ON WEIGHTING MULTIGROUP CROSS SECTIONS

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Media in reactors contain in general many heavy resonant nuclides. To prepare multigroup cross sections for these nuclides, we often, consider an infinite homogeneous medium containing all these nuclides in order to take into account resonance self-shielding and mutual shielding effects. Following the Livolant-Jeanpierre formalism (1), we first determine the fine flux structure by solving the neutron slowing down equation in an infinite homogeneous medium for point wise neutron energy, then we use the obtained fine flux as a weighting function to evaluate multigroup cross sections. Handling such calculation is simple for few resonant nuclides, but it becomes a tedious task fore a large number of resonant nuclides. That requires a huge computer capacity and a large computational time.

In the present work, we propose a simple model to do these calculations with a large reduction of both calculation time and necessary computing time. Basically only one fictitious heavy resonant nuclide "Equivalent Nuclide" is used in the model, to do the calculations of the appropriate weighting function, instead of all the actual resonant nuclides however is their number. The "Equivalent Nuclide" characteristics are directly deduced from those of the actual ones.

The iterative method (2) is used to solve the neutron slowing down equation in order to determine the fine flux which is used as a weighting function for the multigroup cross sections. The model validation has been done for both continuous neutron energy representation and multigroup approximation in the [2.77eV - 676.45eV] and [1.0E-5eV - 2MeV] energy intervals . We have found that our model conserves the reaction rates with a high accuracy and saves more than 82% of the computation time in cases of five heavy resonant nuclides. That leads also to a huge reduction of the necessary computer memory to do the same calculations.

References

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2) A. JEHOUANI et al., Annals of Nuclear Energy 19, 195 (1992).

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