).

The results of HEU-MET-INTER 006 are given in table VII.

TABLE VII

HEU-MET-INTER 006 (or Zeus experiment) results with TRIPOLI-4®

Case	Exp	σ	JEFF-3.1.1	σ	JEFF-3.2	σ	Δk_{eff}
1	0.99770	80	0.99226	13	0.99735	13	+509
2	1.00010	80	0.99515	12	1.00065	12	+550
3	1.00150	80	0.99805	12	1.00436	12	+631

4	1.00160	80	1.00546	12	1.01218	12	+672

Figure 15 shows the results for all intermediate series, and the HEU-MET-INTER 006 can be seen on the right side of the figure.

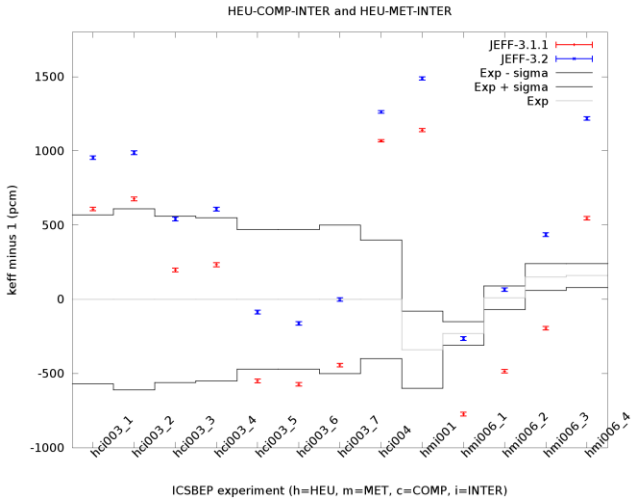

Fig. 15. ICSBEP Uranium benchmarks in intermediate spectrum TRIPOLI-4® results

Other results relative to intermediate spectra experiments are reported in table VIII and table IX.

TABLE VIII

HEU-MET-INTER results with TRIPOLI-4®

Series-Case	JEFF-3.1.1	σ	JEFF-3.2	σ	Δk_{eff}	σ
001	1.01142	11	1.01489	11	+347	16

TABLE IX

HEU-COMP-INTER results with TRIPOLI-4®

Series-Case	JEFF-3.1.1	σ	JEFF-3.2	σ	Δk_{eff}	σ
003-1	1.00608	12	1.00954	12	+346	17
003-2	1.00677	13	1.00988	13	+311	18
003-3	1.00197	13	1.00541	13	+344	18
003-4	1.00233	13	1.00607	13	+374	18
003-5	0.99450	13	0.99913	13	+463	18
003-6	0.99427	12	0.99837	12	+410	17
003-7	0.99557	12	0.99999	12	+442	17
004	1.01068	8	1.01263	8	+195	11
006	0.96859	13	0.97793	13	+934	18

The use of JEFF-3.2 evaluation files leads to an increase of about 300 pcm to 600 pcm in general. For particular ZEUS experiments, the first two cases with 2 keV and 9 keV as mean energies are better calculated. But the third and fourth cases are not. This could indicate that some improvements have been achieved in the lower part of the Unresolved Resonance Range but that some work is still necessary in the upper part.

V. C. Fast spectra

For fast spectra, we consider first the ICSBEP experiments. Results are shown in Table X for U and in Table XV for Pu.

TABLE X

HEU-MET-FAST results with TRIPOLI-4®

Series-Case	JEFF-3.1.1	σ	JEFF-3.2	σ	Δk_{eff}	σ
001	0.99661	9	1.00090	9	+429	13
002-1	1.00108	10	1.00439	10	+331	14
002-2	1.00171	9	1.00494	10	+323	13
002-3	0.99972	10	1.00316	9	+344	14
002-4	0.99933	9	1.00230	9	+297	13
002-5	0.99942	13	1.00272	13	+330	18
002-6	1.00096	14	1.00412	13	+316	19
004	0.99824	12	1.00135	12	+311	17
008	0.99282	8	0.99645	9	+363	12
011	0.99549	12	0.99868	12	+319	17
013	0.99310	9	0.99613	9	+303	13
014	0.99674	9	0.99926	9	+252	13
018	0.99721	9	1.00113	9	+392	13
028	1.00211	9	1.00557	10	+346	13
032-1	1.00324	9	1.00629	9	+305	13
032-2	1.00382	9	1.00689	9	+307	13
032-3	0.99913	9	1.00220	9	+307	13
032-4	0.99921	9	1.00253	9	+332	13

The consistency with experiment is better for JEFF-3.2 library. The impact of this new library is about +300 to +400 pcm as shown in figure 16.

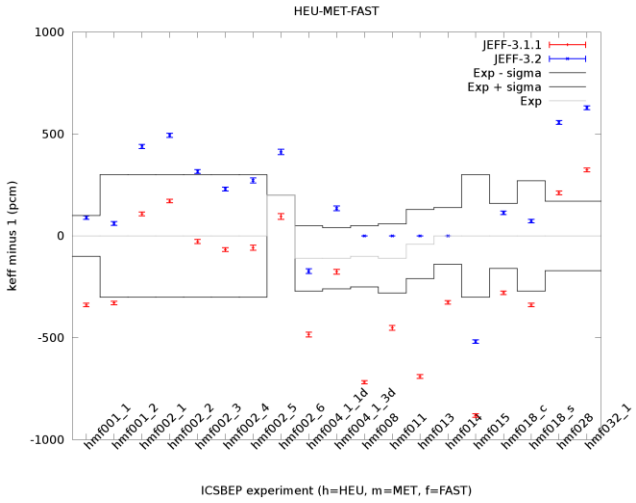

Fig. 16. ICSBEP Highly Enriched Uranium benchmarks in fast spectrum TRIPOLI-4® results

A reactivity effect breakdown is performed for HEU-MET-FAST 001 with SNATCH deterministic code. The calculations have used P5 anisotropy cross sections to ensure a correct calculation of the k_{eff} value. Results show that ^{235}U data have been drastically modified in the High Energy Range.

TABLE XI

SNATCH and TRIPOLI-4® k_{eff} comparison for HEU-MET-FAST 001 (simplified model)

Code	k_{eff}	σ
TRIPOLI-4®	1.00061	9
ECCO/SNATCH	0.99954	-

The discrepancy between SNATCH calculation and TRIPOLI-4® calculation is about 100 pcm. This level is quite acceptable since the problem is very anisotropic and the spectrum very hard. The JEFF-3.1.1 to JEFF-3.2 effect is about +429 pcm (+/- 13 pcm) for TRIPOLI-4® whereas it is +394 for SNATCH code. This result is very satisfying. The reactivity effect breakdown is shown in Table XII.

TABLE XII

JEFF-3.1.1 to JEFF-3.2 reactivity effect breakdown calculated for HEU-MET-FAST 001 with SNATCH

Iso.	(n,γ)	ν	χ	(n,f)	(n,n)	(n,n')	(n,xn)	Sum
^{234}U	+6	+1	0	+5	-4	-5	0	3
^{235}U	+104	-829	+2	+317	+62	+562	+183	+401
^{238}U	+3	-28	0	+21	-13	+7	+7	-3

The effect, due to ^{235}U as it is 95% of the material content, is due to fission, inelastic scattering and (n, xn) diffusions for the positive part (about +1200 pcm) and to ν for the negative part (about -800 pcm).

Some intermediate enrichment benchmarks have also been calculated. The main ones are the JEMIMA experiment INTER-MET-FAST 001 and the BIG TEN experiment INTER-MET-FAST 007.

TABLE XIII

INTER-MET-FAST results with TRIPOLI-4®

Series-Case	JEFF-3.1.1	σ	JEFF-3.2	σ	Δk_{eff}	σ
001-2	0.99813	9	1.00137	9	+324	13
001-3	0.99744	9	1.00085	9	+341	13
001-4	0.99818	9	1.00200	9	+382	13
007	0.99855	9	1.00507	9	+652	13
010	0.99195	2	0.99801	9	+606	13
012	1.00250	9	1.00657	9	+407	13

The increase in k_{eff} values can be partially attributed to the ^{235}U new nuclear data in the high energy range. This effect is very close to the +300 / +400 pcm observed in the HEU-MET-FAST benchmarks. The effect is even higher for INTER-MET-007 and 010 where it can reach more than 600 pcm. To better understand the origin of the effect, some TRIPOLI-4® calculations have been performed mixing JEFF-3.1.1 library and JEFF-3.2 library. First, ^{235}U from JEFF-3.2 has been introduced and then ^{235}U and ^{238}U together.

TABLE XIV

JEFF-3.1.1 to JEFF-3.2 reactivity effect breakdown calculated for INTER-MET-FAST 007 experiment with TRIPOLI-4®

Isotope	k_{eff}	σ	Δk_{eff}	σ
JEFF-3.1.1	0.99855	9		
^{235}U JEFF-3.2	0.99969	9	+114	13
^{238}U JEFF-3.2	1.00367	9	+512	13

The ^{235}U effect is consistent with the lower enrichment compared to HEU-MET-FAST benchmarks. It can be observed that ^{238}U part is very important.

Some correlated samples calculations show that the (n, f) and (n, γ) cross sections modifications for ^{238}U have a huge positive impact. The more important cross section is the capture cross section, leading to about +950 pcm.

Figure 17 shows the fission cross sections for both libraries between 500 keV to 10 MeV. Figure 18 shows the capture cross sections for JEFF-3.1.1 and JEFF-3.2 from 100 keV to 2 MeV. The discrepancy in this energy range is important enough to produce a reactivity effect. The benchmark INTER-MET-FAST 007 has a very hard spectrum with 80% of fissions produced above 100 keV and more than 12% of fissions coming from ^{238}U. This explains the importance of ^{238}U fission cross section.

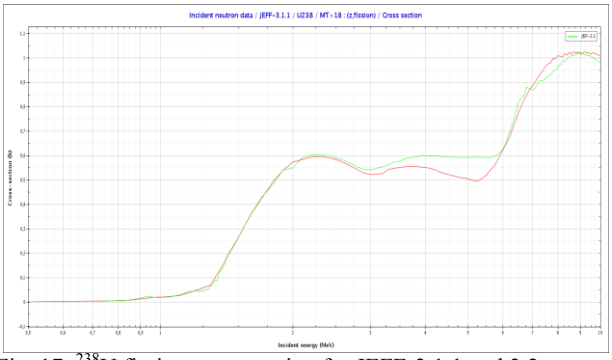

Fig. 17. ^{238}U fission cross section for JEFF-3.1.1 and 3.2

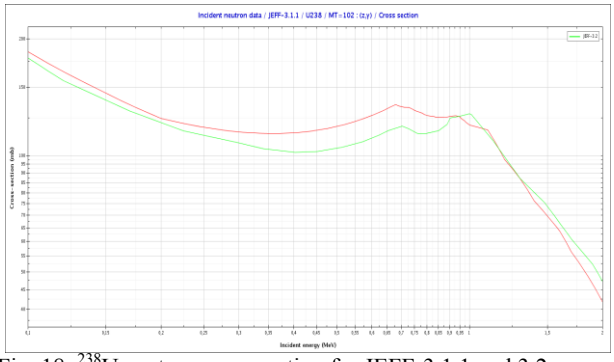

Fig. 18. ^{238}U capture cross section for JEFF-3.1.1 and 3.2

The following figure present the results obtained with TRIPOLI-4® and the two libraries for all the INTER-MET-FAST series.

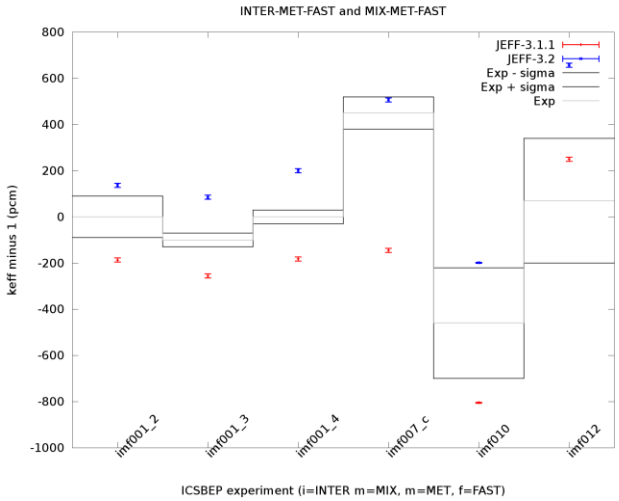

Fig. 19. ICSBEP Intermediate Enriched Uranium benchmarks in fast spectrum TRIPOLI-4® results

For Pu cases, the effect of the JEFF-3.2 library is lower.

TABLE XV

PU-MET-FAST results with TRIPOLI-4®

Series-Case	JEFF-3.1.1	σ	JEFF-3.2	σ	Δk_{eff}	σ
001	1.00023	8	1.00052	8	+29	11
002	1.00433	8	1.00283	8	-150	11
006	1.00340	9	1.00318	9	-22	13
010	1.00158	8	1.00135	9	-23	12
019	0.99964	9	0.99896	9	-68	13
020	1.00066	9	1.00010	9	-56	13
022	0.99811	8	0.99891	8	+80	11
023	0.99860	8	0.99922	8	+62	11
024	0.99981	9	1.00066	9	+85	13
028	1.00031	8	1.00031	9	+0	13
029	0.99727	8	0.99679	8	-48	11
037-1	1.00177	2	1.00232	2	+55	3
037-2	0.99925	2	0.99995	2	+70	3
037-3	0.99718	2	0.99786	2	+68	3
037-4	1.00053	2	1.00124	2	+71	3
037-5	0.99903	2	1.00012	2	+109	3
037-6	0.99843	2	0.99958	2	+115	3
037-7	0.99960	2	1.00082	2	+122	3
037-8	0.99912	2	1.00046	2	+134	3
037-9	0.99780	2	0.99917	2	+137	3
037-10	1.00093	2	1.00217	2	+124	3
037-11	0.99374	2	0.99438	2	+64	3
037-12	0.99783	3	0.99900	3	+117	4
037-13	0.99870	3	1.00012	3	+142	4
037-14	1.00220	3	1.00356	3	+136	4
037-15	1.00121	3	1.00252	3	+131	4
037-16	0.99531	3	0.99643	3	+112	4
041	1.00892	9	1.00799	9	-93	13

The effect of the JEFF-3.2 library is generally low, lower or close to 100 pcm.

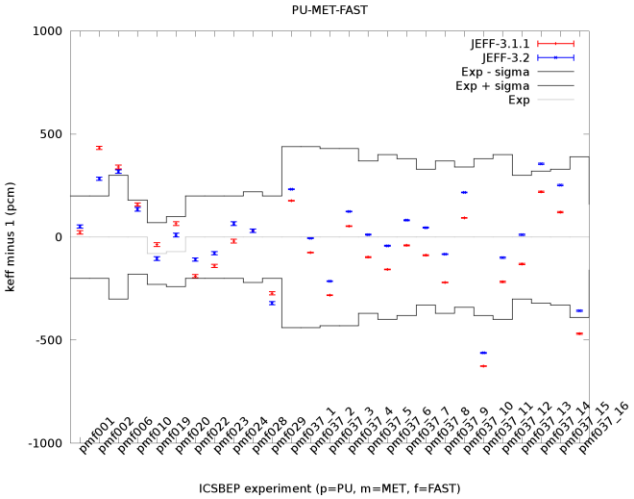

Fig. 20. ICSBEP Plutonium benchmarks in fast spectrum TRIPOLI-4® results

In order to better understand which nuclear data is involved in the effect and to be sure that compensations do not occur, a reactivity effect breakdown is proposed in Table XVI for PU-MET-FAST 001 series.

TABLE XVI

JEFF-3.1.1 to JEFF-3.2 reactivity effect breakdown calculated for PU-MET-FAST 001 with SNATCH

Iso.	(n,γ)	ν	χ	(n,f)	(n,n)	(n,n')	(n,xn)	Sum
^{239}Pu	-73	+100	+5	-63	-54	+150	+29	+94
^{240}Pu	-14	-70	-2	+27	-5	+7	0	-55

The global difference in k_{eff} for SNATCH calculations is 63 pcm because Ga isotopes have not been taken into account in the previous table. As for HEU-MET-FAST 001

calculations, SNATCH calculation is also about 100 pcm lower than TRIPOLI-4® one.

Considering mock-up experiments (MASURCA or IRPHE), TRIPOLI-4® and SNATCH calculations have been performed.

TABLE XVII

Fast spectra mock-up TRIPOLI-4® results

Experiment	JEFF-3.1.1	σ	JEFF-3.2	σ	Δk_{eff}	σ
MASURCA1B	1.00458	21	1.00948	21	+490	29
PRERACINE2B	1.00402	3	1.00068	3	-334	4
RACINE1A	1.00028	2	0.99456	2	-572	3
ZONA2B	1.00803	3	1.00057	4	-746	5
SNEAK7A-RZ	1.01019	9	1.00365	9	-654	13
SNEAK7B-RZ	1.00459	8	0.99992	8	-467	11
ZPPR10A	1.00297	2	0.99540	2	-757	3

To note that like for EOLE experiments the pilot rod has not been modelled in TRIPOLI-4® and its weight is about tens of pcm.

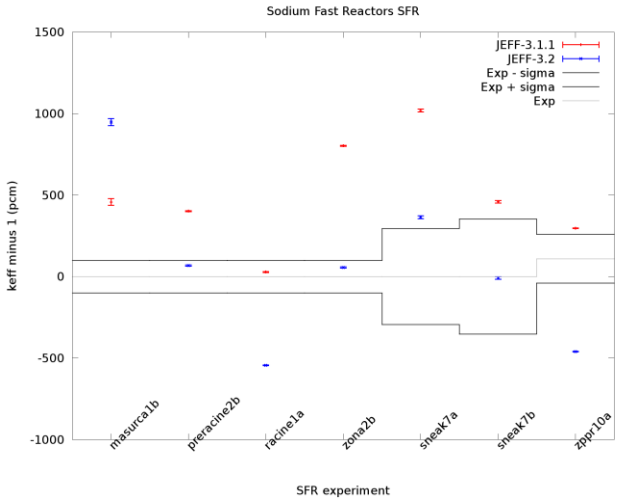

Fig. 21. MASURCA and IRPHE benchmarks in fast spectrum TRIPOLI-4® results

For Uranium configuration MASURCA-1B, the same positive effect as the one for ICSBEP fast spectra benchmarks applies: about +400 pcm. On the contrary, large negative discrepancies can be observed between JEFF-3.1.1 results and JEFF-3.2 results for MOX configurations. The effect is negative and results are closer to the experimental results (experimental k_{eff} is 1.). The main isotopes involved in this effect are ^{239}Pu, ^{23}Na, ^{238}U and ^{240}Pu. Tables XVIII and XIX show the reactivity effect breakdown obtained with SNATCH for SNEAK-7A and ZPPR-10A experiments. Tables XX and XXI show isotope effects obtained with TRIPOLI-4® code for the PRERACINE-2B and ZPPR-10A experiments: isotopes are one by one taken from the new library, first ^{23}Na, then ^{239}Pu, ^{238}U and ^{240}Pu.

TABLE XVIII

JEFF-3.1.1 to JEFF-3.2 reactivity effect breakdown calculated for SNEAK-7A with SNATCH

Iso.	(n,γ)	ν	χ	(n,f)	(n,n)	(n,n')	(n,xn)	Sum
^{239}Pu	-168	+435	+5	-561	+7	-1	0	-283
^{238}U	-35	-397	-4	+236	+85	+6	-45	-154
^{240}Pu	-41	-52	-2	+28	0	-1	0	-68

TABLE XIX

JEFF-3.1.1 to JEFF-3.2 reactivity effect breakdown calculated for ZPPR-10A with SNATCH

Iso.	(n,γ)	ν	χ	(n,f)	(n,n)	(n,n')	(n,xn)	Sum
^{23}Na	-21	0	0	0	-88	-204	0	-313
^{239}Pu	-125	+482	+4	-407	0	-7	-2	-55
^{238}U	-78	-409	-2	+230	-14	+5	-89	-357
^{240}Pu	-58	-63	-2	+37	0	-2	-1	-89

Note that reactivity effect breakdown is consistent with difference of direct calculated k_{eff} except when inserting ^{53}Cr which introduces a bias close to 40 pcm. This should be checked in future works.

TABLE XX

JEFF-3.1.1 to JEFF-3.2 reactivity effect breakdown calculated for PRERACINE-2B experiment with TRIPOLI-4®

Isotope	k_{eff}	σ	Δk_{eff}	σ
JEFF-3.1.1	1.00402	3		
^{23}Na JEFF-3.2	1.00131	3	-271	4
^{239}Pu JEFF-3.2	1.00347	3	-55	4
^{238}U JEFF-3.2	1.00331	3	-71	4
^{240}Pu JEFF-3.2	1.00300	3	-102	4

TABLE XXI

JEFF-3.1.1 to JEFF-3.2 reactivity effect breakdown calculated for ZPPR-10A experiment with TRIPOLI-4®

Isotope	k_{eff}	σ	Δk_{eff}	σ
JEFF-3.1.1	1.00297	2		
^{23}Na JEFF-3.2	0.99996	2	-301	3
^{239}Pu JEFF-3.2	1.00121	2	-176	3
^{238}U JEFF-3.2	1.00170	2	-127	3
^{240}Pu JEFF-3.2	1.00204	2	-93	3

These results are not always consistent with the SNATCH reactivity effect breakdown (for ^{239}Pu and ^{238}U). The method is not the same to evaluate each isotope contribution and can explain these inconsistencies.

All the results show anyway that sodium plays a huge role in the k_{eff} decrease and that the other important isotopes are ^{238}U, ^{239}Pu and ^{240}Pu.

Finally, a critical mass benchmark for Superphénix reactor has been calculated with TRIPOLI-4® and the two libraries. The results are shown in Table XXII.

TABLE XXII

Superphénix start-up tests TRIPOLI-4® results

Experiment	JEFF-3.1.1	σ	JEFF-3.2	σ	Δk_{eff}	σ
Superphénix	1.00733	3	0.99947	2	-786	4

Like for the previous fast neutron SFR configurations, the calculated k_{eff} value is highly improved. Some further analyses on the whole commissioning tests will validate more precisely the JEFF-3.2 library (control rod worth, power density distribution).

V. D. Radiation shielding benchmarks

For the REPLICA experiment, no main change is expected. Actually, water data and steel data have not changed much in the JEFF-3.2 library. Results obtained with the different nuclear data are shown in Table XXIII (D is the distance to the fission plate).

TABLE XXIII

REPLICA reaction rates TRIPOLI-4® results

D (cm)	JEFF-3.1.1	σ (%)	JEFF-3.2 -JEFF-3.1.1 (%)	σ (%)
^{103}Rh(n, n') Reaction rate (Bq/at)				
1.91	1.71E-020	0.02	-0.01	0.03
7.41	3.35E-021	0.03	-0.39	0.04
12.41	1.22E-021	0.02	-0.66	0.03
14.01	1.06E-021	0.03	-0.56	0.04
19.91	4.00E-022	0.03	-0.01	0.05
25.41	1.00E-022	0.03	-0.04	0.04
30.41	4.07E-023	0.04	-0.22	0.06
39.01	1.91E-023	0.04	-0.33	0.06
49.61	5.41E-024	0.06	-0.36	0.08
58.61	1.64E-024	0.04	-0.28	0.05
^{115}In(n, n') Reaction rate (Bq/at)				
39.01	3.71E-024	0.05	-0.31	0.07
49.61	7.71E-025	0.06	-0.27	0.09
58.61	2.19E-025	0.04	-0.22	0.05
^{32}S(n, n') Reaction rate (Bq/at)				
39.01	9.32E-025	0.05	-0.26	0.08
49.61	1.34E-025	0.06	-0.09	0.12
58.61	3.66E-026	0.04	-0.10	0.09

The JEFF-3.1.1 and JEFF-3.2 results are compared to the experimental values in figure 22. The C/E values are plotted with one standard deviation. This standard deviation is a combination of the statistical uncertainty and the experimental uncertainty.

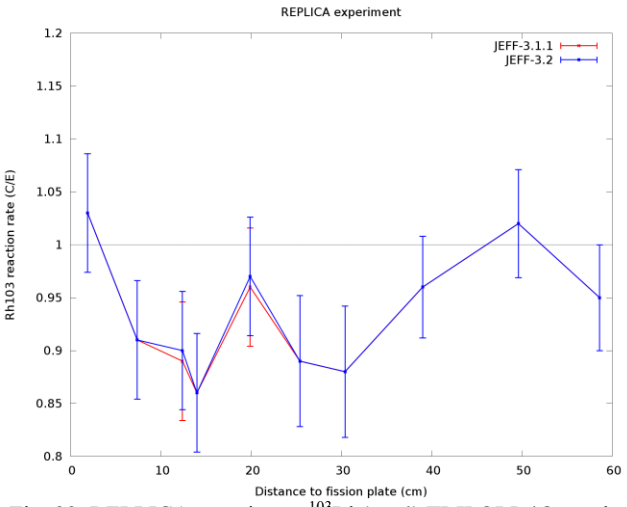

Fig. 22. REPLICA experiment ^{103}Rh(n, n') TRIPOLI-4® results

For the JANUS-8 experiment, more changes are expected because ^{23}Na has been re-evaluated in the whole energy domain. Figures 23 and 24 show TRIPOLI-4® results compared with experiment.

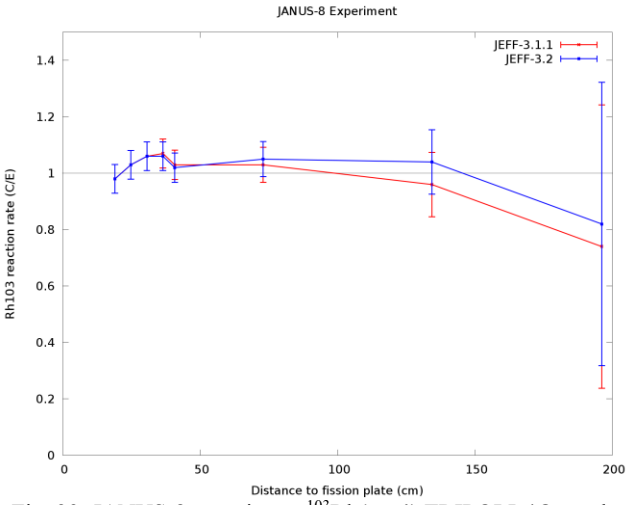

Fig. 23. JANUS-8 experiment ^{103}Rh(n, n') TRIPOLI-4® results

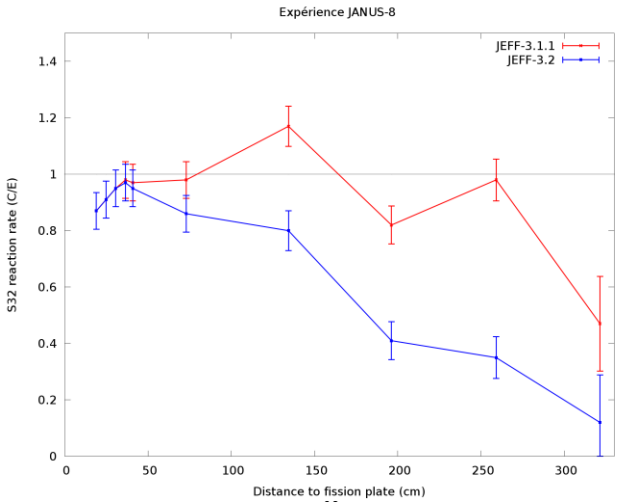

Fig. 24. JANUS-8 experiment ^{32}S(n, p) TRIPOLI-4® results

The ^{103}Rh reaction rate results do not significantly change while changing nuclear data library. ^{32}S reaction rates are very different. The JEFF-3.2 results are far lower than the JEFF-3.1.1 ones and the difference increases with neutron penetration through sodium. The explanation is the modification of the inelastic scattering cross section above 2 MeV. Figure 1 shows that these cross sections have been increased by 30-35% above 2 MeV where the sulfur detector is very sensitive. While crossing the sodium tanks, neutrons are more slowed down by inelastic scattering and their energy falls under the 2 MeV more quickly with the JEFF-3.2 nuclear data. After crossing 3 meters sodium, the result is about 0.2 times the experimental value. For deep penetration of fast neutrons in sodium, the new evaluation does not improve JANUS-8 results.

VI. CONCLUSION

The JEFF-3.2 evaluation has been shortly described and widely tested in this study. Some criticality experiments, some reactor physics experiments and finally some radiation shielding experiments have been calculated with the libraries JEFF-3.1.1 and JEFF-3.2 to evaluate the impact of the new nuclear data provided by the JEFF group. Different benchmarks related to various neutron spectra from thermal to fast have been investigated, even if the aim of the work is to focus on SFR like benchmarks, at least fast spectra benchmarks.

Concerning thermal spectra benchmarks, either for ICSBEP or EOLE configurations, it has been shown that the new JEFF-3.2 library does not degrade the good results obtained with the JEFF-3.1.1 library. For PWR radiation shielding benchmark, no impact has been observed.

For fast spectra benchmarks, the impact of changing the library is much more important, either for Uranium cases or for Plutonium cases. For Uranium, many ICSBEP configurations calculations behave better with the new data. Actually, the GODIVA experiment or BIG TEN are better predicted with JEFF-3.2 because of changes in the high energy range of ^{235}U and ^{238}U. For Plutonium, many MASURCA mockup experiments had calculation results too far from the experimental values for k_{eff}. With the new library, both ^{23}Na, ^{238}U, ^{239}Pu and ^{240}Pu contribute to the decrease of this one and the better prediction. For JANUS-8 radiation shielding experiment, the high energy detector ^{32}S calculation is severely degraded and more investigations will be necessary to state on the quality of the data in this energy range.

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