Annals of Nuclear Energy 110 (2017) 928–940

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Cross section adjustment for fast reactor design: ATCROSS code and its application

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ARTICLE INFO

Article history:
Received 20 February 2017
Received in revised form 26 July 2017
Accepted 7 August 2017

Keywords:
Similarity and uncertainty assessment
Similarity analysis
PGSFR reactor
Adjustment method

ABSTRACT

For a competitive design of a fast reactor core in the Republic of Korea, the conventional cross section adjustment approach was chosen as a promising method to improve prediction accuracy of target core parameters. The new ATCROSS code has been developed which together with the APSTRACT code could serve as effective computational systems in the future SFR design. The determination of cross section uncertainty propagation on evaluated final parameter was implemented into these codes. As a result, the potential impact of inaccurate quantities in evaluated nuclear data, such as microscopic cross sections, can be determined and quantified.

The results of uncertainty analyses performed for the Korean PGSFR reactor are in a good accordance with those already published for the similar designs; thus the calculated target cross section uncertainties could be seen as realistic. The uncertainty of k_{eff} induced by cross section data reaches approximately 2%. The uncertainty induced by cross section data for fuel temperature and loss of coolant reactivity feedback effects reach 6% and 13%, respectively. These uncertainties may be reduced by applying the conventional cross section adjustment method in conjunction with sufficient amount of high-quality experimental data. The first application of conventional cross section method to PGSFR core design is introduced in the paper.

1. Introduction

In design studied of an innovative fast breeder nuclear reactor it is desirable to estimate and reduce uncertainties of design parameters. Validated and comprehensive sensitivity processing codes in conjunction with nuclear data covariances may serve as a powerful tool to compute uncertainty due to inaccurate data for quantities of applied interest. Data deviations in measured cross sections arise from random and systematic errors. Defective data can originate in discrete computer processing, important corrections can be overlooked, equipment might fail or it might be improperly calibrated and low statistic might be used for the measurement. The reasonable estimates of the underlying cross section (XS) probability distribution can be made due to application of the central limit theorem, therefore in the present analysis we assume the cross sections are distributed according to the normal distribution. The first ideas proposing the use of integral experiments in order to improve cross section data were presented and discussed at the UN Conference on Peaceful Use of Nuclear Energy in Geneva in 1964 (Cecchini et al., 1964). Due to the lack of covariance data between the various nuclear data types and the computational complexity the real application of various adjustment methods is not solved until today. Further application of generalized method of least squares combined with integral experimental data measured in critical assemblies and experimental reactor cores may improve the used cross section evaluation in order to achieve better prediction accuracy of the target core responses and parameters. The conventional cross section adjustment (CA) method has been considered as a promising design method suitable to achieve this goal (Salvatores et al., 2013).

In the past three years, within the framework of cross section uncertainty and adjustment program at KAERI (Korea Atomic Energy Institute) two new codes were developed. The sensitivity and perturbation analysis code APSTRACT (Luley et al., 2014) and ATCROSS (Vrban et al., 2013) code intended to evaluate the response uncertainty due to deviations in cross section data and the cross section adjustment tool in parallel may serve as a backbone of the computational system in SFR core design development.

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http://dx.doi.org/10.1016/j.anucene.2017.08.008
0306-4549/

In our scheme the self-shielded cross section libraries for transport or diffusion calculation are prepared by TRANSX code (MacFarlane and George, 1995) in ISOTXS format. PARTISN code (Alcouffe et al., 2008) as a discrete ordinates transport solver, suitable for 1D, 2D and 3D geometries, is used for the models of integral experiments. Moreover, PARTISN produces weighting function suitable for group collapsing procedure available in TRANSX code. For the target core, the fast hexagonal diffusion multigroup nodal option of DIF3D (Derstine, 2011) code serves as flux and eigenvalue solver. To transform the ENDF-6 formatted (Herman and Trkov, 2010) cross sections into energy-averaged cross section covariances the ERROR module of the NJOY 2012 code (MacFarlane et al., 2012) is used. ERROR can process the covariance data of cross sections including resonance parameters as well as angular and energy distributions of secondary neutrons.

2. Applied techniques and methods

2.1. Calculation scheme

In order to give a thorough overview of incorporated methods and data flow used in our approach, the simplified calculation scheme is presented in the Fig. 1. This scheme is described in the following paragraphs.

As shown in the calculation scheme, the cross section data required for transport and diffusion calculations were taken from ZZ-KAFAX-E70 (Kim et al., 2008), ZZ-KAFAX-F31 (Kim et al., 2009a) and ZZ-KAFAX-J33 (Kim et al., 2009 b) MATXS format nuclear data libraries. These libraries use 150 group structure for neutron (see Table 1) and 12 group structure for photon data. They were prepared based the ENDF/B-VII.0 (Chadwick et al., 2006), JEFF 3.1.1 (Santamarina et al., 2009) and JENDL-3.3 (Shibata et al., 2002) evaluated data file weighted by KALIMER-150 (Korean Advance LIquid MEtal Reactor) equilibrium core flux. On a case by case basis the self-shielded ISOTXS libraries were generated from MATXS libraries by TRANSX code in 25 energy group (see Table 1 – bolded and underlined values) structure using its own pre-calculated 150 group zone integrated neutron flux as the weighting function. All transport calculations were performed by PARTISN and DIF3D was used as the diffusion solver. For each experiment, its own 150 group neutron flux was used as the weighting function for energy group condensation.

The transport and diffusion solvers included in PARTISN and DIF3D codes provide the necessary information (including material composition, geometry description, forward and adjoint angular

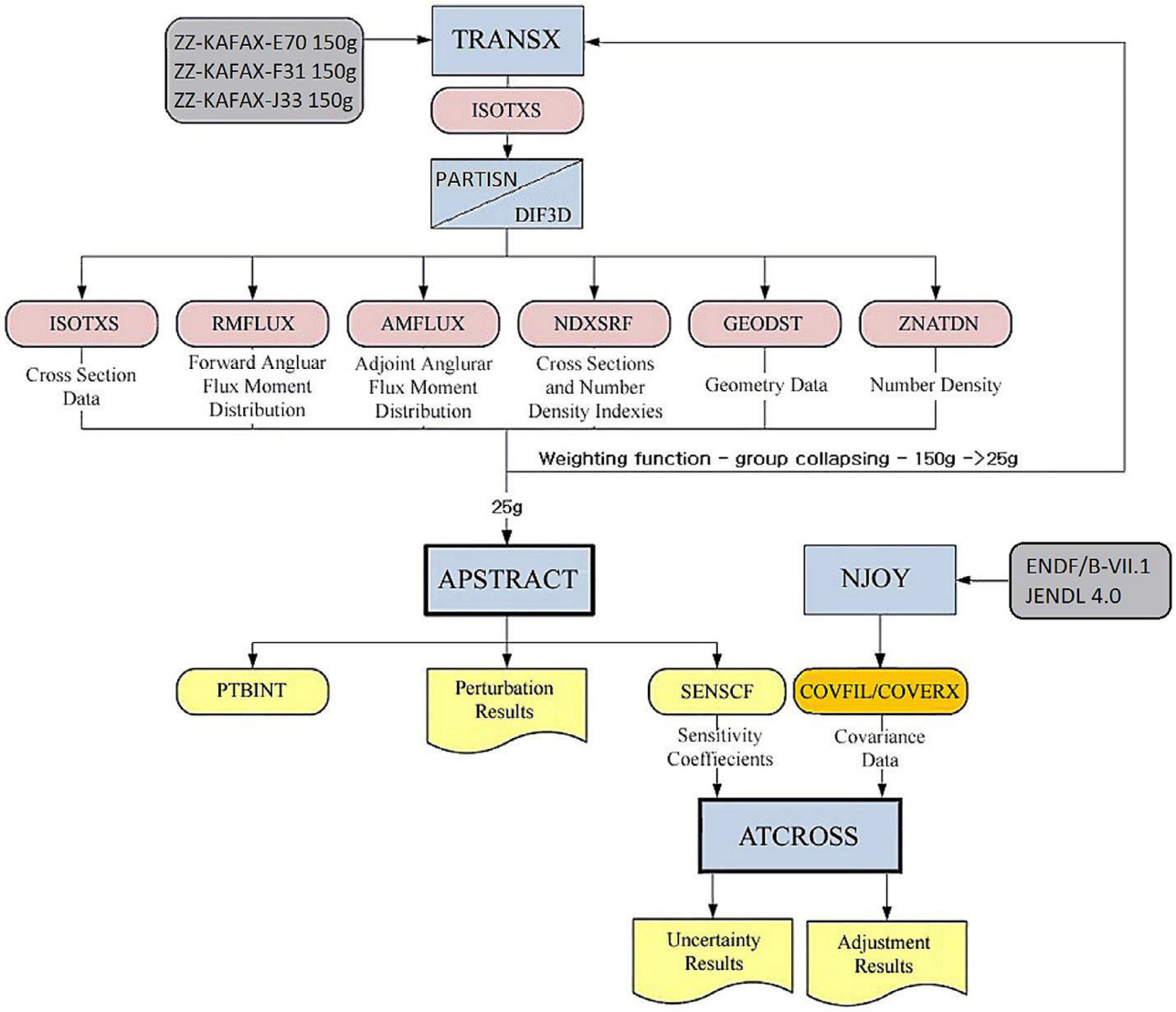

Fig. 1. Adopted calculation scheme.