Nuclear Data Needs for Fast Reactors

Go Chiba

Japan Atomic Energy Agency
e-mail:chiba.go@jaea.go.jp

In the present paper, we show that the neutronics parameter uncertainties expected in current design studies of fast reactors are reasonable when both differential data and integral data are taken into consideration. This conclusion is based on an assumption that cross section covariance is properly evaluated. We attempt to verify the cross section covariance of JENDL-3.3 with the integral data, which were obtained at critical assemblies at Los Alamos National Laboratory. As a result, we suggest that uncertainty of P1 coefficient of elastic scattering cross sections of U-238 seems to be underestimated.

I Introduction

Various researches on nuclear data for fast reactor applications have been carried out until now. Currently, targets of nuclear data researches for fast reactor applications shift to improvement of the nuclear data quality of minor actinides and fission products. In the present paper, we will discuss necessities of nuclear data researches for fast reactor applications except for those on minor actinides and fission products, and attempt to obtain its conclusion.

II Nuclear data needs for fast reactors

'Nuclear data needs for fast reactors' are motivations to improve the prediction accuracies for neutronics parameters of fast reactors. **Table 1** shows uncertainties in neutronics parameters with 1σ reliability, which are expected in the current design studies for fast reactors. These uncertainties are composed of uncertainties induced by nuclear data and numerical simulations for neutron transport.

Before we discuss a necessity to reduce these uncertainties, we have to show that these uncertainties are reasonable. This is the main target of the present paper. The necessity to reduce uncertainties is a future topic.

Neutronics parameter uncertainties induced by nuclear data uncertainties can be estimated using covariance data given in nuclear data files and sensitivity coefficients. **Table 2** shows an example of this estimation for a 1,500MWe fast reactor(1). This result shows

Table 1: Neutronics parameter uncertainties expected in fast reactor design studies

	Uncertainties (%)
Criticality	0.4
Sodium-voided reactivity	7.5
Doppler reactivity	7.5

that the current nuclear data satisfies the expectation for prediction accuracies of the sodium-voided reactivity and the Doppler reactivity. However, uncertainty in criticality is much larger than the expectation. **Table 3** shows the component-wise uncertainties in criticality. It is desirable to improve nuclear data for such nuclide and reaction, if possible.

It has been shown above that we cannot satisfy the expectation of prediction accuracies of neutronics parameters in the current design studies only with the differential data, which are information on evaluated nuclear data files. Hence, we have utilized also the integral data, which are, for example, multiplication factor or spectrum indices obtained at critical assemblies or power reactors via the cross section adjustment technique based on the Bayesian theory. **Table 4** shows neutronics parameter uncertainties when using both the differential and integral data. It is shown that the expectation in the design studies is satisfied using these information.

We have shown above that the uncertainties expected in the current design studies are reasonable. However, it should be noted that the above conclusion is based on the following assumptions:

- Uncertainties induced by numerical simulations for neutron transport are 'properly' estimated.
- Covariance data for nuclear data are 'properly' estimated.

The current numerical simulations for neutron transport are based on the deterministic

Table 2: Nuclear data-induced neutronics parameter uncertainties

	Uncertainties (%)
Criticality	1.0
Sodium-voided reactivity	6.0
Doppler reactivity	8.0

Table 3: Nuclide- and reaction-wise uncertainties in criticality

	Uncertainties (%)
Pu-239, χ	0.4
Pu-239, (n,f)	0.5
U-238, (n,n')	0.3
Fe, (n,n')	0.5

Table 4: Nuclear data-induced neutronics parameter uncertainties with differential and integral data

	Uncertainties (%)
Criticality	0.26
Sodium-voided reactivity	4.0
Doppler reactivity	7.0

theory. Hence, it is not easy to quantify uncertainties induced by numerical simulations. Especially, it is difficult to quantify a correlation between the uncertainties for critical assemblies and those for power reactors since the structure of unit lattice (fuel assembly) is totally different from each other. If it is possible to utilize the Monte-Carlo method for neutronics simulations, the uncertainties induced by numerical simulations may be easily estimated.

Since evaluated nuclear data is 'evaluated' by a person, evaluated nuclear data depend on the person. The evaluated nuclear data is verified through its application into integral data. This procedure can be also applied to covariance of nuclear data. In the next chapter, we will show an example to verify the covariance data with integral measurement data.

III Verification of evaluated covariance data with integral measurement data

In this chapter, we will utilize experimental data obtained at fast critical assemblies in Los Alamos National Laboratory. The features of these assemblies are shown in **table 5**.

Table 5: Features of critical assemblies

Name	Fuel	U-reflector	Radius (cm)	Exp. error ($\Delta k/kk'$)
JEZEBEL	Pu	No	6.3849	0.002
JEZEBEL-240	Degraded Pu	No	6.6595	0.002
GODIVA	U	No	8.7407	0.001
FLATTOP-Pu	Pu	Yes	24.142 (Fuel:4.5332)	0.003
FLATTOP-25	U	Yes	24.1242 (Fuel:6.1156)	0.003

Figure 1 shows C/E values of criticalities of these assemblies with the latest nuclear data files. The error bars in this figure refer to 1σ uncertainties of measurement data. These neutron transport calculations are carried out with the continuous-energy Monte-Carlo code.

The nuclear data-induced uncertainties in these criticalities, V_k, can be estimated as

$$V_k = \vec{G}\vec{M}\vec{G}^t \quad (1)$$

where $\vec{G}$ is sensitivities of nuclear data to k_{eff} and $\vec{M}$ is covariance matrix of nuclear data. In the present study, we calculate $\vec{G}$ with the discrete ordinates transport method and use covariance data given in JENDL-3.3. The calculated uncertainties, *i.e.*, standard deviations and correlation matrix, are shown in **table 6** and **7**.

To verify deviations of these C/Es from 1.0 and uncertainties in C/Es, we calculate χ^2 value defined as

$$\chi^2 = (\vec{CE} - \vec{1.0}) \cdot \vec{V}^{-1}(\vec{CE} - \vec{1.0})^t \quad (2)$$

where $\vec{CE}$ refers to a vector of C/E values, $\vec{1.0}$ a vector whose elements are 1.0 and $\vec{V}$ covariance matrix defined as

$$\vec{V} = \vec{V}_k + \vec{V}_e + \vec{V}_m \quad (3)$$

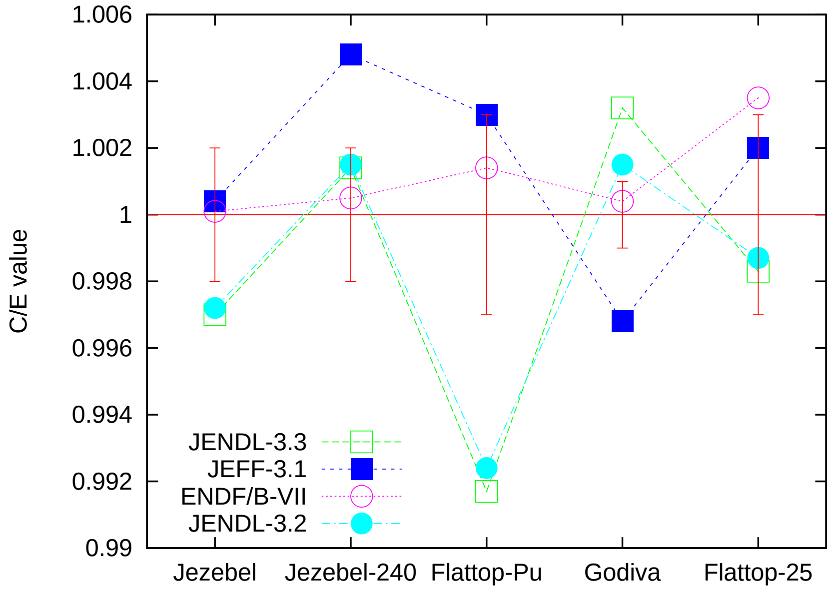

Figure 1: C/E values of criticalities of LANL small-sized fast critical assemblies

Table 6: Standard deviations in criticalities induced by nuclear data uncertainties

Core	Standard deviation	C/E
JEZEBEL	0.0054	0.9970
JEZEBEL-240	0.0057	1.0014
FLATTOP-Pu	0.0064	0.9917
FLATTOP-25	0.0052	0.9984
GODIVA	0.0041	1.0032

Table 7: Correlation matrix in criticalities induced by nuclear data uncertainties

	JEZ	JEZ240	FLAT-Pu	FLAT-25	GODIVA
JEZ	1.00	0.98	0.85	0.07	0.09
JEZ240		1.00	0.85	0.05	0.07
FLAT-Pu			1.00	0.31	0.08
FLAT-25				1.00	0.77
GODIVA					1.00

where $\vec{V}_e$ corresponds to uncertainties in experimental data and $\vec{V}_m$ statistical errors in calculated values. We obtain 6.8 of this χ^2 value in the present case. A value, that χ^2 is divided by the degree of freedom (5 in this case), becomes about 1.4. This result suggests that nuclear data covariance, or uncertainty in experimental data or statistical errors in Monte-Carlo calculations are slightly underestimated.

In the results obtained with JENDL-3.3, C/E values for U-reflected assemblies (FLATTOP-

Pu and FLATTOP-25) are much smaller than those of bare assemblies (JEZEBEL and GODIVA). However, as shown in table 7, nuclear data-induced uncertainties of JEZEBEL and FLATTOP-Pu (GODIVA and FLATTOP-25 also) have strong correlations to each other. Hence, it is difficult to describe this 'reflector-bias' with nuclear data uncertainty.

This bias is not observed in the ENDF-VII result at all. Through sensitivity analyses, this difference is caused by a difference in the P_1 coefficients of elastic scattering cross sections of U-238. **Figure 2** shows this coefficient. **Figure 3** shows differences of the P1

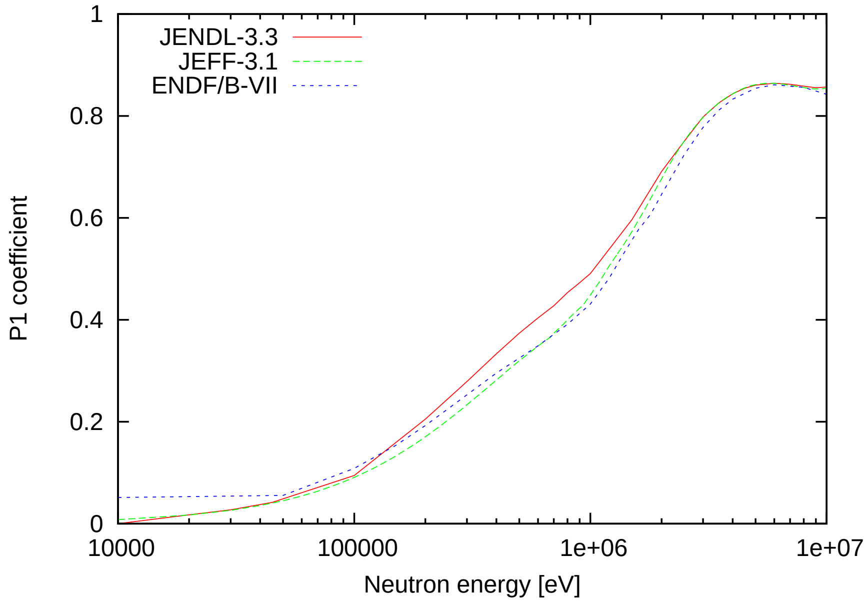

Figure 2: P_1 coefficients of elastic scattering cross sections of U-238

coefficients of ENDF/B-VII and JEFF-3.1 to that of JENDL-3.3. JENDL-3.3 evaluates this cross section larger about 10% systematically than the other data files.

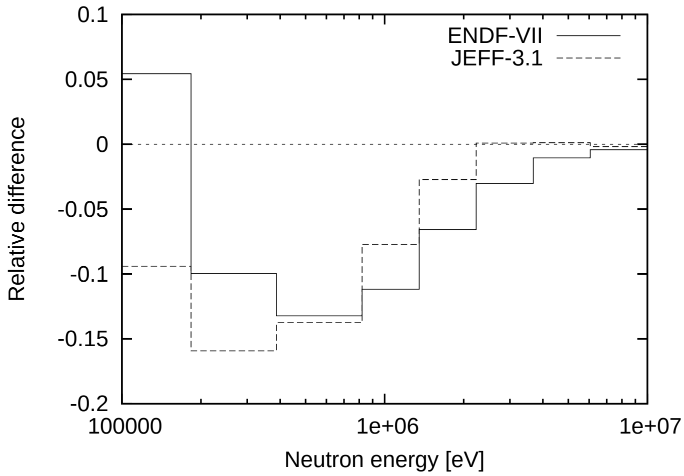

Figure 3: Difference in P1 elastic scattering cross sections of U-238 to JENDL-3.3

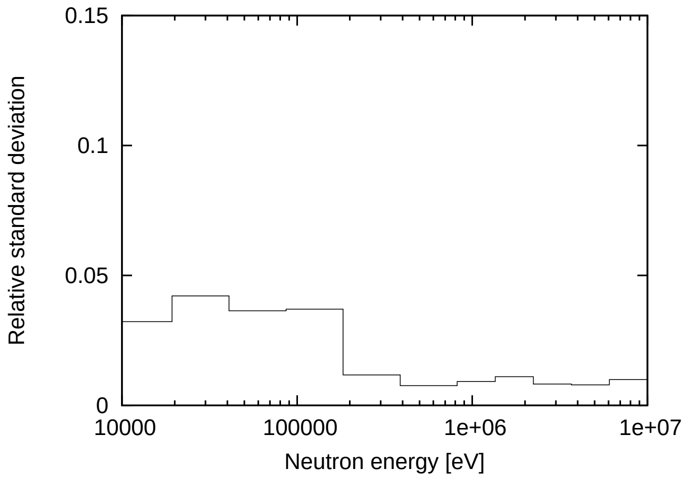

Figure 4: Standard deviation of P1 elastic scattering cross sections of U-238

Figure 4 shows standard deviations of this cross section based on JENDL-3.3. This uncertainty is much smaller than the difference between different nuclear data files.

With this comparison and the reflector-bias observed in the JENDL-3.3 results, it can be said that the uncertainty for P1 coefficients of elastic scattering cross sections of U-238 seem to be underestimated in JENDL-3.3 evaluations.

IV Conclusion

We have shown that the neutronics parameter uncertainties expected in the current design studies of fast reactors are reasonable when both differential and integral data are taken into consideration. We also pointed out that this conclusion is based on an assumption that cross section covariance is properly evaluated. We have attempted to verify the evaluated cross section covariance with the integral data.

References

[1] T.Hazama, *et al.*, 'Development of the unified cross-section set ADJ2000R for fast reactor analysis,' JNC TN9400 2002-064 (2002) [in Japanese].