ANALYSIS OF LWR BENCHMARKS BASED ON DIFFERENT METHODS AND NUCLEAR DATA EVALUATIONS

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ABSTRACT

For reactor physics applications, cross section data sets based on the latest versions of the available evaluated nuclear data files JEF-2.2, JEFF-3, ENDF/B-VI up to release 5, and JENDL-3.2 were generated for continuous Monte Carlo and multigroup transport calculations. The libraries contain nuclides for actinides, structure materials, and fission products, necessary for LWR problems with fresh or irradiated fuel for the temperature range from 293 K up to 3000 K. They were prepared with NJOY version 97 using a reconstruction accuracy of 0.1 % for all data sets. The data libraries mainly will be used for calculating safety related reactor physics parameters for standard and innovative fuel in core configurations under operational and accidental conditions, and criticality safety as well. As transport codes, the continuous Monte Carlo code MCNP and multigroup codes based on S_N (e.g. XSDRN) or Monte Carlo (e.g. KENO-Va or KENO-VI) methods will be used. The calculation of weighted cross sections for the multigroup codes were performed by the first collision spectral code RESMOD solving the slowing down equation for 26,000 energy points. Since today several data evaluations are available, for main nuclides data sets were generated for different evaluations to see their influence on integral reactor parameters. To verify the generated libraries, numerous benchmark experiments relevant for LWR systems were analysed and compared with measured values. The conclusion of these comparisons is that for homogeneous systems, generally JENDL-3.2 based calculations give higher k_{eff} values than JEF-2.2 or ENDF/B-VI based values. A number of heterogeneous UOX and MOX systems agree very well with experiments if JENDL-3.2 based data were used, JEF-2.2 and ENDF/B-VI based calculations lie always at the lower bound of the given experimental uncertainty. Calculations for heterogeneous UOX systems based on the new release of data for U-235 and U-238 for ENDF/B-VI (Release 5) generally show too low k_{eff}

values (up to 1 %). For some homogeneous highly enriched uranium nitrate water solutions as well as for plutonium nitrate water solutions, the JENDL based calculations overpredict the measured values, while the JEF-2.2 and ENDF/B-VI based calculations agree very well with experimental data.

1. INTRODUCTION

For solving safety related reactor physics problems, adequate methods for the solution of the neutron transport equation and nuclear cross section libraries are required. Both nuclear data and calculational methods must serve as a tool for accurate and realistic calculation of safety parameters for reactors and other configurations containing fissionable material. This must be also true for complex geometry and material composition even for configurations under accidental conditions, such as

- Reactivity changes for redistributed absorbers in boiling water reactors.
- Deboration of primary coolant and corresponding reactivity increase.
- Recriticality after redistribution of fuel material and modified moderator density.
- Influence of burnable poisons under accidental conditions.

For all these conditions, criticality and reactivity coefficients must be known as accurately as possible. Furthermore, the calculational system should be able to reliably perform verification of fuel assembly designs for high burn-up, multiple recycled Pu, and minor actinides, or for unusual lattice parameters, as well as for new fuel assembly designs. Only for a part of these tasks experimental data are available, so that the calculation of systems not covered by experiments must be performed from first principles and must be based on the best cross section libraries available.

For the calculation of reactor physics parameters, all relevant nuclear reactions within the total energy range have to be taken into account. In particular, the resonance structure of the cross sections and the spatial and spectral interactions and shielding effects have to be considered explicitly, even for complex compositions and geometry. This can be done for arbitrary geometry by the point-wise Monte Carlo method. For regular lattices, methods based on a combination of a spectral and resonance shielding calculation for a lattice cell, followed by a multigroup transport calculation, can be applied. Necessary for such calculations is a validated cross section data library, containing the reactivity relevant nuclides and covering the temperature range from room temperature to temperatures at accident conditions. Therefore, cross section libraries for the point-wise Monte Carlo method (realized in the MCNP [1] code) and for multigroup applications in the AMPX format, as used in the SCALE code package [2], with 292 energy groups were processed. These libraries contain the cross sections of the most important nuclides such as actinides, structure materials, neutron poisons, and moderators, as well as fission products, covering the requested temperature range. For the application of the multigroup method, the resonance shielding and spectral module RESMOD [3] with JEF-2.2 [4] data was used, followed by the SCALE transport modules XSDRN and KENO. RESMOD solves the slowing down equation for an arbitrary number of energy points (e.g. 26,000 points) for one-dimensional cells,

taking into account overlapping of resonances in any material zone. For special cases, collision probabilities for 3D cells can be used, too.

The cross section libraries for MCNP and RESMOD/SCALE were generated with the nuclear data processing system NJOY-97 [5]. These libraries are mainly based on JEF-2.2 and ENDF/B-VI.5 [6] data. They contain data for all relevant nuclides for temperatures from 300 K to 3000 K. The resonance reconstruction and linearization accuracy in generating the data was generally 0.1 %. For additional comparisons, also data sets based on JENDL-3.2 [7] and partly JEFF-3 [4] were created.

For validation of the cross section libraries, a number of criticality calculations for benchmarks with LWR uranium and MOX lattices, MTR lattices as well as homogeneous Pu nitrate and U nitrate solutions were performed. These benchmarks are based on experiments, and are well documented for re-calculation. Benchmarks documented in [8], KRITZ experiments [9] and CSEWG [10] benchmarks not contained in [8] were selected. The benchmarks are sensitive for the principal fissile and fertile materials, structure materials, and moderator for LWR. Mainly the critical configuration (k_{eff}) was calculated. For some benchmarks, also integral parameters such as reaction rate ratios were given. Both configurations at room temperature and higher temperatures were analysed to verify the data sets generated for higher temperatures also.

2. ANALYSIS OF HOMOGENEOUS SYSTEMS

For homogeneous configurations, fast systems with uranium and plutonium, GODIVA (U-235) and JEZEBEL (Pu-239, benchmark PU-MET-FAST-001), as well as thermal systems with highly enriched uranyl nitrate (HEU-SOL-THERM-013, unreflected spheres measured at ORNL), plutonium nitrate (PU-SOL-THERM-001 to PU-SOL-THERM-006) and mixed uranium plutonium nitrate solutions (MIX-SOL-THERM-004) were regarded. Results are given for the fast systems in Table I. The specification of these benchmarks are given in [8].

The calculated k_{eff} values for GODIVA are underestimated by about 0.4 % for ENDF/B-VI.5 and JEF-2.2 and by about 0.2 % for ENDF/B-V. Using JENDL-3.2 data, the agreement with the experiment looks perfect. The measured central reaction rate ratio σ_f (U-238) / σ_f (U-235) agrees inside the experimental error bounds with the calculations based on ENDF/B-VI.5, JEF-2.2, and JENDL-3.2. ENDF/B-V is slightly overestimating. For JEZEBEL, the calculations based on ENDF/B-V, ENDF/B-VI.5, JEF-2.2, and JEFF-3 agree very well with the experimental values, however, with JENDL-3.2 k_{eff} is slightly underestimated. If JEFF-3 [4] data were used for the Pu isotopes instead of JEF-2.2, k_{eff} for JEZEBEL increases to 1.0012.

The results for the highly enriched uranyl nitrate water solutions (ORNL experiments HEU-SOL-THERM-0013) from [8] are shown in Fig. 1. The benchmarks are sensitive to U-235. Calculations were based on ENDF/B-VI (release 5), JEF-2.2, and JENDL-3.2. In this figure, the results are shown together with the experimental values. The ENDF/B-VI and JEF results are nearly identical, but underestimate k_{eff} compared to experimental values up to 0.4%. JENDL-3.2

based results lie inside the estimated experimental error bounds. No significant changes in k_{eff} were found taking ENDF/B-VI release 5 compared to release 0.

Table I. Fast homogeneous systems from the International Handbook of Evaluated Criticality Safety Benchmark Experiments [8], calculated with MCNP with different data libraries.

a) HEU-MET-FAST-001 (highly enriched uranium metal sphere, GODIVA experiment)

Parameter	Experiment Mean (std dev)	ENDF/B-V Mean (std dev)	ENDF/B-VI.5 Mean (std dev)	JEF-2.2 Mean (std dev)	JENDL-3.2 Mean (std dev)
k_{eff}	1.0000 (0.001)	0.9979 (0.0003)	0.9966 (0.0003)	0.9952 (0.0003)	1.0008 (0.0003)
σ_f(U-238) / σ_f(U-235)	0.1647 (0.0018)	0.1722 (0.0015)	0.1597 (0.0016)	0.1616 (0.0015)	0.1623 (0.0016)

b) PU-MET-FAST-001 (Pu-239 metal sphere, JEZEBEL experiment)

Parameter	Experiment Mean (std dev)	ENDF/B-V Mean (std dev)	ENDF/B-VI.5 Mean (std dev)	JEF-2.2 Mean (std dev)	JENDL-3.2 Mean (std dev)
k_{eff}	1.0000 (0.002)	0.9978 (0.0003)	0.9978 (0.0002)	0.9978 (0.0003)	0.9963 (0.0003)
σ_f(Pu-239) / σ_f(U-235)	1.4480 (0.029)	1.4120 (0.008)	1.4259 (0.008)	1.4344 (0.008)	1.4210 (0.008)
σ_f(U-238) / σ_f(U-235)	0.2137 (0.0023)	0.2050 (0.0014)	0.2092 (0.0014)	0.2087 (0.0014)	0.2115 (0.015

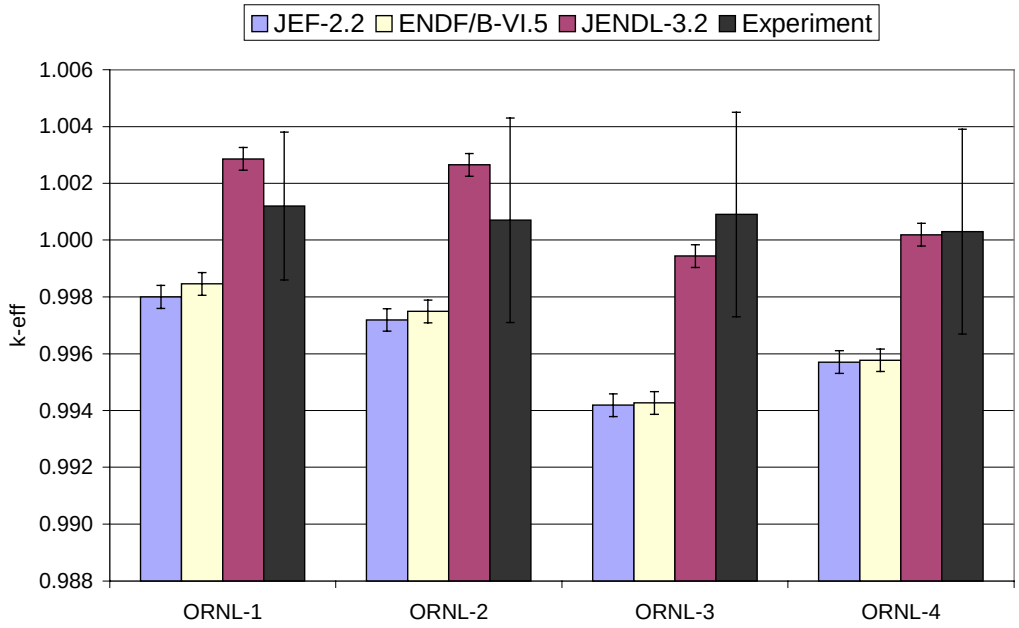

Figure 1. k_{eff} for ORNL Experiments 1–4: MCNP calculations with different data libraries compared to measured values.

The benchmarks PU-SOL-THERM (also known as PNL experiments) from [8] represent Pu nitrate/water solutions with Pu-239 as mainly fissionable material, but they are also sensitive to Pu-240. Several water reflected spheres with Pu nitrate/water solutions, which also represent experiments in the environment of minimal critical mass, were selected. The results are shown in Fig. 2 for the benchmarks PU-SOL-THERM-001 to PU-SOL-THERM-006. For these benchmarks, the average of JEF-2.2 and ENDF/B-VI.5 based MCNP results for k_{eff} agree well inside the estimated error bounds of experiments (0.3 % – 0.5 %). The mean JENDL based results lie about 0.46 % higher. Compared to earlier results based on ENDF/B-IV, JEF-1, or ENDF/B-V [8], the JEF-2.2 and ENDF/B-VI.5 results are up to 1 % lower and agree now much better with the experimental values.

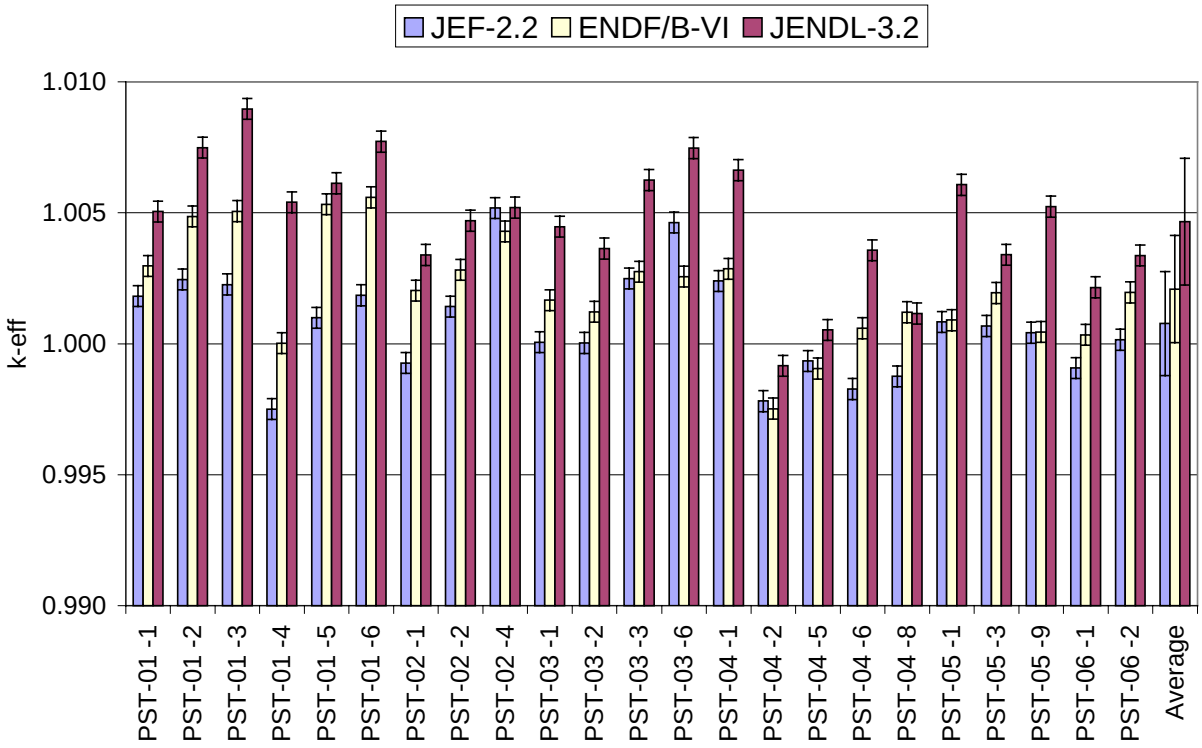

Figure 2. k_{eff} for the benchmarks PU-SOL-THERM-001 – 006 (PNL-Experiments) calculated with MCNP and different nuclear data.

For systems with mixed uranium and plutonium nitrate and water, benchmarks from MIX-SOL-THERM-004 [8] were analysed. The results are shown in Fig. 3. For these mixtures, the results calculated with JEF-2.2 (and ENDF/B-VI) are about 0.5 % lower than the average experimental values. The JENDL based MCNP results agree here, opposite to the pure Pu solutions, rather well with the experimental values. The ENDF/B-V values are quite comparable to the JENDL-3.2 values.

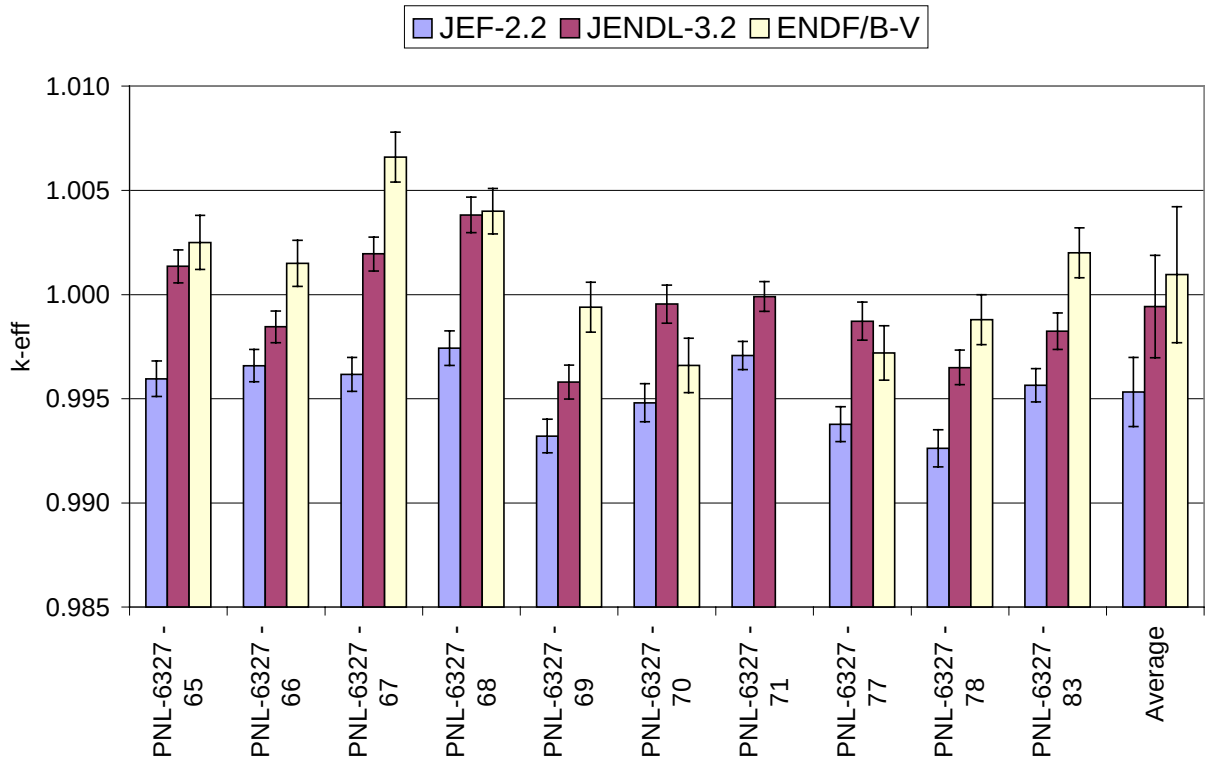

Figure 3. k_{eff} for benchmarks MIX-SOL-THERM-004 (PNL-Experiments). MCNP calculations based on different cross sections. ENDF/B-V results are taken from [8].

3. ANALYSIS OF HETEROGENEOUS SYSTEMS

For MTR type fuel elements of research reactors (thin plates with uranium/aluminum fuel cladded by aluminum), there exist several experiments with different configurations of fuel assemblies. The benchmarks are described in detail as HEU-MET-THERM-006 (SPERT-D experiments, highly enriched uranium fuel) in [8]. Calculations were performed by MCNP based on the three evaluations ENDF/B-VI.5, JEF-2.2 and JENDL-3.2. Results are shown in Fig. 4. Comparable to results for homogeneous solutions with highly enriched uranium, the ENDF/B-VI.5 and JEF-2.2 results are close together, and the JENDL-3.2 results are 0.4 to 0.7 % higher. Due to the thin fuel plates, the heterogeneity of these experiments is only weak, and therefore there is no reason for principally other results than for the homogeneous cases (ORNL benchmarks). However, for these series of experiments, the average value of all calculated cases agrees better with the experimental value (k_{eff} = 0,9997 ± 0.004 for JEF-2.2 and 0.9980 ± 0.004 for ENDF/B-VI.5 data). With JENDL-3.2 the average of calculations is about 0.6 % higher than the average of the selected experiments.

For heterogeneous systems with low enriched uranium fuel, many experiments have been performed for LWR lattices. Benchmarks for testing computational systems can be taken again from [8], e.g. from the LEU-COMP-THERM series or from other sources like [11] (ANS

benchmarks), KRITZ [9] or CSEWG[10] (e.g. TRX-1 – 4 experiments with metallic fuel for which the benchmarks are not yet specified in [8]).

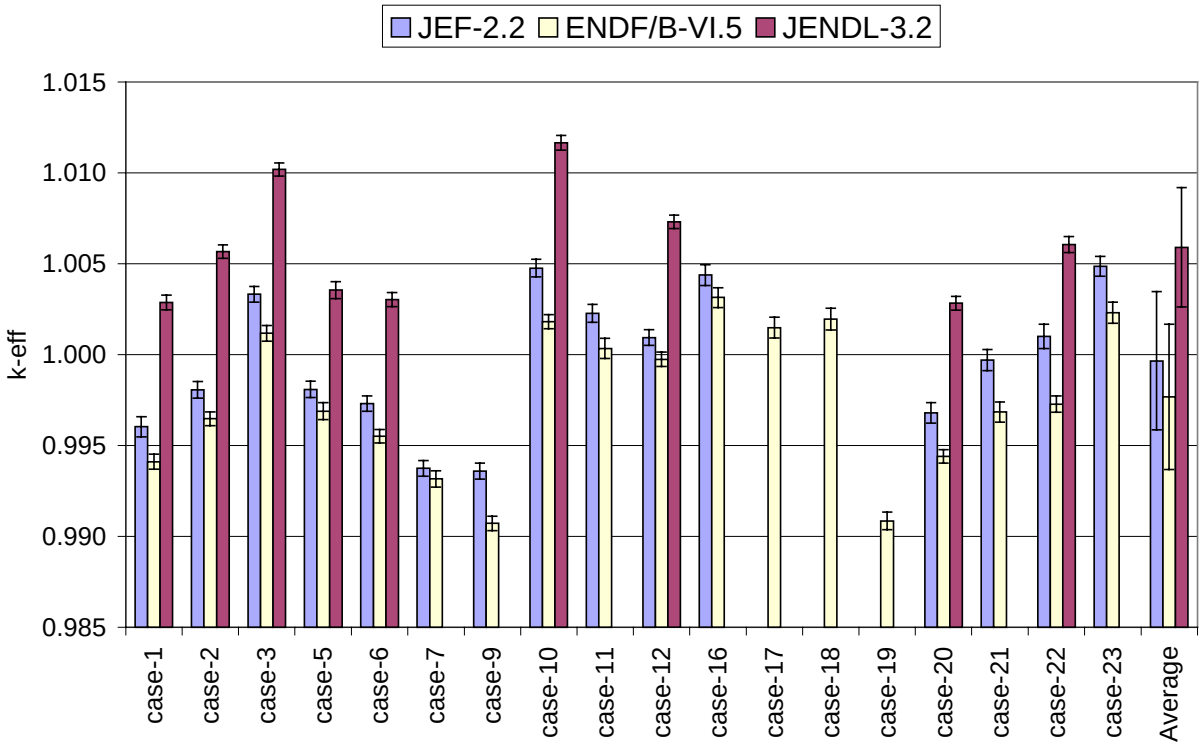

Figure 4. k_{eff} for SPERT-D Experiments, HEU-MET-THERM-006: MCNP calculations based on different cross section libraries.

First, benchmarks from LEU-COMP-THERM-001, -002 and -006 (enrichment 2.35 w/o, 4.31 w/o and 2.6 w/o, respectively) series were calculated with MCNP based on JEF-2.2, and compared with MCNP results based on JENDL-3.2 or ENDF/B-V given in [8]. Additional calculations were made for some of these benchmarks based on ENDF/B-VI release 5. The k_{eff} values of these calculations are shown in Fig. 5. The results show that JENDL based values lie about 0.4 to 0.7 % higher than JEF-2.2 based results. ENDF/B-V values are closer to JEF-2.2 results. The JENDL based values, however, agree with corresponding experiments well inside the given experimental uncertainties, JEF-2.2 based values partly are below the lower limit of the band of uncertainty.

The comparison of ENDF/B-VI.5 based calculations with experiments show a strong underprediction of up to 0.9 %. The main reason for this effect is due to the increased capture cross section of U-235 which is not compensated. Even considering some improved results using the new evaluated U-235 cross sections (e.g. for dry powders and U-236 buildup [12, 13]), for the low enriched LWR lattices, the results are not acceptable neither for design calculations nor for criticality safety. If the U-235 cross sections as in release 5 are correct, then additional modifications in other cross sections (U-238) are necessary for compensating the increased capture.

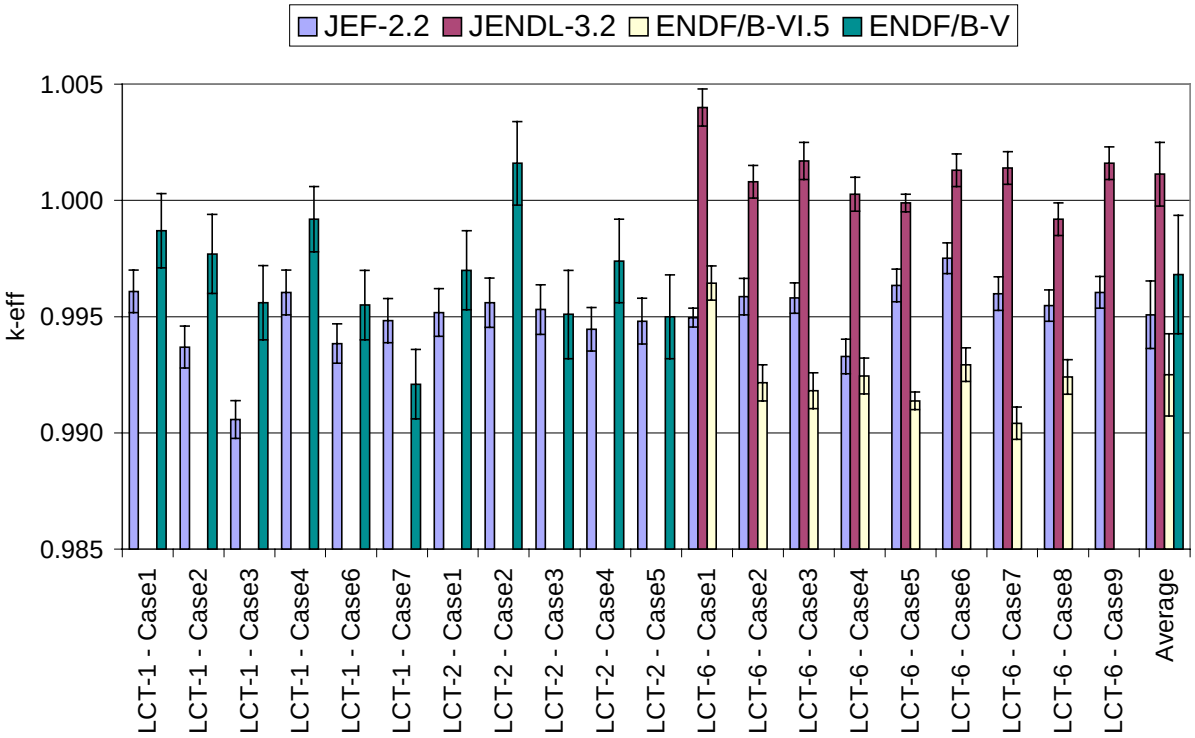

Figure 5. k_{eff} calculated by MCNP with different nuclear data for LEU-COMP-THERM benchmarks (UOX LWR lattices). Values for ENDF/B-V and JENDL-3.2 from [8].

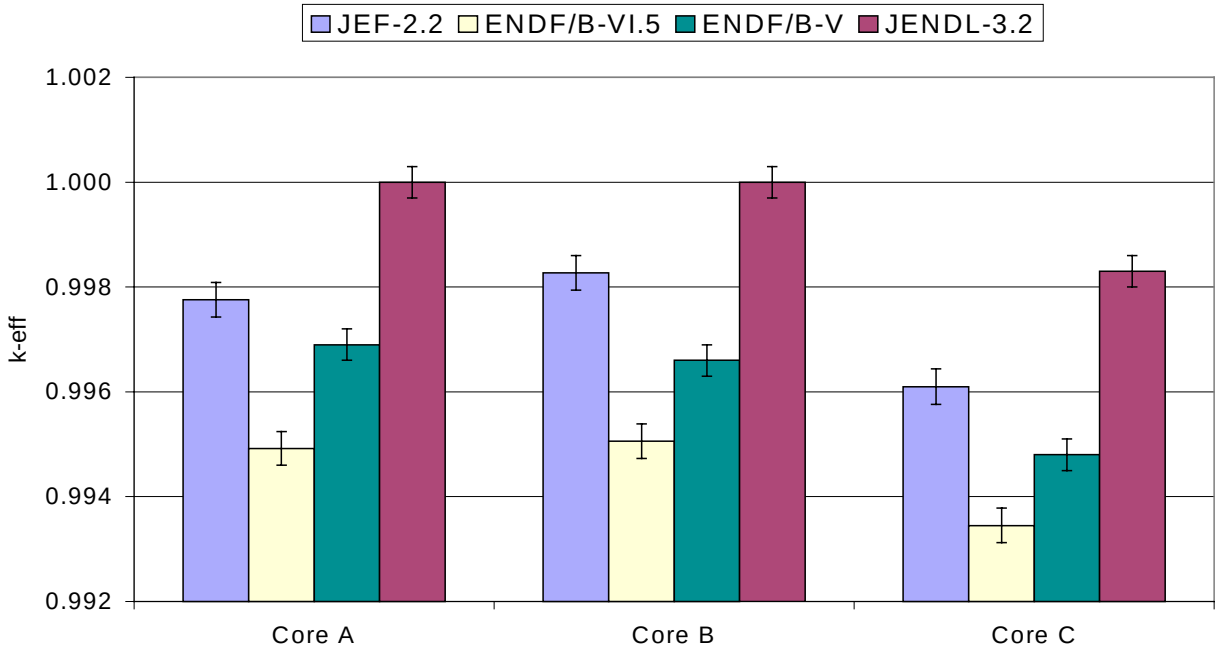

Figure 6. MCNP results for ANS benchmarks core A – C with different data evaluations.

In addition, the principal influence of JEF-2.2, ENDF/B-VI.5, ENDF/B-V, and JENDL-3.2 was analysed for three cores of an ANS benchmark [11]. The results are given in Fig. 6. Here, the same tendency of results as for LEU-COMP-THERM can be observed. JENDL shows the highest values well agreeing with the experiments. ENDF/B-VI.5 based values are about up to 0.6 % lower and below the lower limit of experimental uncertainty. JEF-2.2 based results, but also ENDF/B-V based results, lie about 0.2 % and 0.4 % lower than the experimental values, but inside the experimental uncertainty. Here, too, calculations with U-235 from ENDF/B-VI.5 lead to non acceptable underpredictions.

Hexagonal LWR lattices were analysed for the benchmarks LEU-COMP-THERM-026 and -032 [8]. The enrichment of the fuel was 4.95 w/o for LCT-026 and 10 w/o for LCT-032. For LCT-26, experiments for room temperature and temperatures from 200 to 240 °C are specified. The results are shown in Fig. 7. Only JEF-2.2 and ENDF/B-VI.5 based calculations are compared. The cases 1, 3, and 5 are at room temperature and therefore with full moderator density. Cases 2, 4, and 6 are at about 220 °C with reduced moderator density. For LCT-032, only the case at room temperature was calculated. With the exception of case 3, the JEF-2.2 based calculations agree very well with the experiments, the ENDF/B-VI.5 values lie up to 0.9 % lower than the JEF-2.2 values. The values for LCT-032 (1 0% U-235) differ by about 0.3%. This was also the case for an experiment with 17 % enriched uranium fuel (but no LWR lattice), IEU-COMP-THERM-002, case 1. For this benchmark, MCNP calculations gave for JEF-2.2 k_{eff} = 0.9962, and for ENDF/B-VI.5 k_{eff} = 0.9928. Comparing low enriched and higher enriched uranium systems, one can conclude that the strongest discrepancies between JEF-2.2 and ENDF/B-VI.5 are for low enriched systems for a broad variety of spectra.

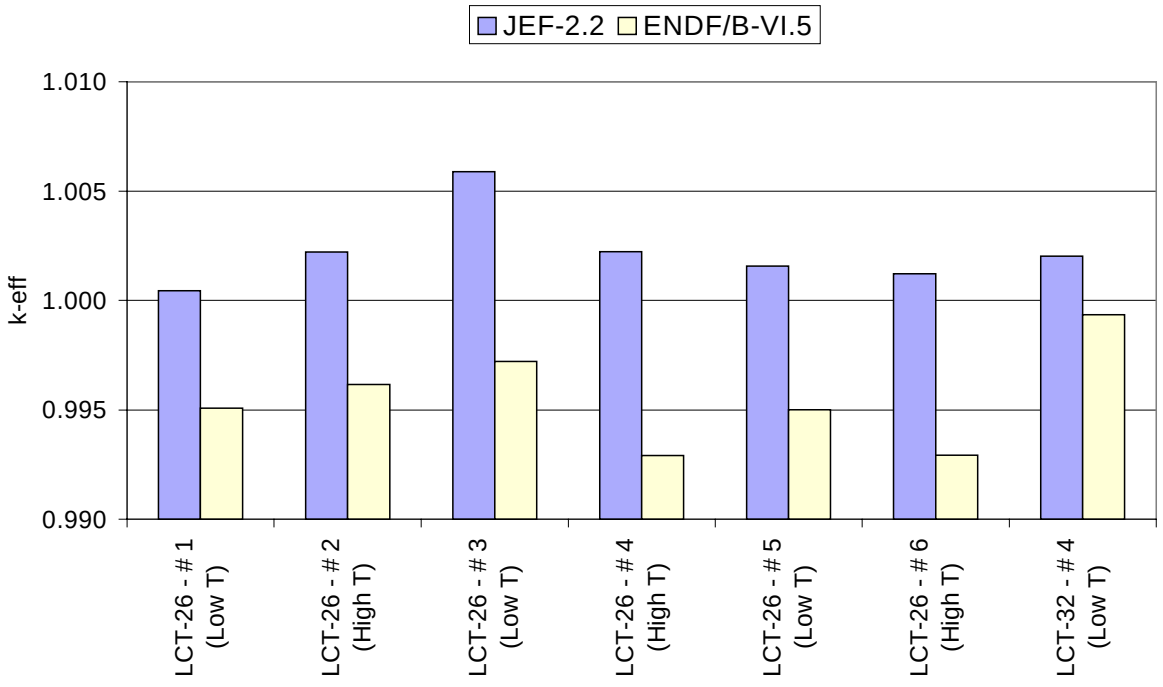

Figure 7. MCNP results for hexagonal LEU LWR lattices at room and higher temperatures.

Further hexagonal lattices, TRX-1 to TRX-4 [10], with metallic Uranium as fuel (1.3 % U-235) were analysed because these lattices are very sensitive to capture, fast fission and inelastic scattering of U-238 as well as for the U-235 fission spectrum. TRX-3 and TRX-4 consist of two region lattices with metallic fuel surrounded by UOX fuel. Both k_{eff} and reaction rate ratios were calculated for full core models by MCNP based on three nuclear data evaluations. The full core models were prepared as proposed in [14]. For comparison, lattice cell and B_3 leakage calculations using the experimental bucklings were performed with the RESMOD code [3] and JEF-2.2 data. The results are given in Table II. For the metallic fuel lattices TRX-1 and TRX-2, k_{eff} shows the same relation as for the UOX lattices analysed before. The JEF-2.2 results lie lower than

Table II. TRX benchmarks calculated with MCNP and RESMOD. With MCNP, different data libraries were used. Experimental errors and statistical uncertainties are given in brackets. ρ^{25} denotes the epithermal/thermal U-238 capture ratio, δ^{25} the epithermal/thermal U-235 fission ratio, δ^{28} the U-238/U-235 fission ratio, and C* the ratio of U-238 captures to U-235 fissions.

		Experiment	MCNP ENDF/B-VI.5	MCNP JEF-2.2	MCNP JENDL-3.2	RESMOD JEF-2.2
TRX-1	k_{eff}	1.0000	0.9906 (0.0002)	0.9936 (0.0002)	0.9943 (0.0002)	0.9964
	ρ^{25}	1.320 (0.021)	1.3406 (0.0118)	1.3400 (0.0118)	1.3360 (0.0118)	1.3400
	δ^{25}	0.0987 (0.0010)	0.0979 (0.0008)	0.0986 (0.0008)	0.0978 (0.0008)	0.0982
	δ^{28}	0.0946 (0.0041)	0.1003 (0.0007)	0.1003 (0.0007)	0.1012 (0.0007)	0.0973
	C*	0.797 (0.008)	0.7973 (0.0056)	0.7968 (0.0056)	0.7926 (0.0055)	0.7970
TRX-2	k_{eff}	1.0000	0.9911 (0.0002)	0.9930 (0.0002)	0.9947 (0.0002)	0.9952
	ρ^{25}	0.837 (0.016)	0.8479 (0.0069)	0.8475 (0.0069)	0.8449 (0.0069)	0.835
	δ^{25}	0.0614 (0.0008)	0.0610 (0.0004)	0.0615 (0.0004)	0.0611 (0.0004)	0.0603
	δ^{28}	0.0693 (0.0035)	0.0723 (0.0004)	0.0724 (0.0004)	0.0730 (0.0005)	0.0697
	C*	0.647 (0.006)	0.6456 (0.0039)	0.6459 (0.0039)	0.6423 (0.0039)	0.642
TRX-3	k_{eff}	1.0000	0.9983 (0.0002)	1.0002 (0.0002)	1.0013 (0.0002)	
	ρ^{25}	3.03 (0.05)	2.9947 (0.1078)	2.9938 (0.1078)	2.9844 (0.1077)	
	δ^{25}	0.231 (0.003)	0.2267 (0.0074)	0.2280 (0.0075)	0.2266 (0.0075)	
	δ^{28}	0.167 (0.008)	0.1884 (0.0056)	0.1879 (0.0055)	0.1898 (0.0056)	
	C*	1.255 (0.011)	1.245 (0.038)	1.241 (0.038)	1.235 (0.037)	
TRX-4	k_{eff}	1.0000	0.9986 (0.0001)	0.9999 (0.0001)	1.0016 (0.0002)	
	ρ^{25}	0.481 (0.011)	0.4801 (0.0175)	0.4798 (0.0176)	0.4783 (0.0175)	
	δ^{25}	0.0358 (0.0005)	0.0332 (0.0009)	0.0335 (0.0009)	0.0332 (0.0009)	
	δ^{28}	0.0482 (0.0020)	0.0480 (0.0011)	0.0481 (0.0011)	0.0485 (0.0011)	
	C*	0.531 (0.004)	0.5276 (0.0110)	0.5283 (0.0110)	0.5254 (0.0110)	

JENDL-3.2, the lowest values are for ENDF/B-VI.5. The RESMOD results lie slightly higher than the MCNP JEF-2.2 results. The values are too low compared to experiment. A question, however, is the validity of the 3D model which was taken from [14]. For the two region lattices TRX-3 and TRX-4, all k_{eff} calculations agree very well with experiment. In addition to $k_{eff,}$ reaction rate ratios for the core center are calculated for and compared with experimental values. Also for these calculations the 3D model was used. The corresponding results are given in Table II, too. Most of the calculated parameters agree in the frame of the experimental errors with the measured ones. For tight lattices (TRX-3, TRX-1), however, the reaction rate ratio δ^{28} between U-238 fission and U-235 fission calculated by MCNP is remarkably higher than the experimental value for all data evaluations. The RESMOD calculations show good agreement for this parameter. It may be that the 3D model chosen was not quite adeqate for interpretation of experimental values.

4. ANALYSIS OF LWR LATTICES WITH MOX FUEL

For lattices with MOX fuel, there are several experimental series defined as benchmarks described in [8]. From these series, the benchmarks MIX-COMP-THERM-002 (PNL cases 30-35, 2 w/o PuO_2), MIX-COMP-THERM-003 (SAXTON, 6.6 w/o PuO_2) and MIX-COMP-THERM-004 (TCA, 3.01 w/o PuO_2) were analysed using MCNP with JEF-2.2, ENDF/B-VI, and JENDL-3.2 data. The results are shown in Fig. 8. Like for some UOX lattices, the JENDL-3.2 based results are very close to the experimental values.

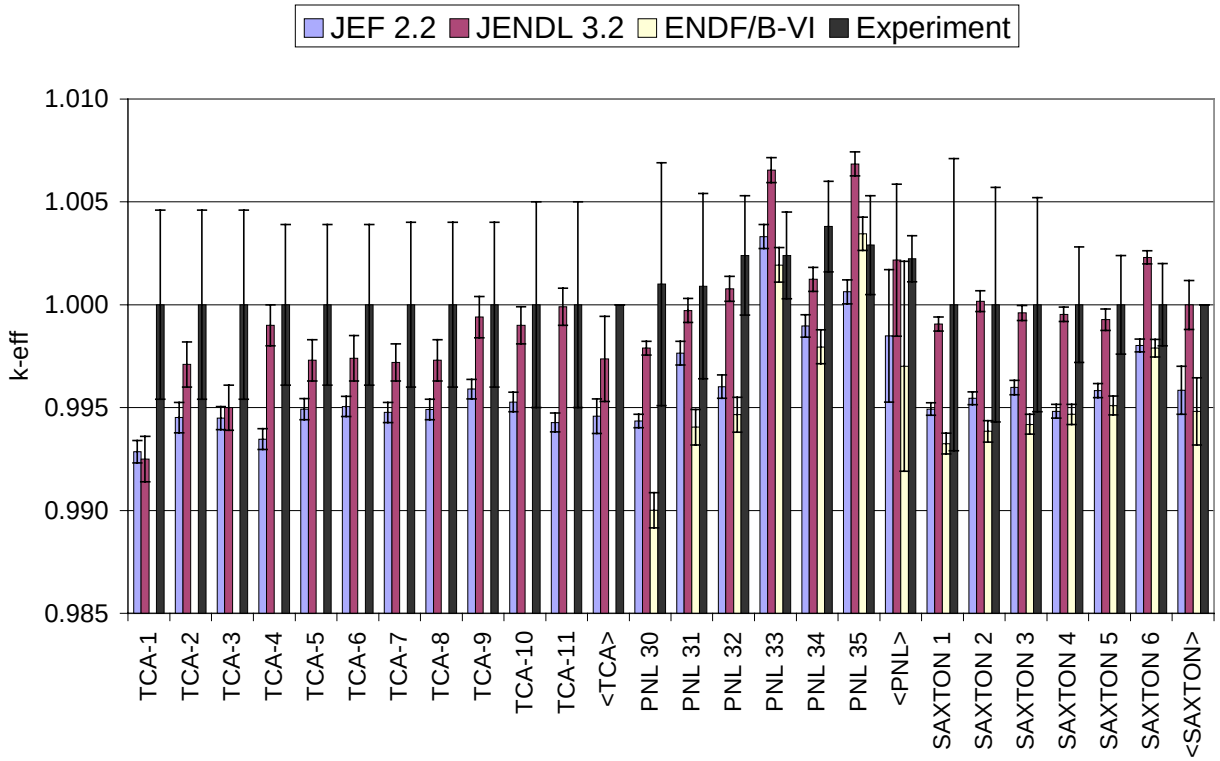

Figure 8. k_{eff} calculated by MCNP with different data libraries for MIX-COMP-THERM benchmarks (MOX LWR lattices) compared with experiment.

The best agreement was found for the TCA and SAXTON benchmarks. For PNL benchmarks, JENDL-3.2 based results tend to overestimate the k_{eff}. The JEF-2.2 results lie up to 0.4 % lower and ENDF/B-VI about 0.5 % lower (averaged over all cases) than experimental values (at the lower bound of experimental uncertainty). A sensitivity analysis showed that the cross section sets of oxygen, U-238 and Pu-239 of JENDL-3.2 data compared to JEF-2.2 or ENDF/B-VI data are responsible for the total difference. For example, the O-16 cross section alone leads to differences in k_{eff} up to 0.25 % due to different elastic scattering in the slowing down region and absorption in the MeV region. No compensating effects were found by exchanging the data sets for these nuclides from JEF-2.2 to JENDL-3.2. Since the U-235 plays no important role for the MOX fuel, the differences between ENDF/B-VI.5 and JEF-2.2 based results are not so significant as for lattices with uranium fuel.

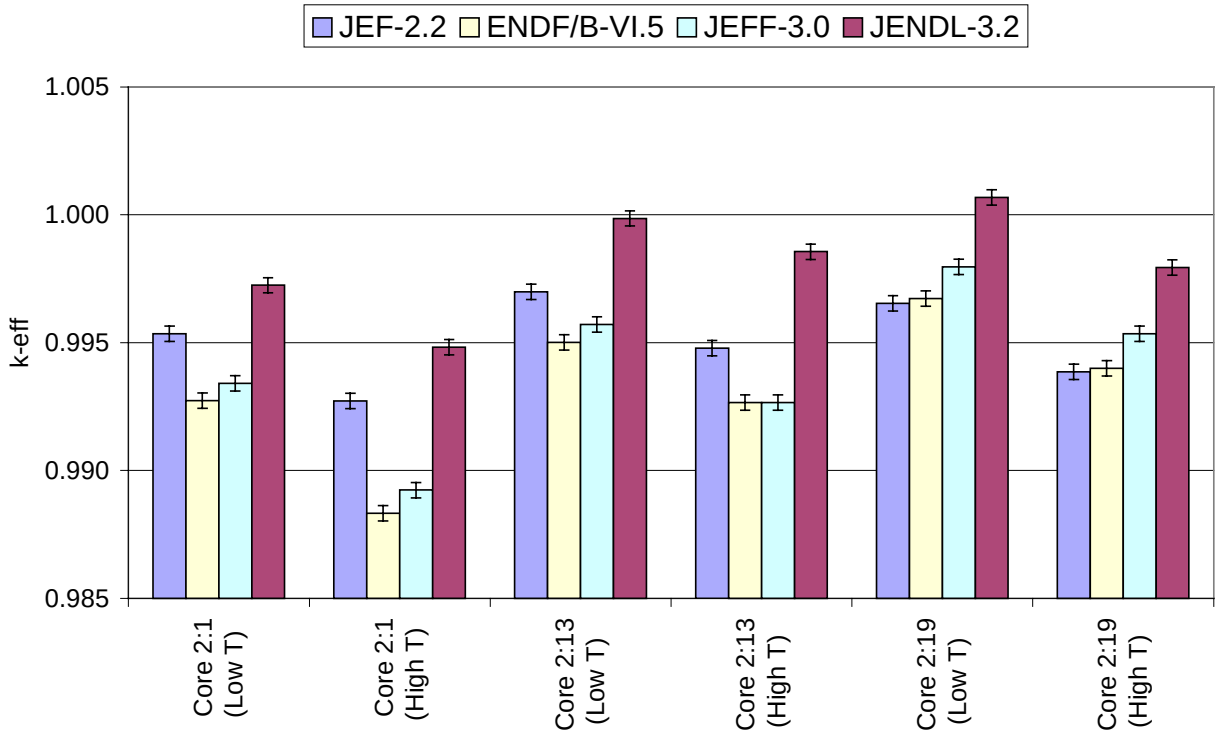

Figure 9. MCNP results for KRITZ benchmarks core 2:1, 2:13, and 2:19, each at room and high temperatures, with different data evaluations.

Further interesting experiments for testing nuclear data are the KRITZ series [9]. Here core 2:1, 2:13, and 2:19 were analysed for both room temperature and high temperature (245 °C). Core 2:1 and 2:13 are UOX cores (1.86 % U-235), core 2:19 contains MOX fuel. The results (k_{eff}) are given in Fig. 9. Here, MCNP results for the data sets mainly based on JEF-2.2, JEFF-3, ENDF/B-VI.5 and JENDL-3.2 are presented. For the UOX cores, the same relations as for the LEU-COMP-THERM and ANS benchmarks can be seen. The JENDL-3.2 based results fit the experimental results sufficiently well. The JEF-2.2 results are lower but acceptable. The ENDF/B-VI.5 results and the JEFF-3 results are significantly too low, especially for core 2:1 at high temperature. This can be explained by the identity of U-235 cross sections in these libraries. For the MOX core 2:19, the JENDL-3.2 results agree quite well with experiments. JEFF-3 based

results are lower than JENDL-3.2 but still higher than JEF-2.2 and ENDF/B-VI.5 based results. With the exception of core 2:1 at high temperature, there is obviously no increased uncertainty in calculating systems and nuclear data for higher temperatures. The same was found for the high temperature cores of LEU-COMP-THERM-026.

5. COMPARISON OF METHODS

Most of the analysis was performed with the MCNP-4B code. To obtain a comparison with other methods, computational benchmarks for UOX and MOX lattices [11] were analysed by MCNP and the cell code RESMOD together with the transport codes XSDRN and KENO-VI from the SCALE system. The N292 group library in AMPX format based on JEF-2.2 [3, 4] was used for RESMOD. The resonance calculation in the resolved resonance range was performed for 26,000 energy points. RESMOD generates averaged 292 group cross section for fuel, clad, and moderator, which were used for XSDRN and KENO-VI transport calculations with P_3 order for anisotropic scattering. The MCNP calculations are based on JEF-2.2, too. First, k_{inf} for the specified cases are given in Table III. Four cases for the UOX cell and two cases for two MOX cells with multiply recycled Pu (fuel 1) and standard fuel (fuel 2) at different fuel and moderator temperatures were regarded. The k_{inf} results from RESMOD (first collision calculation) are 0.2 to 0.3 % higher than MCNP for the UOX cell and agree within ± 0.1 % for the MOX cells. Since RESMOD and XSDRN are one-dimensional codes for cylindrical geometry, the white boundary condition was used for all cell calculations (also for KENO-VI and MCNP). The XSDRN and KENO-VI results are 0.1 to 0.27 % lower than MCNP for the UOX cell, about 0.35 % lower than MCNP for the MOX cell with fuel 1 and about 0.2 % lower than MCNP for the MOX cell with fuel 2. This seems to be a sufficiently good agreement between the two methods.

Table III: k_{inf} results for ROWLANDS benchmarks [15] (calculations based on JEF-2.2). Statistical uncertainties of the Monte Carlo results are given in brackets.

		RESMOD	XSDRN	KENO	MCNP
UOX	1 (293 K)	1.3894	1.3851	1.3856 (0.0004)	1.3865 (0.0004)
	2 (reduced H_2O density)	1.3374	1.3326	1.3327 (0.0004)	1.3336 (0.0004)
	3 (fuel at 900 K)	1.3057	1.2997	1.2995 (0.0004)	1.3030 (0.0004)
	4 (550 K)	1.3199	1.3145	1.3145 (0.0004)	1.3164 (0.0004)
MOX	1a (fuel 1, 300 K)	1.2189	1.2155	1.2160 (0.0004)	1.2200 (0.0005)
	1b (fuel 1, 560 K)	1.2048	1.2011	1.2010 (0.0004)	1.2054 (0.0005)
	2a (fuel 2, 300 K)	1.2631	1.2593	1.2593 (0.0004)	1.2621 (0.0005)
	2b (fuel 2, 560 K)	1.2498	1.2456	1.2459 (0.0004)	1.2484 (0.0005)

To test the temperature dependency of the generated data libraries (multigroup and MCNP) for the three benchmark cells, the fuel temperature was increased up to 2000 K for the cells with hot

fuel temperature. The reactivity change due to increased fuel temperature (reference temperature 600 K) is shown in Fig. 10. For the UOX cell, the average fuel temperature coefficient is -1.80 pcm/K for MCNP, -1.9 pcm/K for RESMOD and -2.0 pcm/K for XSDRN and KENO-VI. For the MOX cell with recycled Pu the average fuel temperature coefficient agree very well (-2.3 pcm/K), for the MOX cell with standard Pu the average fuel temperature coefficient varies from -2.0 pcm/K (MCNP) to -2.05 pcm/K (RESMOD) to -2.1 pcm/K (XSDRN/KENO-VI). The differences for the UOX cell will be further analysed, but the results show a very sufficient agreement. This means that the treatment of temperature dependency in both the multigroup and MCNP libraries is confirmed.

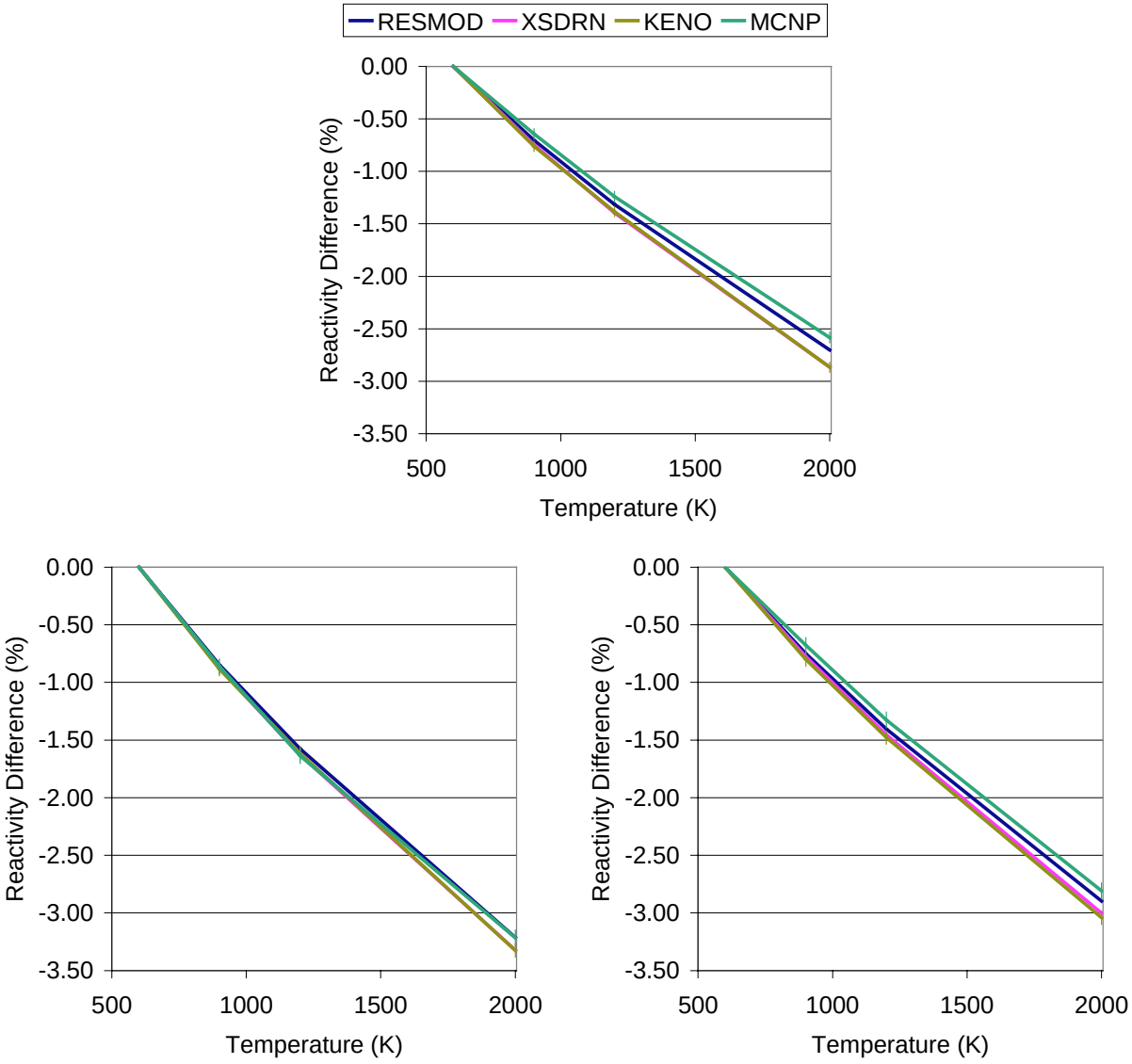

Figure 10. Reactivity change as a function of fuel temperature for pin cells defined in the ROWLANDS benchmark. Calculations were performed with JEF-2.2 data.
Top: UOX cell. Bottom left: MOX cell type 1. Bottom right: MOX cell type 2.

CONCLUSIONS

The calculation of the very different benchmarks based on the evaluations of JEF-2.2 (partly JEFF-3), ENDF/B-VI (incl. release 5), and JENDL-3.2 showed for most of the regarded configurations reasonably good results compared with experiments. The status of modern cross section evaluations such as ENDF/B-VI, JEF-2.2 (JEFF-3), and JENDL-3.2 is generally good, but there are some differences in important nuclides which lead to different results as discussed for the single benchmark series. A compilation of the mean k_{eff} results of the analysed benchmark series is shown in Fig. 11.

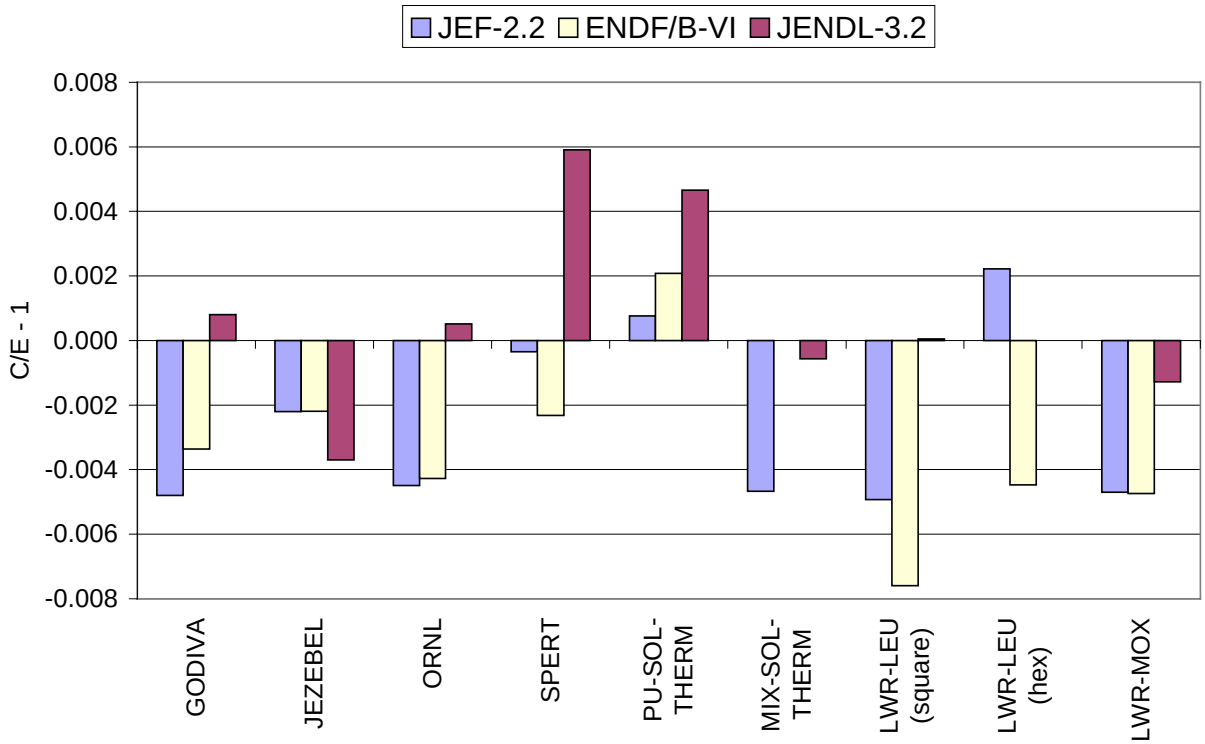

Figure 11. Relative differences between MCNP calculations and measurements for different data evaluations and benchmark series.

In this figure the relative difference between calculated results and experiments (as C/E – 1) are shown for calculations based on JEF-2.2, ENDF/B-VI, and JENDL-3.2. The calculations were performed with MCNP-4B, that means without explicit treatment of the unresolved resonance range. There may be a small bias from this approximation for the systems containing U-238, especially for the TRX benchmarks which have fuel of the highest uranium density and hence the strongest self shielding effects. In [16] changes of about 0.1 % in k_{eff} were reported for TRX-1 if self-shielding in unresolved resonance range is regarded. This small effect lies inside the general uncertainty and is not important for the MCNP calculations performed for this work. Discussing the mean results one can conclude that JEF-2.2 based results are slightly too low for UOX and MOX lattices, U-235 solutions and mixed U-Pu solutions. For Pu solutions alone, the JEF-2.2 and ENDF/B-VI.5 based results agree well with the corresponding experiments and show an important improvement against ENDF/B-V and JEF-1. JENDL-3.2 based results agree well for

U solutions and mixed U-Pu solutions as well as for many experiments with UOX and MOX fuel. They are slightly (up to 0.5 %) too high for the SPERT-D series (MTR type fuel) and the Pu nitrate solutions. ENDF/B-VI.5 based calculations show similar results as JEF-2.2 based calculations for the homogeneous systems, the SPERT-D series and MOX fuel. For LWR UOX fuel, however, there is a too strong underprediction of k_{eff} if ENDF/B-VI release 5 for U-235 is used. Since no clear dependence of the underprediction on enrichment can be observed (except for highly enriched systems), one could conclude that the problem lies in the U-235 cross section itself. Improvements which can be found using ENDF/B-VI.5 for U-235 (see for example [12, 13]) should not influence the important systems with LWR lattices. Comparisons with other methods which are accurate at least for regular lattices show that MCNP works correctly in the complete energy and temperature ranges and can be used as a tool for best estimate calculations of present but also future core and fuel designs. For the nuclear data, a convergence between the evaluated data at least for the important and reactivity relevant nuclides would be necessary to have both a best estimate method and a really problem independent nuclear data library for the solution of all kinds of reactor physical problems.

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