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Criticality safety benchmark calculations using the new evaluated data libraries

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Abstract. The criticality safety benchmark calculations with the RTS&T general purpose Monte Carlo code were extensively used to validate the ^{233,235}U and ^{239}Pu evaluated data files. The benchmarks employed are from the International Criticality Safety Benchmark Evaluation Project (ICSBEP). We have performed calculations using the newly evaluated data files from ENDF/B-VII.0 and special purpose Minsk Actinides Library. The influence of the prompt fission neutron spectra of major actinides on the k_{eff} for the selected fast spectrum benchmarks is investigated.

1 Introduction

This paper presents the results of the criticality safety calculations for the fast spectrum benchmarks from the International Handbook of Criticality Safety Benchmark Experiments [1]. All results reported in this paper were obtained by use of the general purpose Monte Carlo code RTS&T [2,3] and continuous-energy evaluated neutron data libraries ENDF/B-VII.0 [4] and Minsk Actinides Library [5].

2 Brief description of the RTS&T data-driven transport model

The RTS&T code uses the continuous-energy nuclear and atomic evaluated data files to simulate of radiation transport and discrete interactions of the particles in the energy range from thermal energy up to 150 MeV. In contrast with the MCNP the ENDF-data driven model of the RTS&T code has access evaluated data directly. The RTS&T code uses continuous-energy nuclear and atomic evaluated data files to simulate of radiation transport and discrete interactions of the particles in the energy range from thermal energy up to 20/150/3000 MeV. In contrast with the MCNP the ENDF-data driven model of the RTS&T code has access evaluated data directly. A first model version was originally develoved in 1997 to simulate of low-energy (up to 20 MeV) neutron transport using the neutronic ENDF/B-VI evaluated data library. In current model development all data types provided by ENDF-6 format are takes account in the coupled multi-particle radiation transport modelling. Universal data reading and preparation procedure allows to use various data library written in the ENDF-6 format (JENDL, FENDL, CENDL, JEF, BROND, LA150, ENDF-HE/VI, IAEA Photonuclear Data Library, etc.). ENDF data pre-processing (linearization, restoration of the resolved resonances, temperature dependent Doppler broadening of the cross sections and checking and correcting of angular distributions and Legendre coefficients for negative values are produced automatically with the Cullen's ENDF/B Pre-processing codes [6] LINEAR, RECENT (RECEN-DD for Reich-Moore parameters of several isotopes of JENDL library only), SIGMA1 and LEGEND rewritten in ANSI standard FORTRAN-90. ENDF-recommended interpolation laws are used to minimize the amount of data. For data storage in memory and their further use the dynamically allocated tree of objects is organized. All types of reactions provided by ENDF-6 format are taken into account for the particle transport modelling: elastic scattering, radiative capture and production of one neutron in the exit channel, absorption with production of other type particles (with division on excited states of the residual nucleus), the fission with separate yields of prompt and delayed neutrons and residual nucleus simulation by MF = 8 data, etc. The energies and angles of emitted particles are simulated according to the distributions from MF = 4, 5, 6, 12, 13, 14 and 15 files. For example, the following representations of outgoing energy-angle distributions for secondary particle can be used: tabular energy distributions, angular distributions via equally-probable cosine bins, Kalbach-Mann systematics for continuum energy-angle distributions (44 ENDF law), discrete two-body scattering, N-body phase-space energy distributions.

3 Description of the criticality benchmarks

3.1 ^{233}U fast spectrum benchmarks

U233-met-fast001 [umf-001] (1 cases)

The ^{233}U enrichment was 98.13%. The isotopes in these benchmark models are ^{233,234,235,238}U. A 16.535 g sphere ^{233}U with a density of 18.424 g/cm^3 has a radius of 5.9838 cm.

U233-met-fast003 [umf-003.1] (2 cases)

Highly enriched ^{233}U sphere, reflected by ^{238}U. The mass of the uranium-233 core was 10 kg (case 1) and 7.6 kg (case 2). The density of the 7.6 kg core was 18.644 g/cm^3 and 18.621 g/cm^3 for the 10 kg core. The radius of the 10 kg core was 5.0444 cm. The 10 kg sphere was reflected by 2.3012 cm of normal uranium (outer radius of 7.3456 cm). The radius of

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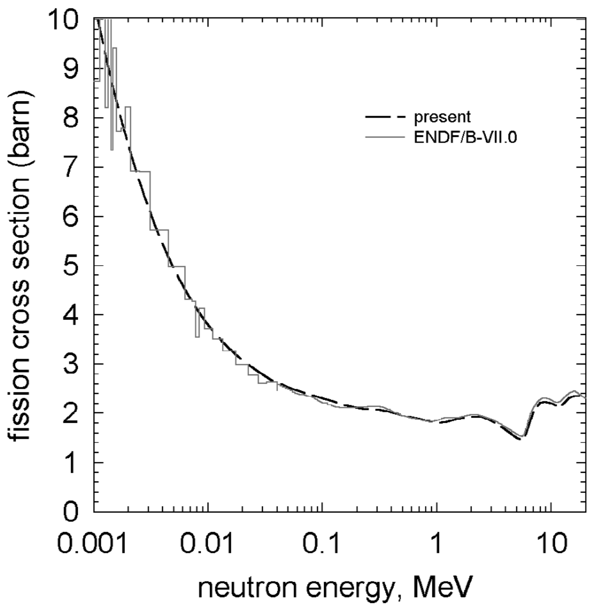

Fig. 1. Fission cross section of ^{233}U(n,f).

the 7.6 kg core was 4.5999 cm. The 7.6 kg sphere was reflected by 5.3086 cm of normal uranium (outer radius of 9.9085 cm).

U233-met-fast005 [umf-003.2] (2 cases)

Highly enriched ^{233}U sphere, reflected by berillium. The mass of the uranium-233 core was 10 kg (case 1) and 7.6 kg (case 2). The density of the 10 kg core was 18.621 g/cm^3 and 18.644 g/cm^3 for the 7.6 kg core. The radius of the 7.6 kg core was 4.5999 cm. The 10 kg sphere was reflected by 2.0447 cm of berillium (outer radius of 7.0891 cm). The radius of the 10 kg core was 5.0444 cm. The 7.6 kg sphere was reflected by 4.1961 cm of berillium (outer radius of 8.7960 cm).

3.2 ^{239}Pu fast spectrum benchmarks

PU239-met-fast001 [pmf-001] (1 case)

A bare sphere of plutonium. The ^{239}Pu enrichment was 95.2%. The isotopes in these benchmark models are ^{nat}Ga, ^{239,240,241}Pu. A 17.020 g sphere of plutonium alloy with a density of 15.61 g/cm^3 has a radius of 6.3849 cm.

PU239-met-fast002 [pmf-002] (1 case)

A bare sphere of plutonium. The plutonium enrichment was 76.4% ^{239}Pu and 20.1% ^{240}Pu. The isotopes in these benchmark models are ^{239,240,241,242}Pu. A 19.460 g sphere of plutonium alloy with a density of 15.73 g/cm^3 has a radius of 6.6595 cm.

PU239-met-fast005 [pmf-005] (1 case)

Highly enriched ^{239}Pu sphere, reflected by tungsten. The plutonium enrichment was 94.79% ^{239}Pu, 4.90% ^{240}Pu and 0.10% ^{241}Pu. The density of the core alloy is 15.778 g/cm^3. The radius of the 8471 g delta-phase plutonium sphere was

Table 1. Results for high-enriched uranium benchmarks with a fast spectrum.

Benchmark	Exp.	ENDF/B-VII.0	Minsk Actinides
umf-001	1.00000(100)	0.99937(16)	0.99965(18)
umf-003.1	1.00000(100)	0.99781(21)	0.99725(12)
umf-003.2	1.00000(100)	0.99742(24)	0.99725(12)
umf-005.1	1.00000(90)	0.99618(17)	0.99480(24)
umf-005.2	1.00000(60)	0.99341(14)	0.99418(21)

Table 2. Results for high-enriched plutonium benchmarks with a fast spectrum.

Benchmark	Exp.	RTS&T / ENDF/B-VII.0	MCNP-4C3/ ENDF/B-VII.0 [7]
pmf-001	1.00000(200)	1.00012(21)	1.00016(19)
pmf-002	1.00000(200)	1.00011(72)	1.00019(65)
pmf-005	1.00000(130)	1.00130(81)	1.00825(74)

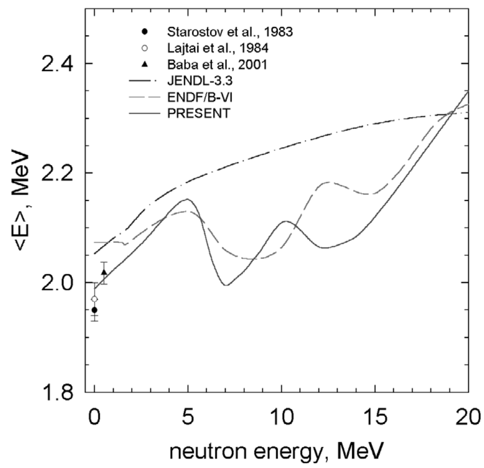

Fig. 2. ⟨E⟩ of PFNS for ^{233}U(n,f).

5.0419 cm. The sphere was reflected by 4.699 cm tungsten alloy (outer radius of 9.7409 cm) for the isotopic composition: 91.3% $^{182-184,186}$W, 5.5% $^{58,60-62}$Ni, 2.5% ^{63,65}Cu, and 0.7% $^{90-92,94,96}$Zr.

4 Results of criticality calculations

The RTS&T sample calculation was run with 110 generations of 2500 histories per generation and the first 10 generations skipped for a total of 250,000 active histories. The values of k_{eff} quoted from the three-combined estimator, and all calculated uncertainties listed are estimated statistical uncertainties at the 1 σ level. The results from calculations based on ENDF/B-VII.0 [4] and Minsk Actinides [5] files for ^{233}U benchmarks with a fast spectrum are given in table 1. The results from calculations by use the RTS&T and MCNP codes based on ENDF/B-VII.0 for plutonium benchmarks are given in table 2.

Table 1 shows, that notwithstanding the ^{233}U(n,f) fission cross section differences in unresolved resonance and fast neutron energies (see figure 1) the k_{eff} are quite similar. In [11] fission cross section estimate in fast neutron energy range is based on absolute fission cross section measurements. It seems that the fission cross section differences are compensated by the differences of the prompt fission neutron spectra. Figure 2 shows that even the average energies of the prompt fission neutron spectra are rather different (in ENDF/B-VII.0 the PFNS of ENDF/B-VI are adopted). Present estimate is supported by the experimental data of [8–10]. The present prompt fission neutron spectra shapes [11] much differ from those of ENDF/B-VII, just in the manner as it was shown for the ^{235}U(n,f) reaction [12]. That criticality comparison for the ^{233}U benchmarks might be considered as a hint for removing various inconsistencies in ^{235}U benchmarks [4].

5 Conclusion

The validation of the evaluated data libraries ENDF/B-VII.0 and Minsk Actinides Library has been performed against criticality safety benchmark experiments for high-enriched ^{233}U and plutonium spheres with a fast spectrum. The results demonstrated a good agreement with experimental data of the present Monte Carlo code calculations. The prompt fission neutron spectra representation based on differential data analysis and systematics, proven in case of ^{235}U(n,f) fission data, gives quite good description of k_{eff}. The RTS&T results reasonably agrees with MCNP criticality calculations.

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