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OECD/NEA INTERNATIONAL BENCHMARK ON THE VENUS-2 MOX CORE MEASUREMENTS

Byung-Chan Na and Enrico Sartori
OECD Nuclear Energy Agency
Le Seine Saint-Germain, 12, Boulevard des Iles
92130 Issy-les-Moulineaux, France
na@nea.fr; sartori@nea.fr

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ABSTRACT

Within the framework of the NEA Nuclear Science Committee, theoretical physics benchmarks and multiple recycling issues related to various MOX-fueled systems have been studied. Many improvements and clarifications in nuclear data libraries and calculation methods have been achieved from the results of the theoretical benchmarks performed. However, it was felt that there was also a need to link these findings to data from experiments. Hence, a blind international benchmark exercise based on the two-dimensional VENUS-2 MOX core measurement data was carried out. Twelve participants from ten countries participated in the benchmark. Both the deterministic and the Monte Carlo methods were applied with different nuclear data sets. This paper provides a comparative analysis between calculated and measured results. Comparison with experimental results identified the origins of discrepancies between calculations and measurements and enabled the quantitative comparison of the relative merits of the different calculation methods.

1. INTRODUCTION

The management of excess plutonium produced during the operation of commercial power plants has become important to the nuclear community. Many possible treatments of this issue have been envisaged. The options include vitrification of the spent fuel and geological disposal with other nuclear waste without prior reprocessing, and burning of the plutonium fuels in nuclear power plants for the production of electricity. The preferred option depends on whether plutonium is seen as a waste or a high quality nuclear fuel.

Several countries use plutonium recovered from spent fuel in the form of mixed-oxide (MOX) uranium and plutonium fuel in existing power plants, a well established technology with many years of experience. Nevertheless, re-use of plutonium in PWRs is limited in quantity as well as in quality. Loading of plutonium assemblies in a PWR core is recommended up to 30% of the total loading to avoid any problem which might be caused by power peaking or degradation of safety parameters. The quality of plutonium decreases after a few recycles (Rome 1991). The actual practice for MOX fuel use is therefore not sufficient to stabilize the stockpile of plutonium extracted from spent fuel.

For managing larger quantities through multiple recycle and high burn-up of plutonium in PWRs, it is still necessary to validate both the basic nuclear data and calculational methods to better understand the behavior of MOX fuel in challenging situations and to identify possible improvements in nuclear data and physics modeling methods.

For this purpose, the NEA Working Party on the Physics of Plutonium Fuels and Innovative Fuel Cycles (WPPR) has in recent years commissioned theoretical physics benchmarks related to multiple recycling of plutonium in PWRs and BWRs, and burning/ multi-recyling of plutonium in fast reactors (OECD/NEA 1995a, 1995b, 1995c, 1995d, 1996, 1999a). More recently, a Task Force on Reactor-Based Plutonium Disposition (TFRPD) was set up to undertake a technical evaluation assessing the implications of moving from low enriched uranium to weapons-grade MOX in western and VVER reactors (OECD/NEA 1999b). Many improvements and clarifications in nuclear data libraries and calculation methods have been achieved from the results of these benchmarks. However, it was felt that there was a need to relate these findings to data from experiments. The two-dimensional VENUS-2 MOX core experimental data have been released for this purpose by SCK•CEN, Mol, Belgium. The VENUS-2 benchmark exercise was a blind test, and hence the measured fuel pin power values at specified VENUS-2 locations were not revealed to the participants. Ten institutions worldwide participated in the benchmark. The calculated pin power distributions using combinations of data and methods were compared with the experimental results. Various nuclear data sets such as ENDF/B-V, ENDF/B-VI, JEF-1, JEF-2.2 and JENDL-3.2 were investigated. For power distribution calculations, the deterministic codes used were two versions of the collision probability code HELIOS, the collision probability code BOXER and 2-D S_N code DORT. The continuous energy Monte Carlo codes used were MCNP-4B, MVP and MCU-B. A diffusion nodal code GNOMER was also applied. This paper provides a summary of the comparative analysis between calculated and measured results. The detailed analysis and results can be found in the reference (Na 2000a).

2. BENCHMARK MODEL

The VENUS facility is a zero power critical PWR mock-up located at SCK•CEN in Belgium. As shown in Figure 1, the core comprises 12 "15×15" subassemblies, instead of the "17×17" type (the pin-to-pin pitch remains typical of the "17×17" subassembly). The central part of the core (four 15×15 assemblies) consists of fuel pins 3.3 wt.% enriched in ^{235}U. There are five Pyrex pins in 1/8 of the core. Of the eight assemblies on the periphery of the core, all of which contain fuel pins 4.0 wt.% enriched in ^{235}U, eight rows of the most external fuel pins have been replaced by mixed-oxide fuel pins (UO_2-PuO_2) enriched 2.0 wt.% in ^{235}U and 2.7 wt.% in high grade plutonium (SCK•CEN).

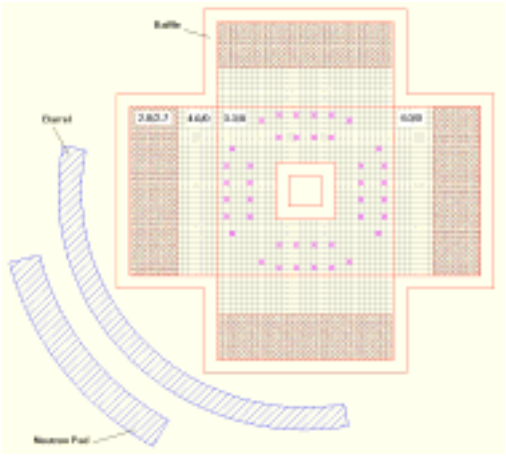

Fig. 1 VENUS-2 core geometry.

The average fission rate in the core corresponding to absolute reference irradiation is 1.87E+08 fissions/cm/sec at the mid-plane. This average fission rate corresponds to a power of 595 watts. One hundred twenty-eight (128) fuel rods in the core were measured after an irradiation of 13.5 h at 90% of the VENUS maximum power. The 1/8 of the core comprises 325 fuel rods in which the pin powers of 121 fuel rods were directly measured and the pin powers of 204 fuel rods were interpolated from the measured values. The additional seven fuel pins measured were located in symmetric positions out of 1/8 of the core. The measured and interpolated positions of the fuel pins are shown in Figure 2. The pin power values were taken from the measured gamma activity of the ^{140}La and normalized to a core averaged fission rate = 1 fission/sec/fuel cell.

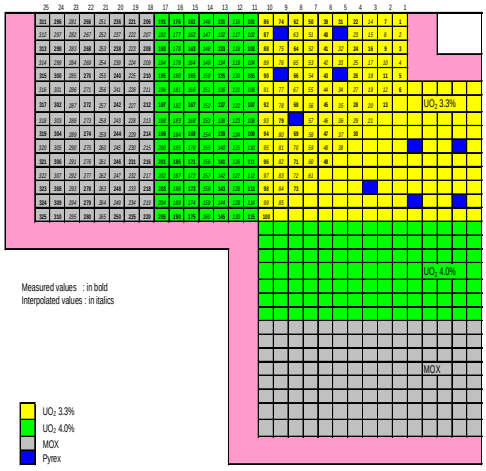

Fig. 2 Measured and interpolated pin power positions in VENUS-2.

Along with all geometry and material data required to develop the detailed computational model of the 1/4 fraction of the VENUS-2 reactor core, the isotopic concentrations of each medium were provided to the participants to minimize the discrepancies from the atomic density calculations. The experimental results of axial buckling measurement were also provided for 2-D calculations (Na 1999).

From each fuel cell calculation (UO_2 3.3%, UO_2 4.0%, MOX), k_∞, absorption and fission reaction rates per isotope (energy integrated and in three groups involving the 5 keV and 4 eV boundaries) were requested. From core calculations, it was requested to report k_{eff} and normalized pin power (i.e. fission rate) distribution on 1/8 of the core which consists of 325 fuel pins (see Figure 2). Normalization was to be made to a core average fission rate = 1 fission/sec/fuel cell.

3. RESULTS AND DISCUSSIONS

3.1 Cell calculations

Three types of pin cells have been studied: two cells fueled with UO_2 3.3 wt.% and 4.0 wt.% enriched in ^{235}U and one cell fueled with MOX 2.0 wt.% enriched in ^{235}U and 2.7 wt.% enriched in high quality plutonium. In cell calculations, k_∞ and reaction rates (fission and absorption) in one group and in three groups were investigated.

Infinite multiplication factor

The average k_∞ values submitted by the participants are 1.40593 ± 0.00393 for UO_2 3.3%, 1.33726 ± 0.00553 for UO_2 4.0% and 1.25673 ± 0.00607 for MOX cell. The results are summarized in Table 1. Although the maximum spreads in k_∞ from the average values are 0.9% for UO_2 3.3%, 1.3% for UO_2 4.0% and 1.3% for MOX cells, most of the results report a deviation of less than 0.5%. In particular the Monte Carlo calculations give results with a deviation of less than 0.2%, which is the claimed uncertainty on reactivity in current nuclear design methods. The spreads among the results are less important in Monte Carlo calculations than in deterministic calculations.

The difference between the NEA and SCK•CEN cell calculations comes mainly from the differences in the libraries provided by SCALE-4.4, keeping in mind that their calculational methodologies are the same ($S_{16}P_3$ and cylindrical cell model). SCK•CEN used the 238 group library (238 *GROUPNDF5*) whereas the NEA used the 44 group library (44 *GROUPNDF5*). The latter gives higher k_∞ values than the former: 470 pcm for UO_2 3.3%, 540 pcm for UO_2 4.0% and 450 pcm for MOX cell. Although the 44 group results are closer to the average values than those from the 238 group results, lower energy resonances may not have been properly treated in the 44 group library.

The basic library used by KAERI and ORNL is the same, i.e. ENDF/B-VI. However, KAERI used a 35 neutron group data library condensed by using HEBE code from the master library which is a 190 neutron group library generated by using NJOY91.13 based on ENDF/B-VI. ORNL, however, used nuclear data libraries prepared by Studsvik Scanpower Inc. for 34, 89 and 190 groups based on ENDF/B-VI. These libraries take into account resonance capture effects in ^{238}U. KAERI and ORNL results are quite similar.

Table 1 k_∞ values of cell calculations.

Institution	Method	Basic library	UO_2 3.3%	Deviation* (%)	UO_2 4.0%	Deviation* (%)	MOX	Deviation* (%)
NEA	SCALE-4.4 (44g)	ENDF/B-V	1.40385	-0.15	1.33366	-0.27	1.25345	-0.26
KAERI	HELIOS-1.5 (35g)	ENDF/B-VI	1.40904	0.22	1.34306	0.43	1.26339	0.53
ORNL	HELIOS-1.4 (190g)	ENDF/B-VI	1.40847	0.18	1.34333	0.45	1.26254	0.46
PSI	BOXER	JEF-1	1.40548	-0.03	1.33797	0.05	1.26399 **H**	**0.58**
SCK•CEN	SCALE-4.4 (238g)	ENDF/B-V	1.39917	-0.48	1.32829	-0.67	1.24894	-0.62
IJS	WIMS-D (69g)	JEF-2.2	1.40358	-0.17	1.33298	-0.32	1.24858	-0.65
	WIMS-D (69g)	ENDF/B-VI	1.39840 **L**	**-0.54**	1.32760 **L**	**-0.72**	1.24730 **L**	**-0.75**
NEA+KAERI	MCNP-4B	ENDF/B-VI	1.40479 (±0.00053)	-0.08	1.33635 (±0.00056)	-0.07	1.25447 (±0.00061)	-0.18
JAERI	MVP	JENDL-3.2	1.41115 (±0.00044)	0.37	1.34549 **H** (±0.00043)	**0.62**	1.26313 (±0.00051)	0.51
KI	MCU-B	ENDF/B-VI JENDL-3.2 BROND	1.40710	0.08	1.33650	-0.06	1.25490	-0.15
KFKI	MCNP-4B	ENDF/B-VI	1.40650 (±0.00086)	0.04	1.33640 (±0.00088)	-0.06	1.26310 (±0.001)	0.51
GRS	MCNP-4B (square cell)	JEF-2.2	1.41120 **H** (±0.0004)	**0.37**	1.34370 (±0.0004)	0.48	1.26140 (±0.0004)	0.37
	MCNP-4B (cylindrical cell)	JEF-2.2	1.40950 (±0.0004)	0.25	1.34020 (±0.0004)	0.22	1.25430 (±0.0004)	-0.19
IJS	MCNP-4B	ENDF/B-VI	1.40480 (±0.0002)	-0.08	1.33610 (±0.0002)	-0.09	1.25470 (±0.0002)	-0.16
Average			1.40593		1.33726		1.25673	

* Deviations from the average k_∞ values.

ORNL investigated the impact of different energy group structures and leakage treatment on k. PSI also calculated k_∞ with different leakage treatments for 70 energy groups. In UO_2 3.3% and UO_2 4.0%, the 89 and 34 group results are similar and closer to the average value (0.11% and 0.35% of deviations from the averages), whereas the 190 group results are somewhat higher (0.23% and 0.48% of deviations from the averages). For the MOX cell, the 190 and 89 group results are similar and closer to the average value (0.48% of deviation from the average), and the 34 group result is somewhat higher (0.67% of deviation from the average). The differences between k_∞ (zero-leakage) and k_∞ calculated from k_{eff}(cell) = 1 are 912, 571 and 220 pcm for UO_2 3.3%, UO_2 4.0% and MOX fuel cells, respectively. IJS WIMS-D results show the impact of the differences in basic libraries. The JEF-2.2 library produces higher k_∞ values than ENDF/B-VI: 518 pcm for UO_2 3.3%, 540 pcm for UO_2 4.0% and 128 pcm for MOX cell. The JEF-2.2 results are closer to the average values.

The effect of square and cylindrical cell models is well known. The differences between two cell models found by GRS for the VENUS-2 cells are as follows: 170 pcm for UO_2 3.3%, 350 pcm for UO_2 4.0% and 710 pcm for the MOX cell. This is consistent with the results of previous studies (Rowlands 1999, DeHart 1999). Even though the cylindrical model provides equivalent moderator volumes in a cell, the effective moderation is higher in the square cell than in the cylindrical cell. This results in more thermal flux in the square cell than in the cylindrical cell. Uranium is not as sensitive to this small spectral change, therefore the effect of the cylindrical cell is rather small. However, for MOX cells, the largest effect observed in the previous studies by going from the cylindrical cell to the square cell is a decrease of resonance absorption (mainly by ^{238}U). This leads to an increase in the thermal flux promoting fissions primarily by ^{239}Pu at that energy region. Therefore the square cell gives a larger increase in multiplication factor.

In general, the Monte Carlo results, which describe the “true” square cell geometry, have larger k_∞ values whereas the deterministic calculations (cylindrical cell geometry) give smaller values (except KAERI results) compared with the average values. The differences among Monte Carlo results are very small, but it is worth noting that the JENDL-3.2 library always produces slightly higher k_∞ values compared with the MCNP results using the other libraries.

Absorption and fission reaction rates

For simplicity of comparison, the reported reaction rates were normalized to the total fission equal to 1.0, although the usual procedure is to normalize the total absorption equal to 1.0. The average values of all results were calculated and deviations from the average values were used for comparison.

For ^{235}U in the MOX cell, the results are consistent. A good agreement among results provided is observed for both total absorption and total fission rates (less than 1% and less than 0.5% of deviations from the average values for absorption and fission rates, respectively). However the deviations are slightly higher in KFKI MOX cell calculations (+2.5% and +3.5% in total absorption and total fission rates, respectively). Between the

different groups, the compensation effect can also be seen, for example, in KAERI MOX cell calculations, a 5% underestimation in Group 1 is compensated by a 5% overestimation in Group 2. Knowing that KAERI and ORNL used the same method and the same basic library, the difference between them could come from the difference in nuclear data reduction methods. In the UO_2 3.3% cell, the agreement is even better. Most results show deviations of less than 0.2% and about 0.5% for the total absorption and fission rates, respectively. PSI reports high deviations (about 14%) in Groups 1 and 2 for both reaction rates. In the UO_2 4.0% cell, the agreement is excellent. Most results show the deviations of less than 0.3% and about 0.2% for the total absorption and fission rates, respectively.

For ^{238}U in the MOX cell, most of the calculations report less than 3% and 2% of deviations from the average values for total absorption and total fission rates, respectively. WIMS-D gives about 8% higher absorption in Group 2 and 4% lower fission rates. The differences in fission rates in Groups 2 and 3 are high (about 15%), but their absolute contributions to the total values are small. In the UO_2 3.3% cell, except for the BOXER results (deviations of 10% in absorption and 16% in fission), the results are consistent as in the MOX cell. In the UO_2 4.0% cell, most of the results show a deviation of less than 1% in total absorption, but it becomes worse in total fission rates (about 3%). WIMS-D calculations overestimate both reaction rates by about 4%. BOXER gives 10% higher total fission rates due to the overestimation of fission in Group 1.

For ^{239}Pu, most of the participants reported less than 0.5% deviations for total absorption and total fission rates. WIMS-D overestimates reaction rates in Groups 1 and 2 (up to 6% in absorption and 5% in fission), but the total values are consistent. KFKI slightly underestimates reaction rates in all three groups, resulting in total absorption and fission rates being underestimated by 3% and 1.5%, respectively.

The trends observed in the cases of ^{240}Pu and ^{241}Pu are very similar, but the differences observed among the results are less pronounced in ^{241}Pu than in ^{240}Pu. The relatively higher differences observed in SCALE-4.4 calculations are probably due to the basic library used (ENDF/B-V). For ^{240}Pu, the NEA reports a 3.5% overestimation of the total absorption and a 1.7% underestimation of the total fission rates, and SCK•CEN gives a 2.5% overestimation of the total absorption and a 1.5% underestimation of the total fission rates. High overestimations of reaction rates (more than 5% for both ^{240}Pu and ^{241}Pu) are observed in WIMS-D calculations. This is more pronounced with the JEF-2.2 library. In the case of ^{240}Pu absorption rates, PSI, KI and KFKI calculations show a high underestimation caused mainly by an underestimation in Group 3. For ^{241}Pu fission rates, KAERI reports a 5% overestimation, whereas JAERI reports a 3% underestimation of the total fission rates. KI and KFKI also report an underestimation of the total fission rates.

For ^{242}Pu, the deviations among the results provided become higher. Many results show absorption rates with up to 10% of deviations and fission rates with up to 6% of deviations. However, since the absolute contributions of ^{242}Pu to the total neutronic balance are very small compared with other isotopes, the large deviations observed for ^{242}Pu would not play an important role. A very big spread (up to a factor of 2.5) is observed in the fission rates in Group 3 even though the neutronic contribution of fission

in Group 3 is small. This is seen in all libraries and methods. JAERI also reports a 23% underestimation of fission rates in Group 2. Two SCALE-4.4 calculations show the same trend (7.5% overestimation of total absorption and 1.5% underestimation of total fission).

For ^{241}Am, about 5% of deviations are observed in total absorption and fission rates, but their absolute contributions to the whole neutronic balance are small. Total absorption rates reported by KI show an 8% overestimation and WIMS-D calculations with JEF-2.2 report 5% and about 8% overestimations of total absorption and fission rates, respectively.

In general, all the participants reported consistent reaction rates values for one group and for three groups, although they used different libraries and cell calculation codes. These results are also consistent with the discrepancies observed in previous benchmarks such as the benchmark on power distribution within UO_2 and MOX assemblies (Na 2000b, Cathalau 2000).

3.2 Core calculations

For the core calculations, k_{eff} and pin power (i.e. fission rates) distribution of the 1/8 core were investigated. Concerning the experimental data, the powers of 121 fuel rods in the 1/8 core were directly measured and the powers of the remaining fuel rods were interpolated from the measured values. In total, the power distribution of 325 fuel rods was reported. The reported uncertainty of the measured data (1σ) is ±1.0% in UO_2 and ±1.5% in MOX pins and that of the interpolated data (1σ) is ±2%. The calculated power distributions were compared with the 121 measured data as C/E values.

Effective multiplication factor

The calculated k_{eff} values are presented in Table 2. All reported k_{eff} show very good agreement with the experimental value which is k_{eff} = 1. The average k_{eff} is 0.99758 ± 0.0045. The deterministic transport calculations give an average of 0.99750 ± 0.0044 and the Monte Carlo calculations lead to an average of 0.99983 ± 0.0037. The maximum discrepancy in k_{eff} (-1.0%) is reported by the diffusion calculation with ENDF/B-VI. Except for the PSI result, all deterministic calculation results show a slight underestimation of k_{eff} (but less than 0.5%). The differences among Monte Carlo calculations seem to depend on the different approximations on the geometry models used.

In the three calculations of ORNL with different energy group structures, there is not much difference in k_{eff} between calculations with the 190 and 89 group nuclear data libraries, but with the 34 group library, there is a sizable increase in the calculated k_{eff}. The same trend can be observed in the two DORT calculation results from the NEA and SCK•CEN. For their core calculations, the NEA directly used 44 group cross-sections generated from cell calculations whereas SCK•CEN performed an additional 1-D core calculation using the module XSDRNPM in SCALE-4.4 to collapse 238 group cross-sections obtained from cell calculations into 44 group cross-sections. The latter procedure allows to take into account region-dependent fluxes in the core in the cross-section collapsing.

Table 2 k_{eff} values of core calculations.

Institution	Code	k_{eff}	Deviation (%)	Comments
NEA	DORT	0.99452	-0.55	44 groups
KAERI	HELIOS-1.5	0.99817	-0.18	35 groups
ORNL	HELIOS-1.4	0.99870	-0.13	190 groups
		0.99907	-0.09	89 groups
		1.00150	0.15	34 groups
PSI	BOXER	1.00378	0.38	21 groups
SCK•CEN	DORT	0.99233	-0.77	44 groups collapsed from 238 groups taking into account region-dependent fluxes
IJS	GNOMER (JEF-2.2)	0.99450	-0.55	Diffusion nodal method (4 groups)
	GNOMER (ENDF/B-VI)	0.98977 **L**	-1.02	Diffusion nodal method (4 groups)
NEA+KAERI	MCNP-4B	1.00213 (±0.00013)	0.21	Histories : 50×10^6
JAERI	MVP	No k_{eff} calc.	None	Histories: 20×10^6
KI	MCU-B	0.99650 (±0.00025)	-0.35	Histories: 10×10^6
KFKI	MCNP-4B	1.00050 (±0.00026)	0.05	Histories: 50×10^6
GRS	MCNP-4B	1.00430 **H** (±0.0002)	0.43	Histories: 15×10^6
IJS	MCNP-4B	0.99570 (±0.0001)	-0.43	Histories: 50×10^6

IJS GNOMER diffusion calculation results give the difference from the basic libraries. The k_{eff} value with JEF-2.2 (0.99450) is closer to the experimental value than that with ENDF/B-VI (0.98977).

Pin power distribution

For relative comparison of these results, C/E values of pin powers for 121 measured positions are plotted in Figures 3-5 for UO_2 3.3%, UO_2 4.0% and MOX, respectively. In the analysis below, the uncertainties the experimental data (1σ) of ±1.0% in UO_2 and ±1.5% in MOX pins should be taken into account.

For two types of UO_2 fuel rods, the calculated pin power results from both deterministic transport and Monte Carlo methods show an excellent agreement with the experimental values. An average deviation from the measured values is less than 2.5% for UO_2 3.3% and less than 1.0% for UO_2 4.0% in most of the calculated results reported. Nevertheless for most calculated results, a trend of slight underestimation is dominant

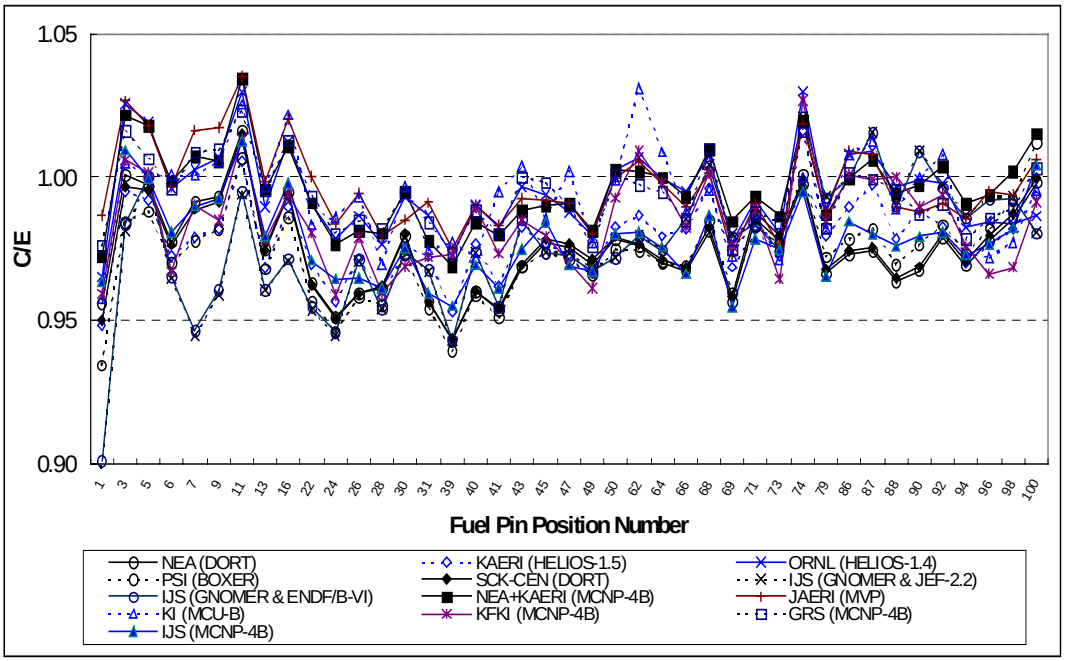

Fig. 3 UO_2 3.3% fuel pin power distribution.

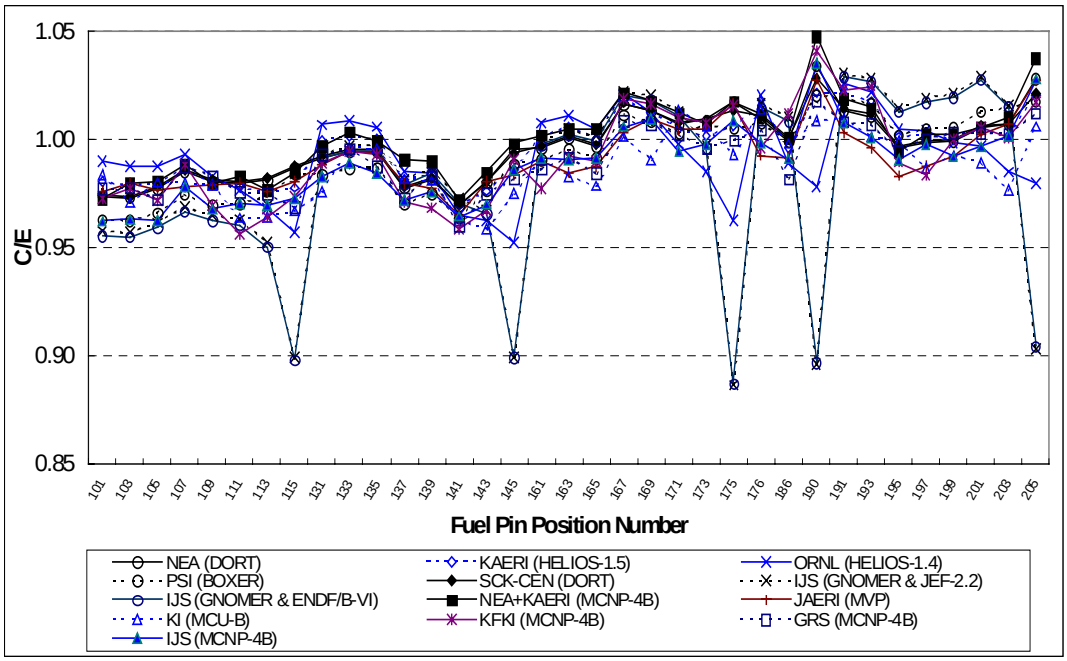

Fig. 4 UO_2 4.0% fuel pin power distribution.

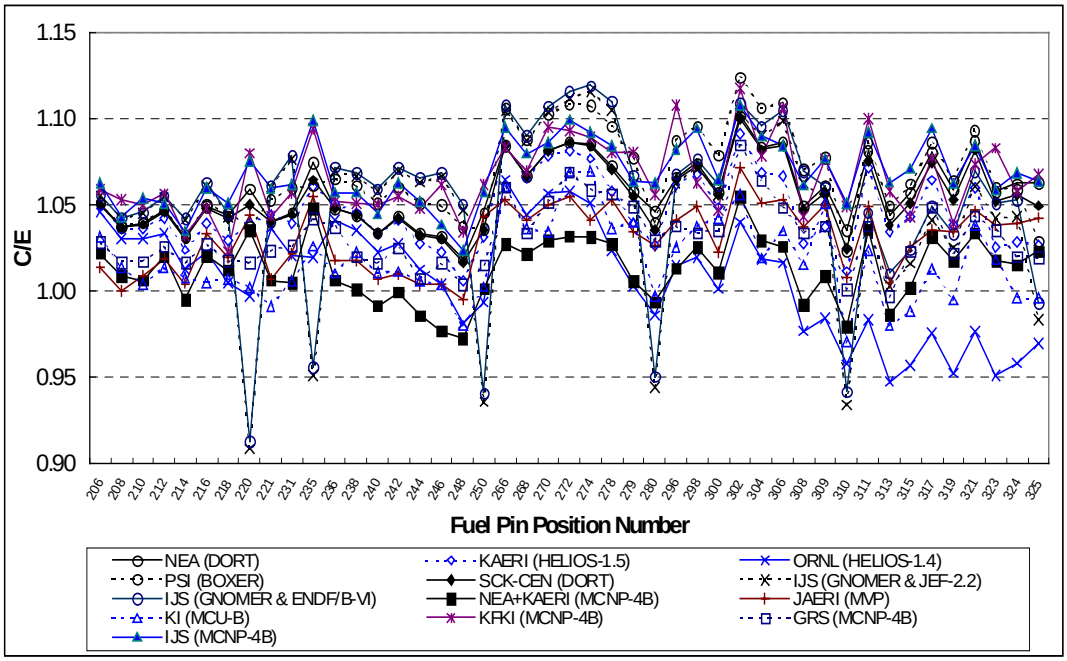

Fig. 5 MOX fuel pin power distribution.

except for a few positions. For UO_2 4.0%, although IJS results from the diffusion nodal code GNOMER show a good agreement for most of the fuel pin positions, a wider scatter band (10%) can be observed for fuel pins located near the outer baffle. This is due to the well-known limits of the diffusion approximation. This problem associated with diffusion methods could be corrected by adjusting the reflector and surrounding water thickness to conserve the volume ratios.

For the MOX fuel pins, even though some MCNP results show a good agreement with the experimental data, the average deviation (especially for outer MOX fuel pins) is larger than for UO_2 fuel pin cases in most of the calculations. The same trend can be observed in deterministic calculations. As in UO_2 4.0% pins, IJS GNOMER results give a larger underestimation of pin power near the outer baffle due to diffusion approximations. The general trend in MOX fuel pin power calculations shows an average deviation of more than 4%, mainly due to the five outermost MOX rows. A general overestimation of MOX pin power was also seen in the results of the previous MOX benchmarks (Na 2000b, Cathalau 2000).

Despite limits of the diffusion approximation, the IJS diffusion results using two different libraries (ENDF/B-VI and JEF-2.2) can be used to examine the differences between these two libraries. The two libraries give almost the same results for UO_2 fuel pins. However, for MOX fuel pins, the results from JEF-2.2 are slightly better than those from ENDF/B-VI.

All the results underestimate the pin power by 1.6% for UO_2 3.3% cells and by 0.9% for UO_2 4.0% cells and an overestimation of 4.3% is observed for the MOX cells. However, some calculations show an excellent agreement with the measured values: HELIOS results (ORNL) report an underestimation of 0.51% for UO_2 3.3% cells, 0.82%

for UO_2 4.0% cells and an overestimation of 0.77% for the MOX cells. Concerning the Monte Carlo calculations, NEA+KAERI results give average deviations of -0.41%, 0.02% and 1.28% from the measured values for UO_2 3.3%, UO_2 4.0% and the MOX cells, respectively. In general, the results from Monte Carlo calculations give a better agreement. This is probably due to the square cell description instead of the cylindrical modeling of the cell, especially for the MOX fuel pins.

The deterministic results together with the diffusion results give an average underestimation of 2.1% for UO_2 3.3% and of 1.0% for UO_2 4.0% and an average overestimation of 4.7% for MOX pins. If the corrected diffusion results are considered, a slight improvement of 0.1% in UO_2 3.3% and 0.3% in UO_2 4.0% can be seen, but for MOX pins the results are almost the same (0.03% difference). The deterministic transport results alone show an average underestimation of 2.1% for UO_2 3.3% and of 0.5% for UO_2 4.0% and an average overestimation of 4.7% for MOX pins. The Monte Carlo results give an average underestimation of 1.0% for UO_2 3.3%, 0.9% for UO_2 4.0% and an average overestimation of 3.8% for MOX pins.

In order to investigate the pin power distribution at the MOX/UO_2 interface, the calculated pin power ratios at the interface of MOX and UO_2 4.0% pins, together with the experimental values, are presented in Figure 6. An overestimation of 2.7% is observed in all the calculated values. On average, the overestimation is higher in the deterministic calculations (3.0% against 2.5% in the Monte Carlo calculations) although the shapes from the deterministic calculations follow the trend of the experimental values. In Monte Carlo calculations, the average deviation from the experimental values is less than in deterministic calculations. However, the scatter band is larger, probably due to the statistical perturbation in Monte Carlo calculations; KFKI and IJS Monte Carlo calculations especially show about 5% of overestimation. Although the diffusion calculations show a defect near the outer baffle, the ratios of pin power are quite comparable to the experimental values.

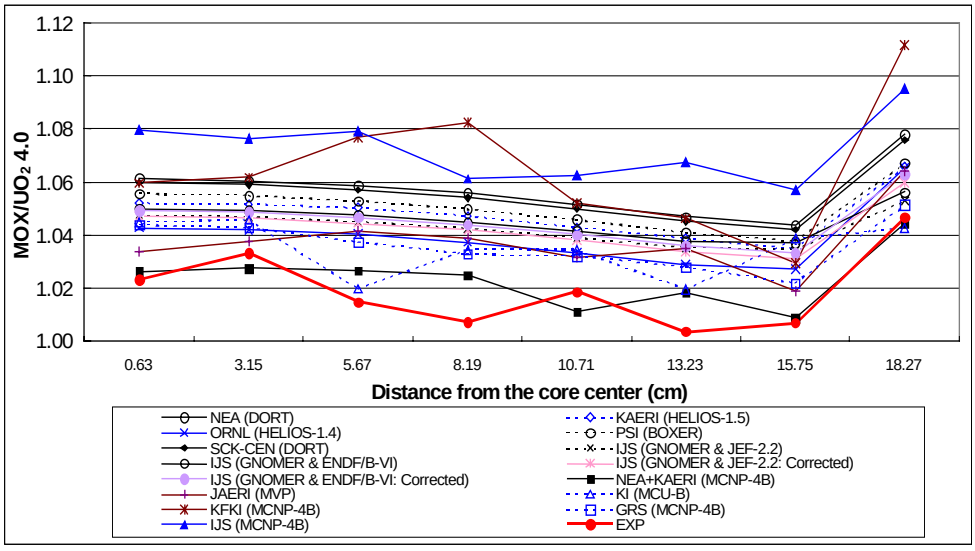

Fig. 6 Pin power ratios at MOX/UO_2 interface

Finally, the reported statistical errors (1σ) associated with Monte Carlo calculation results are satisfactorily small for most of the calculated positions. The average statistical errors are 0.4% in the NEA+KAERI and JAERI calculations (50 and 20 million histories, respectively) and 0.45% in the GRS calculation (15 million histories). KI used 10 million histories giving 0.8 to 1.2% of uncertainties in the fuel pin power calculation.

4. CONCLUSIONS

A blind international benchmark exercise for the prediction of power distribution in the VENUS-2 MOX core experiment was undertaken. Contrary to the previous benchmarks, which have been entirely concerned with theoretical calculations, this benchmark was based on experimental results. Comparison with experimental results allowed to identify the origins of discrepancies between calculations and measurements and to compare quantitatively the relative merits of the different calculation methods. Ten institutions participated in the benchmark, providing more than 14 solutions. Both the deterministic and the Monte Carlo methods were applied with different nuclear data sets. The deterministic codes used were the S_N code DORT together with SCALE/XSDRNPM, the collision probability codes HELIOS and BOXER, and a nodal diffusion code GNOMER in conjunction with the cell calculation code WIMS-D. Various versions of the continuous Monte Carlo codes such as MCNP-4B, MVP and MCU-B were also used.

From the cell calculations, k_∞ and reaction rates were investigated. Although the maximum spreads in k_∞ from the average values are 0.9% for UO_2 3.3%, 1.3% for UO_2 4.0% and 1.3% for MOX cells, most of the results report a deviation of less than 0.5%. In particular the Monte Carlo calculations give results with a deviation of less than 0.2%, which is the claimed uncertainty on reactivity in current nuclear design methods.

Concerning core calculations, k_{eff} and pin power distribution were reported and compared with the experimental values. All reported k_{eff} show a very good agreement with the experimental value. The average k_{eff} is 0.99758 ± 0.0045. The deterministic transport calculations give an average of 0.99750 ± 0.0044 and the Monte Carlo calculations lead to an average of 0.99983 ± 0.0037. The maximum discrepancy in k_{eff} (-1.0%) was reported by the diffusion calculation with ENDF/B-VI.

For two types of UO_2 fuel rods, the calculated pin power results from both deterministic transport and Monte Carlo methods show an excellent agreement with the experimental values. An average deviation from the measured values is less than 2.5% for UO_2 3.3% and less than 1.0% for UO_2 4.0% in most of the calculated results reported. However, for most calculated results, a trend of slight underestimation is dominant except for a few positions. For the MOX fuel pins, even though some MCNP results show a good agreement with the experimental data, the average deviation is larger than for UO_2 fuel pin cases in most of the calculations. The general trend in MOX fuel pin power calculations shows an average deviation of more than 4%, mainly due to the five outermost MOX rows. A general overestimation of MOX pin power was also seen in the results of the previous MOX benchmarks.

The deterministic transport results show an average underestimation of 2.1% for UO_2 3.3%, 0.5% for UO_2 4.0% and an average overestimation of 4.7% for MOX pins. The Monte Carlo results give an average underestimation of 1.0% for UO_2 3.3%, 0.9% for UO_2 4.0% and an average overestimation of 3.8% for MOX pins. The diffusion calculations show quite good results except for pins near the outer baffle. For these regions, a correction method should be applied.

In general, the calculations overestimate the MOX pin powers and slightly underestimate the UO_2 pin powers. This is reflected in all combinations of codes and methods and confirms observations seen in power reactors. Core design calculations for commercial cores with MOX fuel should take this effect into account.

In conclusion, the results are very encouraging and confirm that present methods using the latest nuclear data sets can adequately calculate MOX-fueled systems.

As a follow-up to this two-dimensional benchmark, the launch of a benchmark using three-dimensional VENUS-2 experimental results is in preparation (Van der Meer 2000, Na 2001). This will allow a more thorough investigation into the calculation methods used for MOX-fueled systems.

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