CONVENTIONAL CROSS SECTION ADJUSTMENT METHOD IN NUCLEAR POWER ENGINEERING

Branislav Vrban[1], Jakub Lüley[1], Štefan Čerba[1], Ján Haščík[1]

[1]Institute of Nuclear and Physical Engineering, Slovak University of Technology in Bratislava, Ilkovičova 3, 812 19 Bratislava, Slovakia, EU

E-mail:branislav.vrban@stuba.sk

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1. Introduction

Generation IV International Forum (GIF) is a cooperative international endeavour that is currently trying to define and perform research and development needed to establish feasibility and performance capabilities of the next generation of nuclear systems. The main principles of these systems are rather well understood, however, their optimization, in order to comply more effectively with requirements and their timely deployment, requires the research in nuclear data. Although most nuclear data are by and large available in modern data files, their accuracy and validation is still a major concern. The major source of uncertainty in the calculated response is due to uncertainties in evaluated nuclear data such as microscopic cross sections, fission spectra, neutron yield, and scattering distributions that are contained in cross section evaluations. These uncertainties are governed by probability distributions which are unknown, but the evaluated data values are assumed to represent the mean of the distribution, and the evaluated variance represents a measure of the distribution width. Correlations as well as uncertainties in nuclear data can have a significant impact on the overall uncertainty in the calculated response; thus, it is important to include them in the uncertainty analysis. In order to decrease uncertainty induced by inaccurate nuclear data and to improve prediction accuracy of the target core parameters by using integral experimental data measured in critical assemblies the use of Cross Section Adjustment Method (CA) is considered a promising option.

2. The basic theory

The main principle of the conventional cross section adjustment method (CA) is that adjustments are applied to the evaluated cross section data as much as possible within their error limits and taking into account correlations, in such a way, that a better agreement between calculated results and measured integral data is obtained. An important precondition for the cross section adjustment method is that the linear relation always exists between variations of an integral data *R* (responses for the each experiment) and differential data *T* (cross section set):

$$\delta R = S.\delta T \ , \tag{1}$$

where the symbol *S* denotes the cross section sensitivity coefficients. Assuming that the cross-section set has a Gaussian distribution, and by using Bayesian theorem (basic derivations are described elsewhere [1]in detail) the posterior cross-section set of (CA) method *T'* is derived as:

$$T' = T_0 + MS^T \left[SMS^T + V_e + V_m \right]^{-1} \left[R_e - R_c(T_0) \right], \tag{2}$$

where*M* stands for the prior covariance matrix and $V_e + V_m$are the variances with respect to the experiment and the analysis (calculation) method respectively. The subscripts “e/c” represent if the response was experimentally measured or if it was calculated. Posterior covariance matrix satisfies the Eq. (2):

$$M' = M - MS^T \left[SMS^T + \left(V_e + V_m \right) \right]^{-1}, \quad (3)$$

As a result, adjusted cross section data, adjusted responses and variations for the target and the integral experiments are obtained. In addition, the uncertainty ΔR on the target integral parameter can be evaluated by the well-known sandwich formula:

$$\Delta R^2 = S_R M S_R^T . \quad (4)$$

where the impact of the individual reactions and energy groups can be evaluated separately. The diagonal elements of the resulting matrix (4) represent the relative variance values for each of the system under consideration, and the off-diagonal elements are the relative covariances between given experiments.

3. Calculation specification

In order to demonstrate the capabilities of ATCROSS[2]code developed by authors of this paper in cooperation with KAERI SFR team members, the sensitivity profiles for the specific experiments defined by Subgroup 33[3]were used. All of them were calculated by CEA in 33 energy group format suitable for the fast reactor calculations. The covariance data (JENDL-4.0) for our calculation were prepared by NJOY99.396 code[4]. Covariance data were used only for isotopes present in JENDL-4.0 library namely: ^{10,11}B, ^{16}O, ^{23}Na, ^{56}Fe, ^{58}Ni, ^{235,238}U, ^{238,239,240,241,242}Pu, ^{48}Ti, ^{55}Mn, ^{60}Ni and ^{241}Am. The list of integral simplified experiments, the used associated response values and standard deviations are shown in the Tab. 1. The symbol *C* stands for the calculated value by French well-known ERANOS code and *E* is the experimental (measured) quantity. Due to the fact that ERANOS is the deterministic code, no uncertainty is given to calculated result. The first seven responses correspond to the multiplication factor (k_{eff})and the rest to the measured spectral indices, where the first letter of spectral index describes type of reaction (*C – capture; F - fission*) and the next two numbers are the first and the last digits of atomic number. It can be seen that experimental values of spectral indices are accompanied with higher uncertainty as it is for eigenvalue experiment.Nevertheless spectral indices give a possibility to evaluate neutron spectra in a core and they serve as an additional degree of freedom in CA method application.As a target core ALLEGRO demonstration unit of gas cooled fast reactor with MOX fuel was used. Sensitivity profiles for target were calculated by SCALE system and module TSUNAMI [5] in 238 energy group structure of ENDF/B-VII.0 library and the values were confirmed by Direct Perturbation analyses.

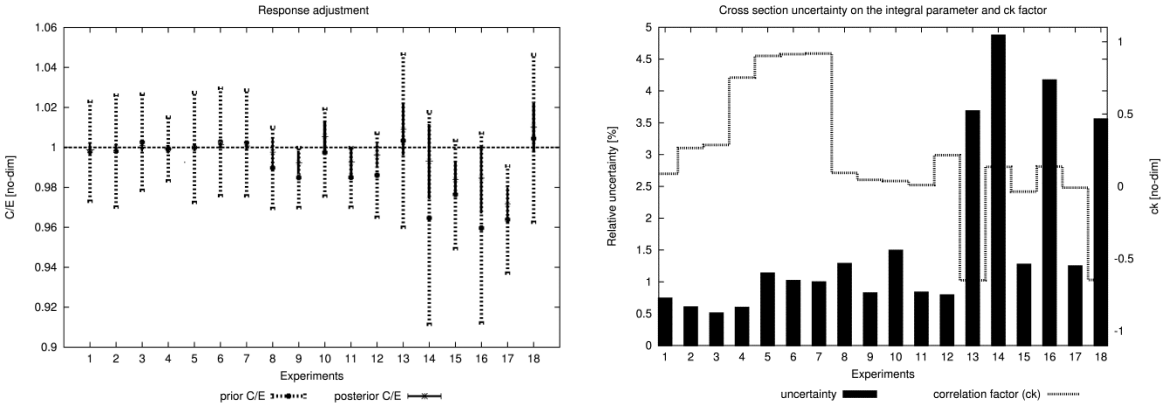
Tab. 1.*The computed and measured results of integral models*

No.	Benchmark problem	C	E	No.	Benchmark problem	C	E
1	FLATTOP	0.99801	1.00000±0.00300	10	JEZEBEL F28/F25	0.2127518	0.2133±0.01
2	JEZEBEL	0.998138	1.00000±0.00200	11	JEZEBEL F37/F25	0.968716	0.9835±0.014
3	JEZEBEL_PU240	1.00256	1.00000±0.00200	12	JEZEBEL F49/F25	1.4405614	1.4609±0.009
4	JOYO	1.0002241	1.00105±0.00180	13	ZPPR9 C28/F25	0.1300342	0.1296±0.019
5	ZPPR9	1.000718	1.00106±0.00117	14	ZPPR9 F28/F25	0.0996832	0.0207±0.027
6	ZPR6-7 Hi Pu	1.0030118	1.00080±0.00220	15	ZPPR9 F49/F25	0.9007318	0.9225±0.02
7	ZPR6-7	1.0032214	1.00051±0.00230	16	ZPR6-7 F28/F25	0.0213998	0.0223±0.03
8	FLATTOP F28/F25	0.1780584	0.1799±0.01	17	ZPR6-7 F49/F25	0.9093764	0.9435±0.021
9	FLATTOP F37/F25	0.8431558	0.8561±014	18	ZPR6-7 C28/F25	0.1328914	0.1323±0.024

These sensitivity profiles were then condensed to appropriate 33 energy group structure. The calculated value of multiplication coefficient of ALLEGRO core design was 1.035125±0.0001. The whole process of sensitivity calculation and basic ALLEGRO core specification can be found elsewhere [6]. The calculation was performed with 6 reactions namely: fission (MT18), nubar (MT452), capture (MT102), elastic (MT2), inelastic (MT4) and n2n (MT16). Total reaction was excluded from our calculation due to the fact that total reaction is basically the summation of all reactions.

4. Results

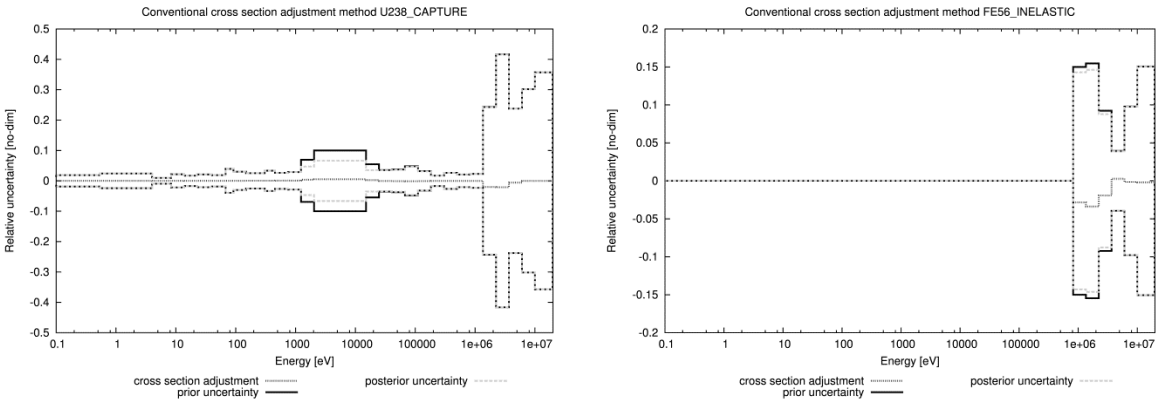
The calculated results are presented mainly in the graphical form and tabulated values are given in case of necessity. The performed uncertainty analysis shows that the biggest contributor to k_{eff} uncertainty induced by cross section data is ^{238}U with reactions capture and inelastic scattering. According to our results the second important contributor is ^{238}Pu with reaction capture. The third one is ^{56}Fe acting as a structural material with reactions of elastic and inelastic scattering. These results are in a good accordance with data presented by Alibertiet. al. **[7]** despite the fact that they investigated large energetic gas cooled fast reactor and in our case we have used different covariance data. The prior and posterior C/E responses with their variances are shown in the next figure. As it is clearly shown in the Fig. 1 a), posterior responses are pushed much closer to the reality and their variances were significantly decreased as well. In the case of experiments number 15 and 17, the original data were greatly biased, and therefore the CA method was not able to reach the correct value within one standard deviation.As can be seen in Fig. 1 b) the highest uncertainty induced by the cross section data defined by Eq. 5 in eigenvalue responses is observed in experiment No. 5 - ZPPR9.



a) b)

Fig.1: a) *Prior and posterior C/E responses; b)Uncertainty and similarity analyses results*

In the case of spectral indices the highest associated cross section induced uncertainty is observed in No. 147 - ZPPR9 F28/F25, where this uncertainty reaches almost 5%. The calculated ALLEGRO core eigenvalue uncertainty based on cross section data reaches almost 1%, but it should be noted that covariance data were not complete and this number correspond only to contribution of used covariances. The similarity analysis results defined by *ck*(correlation coefficient represented by the dot line and the right axis) show the highest similarity *ck = 0.918*between target eigenvalue response and the eigenvalue experiment No. 7 – ZPR6-7. The *ck* values higher than *0.9* for experiments 5 – 7 are acceptable due to the fact, that MOX fuel is rather well known and lots of integral experiments were equipped with similar compositions**.**Finally, as an example, the calculated adjustments for ^{238}U capture and ^{56}Fe reaction are shown in the next Fig. 2. The new predicted target k_{eff} reaches 1.03435.



a) b)

Fig.2: *CA application results* a)^{238}*U capture;* b)^{56}*Fe inelastic*

5. Conclusion

This paper introduces CA method and briefly presents its ability to improve prediction accuracy of the target core parameters. In conjunction with the cross section uncertainty analysis module, it can play a significant role in the future fast reactor development in Slovakia and ALLEGRO project. Clearly a further research in method itself is needed and strong effort should be carried out in order to receive more complex covariance data and related quantities. A lot of questions need to be solved regarding the methodology, used energy group structure, the geometry corrective factors, reactions used for adjustment and statistics itself. Nevertheless the research in this area is ongoing and we believe that will improve the calculation accuracy and our abilities in fast reactor systems.

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