TCA UO_2 / MOX Core Analyses

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In order to examine the adequacy of nuclear data, the TCA UO_2 and MOX core experiments were analyzed with MVP using the libraries based on ENDF/B-VI Mod.3 and JENDL-3.2. The ENDF/B-VI data underpredict k_{eff} values. The replacement of ^{238}U data with the JENDL-3.2 data and the adjustment of $^{235}\nu$-value raise the k_{eff} values by 0.3% for UO_2 cores, but still underpredict k_{eff} values. On the other hand, the nuclear data of JENDL-3.2 for H, O, Al, ^{238}U and ^{235}U of ENDF/B-VI whose $^{235}\nu$-value in thermal energy region is adjusted to the average value of JENDL-3.2 give a good prediction of k_{eff}.

1. Introduction UO_2 and MOX core experiments carried out using TCA of JAERI have been analyzed with the continuous energy Monte Carlo code MVP using the libraries based on ENDF/B-VI Mod.3 and JENDL-3.2 to examine the adequacy of the nuclear data. The effect on k_{eff} with fuel burnup was also evaluated with 2-D transport depletion calculation code PHOENIX-P using a 70-group library based on ENDF/B-VI.

2. Analyses and discussions MVP calculations are performed in three-dimensional core configuration shown in Fig.1. The cores whose moderator to fuel volume ratios are 1.5～3.0 for UO_2 and 2.42～5.55 for MOX are analyzed. The three libraries are examined here : ENDF/B-VI; ENDF/B-VI whose ^{238}U data are replaced with that of JENDL3.2; JENDL3.2. The results are shown in Table-1.

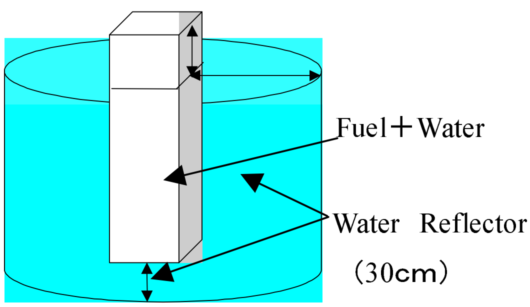

Fig.1 Core Configuration

The followings can be found from the table :

① ENDF/B-VI gives lower k_{eff} by about 0.5% $\Delta k/k$ for UO_2 cores and about 0.25% $\Delta k/k$ for MOX cores,

② ^{238}U of JENDL-3.2 raises k_{eff} by 0.2% $\Delta k/k$ for UO_2 cores and 0.1% $\Delta k/k$ for MOX cores,

③ JENDL3.2 gives higher k_{eff} by 0.2 to 0.4% $\Delta k/k$ for UO_2 cores and 0.1 to 0.2% $\Delta k/k$ for MOX cores.

Table-1 k_{eff} calculated with MVP for TCA U/MOX Cores

Core Name	Fuel rods array	EN/B-VI*	EN/B-VI+ J3.2 ^{238}U	JENDL3.2
1.50U	24X24	0.99508±0.0251%	0.99738±0.0243%	1.00419±0.0232%
1.83U	22X22	0.99448±0.0250%	0.99661±0.0237%	1.00290±0.0244%
2.48U	20X20	0.99448±0.0245%	0.99614±0.0240%	1.00251±0.0243%
3.00U	19X19	0.99444±0.0246%	0.99658±0.0232%	1.00206±0.0230%
2.42Pu	22X22	0.99735±0.0236%	0.99843±0.0235%	1.00095±0.0240%
2.98Pu	21X21	0.99782±0.0239%	0.99882±0.0234%	1.00197±0.0232%
4.24Pu	20X20	0.99788±0.0232%	0.99878±0.0229%	1.00135±0.0224%
5.55Pu	21X21	0.99729±0.0216%	0.99825±0.0215%	1.00187±0.0218%

* A cross-section library based on the ENDF/B-VI Mod.3 is used

The results are also shown in Fig.2 and Fig.3.

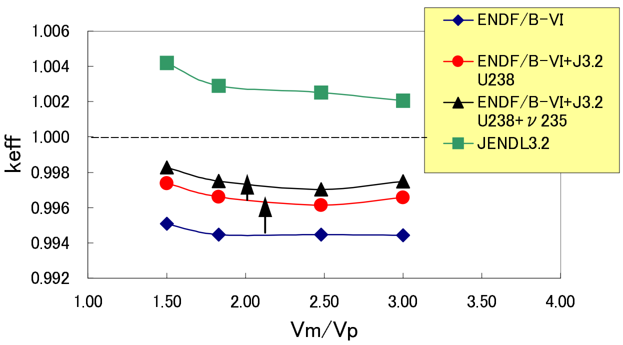

Fig.2 keff of TCA UO2 Core

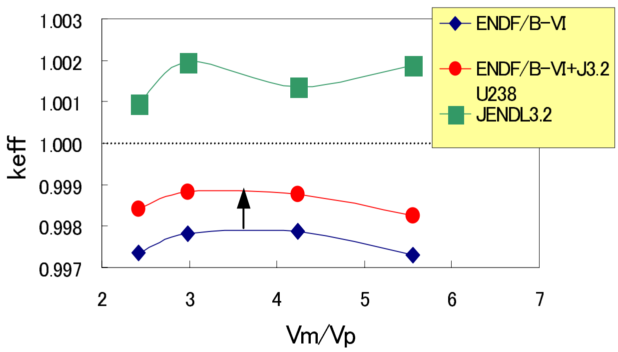

Fig.3 keff of TCA MOX Core

The reason why ENDF/B-VI data underpredict k_{eff} values was investigated. According to H.Takano, the k_{eff} values calculated with ENDF/B-VI are underpredicted and the harder neutron spectrum due to the inelastic scattering of ^{238}U and the ν-value of ^{235}U are the cause of the underprediction[1]. Therefore, we did the check calculations to confirm the effects.

First, we processed inelastic cross-sections and scattering matrices of ^{238}U of ENDF/B-VI and JENDL-3.2 with NJOY. The results are shown in Fig.4 and Fig.5.

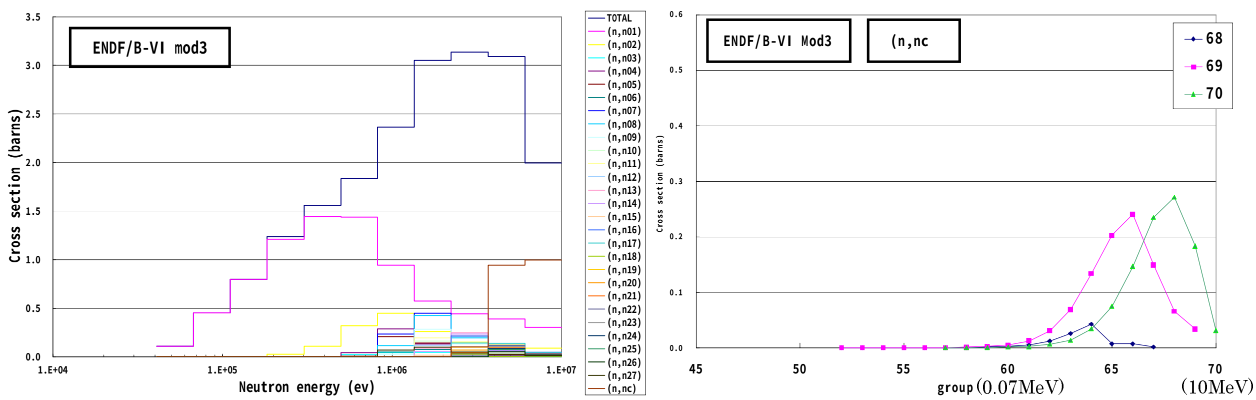

Fig.4 In-elastic Scattering Cross-sections and Matrix of ENDF/B-VI

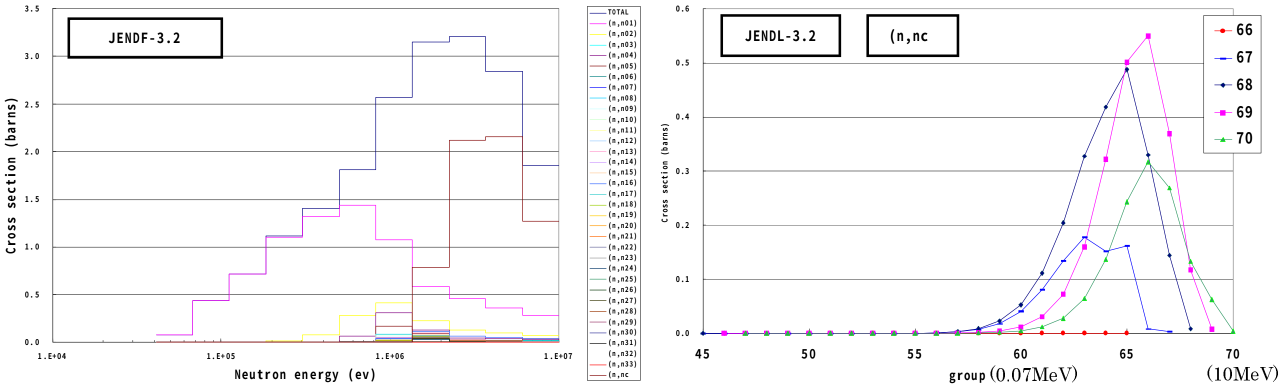

Fig.5 In-elastic Scattering Cross-sections and Matrix of JENDL-3.2

It is obvious from the figures that the continuum scattering matrix of ENDF/B-VI is smaller than that of JENDL-3.2 and gives harder neutron spectrum, and hence more neutron leakage.

In order to confirm that the cause of the underprediction of k_{eff} for the TCA critical experiments is the neutron leakage from the core, the k_{inf} values of MVP were compared for an infinite array of the PWR standard cell shown in Fig.6. As can be seen from Table 2, the replacement of ^{238}U nuclear data gives little effect on k_{inf} of the infinite array system. So, it is clear that the difference of inelastic scattering matrix, especially that of the continuum, affects strongly k_{eff} prediction of experimental cores through the change in neutron leakage from the cores.

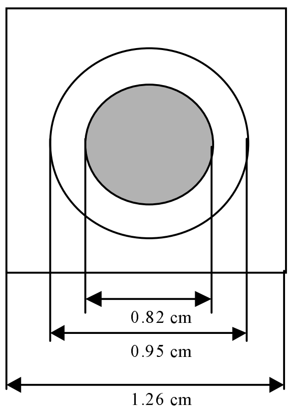

Fig.6 PWR standard cell

Table 2　Effect of ^{238}U on infinite lattice

	k_{inf}
ENDF/B-VI　^{238}U	1.38273±0.0153%
JENDL-3.2　^{238}U	1.38254±0.0153%

Secondly, the fission-reaction rate ($\sigma_f \phi$) averaged ν-value of JENDL3.2 in the thermal energy range below 2.1 eV was calculated and then ν-value of ^{235}U of JENDL-3.2 was found to be bigger than that of ENDF/B-VI by 0.14% and the adjustment to ν-value of ^{235}U of ENDF/B-VI increase multiplication factor by 0.09%Δk/k.

$$\bar{\nu}_t^{JENDL3.2} = \frac{\int \nu\sigma\phi\, E}{\int \sigma\,\phi\, dE} = 2.4372, \quad \bar{\nu}_t^{ENDF/B-VI} = 2.4338, \quad f = \frac{\bar{\nu}_t^{JENDL3.2}}{\bar{\nu}_t^{ENDF/B-VI}} = 1.0014.$$

Energy dependence of $^{235}\nu$-values is presented in Fig.7.

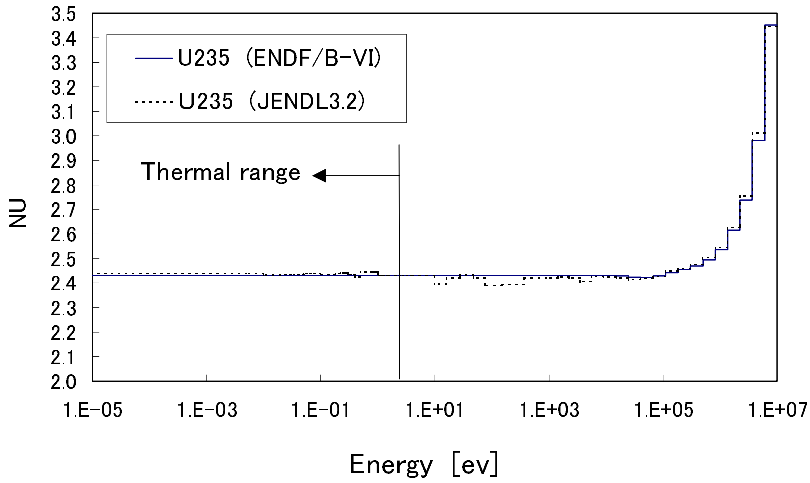

Fig.7 Number of neutrons per fission

The effect of the replacement of ^{238}U is 0.2% Δk/k and the adjustment of $^{235}\nu$ is 0.1% Δk/k. Finally, summing up those effects, we can explain 0.3% difference totally in 0.4～0.5% difference found in Table 1. It is shown in Fig.2 and Fig.3 that the modified library taken into account of them gives almost the same calculation accuracy of k_{eff} for both UO_2 and MOX cores.

The effect on k_{eff} due to the change in ^{238}U and $^{235}\nu$ data with fuel burnup have been also evaluated with 2-D transport depletion calculation code PHOENIX-P. Nuclear data of both libraries were processed and 70 energy-group libraries were generated with NJOY for this purpose. A 17×17 PWR standard assembly was used for the evaluation. The result is shown in Fig.8. It can be found from the figure that the total effect on k_{eff} is 0.1% Δk/k at 0 MWd/t and -0.07% Δk/k at 60 GWd/t for a PWR 17×17 standard assembly with 4.1 w/o enriched UO_2 rods.

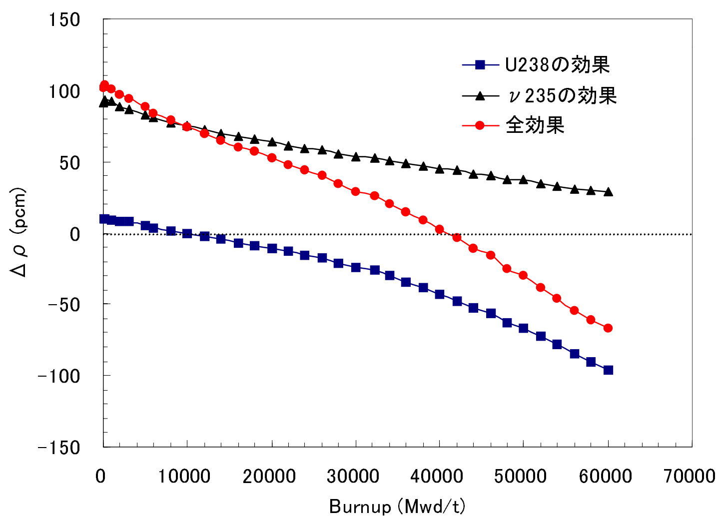

Fig.8 Change in keff with Burnup

The effect of the difference of nuclear data between JENDL-3.2 and ENDF/B-VI on k_{eff} value was investigated for the TCA 15×15 UO_2 core with fuel rods of 2.6w/o ^{235}U. The results are shown in the Table 3 where, e.g., in the case of "JENDL-3.2 base", the reference k_{eff} value is based on all the JENDL-3.2 data and the nuclear data of a nuclide is replaced with the corresponding data of ENDF/B-VI. Considering the result obtained previously with regard to $^{235}\nu$, it can be concluded that the use of the JENDL-3.2 data with ^{235}U of ENDF/B-VI whose $^{235}\nu$ in thermal range is adjusted to the average value of JENDL-3.2 gives good criticality : keff=0.99916+0.0009=1.00006.

Table 3 k_{eff} values calculated with MVP for the TCA 15×15 core

JENDL3.2 Base

Cross-section Library	MVP	
	Keff	Δρ(%)
①J3.2 (H,O,Al,U235,U238)	1.00293±0.0236	0.000
②J3.2 (O,Al,U235,U238) ENDF/B-VI(H)	1.00258±0.0232	-0.035
③J3.2 (H,Al,U235,U238) ENDF/B-VI(O)	1.00275±0.0239	-0.018
④J3.2 (H,O,U235,U238) ENDF/B-VI(Al)	1.00172±0.0239	-0.120
⑤J3.2 (H,O,Al,U238) ENDF/B-VI(U235)	0.99916±0.0239	-0.376
⑥J3.2 (H,O,Al,U235) ENDF/B-VI(U238)	0.99995±0.0238	-0.297
⑦ENDF/B-VI(H,O,Al,U235,U238)	0.99487±0.0234	-0.808

H is the cross-section of H_2O.

ENDF/B-VI Base

Cross-section Library	MVP	
	Keff	Δρ(%)
①ENDF/B-VI (H,O,Al,U235,U238)	0.99487±0.0234	0.000
②ENDF/B-VI (O,Al,U235,U238) J3.2 (H)	0.99593±0.0239	0.107
③ENDF/B-VI (H,Al,U235,U238) J3.2 (O)	0.99626±0.0248	0.140
④ENDF/B-VI (H,O,U235,U238) J3.2 (Al)	0.99607±0.0240	0.121
⑤ENDF/B-VI (H,O,Al,U238) J3.2 (U235)	0.99873±0.0244	0.388
⑥ENDF/B-VI (H,O,Al,U235) J3.2 (U238)	0.99726±0.0243	0.241
⑦J3.2(H,O,Al,U235,U238)	1.00293±0.0236	0.808

H is the cross-section of H_2O.

3. Conclusions UO_2 and MOX core experiments of TCA have been analyzed with MVP using the libraries based on ENDF/B-VI and JENDL-3.2. It has been found that the replacement of ^{238}U data with the JENDL-3.2 data and the adjustment of $^{235}\nu$-value raise the k_{eff} values obtained using ENDF/B-VI based library by 0.3% for UO_2 cores, but still underpredict k_{eff} values. On the other hand, the use of a combined library of JENDL-3.2 for H, O, Al, ^{238}U and ENDF/B-VI for ^{235}U whose $^{235}\nu$ in thermal energy region is adjusted to the average value of JENDL-3.2 seems to give a good prediction of k_{eff}, while the adequacy should be examined from other points of view, e.g., reaction rate ratio measurements, PIE data etc.

Acknowledgements

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Reference

[1] Takano, H., et al: "Reactor Benchmark Testing for JENDL3.2,JEF-2.2,and ENDF/B-VI-2," Int. Conf. Phys. of Nucl. Sci. Technol. (1998).