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Estimation of resonance self-shielding effect in PWR pin cell for different fuel types

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ABSTRACT

The most important factors in any code used for nuclear reactor modeling are the time and precision of the results. In principle, deterministic codes are used in calculations due to the long computational time required by stochastic codes to achieve reliable results. The accuracy of deterministic codes is related to the method used in treating the resonance self-shielding behavior of the cross-sections; hence it is essential to select the best self-shielding method. Both the equivalence in dilution, and the subgroup approaches are used to represent the self-shielding effect in PWR pin cell. The calculations are performed using three classes of subgroup approaches based on physical and mathematical (Ribon extended model and subgroup projection method) probability tables. In the present paper, various cross-section library formats are considered to assess the performance of three PWR fuel types: UO_2, PuO_2–UO_2, and ThO_2–UO_2, using DRAGON4 and WIMS-D5 codes. The accuracy of the calculations is determined by comparison with the stochastic MCNP6 code. The results obtained indicate that the radial sub-divisions and proper resonance treatment are required to treat the rim effect. This is demonstrated in DRAGON code that can accurately calculate the radial distributions for all reaction rates.

1. Introduction

One of the regulatory body goals is to establish a set of computational tools to be used in the review and assessment process. These computational tools have been confirmed to give reliable results for the type of reactor under investigation, and users have sufficient training to use these codes adequately. These codes are applied to assess the points of concern related to safety such as neutronic core calculations for the nuclear reactor.

The best model is selected in such a way to achieve the best results in terms of precision and reasonable computation times. Generally the precision of the results is proportional to the time of the calculation scheme. Both stochastic and deterministic methods are used to calculate the design parameters (Cacuci, 2010). Deterministic codes are preferred to be used because stochastic codes takes long time in calculations (Reuss, 2008).

In any deterministic code, it is necessary to take into account the resonant behavior of the cross-sections within the energy-group where the self-shielding effect is the key element required to represent it. The main problem considered in the resonance self-shielding model is how to use self-shielded factor and probability table information, as retrieved from the cross-section data library (Hebert, 2004). Both equivalence in dilution, and subgroup techniques are implemented in most deterministic codes to represent the resonance self-shielding behavior.

In this paper; calculations are performed using the various self-shielding models. However, the mutual self-shielding due to resonance interference effects between different resonant isotopes and temperature gradient are not considered. It will be interesting to study such effects as future work.

The accuracy of the techniques used for representing the resonance self-shielding has been investigated using DRAGON4 (Marleau et al., 2013) and WIMS-D5 (NEA, 1997) codes, thereafter the results obtained are compared with those of the Monte Carlo MCNP6 code (Pelowitz, 2013).

Calculations are performed for three types of PWR fuel cells: the UO_2, Pu-MOX (PuO_2–UO_2), and Th-MOX (ThO_2–UO_2) fueled cells; each fuel rod is subdivided into six annular rings to study the spatially distributed self-shielding effects within the fuel rod. The pin cell model description is presented in Section 4. Different libraries are considered to study the sensitivity of fuel type to libraries. The resonance self-shielding models and the nuclear data libraries used in this paper are described in Sections 2 and 3 respectively.

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2. Theoretical models

In general, there are two methods for evaluating the neutronic behavior of the system. The first one is the stochastic method, where the Monte Carlo calculation is performed using point-wise cross-sections on the detailed geometry. This method is used in MCNP6 code and the results are the most accurate one but it takes a long computational time (Pelowitz, 2013).

However it is possible to combine different methods to achieve a calculation scheme which is both fast and accurate (Calic et al., 2011). In deterministic codes, such as DRAGON4 (Marleau et al., 2013), WIMS-D5 (NEA, 1997), and SRAC-2006 (Okumura et al., 2007), a resonance self-shielding calculation is required for solving the multi-group neutron transport equation, in order to take into account the resonant behavior of the cross-section in each energy group. The self-shielding models are broadly classified into two groups; equivalence in dilution and subgroup approach.

2.1. Equivalence in dilution

The basic objective is to find a homogeneous medium which has the same self-shielding properties as the actual, heterogeneous geometry. The self-shielding properties are represented by a so-called dilution (background) parameter (Choia and Lee, 2016). Subsequently, self-shielded cross-sections (effective cross-section) are determined from the dilution parameter and a pre-calculated table. Dilution dependent cross-sections must therefore be available in the cross-section data library. Consequently, the cross-section library is interpolated both in temperature and in dilution (Liu and Martin, 2017).

This approach is based on rational approximation of the fuel-to-fuel collision probability (CP). For infinite and homogeneous problems each self-shielded cross-section of each resonant isotope is tabulated against the dilution parameter in the cross-section data library. For heterogeneous problems a heterogeneous resonant situation is replaced with a linear combination of homogeneous resonant problems. In its simplest form this technique reduces to the use of Bell and Dancoff factors. The deterministic code WIMS-D5 is based on this class of model (NEA, 1997).

WIMS-D5 code is a deterministic and comprehensive general lattice cell code released from the OECD/NEA, 1997 data bank (Askew et al., 1966; NEA, 1991). It uses the transport theory to calculate detailed flux spectrum as a function of energy and position in cell. WIMS-D5 includes two main options that are most frequently used to solve the neutron transport equation; the collision probability method and the discrete ordinates method (Hebert, 2009; Bell and Glasstone, 1970).

Further improvements of this model have been proposed by Stamm'ler (Stamm'ler and Abbate, 1983) and later by Hebert and Marleau in DRAGON code (SHIBA module) (Hebert and Marleau, 1991) by increasing the number of terms in the rational expansions, it is known as a generalized Stamm'ler model (GSM). Moreover, the self-shielded averaged neutron fluxes in heterogeneous geometries have been modified using the Livolant and Jeanpierre normalization factorization (LJ) (Hebert, 2009).

The generalized Stamm'ler method was found to be unable to represent the distributed self-shielding effects in a fuel rod (Hebert, 2004). In order to improve the resonance treatment based on the generalized Stamm'ler method, it was proposed to introduce the Nordheim approximation, to gain distributed self-shielding capabilities (GSM+1). The Nordheim approximation (Nordheim, 1961) is used to uncouple the various resonant regions present in the domain and to represent distributed self-shielding effects. The difference from the generalized Stamm'ler model is that the dilution parameter is found for each self-shielding region compared to single value for the earlier model (more details about Nordheim approximation can be found in Ref. (Nordheim, 1961 and Hebert, 2004)). Fuel-to-fuel reduced collision probability for resonant regions is not a single value as was previously considered, but a square matrix (Greene et al., 2006).

Table 1
Geometrical description of pin-cell model used in calculations.

Description	cm
Pellet Outer Diameter	0.41265
Clad Inner Diameter	0.41655
Clad Outer Diameter	0.47435
Pin Cell Pitch	1.2620

Table 2
Isotopic composition of fuel materials used in pin-cell model.

Material	Isotopic Composition
UO_2 fuel	4.9% U-235-enrichment
Pu-MOX fuel	9.7% Pu-enrichment
Th-MOX fuel	73.6% Th-232, and 20.2% U-235 enrichment

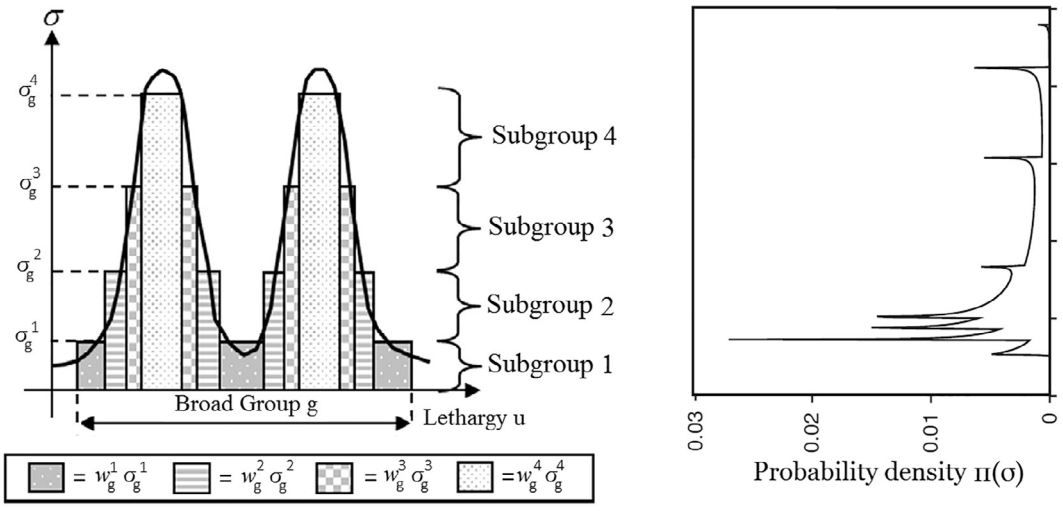

Fig. 1. The subgroup levels and their weights (left), and the probability density (right).