DEVELOPING NEUTRONICS CALCULATION TOOLS FOR MYRRHA

Background

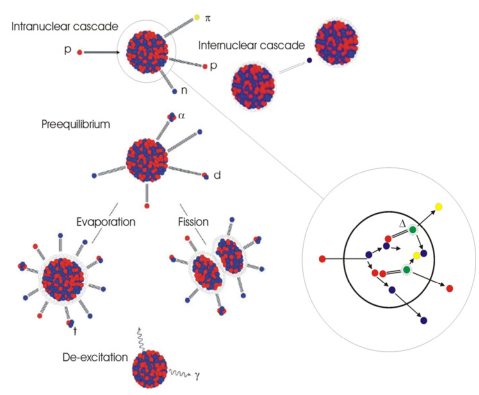
The design of the Accelerator Driven System MYRRHA requires adequate and specialised tools in the field of neutronics calculations. In order to "fill the gaps", several PhD programmes were launched. In 2005 three such PhD projects were running. Each of them focuses on different stages in the computation of a core of MYRRHA. The first project *"Improvements of the spallation reaction model"*, a collaboration with the University of Liège, deals with the characterisation of the spallation neutron source using the INCL (Intra-Nuclear Cascade of Liège) model. Since at high energies, nuclear data are sparse, calculations rely on models. Especially for spallation reactions that occur at proton energies of several hundreds of MeV, models are the only means to evaluate the spallation source in MYRRHA. The second project *"Neutron transport with anisotropic scattering"*, a collaboration with the Université Libre de Bruxelles, works on the development of a neutronics code, CASE-BSM, for systems with highly anisotropic scattering. The presence in large amounts of both lead and bismuth atoms in the MYRRHA core results in a highly anisotropic scattering of the neutrons in the bulk of the coolant. Neglecting this effect has large consequences on both global parameters, like k_{eff}, as well as on local parameters, like the neutron flux seen by the vessel. The third project, *"ALEPH: An integrated Monte Carlo bun-up tool"*, a collaboration with Ghent University, treats the last phase of a core calculation: the depletion of the fuel during irradiation. For an experimental machine like MYRRHA it is of utmost importance to have a fast calculational tool to evaluate the incineration of both isotopes present in the fuel as isotopes present in experimental devices.

Objectives

The main objective is to improve the current quality of the neutronics codes focused on ADS applications and to have this knowledge "in-house".

Principal results

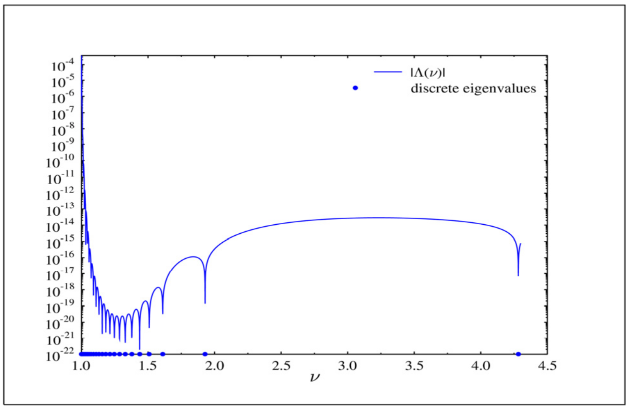
The intranuclear cascade model of Liège (INCL) consists of the tracking in space and time of particles inside the nucleus using Monte Carlo techniques. Given the energy range of validity of this reaction model (200 MeV to 3 GeV), the particles tracked are neutrons, protons, deltas and pions.



Spallation event and intra-nuclear cascade

In the INCL model, nucleons are assumed to move inside a nuclear mean field described by a potential well of constant depth, of about 45 to 50 MeV. However it has been recognised since a long time that the nuclear mean field depends on the nucleon (isospin) and varies with its energy. A better description of the mean field of the nucleons and of the delta-particles is now implemented in the model. Above 300 MeV, pions are progressively produced and can play a non-negligible role to accurately describe spallation reactions. The description of the physics of the pions as implemented in the INCL model is improved by the introduction of an optimal pion mean field and from then reflection or refraction of pions on the nuclear surface and the implementation of pion-nucleon cross section around the delta-resonance.

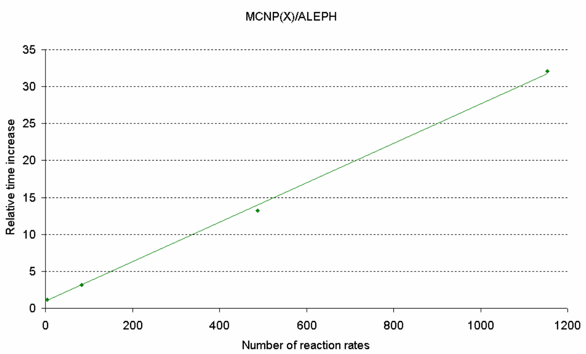
CASE-BSM is a one-dimensional neutron transport module based on the Boundary Sources Method that allows for arbitrary order anisotropic scattering. One can say that the theory for neutron transport with anisotropic scattering really started in 1961 with a paper by Janusz Mika. He clearly identified the duality of the spectrum of the neutron transport operator: the discrete spectrum and the continuum spectrum. In the case of anisotropic scattering there can be a large number of discrete eigenvalues. We have developed a methodology to find all - and this can be guaranteed by the code - eigenvalues for any type of anisotropic scattering. For the continuum spectrum, we have set up a computational framework to evaluate this, based on numerical quadrature.



The discrete eigenvalues are the solutions of the non-linear characteristic equation Λ(ν)=0 (note the logarithmic scale).

All this has been combined in a one-dimensional neutron transport code, CASE-BSM. We found that our code can stand next to the major known benchmark codes when comparing the number of correct digits on neutron fluxes, in- and out-going currents and critical widths for low anisotropic orders. For high anisotropic scattering orders, we believe our code to outperform both on speed as on accuracy the current available codes. This PhD work has been successfully defended on May 12th, 2005 at the Université Libre de Bruxelles.

ALEPH is a C++ interface code between NJOY 99.112, ORIGEN 2.2 and any version of MCNP or MCNPX. Through a deeper understanding of the Monte Carlo game, we have developed an optimum approach to reaction rate calculation which we have called the *multi-group binning approach* where the Monte Carlo code provides a very fine multi-group spectrum that is used to calculate the reaction rates outside the Monte Carlo simulation.



Calculation time increase of MCNP(X) compared to ALEPH for calculating reaction rates in an infinite MOX fuel

This new approach allows us to reduce the calculation time of every burn-up step to that of the basic Monte Carlo simulation process. A time reduction factor up to 900 is shown to be achievable. ALEPH has already been applied to fuel cycle studies and full 3D reactor calculations in the MYRRHA project. The nuclear data for both ALEPH and MCNP(X) is being handled by ALEPH-DLG (Data Library Generator), a utility code to ALEPH. ALEPH-DLG automates the entire process of generating library files with NJOY and verifies the processing

Future work

At intermediate energy (20 MeV to 200 MeV) most theoretical tools are based on various pre-equilibrium models for which the theoretical justification has not yet been provided. Since the INCL model is giving satisfactory results at 100 MeV incident energy, theoretical approaches will be compared with INCL. Finally the previous improvements and comparisons will be applied to the design of spallation target.

Now that the methodology in CASE-BSM has been validated, the module will be included in HEXNODYN, a time-dependent neutron transport code under development at ULB and SCK•CEN. Meanwhile, we are constantly improving our code, both on speed and accuracy.

For now, ALEPH only recalculates the reaction rates for the burn-up calculation. In future versions of ALEPH we will also calculate spectral averaged fission yields and branching ratios for isomer production. The development of ALEPH-DLG will continue in the framework of the NEA-OECD JEFF Nuclear Data Project.

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Main references

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