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Original Article

Processing and benchmarking of evaluated nuclear data file/b-viii.0β4 cross-section library by analysis of a series of critical experimental benchmark using the monte carlo code MCNP(X) and NJOY2016

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ABSTRACT

To validate the new Evaluated Nuclear Data File (ENDF)/B-VIII.0β4 library, 31 different critical cores were selected and used for a benchmark test of the important parameter keff. The four utilized libraries are processed using Nuclear Data Processing Code (NJOY2016). The results obtained with the ENDF/B-VIII.0β4 library were compared against those calculated with ENDF/B-VI.8, ENDF/B-VII.0, and ENDF/B-VII.1 libraries using the Monte Carlo N-Particle (MCNP(X)) code. All the MCNP(X) calculations of keff values with these four libraries were compared with the experimentally measured results, which are available in the *International Critically Safety Benchmark Evaluation Project*. The obtained results are discussed and analyzed in this paper.

1. Introduction

The evaluated nuclear data library Evaluated Nuclear Data File (ENDF)/B-VIII.0β4 was released on February 28, 2017 [1]. The Neutron General Purpose Library contains incident neutron data for 446 isotopes from $^{1}_{1}H$ to $^{255}_{100}Fm$, and the Scattering Thermal Library $S(\alpha, \beta)$ covers 26 different moderators.

The code system used for processing and generating the ACE format (a compact ENDF format for the MCNP) is Nuclear Data Processing Code (NJOY2016) [2] for all isotopes and $S(\alpha, \beta)$ thermal scattering, especially the ACER module (the ACER module prepares libraries in ACE format), for correct processing of the libraries. A locally executable version of NJOY2016 was created using 64 bits under a Linux system.

As part of the validation process, comparisons with the ENDF/B-VI.8, ENDF/B-VII.0, and ENDF/B-VII.1 libraries were carried out.

In this study, Monte Carlo N-Particle (MCNP(X)) v2.6.0 [3] calculations were performed with an HP Pro Intel (Hewlett-Packard company, U.S.A, Product of China) [i5 CPU 3.20 GHz, third generation (4 cores), 4 GB RAM, and 6 MB Cache Memory] under Win 10 system and using MPICH2 [4]. Calculations for keff were performed using these four libraries with 600 iterations on a nominal source size of 60,000 particles per cycle, in order to decrease statistical error estimates. Initial 100 cycles were skipped to ensure homogeneous neutron source distribution.

2. Methodology

2.1. Nuclear data processing converted from evaluated nuclear data to ACE format

NJOY is a modular computer code used for converting evaluated nuclear data in the ENDF format into different types of libraries useful for the calculations of criticality and shielding applications. Since the ENDF format is used worldwide, NJOY gives its users access to a wide variety of the most up-to-date nuclear data [2]. NJOY provides comprehensive capabilities for processing evaluated data, serving applications that include continuous-energy Monte Carlo-like MCNP(X) [3].

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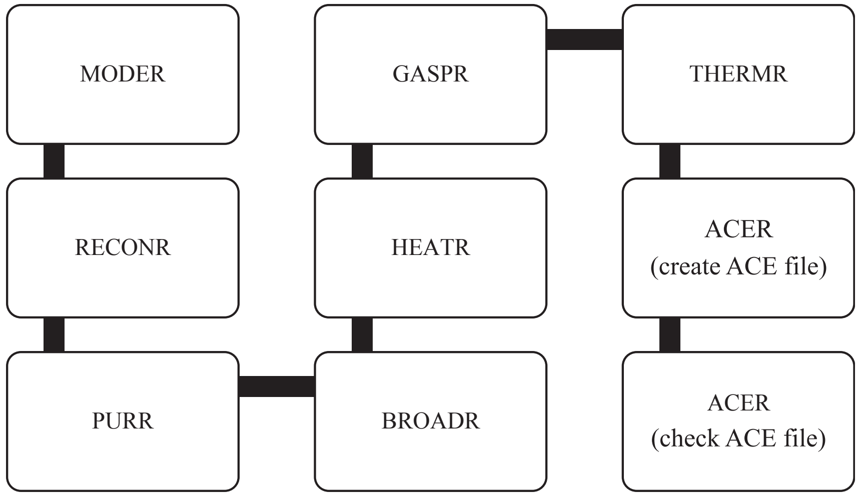

Fig. 1. Flow diagram of NJOY2016 processing for ACE format library construction. ACE: A Compact ENDF format for the MCNP; ACER: The ACER module prepares libraries in ACE format; BROADR, Doppler broadens and thins pointwise cross sections; GASPR, generates gas-production cross sections in pointwise format from basic reaction data in an ENDF evaluation. These results can be converted to multigroup form using GROUPR, passed to ACER, or displayed using PLOTR; HEATR, generates pointwise heat production cross sections (KERMA coefficients) and radiation-damage cross sections; MODER, converts ENDF "tapes" back and forth between ASCII format and the special NJOY blocked-binary format; NJOY, Nuclear Data Processing Code developed at Los Alamos National Laboratory; PURR, generates unresolved-resonance probability tables for use in representing resonance self-shielding effects in the MCNP Monte Carlo code; RECONR, reconstructs pointwise (energy-dependent) cross sections from ENDF resonance parameters and interpolation schemes; THERMR, produces cross sections and energy-to-energy matrices for free or bound scatterers in the thermal energy range.

The ENDF/B-VIII.0β4, ENDF/B-VI.8, ENDF/B-VII.0, and ENDF/B-VII.1 libraries contain continuous-energy neutron cross sections; resonance parameters, angular data, and other data for all the isotopes composing the materials of studied benchmarks have been processed using NJOY2016 in ACE format (compatible with the Monte Carlo code MCNP) at a temperature of 300 K. In addition, probability tables (*ptable*) have been generated for benchmark materials containing unresolved resonance data. In addition, $S(\alpha, \beta)$ thermal scattering has been processed at 300 K [5] for hydrogen bound in polyethylene (poly.10t), graphite (grph.10t), H in H_2O (lwtr.10t), and beryllium (Be) metal (be.10t).

The processing sequence for generating an ACE-formatted library suitable for the MCNP code is shown in Fig. 1.

These modules are explained below. The reason for running the ACER module twice (Fig. 1) is that it is used to evaluate the consistency of the ACE format, which is checked and for which problems are corrected [2].

(1) MODER converts ENDF "tapes" back and forth between ASCII format and the special NJOY blocked-binary format.
(2) RECONR reconstructs pointwise (energy-dependent) cross sections from ENDF resonance parameters and interpolation schemes. The cross-section accuracy in this module is of the order of 0.5% (err = 0.005).
(3) PURR generates unresolved-resonance probability tables for use in representing resonance self-shielding effects in the MCNP Monte Carlo code.
(4) GASPR generates gas-production cross sections in pointwise format from basic reaction data in an ENDF evaluation. These results can be converted to multigroup form using GROUPR; they are then passed to ACER or displayed using PLOTR.
(5) HEATR generates pointwise heat production cross sections (kinetic energy released per unit mass (KERMA) coefficients) and radiation-damage cross sections.
(6) BROADR Doppler broadens and thins pointwise cross sections. The cross-section accuracy in this module is of the order of 0.5% (err = 0.005).

Table 1
ICSBEP abbreviations used [8].

Abbreviation	meaning
Fissile material	
HEU	High enriched uranium ($^{235}U \geq 60$ wt%)
IEU	Intermediate or mixed enrichment uranium (60 wt% > ^{235}U > 10 wt%)
LEU	Low enriched, natural, or depleted uranium ($^{235}U \leq 10$ wt%)
PU	Plutonium
MIX	Mixed uranium and plutonium
U233	Uranium ^{233}U systems
Physical form of fissile material	
MET	Metal
SOL	Solution
COMP	Compound system, e.g., lattice in water
Spectrum	
FAST	Fast system (≥50% of fissions above 100 keV)
THERM	Thermal system (≥50% of fissions below 0.625 eV)

ICSBEP, International Critically Safety Benchmark Evaluation Project.

Table 2
ICSBEP benchmark systems.

ICSBEP benchmark systems	
HEU-MET-FAST-001	IEU-MET-FAST-004-case-2
HEU-MET-FAST-004	LEU-SOL-THERM-001
HEU-MET-FAST-008	LEU-SOL-THERM-002-case-1
HEU-MET-FAST-009-case-1	MIX-MET-FAST-001
HEU-MET-FAST-011	MIX-MET-FAST-003
HEU-MET-FAST-015	MIX-COMP-THERM-002-case-pnl30
HEU-MET-FAST-018	PU-MET-FAST-001
HEU-MET-FAST-019-case-2	PU-MET-FAST-002
HEU-MET-FAST-020-case-2	PU-MET-FAST-005
HEU-MET-INTER-006-case-1	PU-MET-FAST-006
HEU-SOL-THERM-013-case-1	PU-MET-FAST-025
HEU-SOL-THERM-032	PU-MET-FAST-026
IEU-MET-FAST-001-case-1	U233-MET-FAST-002-case-1
IEU-MET-FAST-002	U233-SOL-INTER-001-case-1
IEU-MET-FAST-003-case-2	U233-SOL-THERM-001-case-1
U233-SOL-THERM-008	

COMP, compound system; FAST, fast system; HEU, high enriched uranium; ICSBEP, International Critically Safety Benchmark Evaluation Project; IEU, intermediate enriched uranium; INTER, intermediate; LEU, low enriched uranium; MET, metal; MIX, mixed uranium and plutonium system; PU, plutonium; SOL, solution; THERM, thermal.

Table 3
MCNP(X) calculations of keff values with four data libraries and benchmark keff.

Case name	Benchmark keff	ENDF/B-VI.8	ENDF/B-VII.0	ENDF/B-VII.1	ENDF/B-VIII.0β4
HEU-MET-FAST-001	1.00000 ± 0.00100	0.99656 ± 0.00010	0.99985 ± 0.00010	0.99978 ± 0.00010	0.99994 ± 0.00010
HEU-MET-FAST-004-case-1	1.00200 ± 0.00100	0.99833 ± 0.00010	1.00302 ± 0.00010	1.00305 ± 0.00010	1.00250 ± 0.00010
HEU-MET-FAST-008	0.99890 ± 0.00160	0.99240 ± 0.00010	0.99577 ± 0.00010	0.99573 ± 0.00010	0.99562 ± 0.00010
HEU-MET-FAST-009-case-1	0.99920 ± 0.00150	0.99484 ± 0.00010	0.99507 ± 0.00010	0.99749 ± 0.00010	0.99615 ± 0.00010
HEU-MET-FAST-011	0.99890 ± 0.00150	1.00050 ± 0.00010	0.99881 ± 0.00010	1.00437 ± 0.00010	1.00489 ± 0.00010
HEU-MET-FAST-015	0.99960 ± 0.00170	0.99146 ± 0.00010	0.99491 ± 0.00010	0.99466 ± 0.00010	0.99478 ± 0.00010
HEU-MET-FAST-018-case-2	1.00000 ± 0.00140	0.99599 ± 0.00010	0.99971 ± 0.00010	0.99959 ± 0.00010	0.99936 ± 0.00010
HEU-MET-FAST-019-case-2	1.00000 ± 0.00280	1.00310 ± 0.00010	1.00746 ± 0.00010	1.00713 ± 0.00010	1.00658 ± 0.00010
HEU-MET-FAST-020-case-2	1.00000 ± 0.00280	0.99677 ± 0.00010	1.00087 ± 0.00010	1.00078 ± 0.00010	1.00031 ± 0.00010
HEU-MET-INTER-006-case-1	0.99770 ± 0.00080	0.98567 ± 0.00010	0.99286 ± 0.00010	0.98734 ± 0.00010	0.98805 ± 0.00010
HEU-SOL-THERM-013-case-1	1.00120 ± 0.00260	0.99930 ± 0.00010	0.99872 ± 0.00010	0.99872 ± 0.00010	0.99844 ± 0.00010
HEU-SOL-THERM-032	1.00150 ± 0.00260	0.99880 ± 0.00010	0.99956 ± 0.00010	0.99951 ± 0.00010	0.99850 ± 0.00010
IEU-MET-FAST-001-case-1	0.99890 ± 0.00100	0.99645 ± 0.00010	1.00087 ± 0.00010	1.00073 ± 0.00010	0.99901 ± 0.00010
IEU-MET-FAST-002	1.00000 ± 0.00300	1.00305 ± 0.00010	0.99920 ± 0.00010	0.99876 ± 0.00010	0.99613 ± 0.00010
IEU-MET-FAST-003-case-2	1.00000 ± 0.00170	0.99902 ± 0.00010	1.00251 ± 0.00010	1.00277 ± 0.00010	0.99968 ± 0.00010
IEU-MET-FAST-004-case-2	1.00000 ± 0.00300	1.00362 ± 0.00010	1.00747 ± 0.0001 0	1.00751 ± 0.00010	1.00533 ± 0.00010
LEU-SOL-THERM-001	0.99910 ± 0.00290	1.01004 ± 0.00010	1.01201 ± 0.00010	1.01172 ± 0.00010	1.01164 ± 0.00010
LEU-SOL-THERM-002-case-1	1.00380 ± 0.00400	0.99838 ± 0.00010	0.99993 ± 0.00010	0.99992 ± 0.00010	0.99605 ± 0.00010
IX-MET-FAST-001	1.00000 ± 0.00160	0.99698 ± 0.00010	0.99947 ± 0.00010	0.99952 ± 0.00010	0.99951 ± 0.00010
MIX-MET-FAST-003	0.99930 ± 0.00160	0.99833 ± 0.00010	1.00087 ± 0.00010	1.00067 ± 0.00010	1.00073 ± 0.00010
MIX-COMP-THERM-002-case-pnl30	1.00240 ± 0.00600	0.99207 ± 0.00010	1.00117 ± 0.00010	1.00049 ± 0.00010	0.99957 ± 0.00010
PU-MET-FAST-001	1.00000 ± 0.00200	0.99752 ± 0.00010	0.99985 ± 0.00010	0.99987 ± 0.00010	0.99974 ± 0.00010
PU-MET-FAST-002	1.00000 ± 0.00200	0.99792 ± 0.00010	1.00004 ± 0.00010	0.99981 ± 0.00010	1.00149 ± 0.00010
PU-MET-FAST-005	1.00000 ± 0.00130	1.00753 ± 0.00010	1.00943 ± 0.00010	1.00085 ± 0.00010	0.99940 ± 0.00010
PU-MET-FAST-006	1.00000 ± 0.00300	1.00275 ± 0.00010	1.00123 ± 0.00010	1.00110 ± 0.00010	0.99970 ± 0.00010
PU-MET-FAST-025-case-2	1.00000 ± 0.00200	0.99649 ± 0.00010	0.99868 ± 0.00010	0.99867 ± 0.00010	0.99967 ± 0.00010
PU-MET-FAST-026-case-2	1.00000 ± 0.00240	0.99700 ± 0.00010	0.99866 ± 0.00010	0.99844 ± 0.00010	1.00083 ± 0.00010
U233-MET-FAST-002-case-1	1.00000 ± 0.00100	0.99530 ± 0.00010	0.99897 ± 0.00010	0.99933 ± 0.00010	1.00003 ± 0.00010

Table 3 (*continued*)

Case name	Benchmark keff	ENDF/B-VI.8	ENDF/B-VII.0	ENDF/B-VII.1	ENDF/B-VIII.0β4
U233-SOL-INTER-001-case-1	1.00000 ± 0.00830	0.96033 ± 0.00020	0.98440 ± 0.00020	0.95684 ± 0.00020	0.95268 ± 0.00010
U233-SOL-THERM-001-case-1	1.00000 ± 0.00310	0.99818 ± 0.00010	1.00157 ± 0.00010	1.00139 ± 0.00010	0.99964 ± 0.00010
U233-SOL-THERM-008	1.00000 ± 0.00290	0.99708 ± 0.00010	1.00157 ± 0.00010	1.00149 ± 0.00010	0.99987 ± 0.00010

COMP, compound system; ENDF, Evaluated Nuclear Data File; FAST, fast system; HEU, high enriched uranium; IEU, intermediate enriched uranium; INTER, intermediate; LEU, low enriched uranium; MET, metal; MIX, mixed uranium and plutonium system; PU, plutonium; SOL, solution; THERM, thermal.

(7) THERMR produces cross sections and energy-to-energy matrices for free or bound scatterers in the thermal energy range.
(8) ACER prepares libraries in ACE format for the Los Alamos continuous-energy Monte Carlo code MCNP.

3. Criticality benchmarks

3.1. Brief description of the criticality benchmarks

In this study, all benchmarks [6], which are subdivided into the main categories according to three criteria presented in Table 1, are shown in Table 2; these benchmarks were taken from the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* from an Organisation for Economic Co-operation and Development-Nuclear Energy Agency (OECD-NEA) project—Critically Safety Benchmark Evaluation Project [7].

The meaning of the abbreviations is given in more detail below [9].

(1) Main fissionable isotope: Systems containing uranium-235 are subdivided according to the level of enrichment of ^{235}U: low enriched (LEU: wt% ^{235}U < 10), intermediate enriched (IEU: 10 < wt% ^{235}U < 60), and high enriched (HEU: wt.% ^{235}U > 60) uranium. There are also plutonium (PU) systems, mixed uranium/plutonium systems (MIX) and 233U systems (uranium-233).
(2) Physical form of the fissile material: There are metal (MET), compound (COM), solution (SOL), and miscellaneous systems.
(3) Neutron spectrum: If more than half of the fission occurs for neutrons with energy below 0.625 eV, the spectrum is thermal (THERM); if more than half fission occurs between 0.625 eV and 100 keV, it is intermediate (INTER); and if more than half fission occurs at over 100 keV, it is fast (FAST). If none of these applies, the spectrum is classified as MIXED.

4. Results and analysis

4.1. Results of criticality calculations

In this part, we present all the keff values, which are given in graphical and tabular forms. The columns contain the following items.

(1) Benchmark value of keff [10].
(2) Results obtained from MCNP(X) calculations based on ENDF/B-VI.8. In the four benchmarks, namely, LEU-SOL-THERM-002-case-1, MIX-COMP-THERM-002-case-pnl30, HEU-MET-INTER-006-case-1, and HEU-SOL-THERM-013-case-1, the element Zn was replaced by Cu because the first element is missing in the original library (Release 8, 2001)
(3) Results obtained from MCNP(X) calculations based on ENDF/B-VII.0
(4) Results obtained from MCNP(X) calculations based on ENDF/B-VII.1
(5) Results obtained from MCNP(X) calculations based on the new library ENDF/B-VIII.0β4

Results of keff calculations obtained using MCNP(X) code with ENDF/B-VI.8, ENDF/B-VII.0, ENDF/B-VII.1, and ENDF/B-VIII.0β4 libraries, as well as the benchmark keff values are given in Table 3 and presented in Fig. 2. The value of $\frac{C}{E}-1$ ("*E*" is the expected or benchmark value and "*C*" is the calculated value) and its uncertainty in pcm are shown in Table 4 and presented in Fig. 3 for each of the benchmark systems. The average values of $\frac{C}{E}-1$ (in pcm) per benchmark category are presented in Table 5.

As can be seen in Table 4, deviations from the benchmark values decrease as we move from ENDF/B-VI.8 to ENDF/B-VIII.0β4 library in the nine benchmark cases. In addition, results of calculations prove that the best performance is for the modern ENDF/B-VII.0, ENDF/B-VII.1, and notably ENDF/B-VIII.0β4 libraries because deviations from the benchmark values are <200 pcm in the large majority of cases. In addition, deviations are <100 pcm for the last library in about half of the benchmark cases. However, there are some cases of benchmarks for which the values of $\frac{C}{E}-1$ deviate from the benchmark values by > 1,000 pcm for all libraries. It should be noted that the largest deviations are seen for the U233-SOL-INTER-001-case-1 benchmark for all libraries except the ENDF/B-VII.0 library.

In the "U233-SOL-INTER-001-case-1" benchmark, Be is used as the reflector. It is the authors' opinion that this may be due to Be data, as the performance of the ENDF/B-VIII.0β4, ENDF/B-VII.1, and ENDF/B-VI.8 data libraries is worse for Be reflected systems. In addition, it can be observed that the curves representing the main Be cross sections, as a function of the energies (Fig. A1 in Appendix 1), are almost identical for all libraries, but differ from that of the ENDF/B-VII.0 library, especially for the elastic cross sections in the energy interval (4–6 MeV).

In order to confirm this presumption, we carried out calculations of keff values with the ENDF/B-VIII.0β4, ENDF/B-VII.1, and ENDF/B-VI.8 data libraries, where the Be data and (be.10t) $S(\alpha,\beta)$ thermal scattering in these three libraries were replaced by those in the original ENDF/B-VII.0 library. The results, which are presented in Table 6, show very good agreement with each other according to $\frac{C}{E}-1$ values.

5. Conclusion

The authors have been able to process the ENDF/B-VIII.0β4, ENDF/B-VI.8, ENDF/B-VII.0, and ENDF/B-VII.1 libraries to produce an

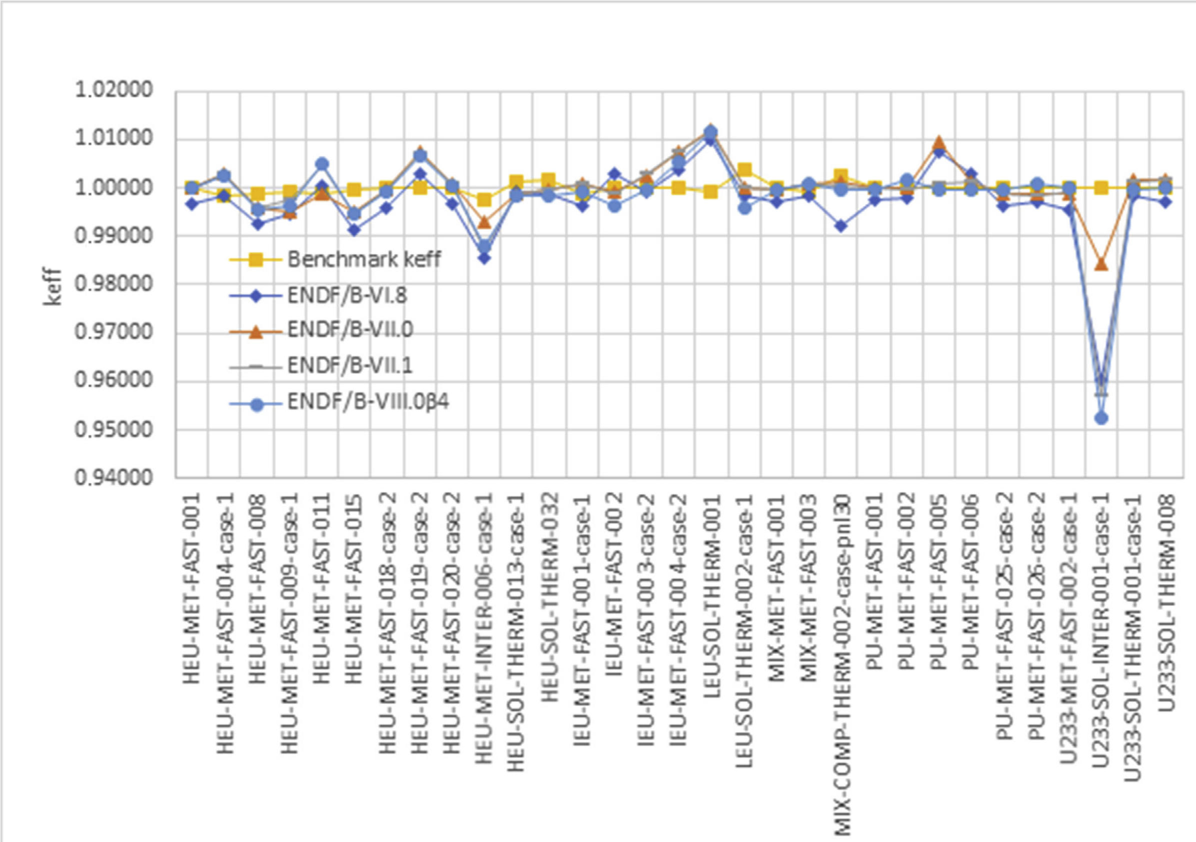

Fig. 2. MCNP(X) calculations of keff values with four data libraries and benchmark keff. COMP, compound system; FAST, fast system; HEU, high enriched uranium; IEU, intermediate enriched uranium; INTER, intermediate; LEU, low enriched uranium; MET, metal; MIX, mixed uranium and plutonium system; PU, plutonium; SOL, solution; THERM, thermal.

Table 4
Value of $\frac{C}{E} - 1$ and its uncertainty (in pcm) for all benchmark cases.

Case name	ENDF/B-VI.8	ENDF/B-VII.0	ENDF/B-VII.1	ENDF/B-VIII.0β4
HEU-MET-FAST-001	−344.0000 ± 0.3457	−15.0000 ± 0.0151	−22.0000 ± 0.0221	−6.0000 ± 0.0060
HEU-MET-FAST-004-case-1	−366.2675 ± 0.3674	101.7964 ± 0.1021	104.7904 ± 0.1051	49.9002 ± 0.0500
HEU-MET-FAST-008	−650.7158 ± 1.0444	−313.3447 ± 0.5029	−317.3491 ± 0.5093	−328.3612 ± 0.5270
HEU-MET-FAST-009-case-1	−436.3491 ± 0.6565	−413.3307 ± 0.6219	−171.1369 ± 0.2575	−305.2442 ± 0.4593
HEU-MET-FAST-011	160.1762 ± 0.2411	−9.0099 ± 0.0136	547.6024 ± 0.8241	599.6596 ± 0.9025
HEU-MET-FAST-015	−814.3257 ± 1.3873	−469.1877 ± 0.7993	−494.1977 ± 0.8419	−482.1929 ± 0.8215
HEU-MET-FAST-018-case-2	−401.0000 ± 0.5628	−29.0000 ± 0.0407	−41.0000 ± 0.0575	−64.0000 ± 0.0898
HEU-MET-FAST-019-case-2	310.0000 ± 0.8685	746.0000 ± 2.0901	713.0000 ± 1.9977	658.0000 ± 1.8436
HEU-MET-FAST-020-case-2	−323.0000 ± 0.9050	87.0000 ± 0.2438	78.0000 ± 0.2185	31.0000 ± 0.0869
HEU-MET-INTER-006-case-1	−1,205.7733 ± 0.9746	−485.1158 ± 0.3920	−1,038.3883 ± 0.8392	−967.2246 ± 0.7817
HEU-SOL-THERM-013-case-1	−189.7723 ± 0.4932	−247.7028 ± 0.6437	−247.7028 ± 0.6437	−275.6692 ± 0.7164
HEU-SOL-THERM-032	−269.5956 ± 0.7004	−193.7094 ± 0.5033	−198.7019 ± 0.5162	−299.5507 ± 0.7782
IEU-MET-FAST-001-case-1	−245.2698 ± 0.2468	197.2169 ± 0.1984	183.2015 ± 0.1843	11.0121 ± 0.0111
IEU-MET-FAST-002	305.0000 ± 0.9155	−80.0000 ± 0.2401	−124.0000 ± 0.3722	−387.0000 ± 1.1616
IEU-MET-FAST-003-case-2	−98.0000 ± 0.1669	251.0000 ± 0.4274	277.0000 ± 0.4717	−32.0000 ± 0.0545
IEU-MET-FAST-004-case-2	362.0000 ± 1.0866	747.0000 ± 2.2422	751.0000 ± 2.2542	533.0000 ± 1.5999
LEU-SOL-THERM-001	1,094.9855 ± 3.1802	1,292.1629 ± 3.7528	1,263.1368 ± 3.6685	1,255.1296 ± 3.6453
LEU-SOL-THERM-002-case-1	−539.9482 ± 2.1523	−385.5350 ± 1.5368	−386.5312 ± 1.5408	−772.0661 ± 3.0775
MIX-MET-FAST-001	−302.0000 ± 0.4841	−53.0000 ± 0.0850	−48.0000 ± 0.0769	−49.0000 ± 0.0786
MIX-MET-FAST-003	−97.0679 ± 0.1557	157.1100 ± 0.2520	137.0960 ± 0.2199	143.1002 ± 0.2296
MIX-COMP-THERM-002-case-pnl30	−1,030.5267 ± 6.1692	−122.7055 ± 0.7346	−190.5427 ± 1.1407	−282.3224 ± 1.6901
PU-MET-FAST-001	−248.0000 ± 0.4966	−15.0000 ± 0.0300	−13.0000 ± 0.0260	−26.0000 ± 0.0521
PU-MET-FAST-002	−208.0000 ± 0.4165	4.0000 ± 0.0080	−19.0000 ± 0.0380	149.0000 ± 0.2984
PU-MET-FAST-005	753.0000 ± 0.9817	943.0000 ± 1.2295	85.0000 ± 0.1108	−60.0000 ± 0.0782
PU-MET-FAST-006	275.0000 ± 0.8255	123.0000 ± 0.3692	110.0000 ± 0.3302	−30.0000 ± 0.0901
PU-MET-FAST-025-case-2	−351.0000 ± 0.7029	−132.0000 ± 0.2643	−133.0000 ± 0.2663	−33.0000 ± 0.0661
PU-MET-FAST-026-case-2	−300.0000 ± 0.7206	−134.0000 ± 0.3219	−156.0000 ± 0.3747	83.0000 ± 0.1994
U233-MET-FAST-002-case-1	−470.0000 ± 0.4724	−103.0000 ± 0.1035	−67.0000 ± 0.0673	3.0000 ± 0.0030
U233-SOL-INTER-001-case-1	−3,967.0000 ± 32.9365	−1,560.0000 ± 12.9519	−4,316.0000 ± 35.8342	−4,732.0000 ± 39.2882
U233-SOL-THERM-001-case-1	−182.0000 ± 0.5645	157.0000 ± 0.4870	139.0000 ± 0.4311	−36.0000 ± 0.1117
U233-SOL-THERM-008	−292.0000 ± 0.8473	157.0000 ± 0.4556	149.0000 ± 0.4324	−13.0000 ± 0.0377

COMP, compound system; ENDF, Evaluated Nuclear Data File; FAST, fast system; HEU, high enriched uranium; IEU, intermediate enriched uranium; INTER, intermediate; LEU, low enriched uranium; MET, metal; MIX, mixed uranium and plutonium system; PU, plutonium; SOL, solution; THERM, thermal.

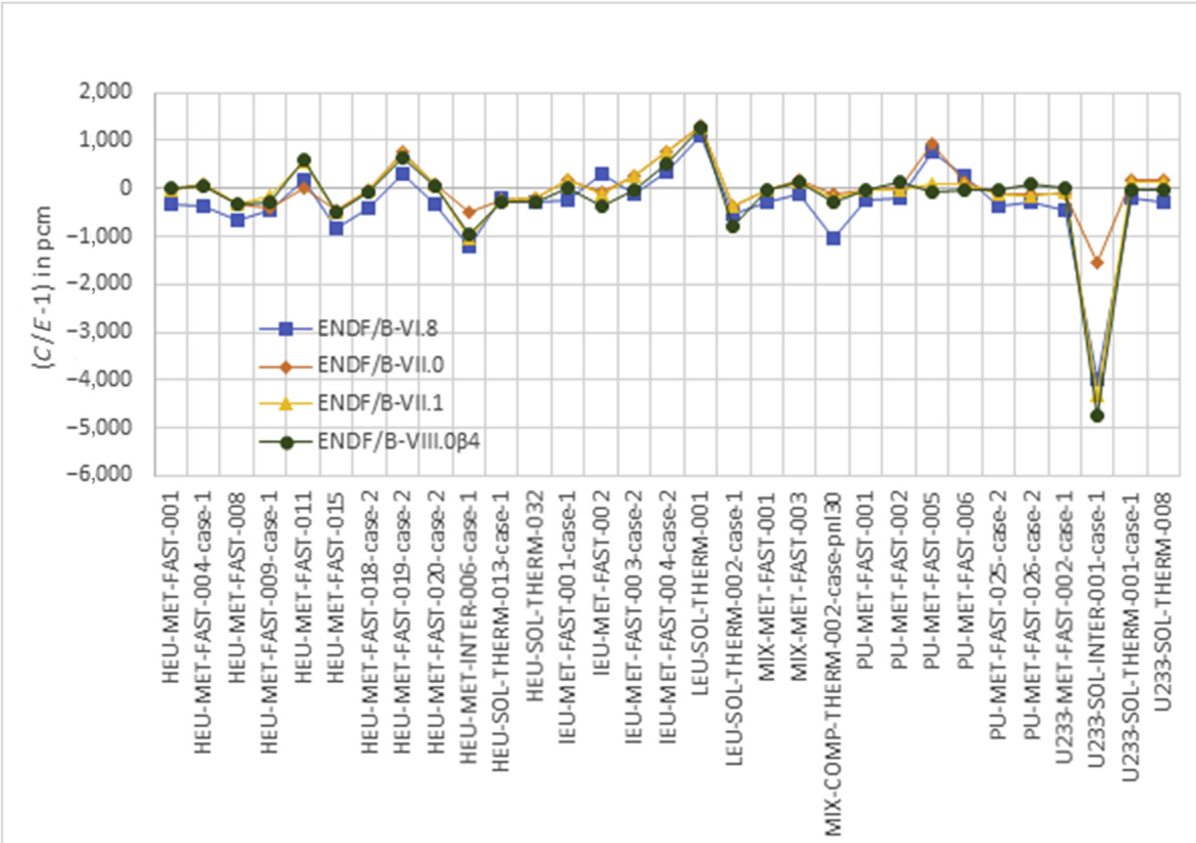

Fig. 3. Value of $\frac{C}{E}-1$ (in pcm) for all benchmark systems. COMP, compound system; ENDF, evaluated nuclear data file; FAST, fast system; HEU, high enriched uranium; IEU, intermediate enriched uranium; INTER, intermediate; LEU, low enriched uranium; MET, metal; MIX, mixed uranium and plutonium system; PU, plutonium; SOL, solution; THERM, thermal.

Table 5
Average values of $\frac{C}{E}-1$ (in pcm) per benchmark category.

Category	N	ENDF/B-VI.8	ENDF/B-VII.0	ENDF/B-VII.1	ENDF/B-VIII.0β4
HEU-MET-FAST	9	−318.3869	−34.8974	44.1899	16.9735
HEU-MET-INTER	1	−1,205.7733	−485.1158	−1,038.3883	−967.2246
HEU-SOL-THERM	2	−229.6839	−220.7061	−223.2024	−287.6099
IEU-MET-FAST	4	80.9326	278.8042	271.8004	31.2530
LEU-SOL-THERM	2	277.5186	453.3140	438.3028	241.5317
MIX-MET-FAST	2	−199.5340	52.0550	44.5480	47.0501
MIX-COMP-THERM	1	−1,030.5267	−122.7055	−190.5427	−282.3224
PU-MET-FAST	6	−13.1667	131.5000	−21.0000	13.8333
U233-MET-FAST	1	−470.0000	−103.0000	−67.0000	3.0000
U233-SOL-INTER	1	−3,967.0000	−1,560.0000	−4,316.0000	−4,732.0000
U233-SOL-THERM	2	−237.0000	157.0000	96.0000	−24.5000

N is the number of benchmarks in the category.
COMP, compound system; ENDF, Evaluated Nuclear Data File; FAST, fast system; HEU, high enriched uranium; IEU, intermediate enriched uranium; INTER, intermediate; LEU, low enriched uranium; MET, metal; MIX, mixed uranium and plutonium system; PU, plutonium; SOL, solution; THERM, thermal.

Table 6
MCNP(X) calculations of keff values—benchmark keff values.

U233-SOL-INTER-001-case-1		
Library/benchmark keff	Keff	$\left(\frac{C}{E}-1\right)$ in pcm
Benchmark keff	1.00000 ± 0.0083	
ENDF/B-VI.8 (Be from ENDF/B-VII.0)	0.98516 ± 0.0002	−1,484.0000 ± 12.3209
ENDF/B-VII.0	0.98440 ± 0.0002	−1,560.0000 ± 12.9519
ENDF/B-VII.1 (Be from ENDF/B-VII.0)	0.98444 ± 0.0002	−1,556.0000 ± 12.9187
ENDF/B-VIII.0β4 (Be from ENDF/B-VII.0)	0.98316 ± 0.0002	−1,684.0000 ± 13.9814

Values of $\frac{C}{E}-1$ (in pcm)—replacement of Be data in the ENDF/B-VIII.0β4, ENDF/B-VII.1, and ENDF/B-VI.8 by those in ENDF/B-VII.0.
ENDF, Evaluated Nuclear Data File; INTER, intermediate; SOL, solution.

ACE format compatible with MCNP and run a significant number of benchmark cases without issue. The three-dimensional and continuous-energy MCNP(X) Monte Carlo code was used in our neutronic calculations with the aim of reducing all errors due to approximations in geometry modeling or cross-section processing.

The obtained ENDF/B-VIII.0β4 results for most cases of benchmarks improved, with the exception of the U233-SOL-INTER-001-case-1 system. Indeed, the calculated values of $\frac{C}{E}-1$ and their uncertainties are better, especially for certain benchmark cases, compared with older evaluations. On the contrary, for the rest of the benchmark cases, the differences with older evaluations are not large. In addition, the improvement of this library can also be observed for calculations of the average values of $\frac{C}{E}-1$ per benchmark category (see Table 5).

The keff values calculated using the Be data in the ENDF/B-VIII.0β4, ENDF/B-VII.1, and ENDF/B-VI.8 data libraries were replaced with those in the original ENDF/B-VII.0 library and show very good agreement with each other.

From this work, we can underline the following remark: Based on the obtained results for the U233-SOL-INTER-001-case-1 benchmark, incident neutron data and thermal neutron scattering data need to be revised for Be in ENDF/B-VIII.0β4 evaluation when this Be is used as a reflector.

Conflict of interest

There is no conflict of interest.

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Appendix 1

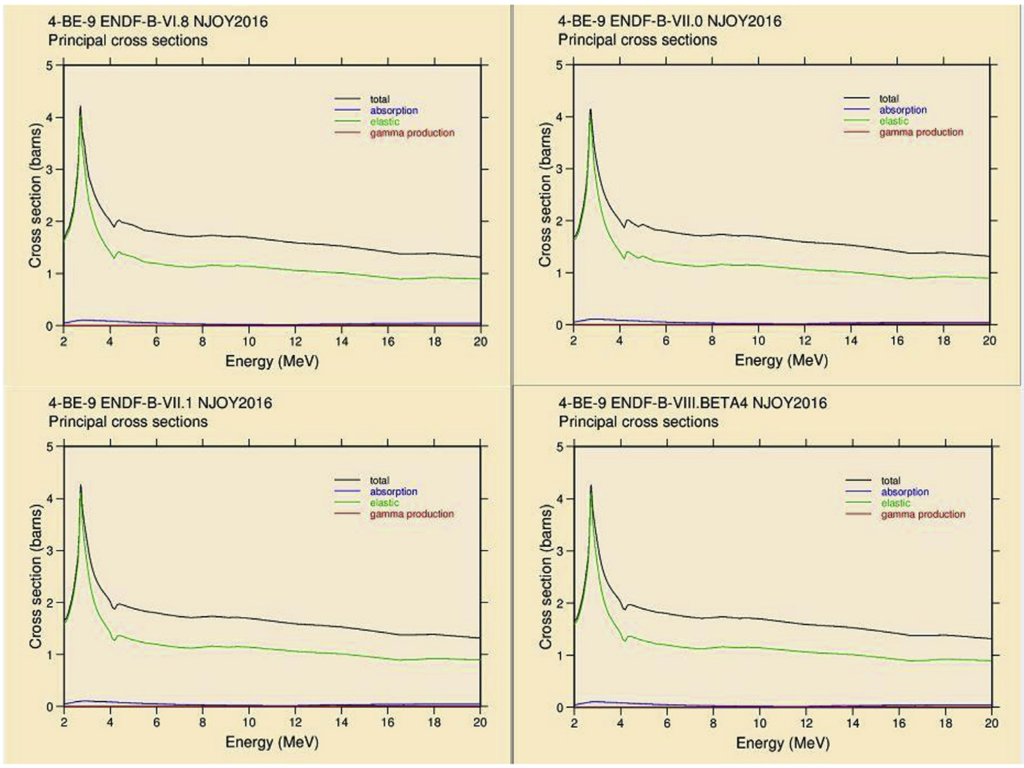

Fig. A1. Principal Be cross sections as a function of energies for the four libraries. ENDF, Evaluated Nuclear Data File.

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