USE OF THE APOLLO-II MULTIGROUP TRANSPORT CODE FOR CRITICALITY CALCULATIONS

by

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I. INTRODUCTION

APOLLO-II is a new-generation multigroup transport code for assembly calculation[1]. The code has been designed to be used as a tool for reactor design as well as for the analysis and interpretation of small nuclear facilities. As the first step in a criticality calculation, the collision probability module of the APOLLO-II code can be used to generate cell or assembly homogenized reaction-rate preserving cross sections that account for self-shielding effects as well as for the fine-energy within cell flux spectral variations. These cross section data can then be used either directly within the APOLLO-II code in a direct discrete ordinate multigroup transport calculation of a small nuclear facility or, more generally, be formatted by a post-processing module to be used by the multigroup diffusion code CRONOS-II[2] or by the multigroup MonteCarlo code TRIMARAN[3]. The standard APOLLIB-II library has a 99 group grid with 47 thermal groups. It contains multigroup data compiled from ENDF and JEF for over 400 isotopes, and pretabulated infinite-medium reaction rate data for selfshielding calculations for a large number of resonant isotopes. This paper discusses two of the main physical models that are used in the above calculational scheme. Namely, the self-shielding models that are used to account for fast flux fluctuations in the resonance energy domain, and the leakage model that is used to represent the macroscopic flux gradient due to the assembly environment. Geometrical approximations that are used to represent cell and assembly configurations, and the reaction-rate preserving homogenization and group collapsing procedures are discussed

elsewhere[1]. As an example calculation, Figure 1 shows the flux spectrum in the Hafnium filter of the SCARABEE BF2 experiment that was obtained using a cylindrical modelization.

II. SELF SHIELDING PROCEDURES

Geometry-dependent multigroup selfshielded cross sections are determined from a reaction-rate preserving equivalence procedure for a fixed-source slowing down problem that, thanks to the modular structure of the code, can incorporate as much geometrical details as necessary. A flat-flux collision probability approximation is used to blend geometrical effects into a matrix formulation of the slowing-down problem:

$$V \Sigma \Phi = P V (R_0\Phi + S) , \qquad (1)$$

where P = collision probability matrix, R_0 = slowing down operator for the resonant isotopes, and **S** = fixed slowing down source. The crucial step in this procedure is the determination of the group-dependent effective resonance integral for each resonant isotope present in the configuration. Symbolically one can write

$$\tau^g_{eff} = \int_{(g)} \Sigma_{a0}(u)\ \Phi(u)\ du = \int_{(g)} \Sigma_{a0}\ A(1-R_0A)^{-1} S\ du, \qquad (2)$$

where $A = (V\Sigma)^{-1}PV$. The integration is carried out numerically and requires the calculation of the collision matrix for several values of the resonant cross section Σ_0.

Besides the three classical models (NR, WR and IR approximations) used for this calculation, a fourth statistical NR approximation has been recently made available[4]. In this approximation the resonant slowing down contribution at a lethargy u is replaced by the group average

$$(R_0\Phi)(u) = \frac{1}{\Delta u_g} \int_{(g)} \Sigma_{s0}(u)\ \Phi(u)\ du\ . \qquad (3)$$

III. LEAKAGE MODELS

To obtain the best cross section data for a transport or diffusion calculation of an optically very thick domain, such as that of a reactor core, one need to determine accurately the energy and spatial flux dependence within the assemblies and at their interfaces. In practice such calculation is replaced by a number of transport calculations for small parts of the core (cells, assemblies, sets of assemblies) in which the action of the environment is represented by a leakage model. In the standard volumetric-leakage model of the APOLLO-II code, leakage due to the environment is calculated in an equivalent infinite medium obtained by flux-homogenization of the assembly. This leakage is then introduced in the heterogeneous assembly calculation under the form of an artificial, spatially uniform absorption cross section. Two new models have been recently developed to overcome the main drawbacks of the simple volumetric model. In the surface-leakage model, the artificial volumetric leakage is introduced as an uniformly incoming isotropic angular flux. However, the spectral shape of this flux is taken to reproduce the energy dependence of the infinite-medium leakage term. A third model, called TIBERE, accounts for anisotropic leakage effects in regular lattices. This model, which is being currently used for the analysis of the EPICURE experiment of water voiding in the central region of the EOLE LWR experimental facility, is based in a low-order approximation of the 'B_1 heterogeneous formalism.' In this formalism the flux in a periodical lattice is factorized into the product of a fast-varying local flux times a macroscopic curvature term that accounts for flux variation in a larger scale:

$$\varphi(\mathbf{r},\Omega,E) \approx \text{Real}\left[\psi(\mathbf{r},\Omega,E)\, e^{i\mathbf{B}\cdot\mathbf{r}}\right]. \qquad (4)$$

This gives two couple transport equations for the real and imaginary components of the local flux $\psi(\mathbf{r},\Omega,E)$. The solutions of this system can be used for accurate evaluations of reaction rates and spatially dependent directional leakage rates. In TIBERE such a complicated solution is avoided with two approximations. First one neglects angular correlation terms. For linearly anisotropic scattering this gives the system of equations:

$$(\Omega\cdot\nabla + \Sigma)\,\psi_s = \Sigma_0\varphi_s - \sum_k B_k^2\,\Omega_k\,\psi_{ak},$$

$$(\Omega\cdot\nabla + \Sigma)\,\psi_{ak} = \Sigma_1\Omega_k\,j_{akk} + \Omega_k\,\psi_s, \qquad (5)$$

where

$$\psi(\mathbf{r},\Omega,E) = \psi_s(\mathbf{r},\Omega,E) - i\sum_k B_k\psi_{ak}(\mathbf{r},\Omega,E), \qquad (6)$$

and where Σ_0 and Σ_1 represent respectively the isotropic and anisotropic transfer operators, φ_s is the real flux and j_{akk} is the current due to the angular flux ψ_{ak}. These equations are further simplified by replacing $\psi_{ak} = 3\,j_{akk}\,\Omega_k$ in the first equation, and $\psi_s = \varphi_s/(4\pi)$ in the second equation. Since the resulting equations represent a low-order B_1 approximation, a multiplicative corrective term is introduced in the second equation to account for the correct Fick's law in the homogeneous limit[5]. The resultant formalism requires only the calculation of classical and directional collision probability matrices. The interest of this leakage model is to provide for low cost spatially and directional dependent leakage consistent corrections that are of extreme importance in accidental core situations.

References

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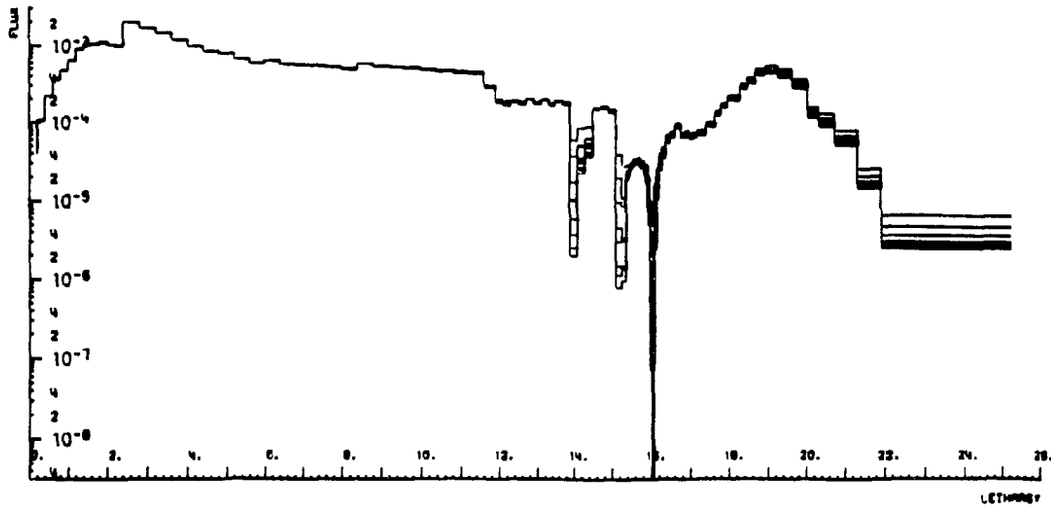
Figure Caption

Figure 1: Flux spectra in the Hafnium filter of the SCARABEE BF2 experiment. This filter is placed between the external thermal feeding region and the internal fuel molten region.


FLUX
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4
10^{-7}
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10^{-8}
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