Error Estimation for ADS Nuclear Properties by using Nuclear Data Covariances

Kasufumi TSUJIMOTO
Center for Proton Accelerator Facilities, Japan Atomic Energy Research Institute
Tokai-mura, Naka-gun, Ibaraki-ken 319-1195
e-mail : ktsuji@omega.tokai.jaeri.go.jp

Error for nuclear properties of accelerator-driven subcritical system by the uncertainties of nuclear data was performed. An uncertainty analysis was done using the sensitivity coefficients based on the generalized perturbation theory and the covariance matrix data. For major actinides and structural material, the covariance data in JENDL-3.3 library were used. For MA, newly evaluated covariance data was used since there had been no reliable data in all libraries.

1. Introduction

The Japanese long-term program called OMEGA has started in 1988 for research and development of new technologies for partitioning and transmutation of minor actinides (MA) and fission products. Under the OMEGA Program, for a dedicated transmutation system, the Japan Atomic Energy Research Institute (JAERI) has been proceeding with the research and development on accelerator-driven subcritical system (ADS)[1].

The uncertainties of nuclear properties of ADS are very significant both on the safety assessment and the economic evaluation. High k_{eff} , for example, implies small radial peaking and low proton beam current but risk of approaching criticality under accidental conditions will increase. Therefore, uncertainties on the subcriticality level must be include in the definition of the design margin. In this study, preliminary error estimation due to uncertainties of the nuclear data for ADS nuclear properties, such as the subcriticality and the coolant lead-bismuth (Pb-Bi) void reactivity, was performed.

2. Uncertainty Analysis

Sensitivity Analysis

Some reactor parameter *R*, such as k_{eff} or a reactivity coefficient, can be represent as a function of cross-section data. Sensitivity coefficients are defined physically such that they represent the effect on the integral parameter *R* due to the change of the cross-section data σ. Based on the generalized perturbation theory, the sensitivity coefficients are formally given by

$$S = \frac{\partial R}{R} \times \frac{\sigma}{\partial \sigma}$$

Sensitivity analysis produce energy-dependent sensitivity coefficients that give the relative change in the integral parameter *R* as a function of relative changes in the cross-section data by nuclide, reaction, and energy. The calculations of the sensitivity coefficients were performed using SAGEP code[2] with JENDL-3.3 library.

The calculated core was a Pb-Bi cooled ADS proposed by JAERI[3] for a dedicated transmutation system of MA. For the core fuel, mixture of mono-nitride of MA (60%) and plutonium (40%) was used with an inert matrix, ZrN. The isotopic composition of MA and Pu were assumed considering the following reprocessing scheme. The spent PWR fuel of 50 GWd/t burnup was reprocessed after 7 years cooling, and MA and Pu were recovered. Before fabrication of the ADS fuel, additional 3-year period after recovery of MA and Pu was assumed[3]. Nitrogen with ^{15}N enriched (100%) was assumed to be used for both (MA,Pu)-nitride and ZrN.

The calculational results of the sensitivity coefficients for k_{eff} by nuclide and reaction are shown in Fig.1. The results show that the sensitivity coefficients of fission cross section and ν-value of the actinide nuclides are significant large. The contributions of capture cross section of MA, such as ^{237}Np and ^{241}Am, are also relatively large. In the structural, coolant and other material, the sensitivity coefficients of elastic scattering of ^{15}N and inelastic scattering of Pb and Bi are important. The energy break down of the sensitivity coefficients are presented in Fig.2 for ^{239}Pu and ^{241}Am which have relatively large sensitivity for k_{eff}. For ^{241}Am, The sensitivity coefficients of fission cross section and ν-value have peak values about 1MeV because the threshold fission cross section, while that of capture cross section are large from 10keV to 1MeV.

Covariance Matrix

Present status of nuclear data for MA is not sufficient, and there is large discrepancy among major nuclear libraries. Although the evaluation is, recently, proceeding with variance-covariance data, such data for MA are practically non-existent in all major libraries. In this study, since the main calculations were performed using JENDL-3.3 library, variance-covariance data provided in JENDL-3.3 library for major actinide ($^{239, 240, 241}Pu$) and structural material (Fe, Cr, Mn, Ni). For MA (^{237}Np, ^{241}Am and ^{243}Am), newly evaluated valiance-covariance data were used. The evaluations of capture, fission and number of fission neutron were done by Nuclear Data Center in JAERI based on JENDL-3.3 library. The variance data (diagonal section of covariance matrix) of capture and fission cross section processed by the ERRORJ code[4] of ^{237}Np, ^{241}Am and ^{243}Am are shown in Fig.3 with discrepancies between other libraries (ENDF/B-VI.8 and JEFF-3.0) and JENDL-3.3. The variance data of fission cross section are relatively small above threshold energy of fission reaction. On the other hand, the variance data of capture cross section for ^{241}Am and ^{243}Am are relatively large in the energy region from 100keV to 1MeV since there is less experimental data in this energy region. The variance data value of ν-value are 0.5-1.0% for these MA nuclides. These values are much smaller than those of capture and fission cross section.

For other structural materials like Pb and Bi, and ^{242}Pu, the covariance data in ENDF/B-VI were used for this study. For the nuclide for which there are no covariance data in all libraries, such as ^{15}N, tentative variance (not covariance) data are estimated from the discrepancies among major nuclear data libraries. The estimations of the covariance data for not only MA but also structural material, such as Pb, Bi, and ^{15}N are desirable for future study.

Error Estimations

The error estimations of the integral reactor parameter were performed using the sensitivity coefficients S and the covariance matrix M. The uncertainties of the integral parameter can then be obtained from GMG^t by matrix calculation. The results of the uncertainty analysis for k_{eff} and Pb-Bi void reactivity are given in Fig.4 together with main contributor for the total uncertainties. Total value is the square root of the sum of the squares. The total value (±0.9%) for k_{eff} is higher than corresponding values for critical fast reactor. The major contributor among the actinide nuclides is ^{237}Np and ^{241}Am, and the capture cross-section of ^{241}Am especially play a major role. For Pb-Bi void reactivity, the capture cross-section of ^{241}Am is much important than Pb and Bi. The reason of little contribution of Pb and Bi for the void reactivity is that the void reactivity for only core region was estimated in this study, so that the leakage effect has relatively lsmall contribution for the void reactivity.

3. Conclusions

The preliminary error estimations for ADS nuclear properties were carried out using the sensitivity coefficients and the covariance matrix. As expected, the important nuclear data for the uncertainties of k_{eff} and Pb-Bi void reactivity were the cross-section data of MA, especially the capture cross section of ^{241}Am. This study is a first step to provide the guideline on priorities for new evaluations or validation experiments. From the results of this study, present status of the calculation accuracies for MA-dominant fueled ADS and required improvement on specific nuclear data were derived.

In this study, as typical nuclear properties, k_{eff} and Pb-Bi void reactivity were analysis. For further work, bunrnup characteristics, such as burnup reactivity change and transmutation performance. However, for this purpose, it will be needed that the generalized perturbation theory should be extended to the system with external source. Moreover, in this study, although covariance data of capture, fission and ν-value for ^{237}Np, ^{241}Am and ^{243}Am, other data, such as fission spectrum, delayed-neutron fraction, and other nuclide will be necessary.

References

[1] T. Mukaiyama, T. Takizuka, M. Mizumoto, Y. Ikeda, T. Ogawa, A. Hasegawa, H. Takada, and H. Takano : "Review of research and development of Accelerator-driven System in Japan for transmutation of long-lived nuclides," *Prog. Nucl. Energy*, ADS special edition, **38**, No. 1-2, p.107, (2001).

[2] A. Hara, T. Takeda, and Y. Kikuchi :"SAGEP: Two-Dimensional Sensitivity Analysis Code Based on Generalized Perturbation Theory", JAERI-M-84-27 (in Japanese).

[2] Tsujimoto K., Sasa T., Nishihara K., Takizuka T., and Takano H.,:"Accelerator-Driven System for Transmutation of High-level Waste," *Prog. Nucl. Energy*, **37**, No. 1-4, p. 339, (2000).

[3] "Accelerator-driven Systems (ADS) and Fast Reactors (FR) in Advanced Nucler Fuel Cycles : A Comparative Study," Nuclear Energy Agency, Organization for Economic Cooperation and Development (2002).

[4] G. Chiba :"ERRORJ — Covariance Processing Code Version2.2", JNC TN9520 2004-003 (2004).

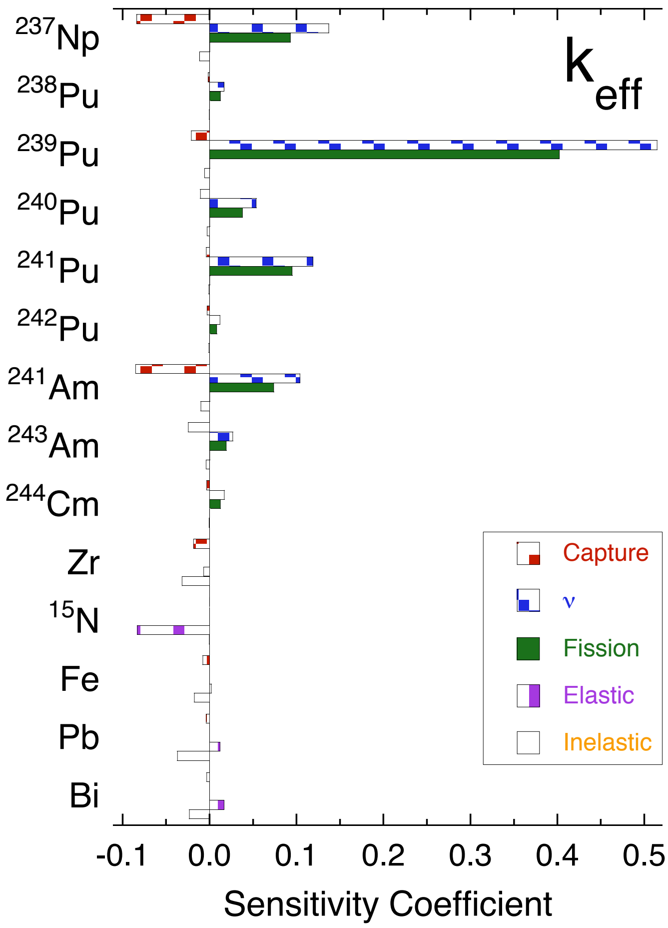


Fig.1 Sensitivity coefficients of k_{eff} by nuclide and reaction.

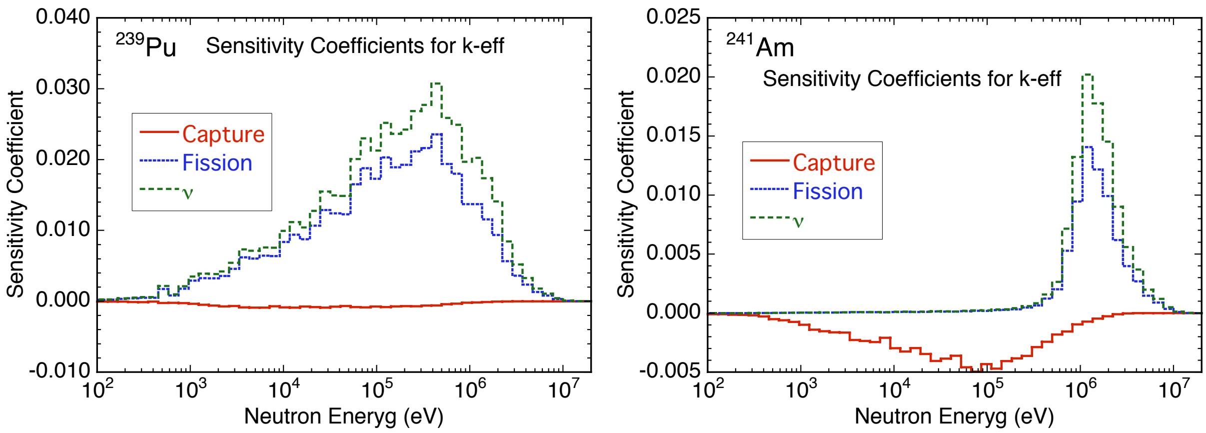
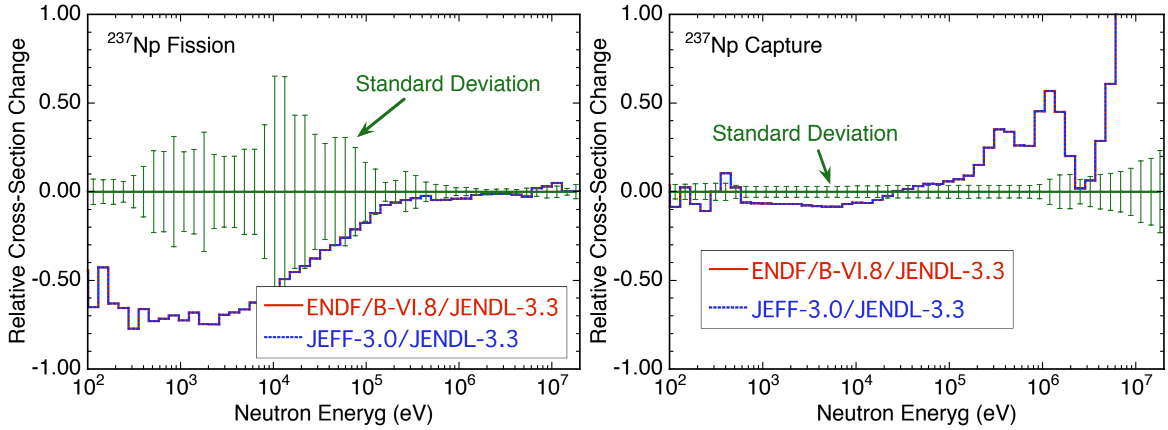
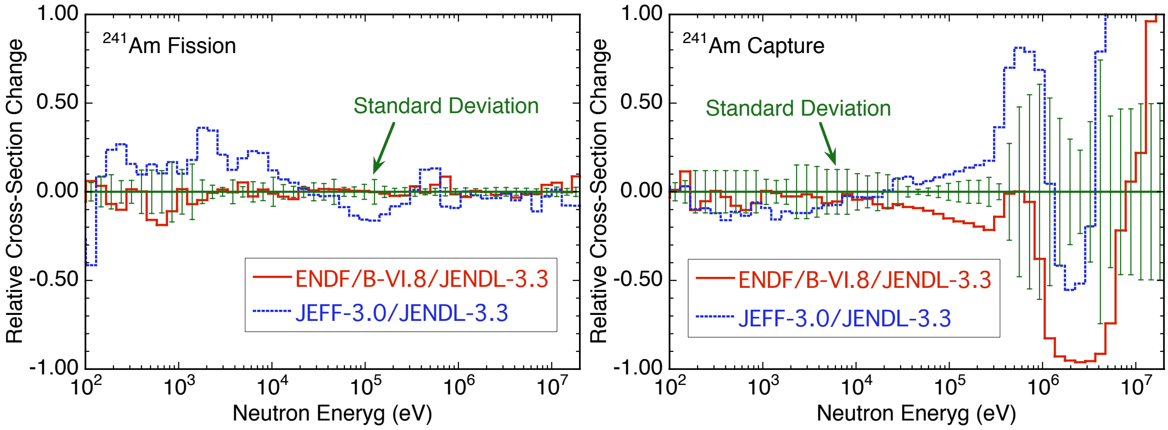


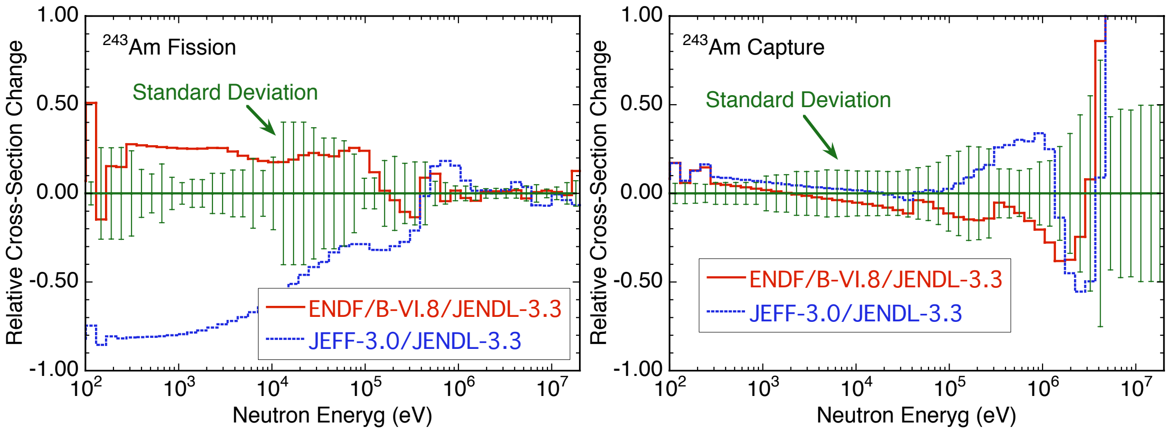
Fig.2 Energy break down of sensitivity coefficients of ^{239}Pu and ^{241}Am.



(a) ^{237}Np



(b) ^{241}Am



(c) ^{243}Am

Fig.3 Newly evaluated variance data of capture and fission cross section for ^{237}Np, ^{241}Am and ^{243}Am. Variance data is diagonal data of covariance matrix proceeded by ERRORJ code.

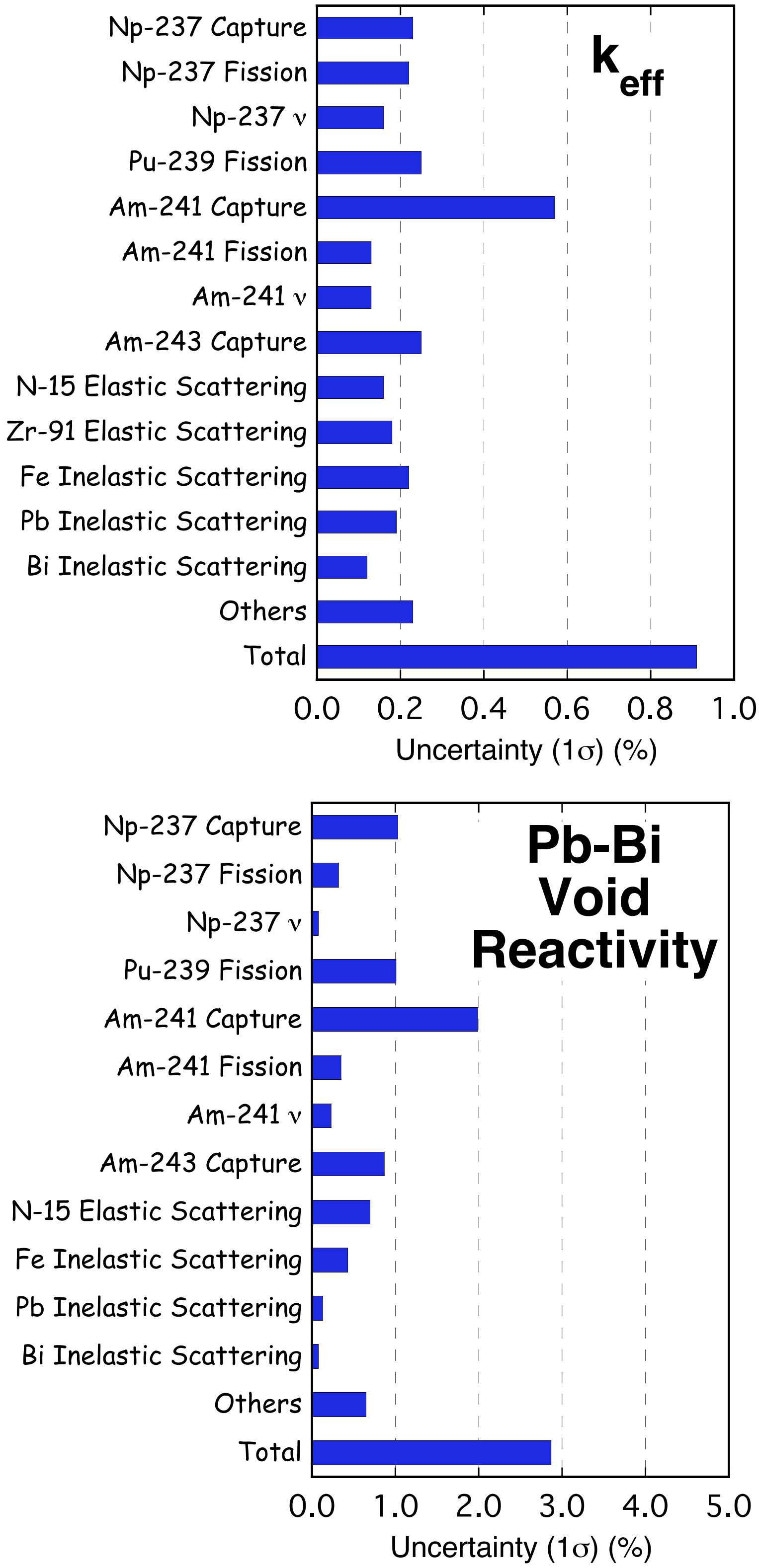


Fig.4 The results of error estimation for keff and Pb-Bi void (core only) reactivity.