THE SELF-SHIELDING MODELS USED IN THE NJOY NUCLEAR DATA PROCESSING SYSTEM

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ABSTRACT

In this paper, the Bondarenko and the flux calculator models used in the NJOY nuclear data processing system to take into account the resonance self-shielding effects are discussed. While the first model is recommended for weighting moderator and structural materials cross sections, the second one is indicated for averaging fission product and actinide cross sections. It is shown for an infinite value of the background cross section both models are governed by the same basic equation.

1. INTRODUCTION

Neutron transport calculations can be performed using stochastic methods together with energy-dependent (pointwise) nuclear data or deterministic methods with multigroup nuclear data. In deterministic methods, a guess for the neutron flux shape, generally known as weighting function, is used to calculate multigroup nuclear data and then the transport equation is solved to obtain the actual neutron flux.

Nowadays, considering the criteria of versatility and completeness [1], the NJOY nuclear data processing system [2,3] is the only one option to generate nuclear data libraries for practical applications. This system can produce pointwise and multigroup nuclear data from a library in ENDF/B format [4,5] and it is composed by a set of twenty three modules, or computer codes, each one performing a well defined nuclear data transformation. The self-shielded multigroup cross sections and group-to-group transfer matrices are generated in the GROUPR module. This module has one pointwise user’s specified and nine built-in weighting function options. However, if resonance absorbers are present in a specific case, self-shielding effects will occur producing a complex change in the weighting function shape. This module utilizes the Bondarenko [2] or the flux calculator [2] models, discussed in next sections, to take into account these effects.

2. THE BONDARENKO MODEL

In the Bondarenko model, which utilizes the narrow resonance approach, the weighting flux is obtained by

$$\Phi(E) = \frac{\sigma_0 C(E)}{\sigma_0 + \sigma_r(E)},$$  \hspace{1cm} (1)
where: \( \sigma_0 \) is the background cross section, which represents a mixture of materials; \( \sigma_t(E) \) is the total microscopic cross section for the material; and \( C(E) \) is a smooth function of energy. If the \( \sigma_0 \) value is larger than the peaks in \( \sigma_t(E) \), the weighting flux \( \Phi(E) \) is approximately proportional to the smooth weighting function \( C(E) \). This situation is called “infinite dilution” and the material cross section, \( \sigma_t(E) \), has little or no effect on the flux. On the other hand, if the \( \sigma_0 \) value is smaller than \( \sigma_t(E) \), the weighting flux will have large dips at the locations where \( \sigma_t(E) \) has peaks and a large self-shielding effect will be expected.

This model is applied for practical fast reactor cases. However, for nuclear systems sensitive to energies from 1 eV to 500 eV, there are many broad and intermediate-width resonance which can not be self-shielded with sufficient accuracy using the Bondarenko model. The flux calculator model was developed to improve the weighting flux calculation in that energy range.

### 3. THE FLUX CALCULATOR MODEL

In the flux calculator model, the weighting flux is obtained solving the integral slowing-down equation

\[
[\sigma_0 + \sigma_t(E)]\Phi(E) = \sigma_0 C(E) + \int_{E}^{E/\alpha} \frac{\sigma_s(E')\Phi(E')dE'}{(1-\alpha)E'},
\]

where \( \sigma_s(E) \) is the elastic scattering microscopic cross sections for the material,

\[
\alpha = \left( \frac{\text{AWR} - 1}{\text{AWR} + 1} \right)^2
\]

and AWR is the atomic weight ratio. It is important to mention that fission source and thermal up-scatter effects can be included in the smooth weighting function \( C(E) \).

### 4. RESULTS AND COMMENTS

As a sample case, consider a nuclear data generation in one energy group for the material \( ^{238}_{92}\text{U} \) using the NJOY nuclear data processing system and the version VI of ENDF/B. The \( ^{238}_{92}\text{U} \) total, elastic scattering and radiative capture cross sections for the neutron energy from 1 eV to 500 eV are presented in Fig. 1. The same cross sections for the energy range near 6.67 eV are shown in Fig. 2.
Figure 1. \(^{238}_{92}\)U total, elastic scattering and radiative capture cross sections from 1 to 500 eV.

Figure 2. \(^{238}_{92}\)U total, elastic scattering and radiative capture cross sections near 6.67 eV.
The “EPRI-CELL LWR” built-in weighting function was selected as the $C(E)$ function and the energy group covers the range from $1.0 \times 10^{-05}$ eV to $2.0 \times 10^{07}$ eV. Please note the main interest here is in the weighting flux calculation and these boundaries cover the entire neutron nuclear data energy range. The weighting function $C(E)$ is presented in Fig. 3.

![Image of C(E) weighting function for the entire energy range.](image)

Figure 3. $C(E)$ weighting function for the entire energy range.

The fluxes calculated with the Bondarenko, $B(E)$, and the flux calculator, $F(E)$, models using a background cross section, $\sigma_0$, of 50 barns/atom for the entire energy range have 44382 and 43838 energy points, respectively, and these fluxes for the energy range from 1 eV to 500 eV are shown in Fig. 4. Fig. 5 presents the same fluxes for the energy range near the $6.67$ eV $^{238}$U absorption resonance. As can be seen from these figures, the weighting flux calculated with the Bondarenko model is quite similar to that obtained with the flux calculator model and, consequently, large differences in the generated multigroup cross sections are not expected.

Finally, attributing an infinite value, $1.0 \times 10^{10}$ barns/atom, to the background cross section, $\sigma_0$, the right hand side of Eq. (2.a) reduces to $\sigma_0 C(E)$ and Eqs. (1) and (2a) became the same basic equation. Clearly, in this situation, the weighting function is described by

$$\Phi(E) = C(E).$$

(3)
Figure 4. $B(E)$ (+) and $F(E)$ (□) fluxes for the energy range from 1 eV to 500 eV.

Figure 5. $B(E)$ (+) and $F(E)$ (□) fluxes for the energy range near 6.67 eV.
5. FINAL COMMENT

If the considered case is sensitive to energies from some eVs to hundred of eVs; it has fission product or actinide materials; and it presents a finite value for the background cross section, then the flux calculator model should be utilized. Otherwise, the Bondarenko model should be employed without loss of quality in the generated multigroup cross sections.

REFERENCES

5. International Atomic Energy Agency Nuclear Data Services. [www-nds.iaea.org](http://www-nds.iaea.org) or [www-nds.ipen.br](http://www-nds.ipen.br).