Uranium-fuel Thermal Reactor Benchmark Testing of CENDL-3

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CENDL-3, the new version of China Evaluated Nuclear Data Library are being processed, and distributed for thermal reactor benchmark analysis recently. The processing was carried out using the NJOY[1]nuclear data processing system. The calculations and analyses of uranium-fuel thermal assemblies TRX-1,2, BAPL-1,2,3, ZEEP-1,2,3 were done with lattice code WIMSD5A. The results were compared with the experimental results, the results of the '1986' WIMS library and the results based on ENDF/B-VI.

I. INTRODUCTION

CENDL-3, the new version of China Evaluated Nuclear Data Library was started in 1996, it's now in progress. According to the plan, CENDL-3 will be completed by the year 2000, and will contain about 200 nuclides. Since then, the work has been done at CNDC (China Nuclear Data Center). So far, the present of CENDL-3 is shown in Table 1.

Table 1. The completed and planned nuclides of CENDL-3

Nuclides	Planned	Completed
Fissile nuclide	15	5(^{233,234,235,238}U, ^{239}Pu)
Structure material	24	15(^{90,91,92,96,0}Zr, ^{58,59,60,61,62,0}Ni, ^{63,65,0}Cu)
Fission products	91	73
Light nuclide	3	2(^{6}Li, ^{9}Be)
Total	133	95

In order to demonstrate the reliability of CENDL-3, it's necessary to subject the nuclear data library to benchmark testing against relevant integral experiments. The WIMS-D/4 lattices code [2] is widely used for thermal reactor and power reactor calculations, so it also was used to do the benchmark testing of CENDL-3.

This report gives the testing results of uranium-fuel thermal assemblies based on CENDL-3 and comparisons among CENDL-3, ENDF/B-VI and the '1986' WIMS library. The processing was carried out using NJOY nuclear data processing system and WIMSD5A code. The materials processed were hydrogen bound in water, oxygen, aluminum, uranium 235 and uranium 238.These materials are crucial components of light water reactor (LWR) and heavy water reactor (HWR).

II. Benchmark Calculations

1. Evaluated nuclear data processing

The CENDL-3 and ENDF/B-VI evaluated nuclear data libraries were selected as the source of basic data. The NJOY-97.0 modular system was applied for nuclear data processing.

The processing options consistently integrate the references[3-9]. A value of 0.2% was used for the resonance reconstruction tolerance, and a 0.2% maximum tolerance criterion for thinning was also applied.

A weighting spectrum for cross section averaging was derived for a PWR fuel cell according to the references [6,8,9]. At thermal energies the shape of the spectrum depends very strongly on the lattice geometry, composition and the operating parameters. In the resonance range between 4.0 and 9118 ev, a 1/E shape was applied to be consistent with the method of WIMS. Above the resonance range the EPRI-CELL LWR spectrum was adopted again.

Energy independent Goldstein-Cohen values were applied. They are shown in Table 2.

Table 2 List of applied Goldstein-Cohen λ values.

Materials	λ values
H	**1.0**
D	**1.0**
O	**1.0**
Al	**1.0**
U-235	**0.2**
U-238	**0.2**

Table 3. Brief Characteristics of TRX-1,2 and BAPL-1,2,3

Lattice	Fuel	Cladding	Moderator	Rod Radius(cm)	Pitch(cm)
TRX-1	1.3 wt. % U-metal	Al	H_2O	0.4915	1.8060
TRX-2	1.3 wt. % U-metal	Al	H_2O	0.4915	2.1740
BAPL-1	1.3 wt. % UO_2	Al	H_2O	0.4864	1.5578
BAPL-2	1.3 wt. % UO_2	Al	H_2O	0.4864	1.6528
BAPL-3	1.3 wt. % UO_2	Al	H_2O	0.4864	1.8057

2. Method of cell calculation

The cell calculations were used with WIMSD5A code. At first, according to real cell composition, intermediate approximation was used to calculate resonance self-shielding. The main transport equation was solved using Sn method, and the cylindrical cell approximation was used to simplify the geometry of the cell. Leakage calculations have been done with input buckling values and B1 method. The reaction rates of ^{235}U and ^{238}U were given in output files in two groups.

3. Benchmark calculations and analysis

All the integral parameters are defined as follows:

K_{eff} finite medium effective multiplication factor, -

ρ^{28} ratio of epithermal to thermal ^{238}U capture reaction rate,

δ^{25} ratio of epithermal to thermal ^{235}U fission reaction rate,

δ^{28} ratio of ^{238}U fission to ^{235}U fission reaction rate,

$\mathbf{C}^*$ ratio of ^{238}U capture to ^{235}U fission reaction rate.

RCR ratio of $\mathbf{C}^*_{lattice}$ to $\mathbf{C}^*_{Maxwellian}$

3.1 The calculations for TRX-1,2, BAPL-1,2,3

TRX-1,2 used Uranium metal fuel in ^{235}U enriched to 1.305 wt. %; BAPL-1,2,3 used Uranium oxide fuel enriched 1.311 wt. %; TRX and BAPL were water(H_2O)-moderated, Details of these lattices are given in Table 3. The comparisons are shown in Table4.

Table 4. Integral parameter comparison

Lattices	Integral parameter	Experiment	WIMS/D	CENDL-3	ENDF/B-VI
TRX-1	K-eff	1.0000(~.30)	1.0023	0.9975	0.98853
	ρ^{28}	1.32(~1.6)	1.279	1.3608	1.377
	δ^{25}	0.0987(~1.0)	0.099	0.09803	0.0977
	δ^{28}	0.0946(~4.3)	0.0965	0.09622	0.0974
	C^*	0.797(~1.0)	0.780	0.7922	0.808
TRX-2	K-eff	1.0000(~.10)	0.9965	0.9998	0.99113
	ρ^{28}	0.837(~1.9)	0.808	0.8530	0.863
	δ^{25}	0.0614(~1.3)	0.061	0.06201	0.0600
	δ^{28}	0.0693(~5.1)	0.0695	0.06811	0.0690
	C^*	0.647(~.93)	0.636	0.6387	0.650
BAPL-1	K-eff	1.0000(~.10)	1.0029	1.0016	0.99431
	ρ^{28}	1.390(~.72)	1.358	1.3923	1.429
	δ^{25}	0.084(~2.4)	0.084	0.08199	0.0824
	δ^{28}	0.078(~5.1)	0.0755	0.07362	0.0751
	C^*	0.0000	0.800	0.7972	0.819
BAPL-2	K-eff	1.0000(~.10)	1.0005	1.0003	0.99459
	ρ^{28}	1.120(~.89)	1.133	1.1602	1.188
	δ^{25}	0.068(~1.5)	0.0687	0.06695	0.0672
	δ^{28}	0.070(~5.7)	0.0652	0.06327	0.0645
	C^*	0.0000	0.732	0.7274	0.746
BAPL-3	K-eff	1.0000(~1.0)	0.9981	1.0007	0.99565
	ρ^{28}	0.906(~1.1)	0.894	0.9130	0.933
	δ^{25}	0.052(~1.9)	0.0529	0.0515	0.0516
	δ^{28}	0.057(~5.3)	0.0538	0.05184	0.0528
	C^*	0.0000	0.657	0.6511	0.666

The results show that the calculated results with CENDl-3are much better than those results with old WIMS library.

The K_{eff} values from CENDL-3 are in good agreement with experiments. Only the value of TRX-1 is lower than 0.1%. The values of ρ^{28} for TRX-2, BAPL-1 and 3 are well predicted within the uncertainty integral of the measurement values. For TRX-1 and BAPL-2, the results are higher than 3% (1.6% uncertainty in measurement) and 3.59% (0.89% uncertainty in measurement) respectively. All the values of δ^{25} are lower from 0.679% to 2.37% than experiment ones. For δ^{28}, the calculated results are generally within the uncertainty integral of the measurement values except for BAPL lattices, for which the prediction values are underestimated about 5.9% to 9% .The agreement for C^* is very good.

3.2 The calculations for ZEEP-1,2,3

ZEEP-1,2,3 used natural uranium fuel. ZEEP were heavy water(D_2O) -moderated. Details of these lattices are given in Table 5 and 6. The $C^*_{Maxwellian}$ given in the work is 0.654.

Table 5. Brief Characteristics of ZEEP-1,2,3

Region	Outer radius (mm)	Isotope	Concentration (10^{24} atoms / cm^2)
Fuel	16.285	^{235}U ^{238}U	3.454E-4 4.760E-2
Air Gap	16.470		5.0E-5
Cladding	17.490		6.025E-2
Moderator		^{1}H ^{2}H O	1.529E-4 6.633E-2 3.324E-2

Table 6. Integral parameter comparison for Heavy water reactor benchmarks

Lattices	Integral parameter	Experiment	WIMS/D4	CENDL-3	ENDF/B-VI
ZEEP-1	$\mathbf{K_{eff}}$	1.0000	0.99398	1.0032	1.0036
	ρ^{28}	-----	0.26026	0.28687	0.282
	δ^{25}	-----	0.0258	0.0256	0.0263
	δ^{28}	0.0675	0.06804	0.06785	0.0682
	RCR	1.16	1.279	1.274	1.281
ZEEP-2	$\mathbf{K_{eff}}$	1.0000	0.9993	0.99993	1.00161
	ρ^{28}	-----	0.4688	0.52713	0.516
	δ^{25}	-----	0.04891	0.048715	0.0502
	δ^{28}	-----	0.07261	0.07192	0.0725
	RCR	-----	1.464	1.489	1.491
ZEEP-3	$\mathbf{K_{eff}}$	1.0000	1.00384	0.99752	1.00089
	ρ^{28}	----	0.62021	0.70397	0.688
	δ^{25}	----	0.06539	0.06538	0.0674
	δ^{28}	----	0.07672	0.07583	0.0764
	RCR	----	1.5968	1.6424	1.640

Although there are no quite enough experimental data with heavy water moderated lattices, the available data have shown that the calculated results based on CENDL-3 are within or close to the experimental uncertainty limits. All the lattice parameters calculated using CENDL-3 are in good agreement with those of experiments and ENDF/B-VI.

III. Conclusions

It is obvious that different evaluated nuclear data library is the cause of the difference of the results between old and new WIMS 69-group libraries. In general, the new library based on CENDL-3 or ENDF/B-VI is more reliable than the '1986' WIMS library.

In the benchmarks testing for both uranium metal fuel lattice assemblies and uranium oxide fuel lattice assemblies, the K_{eff} values calculated with ENDF/B-VI are underestimated. We can see that the excellent K_{eff} values for CENDL-3 are shown in the Table 4.

It is well-known that ^{238}U data affects strongly on the calculated results of reactor physics parameters and their trend, due to high concentration in the uranium fuel reactors. Through the comparisons of group cross sections between old and new library, no obvious difference was found for fission and absorption.

Through the comparisons of scattering cross sections between old and new library, a big difference was found. In the resonance energy region, the values of old library is much bigger than that of the new ones. The difference arises from the evaluation data and resonance self-shielding treatment for scattering cross sections as well. The bigger scattering cross section of old library caused relative lower possibility of absorption, higher slowdown power.

It is the same reason above mentioned, ^{238}U inelastic scattering effect of ENDF/B-VI makes assemblies spectrum harder and underestimates the fission contributions of ^{235}U, so that the K_{eff} values calculated with ENDF/B-VI are underestimated

It is obvious that ^{238}U data of CENDL-3 are better than that of ENDF/B-VI for thermal reactor calculations.

As shown in Table 4 and Table 6, the calculated results based on CENDL-3 are well consistent with the experimental results. It can be concluded that CENDL-3 is reliable for thermal reactor calculations.

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