JEFF-3.1, ENDF/B-VII and JENDL-3.3 Critical Assemblies Benchmarking With the Monte Carlo Code TRIPOLI

Jean-Christophe Sublet

***Abstract*—ENDF/B-VII.0, the first release of the ENDF/B-VII nuclear data library, was formally released in December 2006. Prior to this event the European JEFF-3.1 nuclear data library was distributed in April 2005, while the Japanese JENDL-3.3 library has been available since 2002. The recent releases of these neutron transport libraries and special purpose files, the updates of the processing tools and the significant progress in computer power and potency, allow today far better leaner Monte Carlo code and pointwise library integration leading to enhanced benchmarking studies. A TRIPOLI-4.4 critical assembly suite has been set up as a collection of 86 benchmarks taken principally from the International Handbook of Evaluated Criticality Benchmarks Experiments (2006 Edition). It contains cases for a variety of U and Pu fuels and systems, ranging from fast to deep thermal solutions and assemblies. It covers cases with a variety of moderators, reflectors, absorbers, spectra and geometries. The results presented show that while the most recent library ENDF/B-VII.0, which benefited from the timely development of JENDL-3.3 and JEFF-3.1, produces better overall results, it suggest clearly also that improvements are still needed. This is true in particular in Light Water Reactor applications for thermal and epithermal plutonium data for all libraries and fast uranium data for JEFF-3.1 and JENDL-3.3. It is also true to state that other domains, in which Monte Carlo code are been used, such as astrophysics, fusion, high-energy or medical, radiation transport in general benefit notably from such enhanced libraries. It is particularly noticeable in term of the number of isotopes, materials available, the overall quality of the data and the much broader energy range for which evaluated (as opposed to modeled) data are available, spanning from meV to hundreds of MeV. In pointing out the impact of the different nuclear data at the library but also the isotopic levels one could not help noticing the importance and difference of the compensating effects that result from their single usage. Library differences are still important but tend to diminish due to the ever increasing and beneficial worldwide collaboration in the field of nuclear data measurement and evaluations.**

***Index Terms*—Benchmarking, ENDF/B-VII, JEFF-3.1, JENDL-3.3, Monte Carlo, TRIPOLI-4.4.**

I. Introduction

ENDF/B-VII.0, the first release of the ENDF/B-VII [1] nuclear data library was formally released in December 2006. Prior to this event the European JEFF-3.1 [2] nuclear data library was distributed in April 2005, while the Japanese JENDL-3.3 [3] library has been available since 2002. From the turn of this new millennium, a concerted effort by many people and laboratories worldwide has been dedicated to the improvement of the predictive power of the tools that require nuclear data libraries as part of their input stream. The systematic Kcalc under-prediction of many criticality safety benchmarks with thermal uranium spectrum has received most of the attention, but many other applications or aspects of the use and misuse of parts or all available nuclear data have also been thoroughly investigated and acted upon.

In order to assess the quality and robustness of each library a set of benchmark experiments have been taken principally from the International Handbook of Evaluated Criticality Benchmarks Experiments [4]. This choice reflects the relevance, quality and documentation of this set of experimental values as well as their ability to probe accurately and efficiently certain aspects of the nuclear data that need to be used to predict them. The code TRIPOLI-4.4 [5] based on the Monte Carlo method to solve the Boltzmann transport equation is then used to replicate the experimental assembly set-up and then predict principally the Keff, referenced thereafter as Kcalc because it has been calculated. The code and the Monte Carlo method have been chosen because it is the least invasive, in terms of processing or representation: it has pointwise cross sections, accurate and precise angular distribution sampling, pointwise emitted particle spectra, continuous three-dimensional geometry and high and always determined calculational accuracy in term of low standard deviation. It then allows precise and specific probing of the nuclear data without being impacted by the calculational scheme (it represents the reference route for reactors physics and many other applications).

All results reported here were obtained with the use of the Monte Carlo code TRIPOLI, version 4.4 [5]. The cross section sets were processed, for each library, using NJOY-99 [6] and CALENDF-2005 [7] and a dedicated set of scripts, equivalent for each common isotope of all the libraries. No unique bias could then have been introduced at the processing levels.

II. Processing

A. NJOY Processing

The 337, 381 and 393 general purpose evaluations contained in the JENDL-3.3 (2 elements), JEFF-3.1 (7 elements) [8] and ENDF/B-VII.0 (3 elements) libraries were respectively processed using the most up to date (at the time) NJOY-99.161 code and input parameters set, in agreement with the recommendations of the international NJOY user group and the TRIPOLI team. The ENDF [9] formatted evaluations have

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The author is with the Commissariat a l'Energie Atomique, Cadarache, 13108 St Paul lez Durance, France (e-mail: jean-christophe.sublet@cea.fr).

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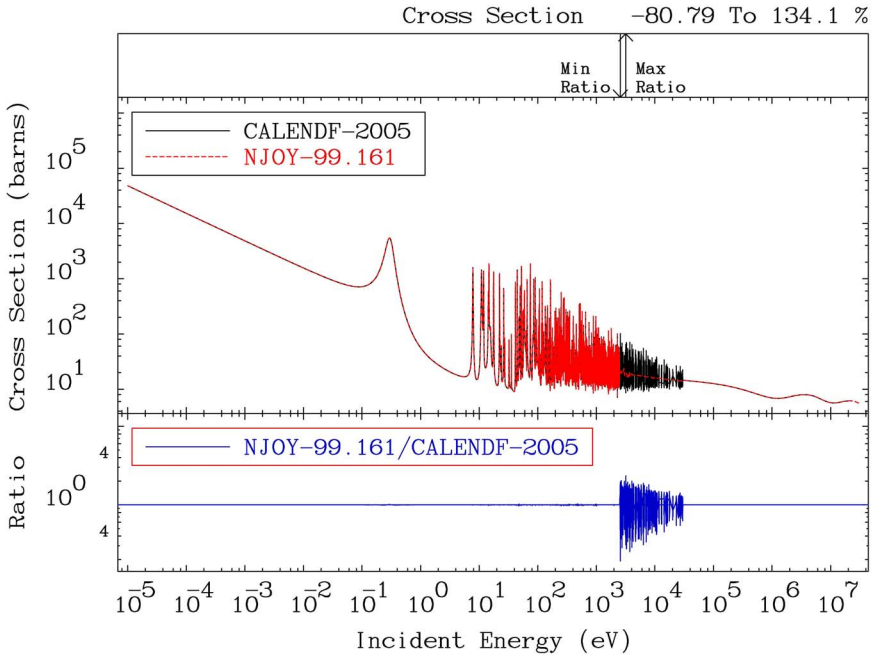

Fig. 1. CALENDF-2005 and NJOY-99 ^{239}Pu pointwise total cross section comparison.

been uniformly processed with a reconstruction criterion of 0.1% (0.001) for linear interpolation mode, at a temperature of 293.6 K, with the computation of unresolved resonance cross sections. A set of thermal quantities has been computed and outputted for verification and QA purposes alongwith graphical plots of cross section. Solely the moder, reconr, broadr and unresr modules of NJOY are used in this straightforward and simple processing scheme.

Furthermore, the 9 and 20 compounds with thermal data of JEFF-3.1 and ENDF/B-VII.0 thermal files have been respectively processed with a modified version of the thermr module of NJOY. This modification (not update) has been implemented and is specific to TRIPOLI-4.4 in order to output thermal tables in terms of 32 equally probable cosine bins up to 4.95 eV. This representation is used to extract more easily the repartition function although it is known to bias the averaged cosine obtained during the sampling process when the function is very anisotropic. The results derived from such a specific processing in the thermal range compare favourably with other Monte Carlo codes implementation when applied to benchmarks of neutrons slowing down in water [10]. Concerning the JENDL-3.3 libraries that do not include any thermal scattering files, those from JEFF-3.1 have been used when needed.

B. CALENDF Processing

The CALENDF [7] nuclear data processing system is used to convert resolved and unresolved resonance parameters, in ENDF-6 structured evaluations, into temperature dependent point-wise cross-sections (this is an alternative processing to that in NJOY). It then forms group averaged cross sections (again an alternative to NJOY). Further, it generates "cross-section probability tables", based on Gauss quadrature and Padé approximants. These represent detailed resonance self shielding within any of the groups and can be used directly in the TRIPOLI-4.4 Monte Carlo code. This feature is one of NJOY's options, replacing advantageously the purr module Levitt style probability tables in the unresolved resonance range of an evaluation (Fig. 1). The probability tables or sub-group parameters are computed for all the 140, 252 and 209 isotopes of JEFF-3.1, ENDF/B-VII.0 and JENDL-3.3 respectively, with an unresolved range. They are outputted in an 11 276 fine group structure, but limited to the unresolved resonance range of each evaluation. The probability tables data supersede the pendf cross section in (and only) the unresolved resonance range of the evaluations that contain one.

C. TRIPOLI-4.4 Processing

The linearized pointwise pendf files are directly handled by TRIPOLI-4.4. A free gas thermal treatment for each nucleus is added internally and automatically within the code itself up to the energy of kT over A. TRIPOLI-4.4 generates file of cross sections tables in a XDR (portable binary) format, a dictionary and angular distribution the first time it reads and uses a simple pendf file alongside its endf originator. The XDR binary file is a translation of the binary outputted by NJOY, however, it could be read on every Operating System which is not the case for OS dependant binary files of NJOY. The dictionary file (an ascii file) gives code numbers for the reactions described in the evaluated file. TRIPOLI-4.4 extracts values contained in the file 4 in the evaluation file and produces its own file for angular distribution. If the angular distributions of the emitted particles are expressed as normalized probability distribution in terms of angles, below 256 bins, it is taken as it is. Another possibility used in evaluation files is to represent the angular distribution as a series of Legendre polynomials. In this case, TRIPOLI-4.4 performs the calculations to provide the distribution in terms of equi-probable cosines allowing the extraction of the repartition functions. A minimum of 16 bins are used, however this number varies to accommodate a maximum error of 0.5% on the bin integral. TRIPOLI-4.4 also checks the normalization of partials and may

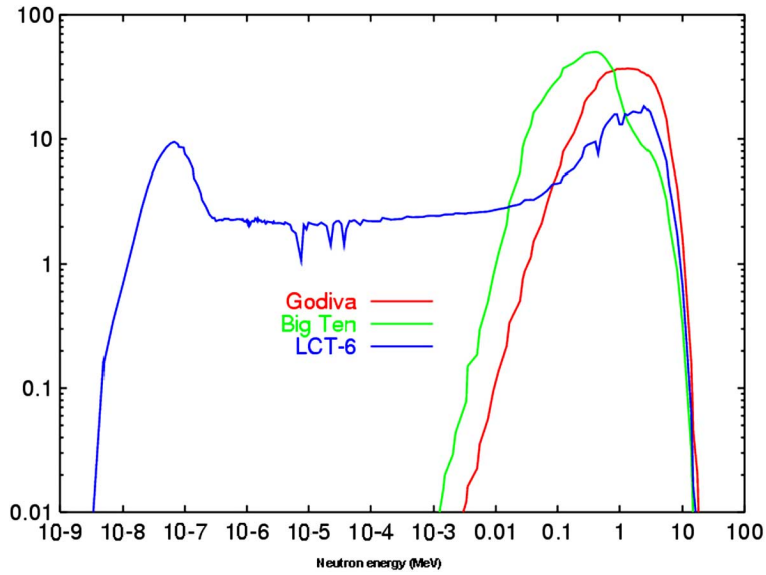

Fig. 2. ICSBEP's thermal, intermediate and fast neutron benchmarks spectra.

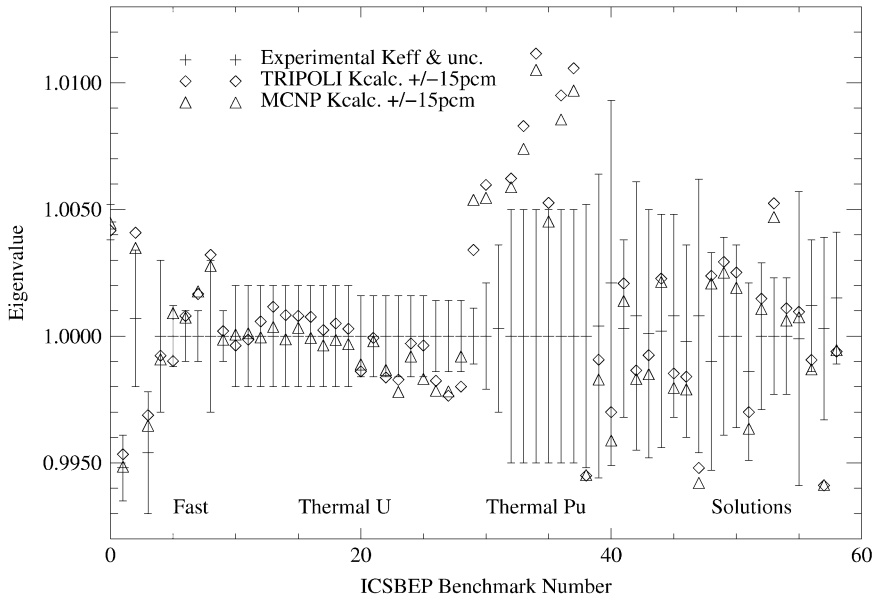

Fig. 3. Experimental Keff and uncertainty and Kcalc values obtain from TRIPOLI-4.4 and MCNP5 on the same ICSBEP benchmarks when using ENDF/B-VII.0 nuclear data.

detect errors in energy mesh, interpolation mode, pendf dictionary or angular distributions, etc. It usually fails to end its own processing in those particular cases when errors are detected, then the original file needs to be amended.

III. Benchmarking and Analyses

The total number of histories for each case and libraries has been determined to render the standard deviation small or negligible relative to the quoted benchmarks uncertainty but also aiming toward a 15 pcm value. This rather small, unusual and costly in computer resource value, allows more confidence to be placed in the results, their interpretation and the libraries comparison. Nota Bene: since the inhour, dollar, and cent all depend upon the constituents and even structure of a reactor, reactivity is also measured in units defined as decimal fraction of unity. There are varying degree of usage and appropriation of the Canadian mill-k, equal to 10^{-3}, the British millinile, the French "pour cent mille", the Italian "per centomila", the US percent millirho, all equal to 10^{-5} and abbreviated pcm.

Each benchmark has been categorized and their naming convention reflects on the physical form of the fissile materials, with fast, intermediate, thermal or mixed spectra systems. The following types have been selected and used in the suite

LCT = Low Enriched Uranium-Compound-Thermal spectrum
PST = Plutonium-Solution-Thermal spectrum
MCT = Mix(Plutonium)-Compound-Thermal spectrum
HST = High Enriched Uranium-Solution-Thermal spectrum

TABLE I
ICSBEP'S TRIPOLI-4.4 & JEFF-3.1, ENDF/B-VII.0 AND JENDL-3.3 RESULTS, ICSBEP AND CROSS SECTION EVALUATION WORKING GROUP CSEWG IDENTIFIERS, THERMAL URANIUM SPECTRUM. $|\Delta k| > \sigma_{exp}$ (SHADED)

Code Library		Experiment		Tripoli-4.4.1 JEFF-3.1 Calculation		Tripoli-4.4.1 ENDF/B-VII Calculation		Tripoli-4.4.1 JENDL-3.3 Calculation	
		Keff	Unc.	Kcalc	S.D.	Kcalc	S.D.	Kcalc	S.D.
ICSBEP	Name								
				Thermal range					
LCT-006									
	c-1	1.0000	200	0.99998	12	1.00058	12	0.99834	12
	c-3	1.0000	200	1.00051	9	1.00116	9	0.99866	9
	c-4	1.0000	200	0.99987	12	1.00083	12	0.99823	12
	c-8	1.0000	200	1.00059	12	1.00080	12	0.99837	12
	c-9	1.0000	200	1.00011	12	1.00076	12	0.99837	12
	c-13	1.0000	200	0.99994	12	1.00024	12	0.99810	12
	c-14	1.0000	200	0.99958	12	1.00050	12	0.99814	12
	c-18	1.0000	200	0.99978	12	1.00029	12	0.99782	12
Average				**1.00005**		**1.00065**		**0.99825**	
Δ (C-E)				**5**		**65**		**-175**	
LCT-007									
Valduc	c-1	1.0000	160	0.99780	10	0.99861	12	0.99653	10
	c-2	1.0000	160	0.99932	10	0.99993	14	0.99833	10
	c-3	1.0000	160	0.99749	10	0.99837	14	0.99720	10
	c-5	1.0000	160	0.99753	10	0.99828	14	0.99619	10
	c-6	1.0000	160	0.99915	10	0.99971	14	0.99845	10
	c-7	1.0000	160	0.99843	10	0.99963	14	0.99793	10
Average				**0.99829**		**0.99909**		**0.99744**	
Δ (C-E)				**-171**		**-91**		**-256**	
LCT-039									
Valduc	c-1	1.0000	140	0.99761	12	0.99824	14	0.99615	12
	c-4	1.0000	140	0.99665	12	0.99765	14	0.99525	12
	c-6	1.0000	140	0.99767	12	0.99801	14	0.99666	12
Average				**0.99731**		**0.99797**		**0.99602**	
Δ (C-E)				**-269**		**-203**		**-398**	
Hiss		1.0000	600	1.01003	13	1.01107	13	1.00580	13
Δ (C-E)				**1003**		**1107**		**580**	
Topsy-NI		1.0000	400	1.00201	17	1.00740	17	1.00593	17
Δ (C-E)				**201**		**740**		**593**	
Topsy-UR		1.0000	400	1.00687	16	1.00747	16	1.00454	16
Δ (C-E)				**687**		**747**		**454**	
LCT-027									
Pb refl.	c-1	1.0000	110	1.00757	12	1.00340	12	1.01003	12
Δ (C-E)				**757**		**340**		**1003**	
LCT-10									
Pb refl.	c-1	1.0000	210	1.00697	12	1.00597	12	1.00768	12
Δ (C-E)				**697**		**597**		**768**	
Pb refl.	c-20	1.0000	280	1.00538	12	1.00531	12	1.00536	12
Δ (C-E)				**538**		**531**		**536**	

IMF = Intermediate Enriched-Metal-Fast spectrum
HMF = High Enriched-Metal-Fast spectrum
PMF = Plutonium-Metal-Fast spectrum

From this reduced set of benchmark experiments. The ICSBEP contains currently 442 evaluations representing 3955 critical, near-critical, or sub-critical configurations and 21 criticality alarm placement/shielding configurations with multiple dose points for each. Some clear patterns and potential improvements can already been seen. Some of the main thermal and intermediate neutron energy spectra extracted from them are shown in Fig. 2.

A. Thermal Uranium Assemblies—LCT

The improvements to the ^{235}U (compared with JEF-2.2) and the ^{238}U cross section and also angular distribution have led for JEFF-3.1 and ENDF/B-VII.0 to a major improvement in the ability to predict accurately the low enriched uranium benchmarks LCT-006 in Table I and in doing so qualifying the libraries for light water reactors. It is interesting to notice that

TABLE II
ICSBEP'S TRIPOLI-4.4 & JEFF-3.1, ENDF/B-VII.0 AND JENDL-3.3 RESULTS, ICSBEP AND CSEWG IDENTIFIERS, THERMAL PLUTONIUM SPECTRUM. $|\Delta k| > \sigma_{exp}$ (SHADED)

Code Library		Experiment		Tripoli-4.4.1 JEFF-3.1 Calculation		Tripoli-4.4.1 ENDF/B-VII Calculation		Tripoli-4.4.1 JENDL-3.3 Calculation	
		Keff	Unc.	Kcalc	S.D.	Kcalc	S.D.	Kcalc	S.D.
ICSBEP	Name								
				Thermal range					
PST-009	48" sphere, Al vessel, bare								
9.54 gPu/l	c-2A	1.0003	330	1.01893	11	1.01923	11	1.02244	11
9.46 gPu/l	c-3A	1.0003	330	1.01927	11	1.01928	11	1.02224	11
Average				**1.01910**		**1.01926**		**1.02234**	
Δ (C-E)				**1880**		**1896**		**2204**	
MCT-004	Mox 3.01 wt% PuO2-UO2 fuel rods,								
2.4 w/f ratio	c-1	1.0000	460	0.99683	13	0.99752	13	0.99760	13
2.9 w/f ratio	c-4	1.0000	390	0.99707	13	0.99772	14	0.99795	13
4.2 w/f ratio	c-7	1.0000	400	0.99779	13	0.99850	13	0.99822	13
5.5 w/f ratio	c-10	1.0000	510	0.99783	13	0.99861	13	0.99899	13
Average				**0.99738**		**0.99809**		**0.99819**	
Δ (C-E)				**-262**		**-191**		**-181**	
PST-001	11.5" sphere, water reflected								
73.0 gPu/l	c-1	1.0000	500	1.00186	12	1.00622	12	1.00874	12
96.0 gPu/l	c-2	1.0000	500	1.00356	12	1.00829	12	1.01054	12
119.0 gPu/l	c-3	1.0000	500	1.00665	12	1.01115	12	1.01305	12
132.0 gPu/l	c-4	1.0000	500	1.00104	12	1.00526	12	1.00712	12
140.0 gPu/l	c-5	1.0000	500	1.00505	17	1.00950	12	1.01117	12
268.7 gPu/l	c-6	1.0000	500	1.00681	12	1.01057	12	1.01237	12
Average				**1.00416**		**1.00850**		**1.01050**	
Δ (C-E)				**416**		**850**		**1050**	
PST-011	16&18" sphere, bare								
34.9 gPu/l	16-1	1.0000	520	1.00669	13	1.01037	12	1.01408	13
43.4 gPu/l	16-5	1.0000	520	1.00337	13	1.00668	12	1.01041	13
Average				**1.00503**		**1.00853**		**1.01225**	
Δ (C-E)				**503**		**853**		**1225**	
22.3 gPu/l	18-1	1.0000	520	0.99134	13	0.99450	12	0.99893	13
27.5 gPu/l	18-6	1.0000	520	0.99708	13	1.00049	12	1.00435	13
Average				**0.99421**		**0.99750**		**1.00164**	
Δ (C-E)				**-579**		**-250**		**164**	
PST-013	256-mm cyl, in air								
115 gPu/l	c-1	0.9980	400	1.00169	12	1.00510	12	1.00823	12
115 gPu/l	c-2	0.9980	400	1.00157	12	1.00528	12	1.00774	12
Average				**1.00163**		**1.00519**		**1.00799**	
Δ (C-E)				**363**		**719**		**999**	
115 gPu/l	c-4	0.9965	520	0.99419	12	0.99747	12	1.00037	12
Δ (C-E)				**-231**		**97**		**386**	

in the term (C-E) the JENDL-3.3 library does indeed perform rather well, and much better than ENDF/B-VIr8 [1] or JEF-2.2 [11]. It is clear that the reasons behind the rather good performances of JENDL-3.3 are different from those due to the remarkable improvement seen for the two other libraries but nevertheless noticeable. Another point to account for in those comparisons is that both isotopic components of the main moderator ^{1}H and ^{16}O have evolved from one library to the other and that their isolated impact should not be considered as insignificant nor systematic.

If one set of LCT's seems to have drastically improved one could not help noticing that the however very well performed and documented Valduc 007 and 039 series do not show such an improvement in all but one library. They indubitably probe different parts of the nuclear data of the major nuclides present, clearly reminding the nuclear physicist that not all is achieved at once nor as yet been unearthed. The difficult but important and reactor relevant Hiss and Topsy set-up [4] are much better predicted than in the past, though still on the high side.

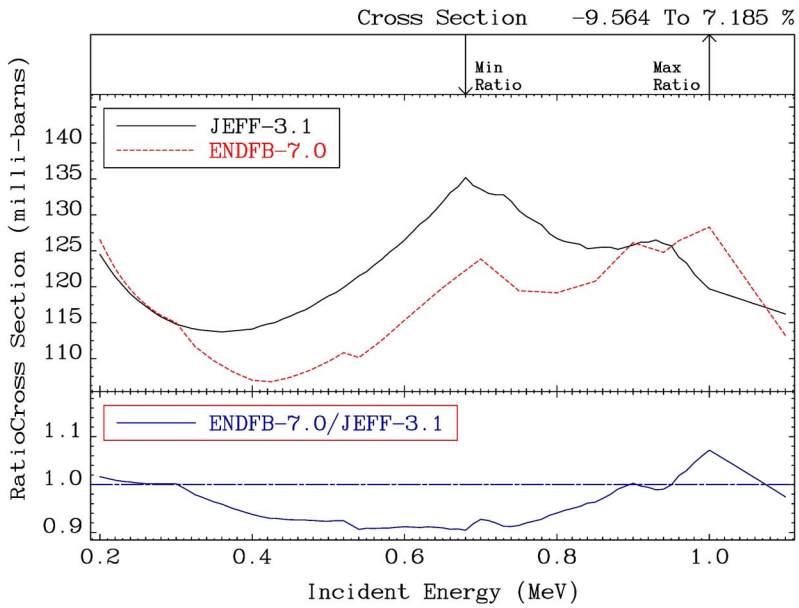

Fig. 4. JEFF-3.1—ENDF/B-VII.0 ^{238}U (n, γ) cross section comparison.

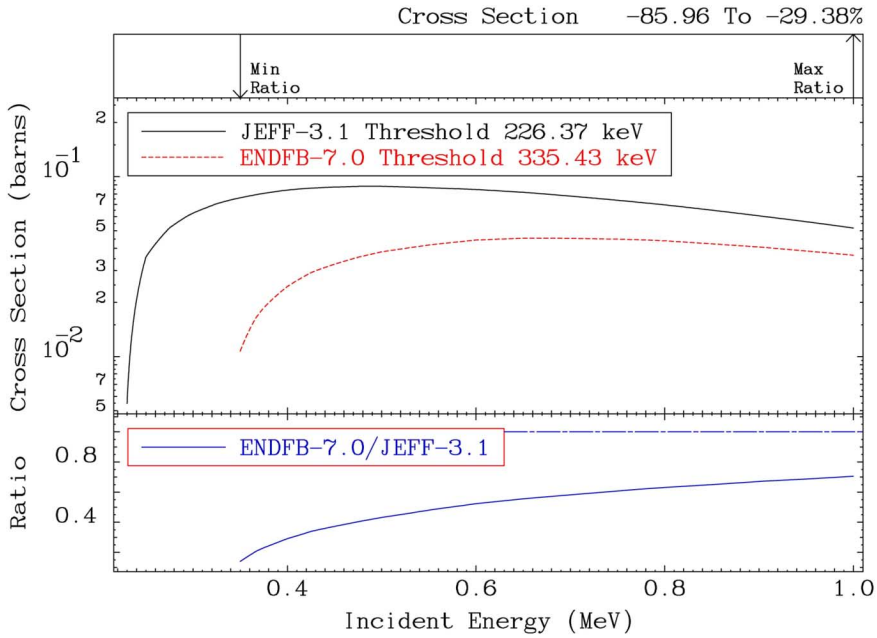

Fig. 5. JEFF-3.1—ENDF/B-VII.0 ^{235}U (n,n') cross section comparison.

If it has been established previously that the assembly and the experimental setup can and have been predicted with satisfaction and constancy one may test other setups by replacing the reflector for example, and by doing so benchmarking the nuclear data of the major isotopes of its constituents. That has been done in the LCT-10 or 27 series that probe the lead isotopes in the thermal and epithermal range with a mix but noticeable success for the work behind the ENDF/B-VII.0 lead.

B. Thermal Plutonium Assemblies—PST, MCT

Table II shows that, whatever the libraries the performances are poor, not to say unacceptable, particularly when Mox fuels are put in reactors. This uniformity is mainly due to the unique ^{239}Pu resonance parameters file present in all evaluations that does not seem to be adequate for the job. It would not be fair to point the finger at only this isotope where significant quantities of ^{240,241}Pu are also present in the fuel of those solution or oxide fuel assemblies and will influence the end results. With unresolved resonance ranges ending at 30–40 keV for those isotopes, but beginning at 300 eV for the ^{241}Pu it seems to indicate, particularly for solid fuel, that some refinement would be necessary for those isotopes in those energy ranges. It goes without saying that this conclusion does not, in any case jeopardise, the excellent performances of the same isotopes at much higher energy for other type of applications. It may come to the mind of a careful reader that some outliers could be spotted. The experimental results of those may not be considered as robust as the

TABLE III
ICSBEP'S TRIPOLI-4.4 & JEFF-3.1, ENDF/B-VII.0 AND JENDL-3.3 RESULTS, ICSBEP AND CSEWG IDENTIFIERS, FAST SPECTRUM. $|\Delta k| > \sigma_{exp}$ (SHADED)

Code Library		Experiment		Tripoli-4.4.1 JEFF-3.1 Calculation		Tripoli-4.4.1 ENDF/B-VII Calculation		Tripoli-4.4.1 JENDL-3.3 Calculation	
		Keff	Unc.	Kcalc	S.D.	Kcalc	S.D.	Kcalc	S.D.
ICSBEP	Name								
				Fast range					
IMF-007									
Big Ten	deta.	1.0045	70	0.99863	13	1.00503	13	1.01040	13
	simp.	1.0045	70	0.99790	13	1.00417	13	1.00950	13
Δ (C-E)				-623		10		545	
	t.z.h.	0.9948	130	0.98830	12	0.99534	12	1.00188	13
Δ (C-E)				-650		54		708	
IMF-012									
ZPR(16%)	c-1	1.0007	270	1.00261	13	1.00408	13	1.00618	13
Δ (C-E)				191		338		548	
IMF-10									
ZPR-U9	c-1	0.9954	240	0.99181	12	0.99688	12	1.00373	13
Δ (C-E)				-359		148		833	
IMF-002									
	c-1	1.0000	300	0.99216	10	0.99923	10	1.00228	10
Δ (C-E)				-784		-77		228	
IMF-001									
Jemima	c-2	1.0000	120	0.99837	12	0.99902	12	1.00702	12
	c-3	1.0000	100	0.99741	12	1.00080	12	1.00526	12
	c-4	1.0000	100	0.99850	12	1.00166	12	1.00625	12
Average				0.99809		1.00049		1.00618	
Δ (C-E)				-191		49		618	
HMF-028									
Flattop-25		1.0000	300	1.00210	11	1.00321	11	1.00385	11
Δ (C-E)				210		321		175	
HMF-001									
Godiva	c1	1.0000	100	0.99645	11	1.00020	11	1.00717	11
	c2	1.0000	100	0.99660	11	1.00027	11	1.00723	11
Average				0.99653		1.00023		1.00720	
Δ (C-E)				-347		23		720	
PMF-001									
Jezebel	c-1	1.0000	200	1.00025	15	0.99963	15	0.99759	15
Δ (C-E)				25		-37		-241	
PMF-002									
Jez. 240	c-1	1.0000	200	1.00430	15	0.99986	15	1.00118	15
Δ (C-E)				430		-14		118	

one of other and it would be most beneficial to seriously consider enlarging the experimental data base in this deep thermal range.

C. Fast Assemblies—IMF, HMF, PMF

It is for those fast assembly critical benchmarks in Table III that the library differences are the more important and at the same time certainly the best explained. The quite striking improvement when ENDF/B-VII.0 is used can directly be linked to the major and successful efforts made in the ^{235}U (Godiva) and ^{238}U (Big Ten) evaluations (Figs. 4 and 5) in terms of cross sections, but as well angular distribution for capture, elastic and inelastic scattering above the KeV region. If the JEFF-3.1 and ENDF/B-VII.0 ^{239}Pu (Jezebel) new evaluations in the fast region led to an excellent agreement, the ^{240}Pu (Jezebel 240) evolution in JENDL-3.3 and JEFF-3.1 did not show the way to the best improvement. It shall be said that in the energy range important for those assemblies, where the unresolved resonance range plays an important role, the impact of the probability tables is important. For all the Big Ten (IMF-007) benchmarks a further underestimation of circa 400 pcm would be seen, leading to an embarrassing -1% for JEFF-3.1, if the probability tables were not plugged in the TRIPOLI-4.4 calculation [8]. The defaults setting for all the results therein presented included the probability tables. Even if the experimental uncertainties with a maximum of ± 300 pcm should allow the calculated and experimental Keff to fit nicely within their range and that this

TABLE IV
ICSBEP'S TRIPOLI-4.4 & JEFF-3.1, ENDF/B-VII.0 AND JENDL-3.3 RESULTS, ICSBEP AND CSEWG IDENTIFIERS, SOLUTIONS. $|\Delta k| > \sigma_{exp}$ (SHADED)

Code Library		Experiment		Tripoli-4.4.1 JEFF-3.1 Calculation		Tripoli-4.4.1 ENDF/B-VII Calculation		Tripoli-4.4.1 JENDL-3.3 Calculation	
		Keff	Unc.	Kcalc	S.D.	Kcalc	S.D.	Kcalc	S.D.
ICSBEP	Name								
				Solutions					
HST001									
Mid	c-1	1.0004	600	0.99908	16	0.99907	16	1.00026	16
Leakage	c-2	1.0021	720	0.99666	16	0.99700	17	0.99772	16
Nitrate	c-3	1.0003	350	1.00237	16	1.00208	17	1.00331	16
	c-4	1.0008	530	0.99929	16	0.99865	16	1.00002	16
	c-5	1.0001	490	0.99974	16	0.99925	16	1.00072	16
	c-6	1.0002	460	1.00314	16	1.00227	16	1.00406	16
	c-7	1.0008	400	0.99882	16	0.99853	16	0.99934	16
	c-8	0.9998	380	0.99890	16	0.99840	16	0.99957	16
	c-9	1.0008	540	0.99483	16	0.99480	16	0.99546	16
Average		**1.0006**		**0.99920**		**0.99890**		**1.00005**	
Δ (C-E)				**-139**		**-169**		**-54**	
HST009									
High	c-1	0.9990	430	1.00064	19	1.00238	17	1.00322	16
Leakage	c-2	1.0000	390	1.00144	16	1.00293	16	1.00309	16
Fluoride	c-3	1.0000	360	1.00099	16	1.00251	16	1.00191	16
	c-4	0.9986	350	0.99559	16	0.99700	16	0.99660	16
Average		**0.9994**		**0.99966**		**1.00120**		**1.00120**	
Δ (C-E)				**26**		**180**		**180**	
HST010									
Fluoride	c-1	1.0000	290	1.00104	16	1.00148	16	1.00192	16
	c-2	1.0000	290	1.00122	16	1.00221	16	1.00224	16
	c-3	1.0000	290	0.99872	16	0.99953	16	0.99973	16
	c-4	0.9992	290	0.99666	16	0.99773	16	0.99769	16
Average		**0.9998**		**0.99941**		**1.00024**		**1.00039**	
Δ (C-E)				**-39**		**44**		**59**	
HST011									
Fluoride	c-1	1.0000	230	1.00473	16	1.00524	16	1.00595	16
	c-2	1.0000	230	1.00062	16	1.00110	16	1.00167	16
Average				**1.00267**		**1.00317**		**1.00381**	
Δ (C-E)				**267**		**317**		**381**	
HST012									
	c-1	0.9999	580	1.00115	16	1.00096	16	1.00193	15
Δ (C-E)				**125**		**106**		**203**	
HST013									
ORNL-1	c-1	1.0012	260	0.99880	16	0.99906	16	0.99982	16
ORNL-2	c-2	1.0007	360	0.99791	16	0.99766	16	1.00167	16
ORNL-3	c-3	1.0003	360	0.99416	16	0.99411	16	0.99519	16
ORNL-4	c-4	1.0003	360	0.99591	16	0.99593	16	0.99702	16
Average		**1.0006**		**0.99669**		**0.99669**		**0.99843**	
Δ (C-E)				**-393**		**-394**		**-220**	
HST018									
Nitrate	c-1	1.0000	340	0.98956	16	0.99093	16	0.99071	16
	c-2	1.0000	460	0.98503	16	0.98570	16	0.98620	16
	c-3	1.0000	420	0.98832	16	0.98949	16	0.98969	16
Average				**0.98764**		**0.98871**		**0.98887**	
Δ (C-E)				**-1236**		**-1129**		**-1113**	
HST019									
	c-1	1.0000	410	0.99691	16	0.99835	17	0.99846	16
Δ (C-E)				**-309**		**-165**		**-154**	
HST032									
ORNL-10		1.0015	260	0.99881	16	0.99939	16	0.99990	16
Δ (C-E)				**-269**		**-211**		**-160**	

fast region always has been considered by some simpler and easier to predict, it seems that only ENDF/B-VII.0 could pretend to fit such large criteria.

D. Solutions—HST

For those highly enriched solutions in thermal spectrum, with usually large experimental uncertainty (above 300 pcm)

TABLE V
EXPERIMENTAL, TRIPOLI-4.4 AND MCNP5 BENCHMARKS RESULTS

Code				TRIPOLI-4.4		MCNP5		
Library				ENDF/B-VII		ENDF/B-VII		
		Experiment		Calculation		Calculation		T4-M5
		Keff	Unc.	Kcalc	S.D.	Kcalc	S.D.	Kcalc
ICSBEP	Name							
IMF-007	simp.	1.0045	70	1.00417	13	1.00445	10	-28
IMF-007	t.z.h.	0.9948	130	0.99534	12	0.99485	7	49
IMF-012	c-1	1.0007	270	1.00408	13	1.00348	10	60
IMF-10	c-1	0.9954	240	0.99688	12	0.99647	10	41
IMF-002	c-1	1.0000	300	0.99923	10	0.99909	10	14
IMF-001	c-2	1.0000	120	0.99902	12	1.00091	11	-189
IMF-001	c-3	1.0000	100	1.00080	12	1.00073	11	7
IMF-001	c-4	1.0000	100	1.00166	12	1.00177	12	-11
HMF-028		1.0000	300	1.00321	11	1.00277	10	44
HMF-001	c1	1.0000	100	1.00020	11	0.99986	10	34
PMF-001	c-1	1.0000	200	0.99963	15	1.00006	10	-43
PMF-002	c-1	1.0000	200	0.99986	15	1.00011	11	-25
LCT-006	c-1	1.0000	200	1.00058	12	0.99995	10	63
LCT-006	c-3	1.0000	200	1.00116	9	1.00036	10	80
LCT-006	c-4	1.0000	200	1.00083	12	0.99988	10	95
LCT-006	c-8	1.0000	200	1.00080	12	1.00033	10	47
LCT-006	c-9	1.0000	200	1.00076	12	0.99993	10	83
LCT-006	c-13	1.0000	200	1.00024	12	0.99964	10	60
LCT-006	c-14	1.0000	200	1.00050	12	0.99985	10	65
LCT-006	c-18	1.0000	200	1.00029	12	0.99969	10	60
LCT-007	c-1	1.0000	160	0.99861	12	0.99889	11	-28
LCT-007	c-2	1.0000	160	0.99993	14	0.99980	11	13
LCT-007	c-3	1.0000	160	0.99837	14	0.99867	10	-30
LCT-007	c-5	1.0000	160	0.99828	14	0.99780	11	48
LCT-007	c-6	1.0000	160	0.99971	14	0.99920	11	51
LCT-007	c-7	1.0000	160	0.99963	14	0.99831	12	132
LCT-039	c-1	1.0000	140	0.99824	14	0.99784	11	40
LCT-039	c-4	1.0000	140	0.99765	14	0.99782	11	-17
LCT-039	c-6	1.0000	140	0.99801	14	0.99920	11	-119
LCT-027	c-1	1.0000	110	1.00340	12	1.00538	11	-198
LCT-10	c-1	1.0000	210	1.00597	12	1.00546	11	51
PST-009	c-3A	1.0003	330	1.01928	11	1.01950	6	-22
PST-001	c-1	1.0000	500	1.00622	12	1.00588	13	34
PST-001	c-2	1.0000	500	1.00829	12	1.00739	13	90
PST-001	c-3	1.0000	500	1.01115	12	1.01051	13	64
PST-001	c-4	1.0000	500	1.00526	12	1.00453	13	73
PST-001	c-5	1.0000	500	1.00950	12	1.00855	13	94
PST-001	c-6	1.0000	500	1.01057	12	1.00969	13	88
PST-011	18-1	1.0000	520	0.99450	12	0.99452	13	-2
HST001	c-1	1.0004	600	0.99907	16	0.99829	10	78
HST001	c-2	1.0021	720	0.99700	17	0.99588	11	112
HST001	c-3	1.0003	350	1.00208	17	1.00139	11	69
HST001	c-4	1.0008	530	0.99865	16	0.99831	11	34
HST001	c-5	1.0001	490	0.99925	16	0.99850	9	75
HST001	c-6	1.0002	460	1.00227	16	1.00214	9	13
HST001	c-7	1.0008	400	0.99853	16	0.99796	10	57
HST001	c-8	0.9998	380	0.99840	16	0.99790	11	50
HST001	c-9	1.0008	540	0.99480	16	0.99421	5	59
HST009	c-1	0.9990	430	1.00238	17	1.00208	14	30
HST009	c-2	1.0000	390	1.00293	16	1.00251	13	42
HST009	c-3	1.0000	360	1.00251	16	1.00191	14	59
HST009	c-4	0.9986	350	0.99700	16	0.99635	14	65
HST010	c-1	1.0000	290	1.00148	16	1.00107	13	41
HST011	c-1	1.0000	230	1.00524	16	1.00470	11	54
HST011	c-2	1.0000	230	1.00110	16	1.00062	11	48
HST012	c-1	0.9999	580	1.00096	16	1.00075	8	21
HST013	c-1	1.0012	260	0.99906	16	0.99869	11	37
HST013	c-3	1.0003	360	0.99411	16	0.99413	12	-2
HST032		1.0015	260	0.99939	16	0.99945	5	-6

the library performances are rather equivalent representative of the benchmarks themselves, in Table IV. It is clear that they were made for and dedicated to a specific purpose rather than LWR benchmarking. However, they are of interest in probing other aspects and parts of the nuclear data: enrichments, nitrate, poisons etc. The HST-018 series is particularly interesting thus

probing the influence and lacuna of gadolinium evaluations at different concentrations in a solution of uranium nitrate.

IV. Replicate Benchmarks for Verification

A number of the ICSBEP benchmarks in this suite have been calculated as well by R.E. MacFarlane and A.C. Kahler of Los Alamos National Laboratory, in fact 60 of them [1]. They have been using the MCNP5 Monte Carlo code with its own sets of input describing the same benchmarks. Both the TRIPOLI-4.4 and MCNP5 Monte Carlo codes results were obtained with the same nuclear data library, ENDF/B-VII.0. The agreement in these 60 common Kcalc, ranging from fast to deep thermal assemblies and solution is excellent. The achieved calculational accuracy, standard deviation of ± 15 pcm does allow placing some confidence in the results of those two Monte Carlo codes and their direct comparison. It is clear that if they access the same nuclear data set, they do not interpret and/or read them in the same ways: different thermal treatment, angular distribution sampling and probability tables generation schemes, etc. It is as well quite interesting to notice from Fig. 3 that when a disagreement exists with the experimental value they both indicate the same trend. However, shaded in Table V, one may spot a handful of outliers, where (Kcalc, TRIPOLI-4.4-MCNP5) lay outside the standard deviation that will require further investigation. One may envisage that the differences could have been induced by the different interpretations of the sometimes long, complex, yearly updated (some experimental values evolved as well) but extremely useful, ICSBEP benchmark documentation.

V. Summary and Conclusion

The new JEFF-3.1, ENDF/B-VII.0 and JENDL-3.3 libraries respectively produced by a handful of EU, US, Japanese and Korean laboratories represent common achievements. They all benefited from appreciable, substantial and timely input from the international nuclear data community that all those laboratories together represent so well. For the first time considerable efforts have been dedicated to the verification, processing and validation of all libraries prior to their open releases. This becomes apparent in the seamless production of many more dedicated (usually multi-group, but also high-energy) cross section libraries for specific applications.

The striking overall improvement of the predictions and simulation for many nuclear technologies (fission reactors, accelerator, transmutation, fusion technology, medicine, earth exploration, astrophysics, passive interrogation techniques etc.) has been the result of concerted, usually voluntary efforts on all the general but also the special purpose files. It does not always involve directly the cross sections themselves but also their angular distribution, particles emitted spectra, and other nuclear data such as yields, Q-values, emitted particles numbers and kinds, half-lives and decay data and schemes.

The specific further improvements, already foreseen through the reported benchmarking activities, suggest that for all libraries the plutonium data need be reviewed in the thermal, epi-thermal range, alongside the more prominent of the Gd isotopes. A revision of the JEFF-3.1 ^{235,238}U and ^{240}Pu evaluations will be necessary to solve the remaining deficiencies in the fast spectrum. It should be noted that at this level of probing the influences of the other isotopes present in the input deck of the calculated benchmarks may have a non negligible impact on the end results. The impact of one secondary specific isotope and/or compound may well compensate the one of another. It is clear that they are library dependant and different in each library but always relate to low Z nuclides ^{1}H, ^{16}O, ^{14}N, the important (for Keff calculation) sets of fission yields and nuclides, the absorbers and cladding materials isotope constituents. It is striking to note as well that newly measured and evaluated data for astrophysics nucleo-synthesis s-process benefit as well reactor physics when the isotope in that case is produced by fission. Those libraries are truly multi-purpose.

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