Ⅳ BENCHMARK TESTING

Benchmark Testing of CENDL-2.1 for Heavy Water Reactor

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Introduction

The new version evaluated nuclear data library of ENDF-B6.5 has been released recently. In order to compare the quality of evaluated nuclear data CENDL-2.1 with ENDF/B-6.5, it is necessary to do benchmark testing for them.

In this work, CENDL-2.1 and ENDF/B-6.5 were used to generate the WIMS-69 group library respectively, and the benchmarks testing was done for the heavy water reactor, using WIMS5A code, WIMSD[1] which is a good lattice physics code and has been widely used in the world.

1 Generating of group constants library

The generation of the new group constant libraries based on CENDL-2.1 and ENDF/B-6.5 was done with NJOY94.35[2] code system, in which the modules MODER、RECONR、BROADR、UNRESR、THERMR and GROUPR were used to produce 69-group data file GENDF from ENDF/B format evaluated nuclear data according to the temperatures, weighting function and so on. The module WIMSR converts the GENDF file to the WIMSD format. The resonance reconstruction tolerance was 0.1%, and the tolerance for thinning of 0.2% was used.

In the old WIMS library, only one fission spectrum was given. In this work, the fission spectrum was composed of 7.5% ^{238}U and 92.5% ^{235}U fission spectrums. In the group averaging cross section calculations, input weight function(IWT=1)[3] was used for weighting.

In the WIMS 69-groups library, only absorption and neutron yield per fission integrals were tabulated, and all other cross sections were entered corresponding to a σ_0, which was chosen from input values. Clearly, the results of benchmark calculations are sensitive to the selection of σ_0. In this work, σ_0 was derived from the calculation according to normal reactor cells. Table 1 shows the selected reference values for each isotope.

The Goldstein-Cohen λ values should be consistent with the resonance integral tabulation. In this work, the selection of λ is similar to the old WIMS library. Table 2 shows the selected reference values for some isotopes.

Table 1 List of applied reference σ_0 values

Material	Reference σ_0 values.
^{1}H	1.E10
$^{16}O_8$	40.
D	1.E10
$^{27}Al_{13}$	1.E10
$^{235}U_{92}$	1.E3
$^{238}U_{92}$	28

Table 2 List of applied reference λ values

Materials	λ values
H	1.0
D	1.0
O	1.0
Al	1.0
^{235}U	0.2
^{238}U	0.2

2 Description of Benchmark Experiments

In order to test the data files, three benchmarks lattices containing ^{235}U and ^{238}U , which include ZEEP-1,2,3, were chosen.

The natural uranium fuel, and heavy water (D_2O)-moderated, are wed in ZEEP-1,2,3. Details of these lattices are given in Table 3.

For these lattices, experimental buckling values are available. It is easy to do leakage calculations using the input buckling values.

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

Region	Outer radius (mm)	Isotope	Concentration (10^{24} atoms / cm^3)
Fuel	16.285	^{235}U	3.454E-4
		^{238}U	4.760E-2
Air Gap	16.470		5.0E-5
Cladding	17.490		6.025E-2
		^{1}H	1.529E-4
Moderator		^{2}H	6.633E-2
		O	3.324E-2

3 Method of cell calculation

The cell calculations were used with WIMS/D4 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 had 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.

4 Results and Discussions

For ZEEP-1,2,3, the results of CENDL-2.1 , old WIMS library and ENDF/B-6.5 are given in Table 4. The $C^*_{Maxwellian}$ given in Ref.[4] is 0.654.

All the integrals parameters are defined as follows:

ρ^{28} — epithermal / thermal captures for ^{238}U

δ^{25} — epithermal / thermal fission for ^{235}U

δ^{28} — ^{238}U / ^{235}U fission

C^* — ^{238}U capture / ^{235}U fission

RCR — $C^*_{lattice} / C^*_{Maxwellian}$

Although there are no quite enough experiment data with heavy water moderated lattices, the available data have shown that the calculated results based on CENDL-2.1 are within or close to the experimental uncertainty limits. All the

lattices parameters calculated with CENDl-2.1 are in good agreement with those values of experiments.

Table 4 Integral parameter comparison for heavy water reactor benchmarks

Lattices	Integral parameter	Experiment	WIMS/D4	CENDL-2.1	ENDF/B-6.5
ZEEP-1	k_{eff}	1.0000	0.99398	1.0032	0.99790
	ρ^{28}	-----	0.26026	0.28687	0.28488
	δ^{25}	-----	0.0258	0.0256	0.02557
	δ^{28}	0.0675	0.06804	0.06785	0.06892
	RCH	1.16	1.279	1.274	1.29226
ZEEP-2	k_{eff}	1.0000	0.9993	0.99993	0.99465
	ρ^{28}	-----	0.4688	0.52713	0.52368
	δ^{25}	-----	0.04891	0.048715	0.04849
	δ^{28}	-----	0.07261	0.07192	0.07332
	RCR	-----	1.464	1.489	1.50807
ZEEP-3	k_{eff}	1.0000	1.00384	0.99752	0.99237
	ρ^{28}	----	0.62021	0.70397	0.69603
	δ^{25}	----	0.06539	0.06538	0.06504
	δ^{28}	----	0.07672	0.07583	0.07748
	RCR	----	1.5968	1.6424	1.66292

5 Conclusions

It is obvious that different evaluation nuclear data library is the cause of the difference of the results between old and new WIMS 69-group library. Through the comparison of ^{238}U 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 and therefore softer neutron spectrum.

Owning to the ^{238}U inelastic cross sections of ENDF/B-6.5 make neutron spectrum of assembly hardened and underestimated neutron moderating power, the neutron production rate in the high energy region is remarkably overestimated and the neutrons number in the low energy region is remarkably overestimated. Consequently, k_{eff} for thermal reactor calculations is underestimated. It seems that the data of ^{238}U in the ENDF/B-6.5 is not the best.

It is obvious that data files of CENDL-2.1 is better than that of old WIMS library for the heavy water reactors calculations, and is in good agreement with those of ENDF/B-6.5.

References

[1] "WIMSD, A Neutron Code for Standard Lattice Physics Analysis", ANSWERS Software Service, AEA Technology, Winfrith, UK, March 1996. See also "Code Package for WIMSD5A", NEA1507/01, OECD Nuclear Energy Agency, Pairs, France.

[2] R. E. MacFarlane and D. W. Muir, " The NJOY Nuclear Data Processing System Version 91", Los Alamos National Laboratory Report LA-12740-M (October 1994). For a summary of updates leading to NJOY94.105, see http://t2.lanl.gov/codes /njoy94/Readme105, prepared by R. E. MacFarlane

[3] S.Gan esan. "Update of the WIMS-D4 Nuclear Data Library Status Report of the IAEA WIMS Library Update Project", Report INDC(NDS)-290 (December, 1993).

[4] T.Zidi, Nuclear Data Processing and Applications, P,214.