Methodology to resolve the transport equation with the discrete ordinates code TORT into KRITZ reactor

A. Bernal[a], A. Abarca[a], R. Miró[a] , T. Barrachina[a], G. Verdú[a]

[a]Institute for Industrial, Radiophysical and Environmental Safety (ISIRYM)
Universitat Politècnica de València - UPV
Camí de Vera s/n, 46022 Valencia
abernal@iqn.upv.es, aabarca@isirym.upv.es, rmiro@iqn.upv.es, tbarrachina@iqn.upv.es, gverdu@iqn.upv.es

Abstract

The resolution of the steady state Neutron Transport Equation in a pool nuclear reactor is usually achieved by means of two different numerical methods: Monte Carlo (stochastic) and Discrete Ordinates (deterministic). Both methods have been extensively tested in the nuclear field and proved as a powerful tool for research, analysis, design, and as well as validation and verification of other codes and experimental results.

Discrete Ordinates method resolves the Transport Equation for a set of selected directions (quadrature sets), obtaining a set of directional equations and solutions for each equation which are the angular flux. The final solution will be the weighted sum of all the directional solutions. In order to deal with the energy dependence, an energy multi-group approximation is commonly performed, obtaining a set of equations depending on the number of energy groups. In this case, the final solution will be the sum of the solutions for each group. With these two discretizations, energy and direction, special cross-sections (interaction probabilities) are needed. On one hand, the production (fission) and total cross-sections are needed in terms of group energy cross-sections, but there is not directional dependence due to the isotropic behavior of them. On the other hand, the scattering cross-sections must be specified by energy and by direction, due to the double differential scattering cross-section. Therefore, the Legendre Expansion can be used to define the directional dependence; thus, scattering cross-section must be defined for each energy group and term of the Legendre Expansion.

Using the deterministic code TORT, the problem is solved for each direction, energy group and zone of the space. However, it requires a binary file that contents the cross-sections defined as it is mentioned before. The methodology implies the generation of the cross-section in ANISN format at the desire temperature, and the code GIP produces the required binary cross-section file for TORT. To obtain the multi-group nuclear cross-sections data, in the fast and thermal energy range, at the temperature of interest, a methodology has been developed based on the capabilities of NJOY99 code. The evaluated nuclear data files of ENDF/B-VII libraries have been used as a started point.