Reactor Physics Group Activities

Valerio Giusti

Università di Pisa
Dipartimento di Ingegneria Civile ed Industriale

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Main Research Activities

- Development of computational codes for nuclear reactor core calculations using a Boundary Element – Response Matrix approach.
- Research on the application of neutrons in medicine. In particular as it concerns the Boron Neutron Capture Therapy (BNCT) for the treatment of malignant tumours.
- Stochastic computational codes:
  - comparison between MCNP6 and Geant4 applied to the transport of low energy (<200 MeV) neutrons and protons;
  - study of the neutronic performance of the nuclear reactor concepts currently considered for the Generation IV systems.

Development of codes - diffusion approximation

The boundary element method reduces the multigroup diffusion equation to an integral equation in terms of the boundary values of the flux and its derivative.

$$D_g \nabla_{\mathbf{r}}^2 \phi_g(\mathbf{r}) + \sum_{g'=1}^{G} \left(\Sigma_{s,g' \to g} - \Sigma_{rg} \delta_{gg'} + \frac{1}{k} \chi_g \nu_{g'} \Sigma_{fg'} \right) \phi_g(\mathbf{r}) = 0$$

$$\Downarrow$$

$$\frac{1}{2} \phi_g(\mathbf{r}_s) + \sum_{g'=1}^{G} \int_s \left(\frac{\partial \tilde{\phi}_{gg'}}{\partial n'_s} (\mathbf{r}_s, \mathbf{r}'_s) \phi_{g'}(\mathbf{r}'_s) - \tilde{\phi}_{gg'}(\mathbf{r}_s, \mathbf{r}'_s) \frac{\partial \phi_{g'}}{\partial n'_s}(\mathbf{r}'_s) \right) dS' = 0$$

After the introduction of the partial currents, an iterative procedure based on the Response Matrix formalism is used to solve it.

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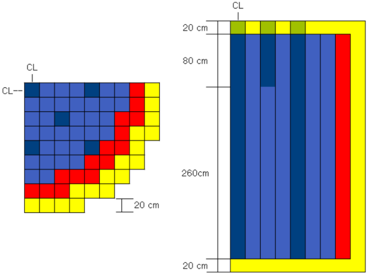
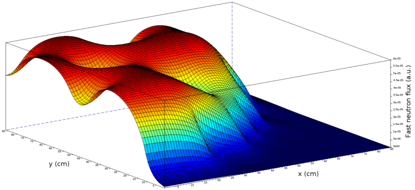
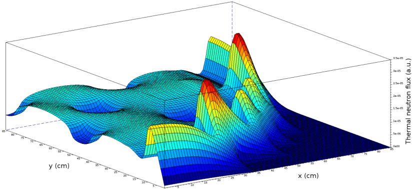
Development of codes - diffusion approximation

The IAEA 3-D PWR problem is an important 2-group benchmark to measure the performance of neutronics calculation methods. The core is made of 177 fuel assemblies and is surrounded by 64 reflector assemblies.







Node size (cm^3)	L	k	Δ_k (pcm)	Assembly power error Max (%)	Average (%)	CPU time (s)
$20\times20\times20$[a]	2	1.029160	-9.1	1.37	0.61	7.9
	3	1.029098	-2.9	0.64	0.27	15.5
	4	1.029076	-0.7	0.18	0.04	32.0
$10\times10\times20$[a]	2	1.029105	-3.6	0.71	0.32	15.9
	3	1.029075	-0.6	0.50	0.21	23.8
	4	1.029069	0.0	0.12	0.06	59.6

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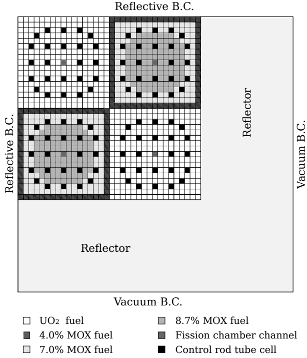
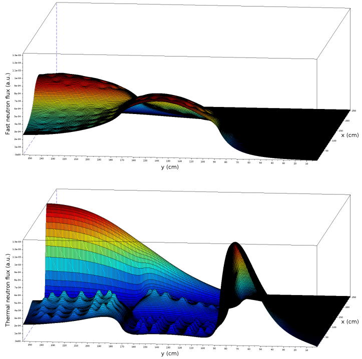
Development of codes - transport approximation

With the A_N method the integral transport equation is approximated by a system of equations analogous to the usual multigroup diffusion systems (with up-scattering).
The A_N approximation has been shown to be equivalent to SP_{2N-1} approximation.





Development of codes - work in progress

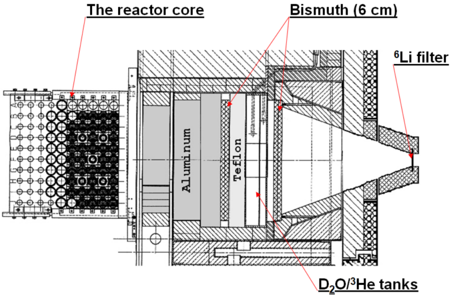
Currently the following works are in progress:

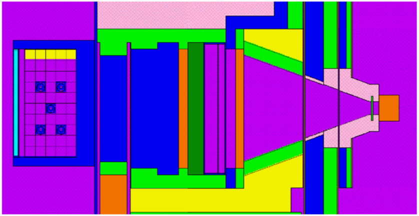
- development of a code to solve the A_N transport equation in hexagonal geometry (the geometry foreseen for the next generation of nuclear reactor, Gen. IV);
- development of a code which solve the A_N transport equation in case of presence of an external source (important for the ADS systems).

Neutron in medicine - BNCT

- Boron Neutron Capture Therapy is a binary treatment modality where the patient is first administered with a drug, containing ^{10}B, which suitably accumulates in the tumour tissue; the tumour region is then irradiated with neutrons of proper energy: $^{10}B + n \rightarrow ^7 Li + \alpha$
- Suitable neutron filter have to be designed in order to produce neutron beam with an energy spectrum enhanced in the epithermal energy region (i.e. from 1-10 eV up to 10-20 keV).

Neutron in medicine - The Studsvik BNCT facility





MCNP model

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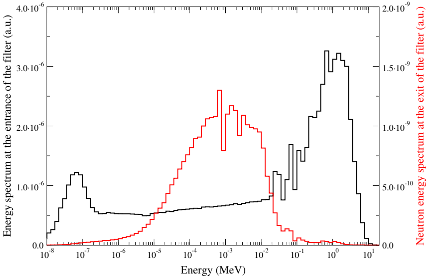
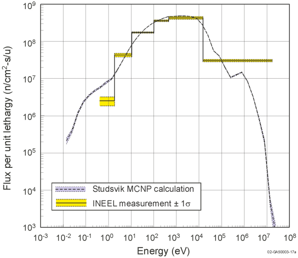
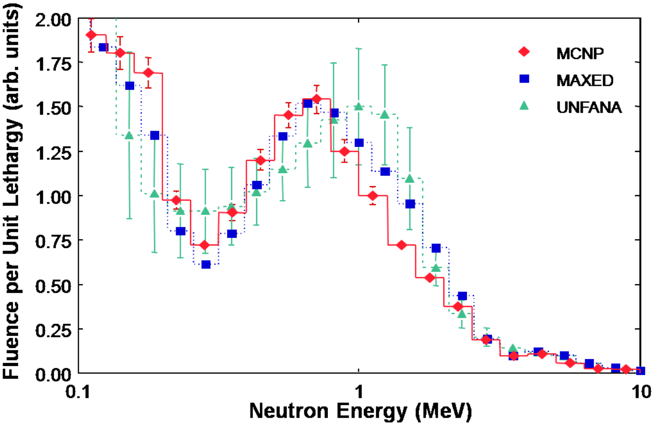
Neutron in medicine - The Studsvik BNCT facility





- At the entrance of the filter the neutron energy spectrum is dominated by the thermal and fission peaks while at the end of the filter it is right enhanced in the energy range of interest.
- The neutron spectrum at the exit matches quite well the measured values (data from INEEL)

Neutron in medicine - The Studsvik BNCT facility



Comparison of the measured and calculated fast (>100 keV) energy spectrum.

Stochastic computational codes

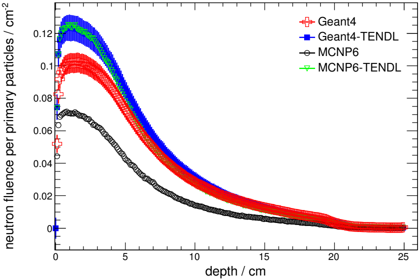
- The group has achieved a 10+ years experience on the use of the Monte Carlo code MCNP.
  This code has been extensively used to study different radiation problems as well as to benchmark the home made deterministic codes.

- Recently an activity aiming at the comparison between MCNP6 and Geant4 has been started in collaboration with the McGill University (Montreal, Canada).

Stochastic computational codes

- Interactions of charged particles with energy of the order of some MeV cannot be suitably reproduced by the physics models implemented in Geant4.
- As a consequence the production of secondary particles from proton-nucleus interactions are not accurate for energy lower than 20 MeV.
- The interest in the cyclotron for the production of medical radioisotopes and the increasing use of protons for therapy has led to the development of a new package in Geant4 which makes use of evaluated cross section databases to reproduce the interactions below 20 MeV.

Stochastic computational codes



Monte Carlo code	Models and Cross sections
Geant4	Proton: default models Neutron: G4NDL4.4 (ENDF/B-VII.1)
MCNP6	Proton: default models Neutron: default models >20 MeV ENDF-B/VII.1 <20 MeV
Geant4-TENDL	Proton: TENDL-2012 Neutron: G4NDL4.4 (ENDF/B-VII.1)
MCNP6-TENDL	Proton: TENDL-2012 Neutron: TENDL-2012

Stochastic computational codes

Recently the MCNP code has been used to study the fission power distribution under different operational condition within the core of the European Sodium fast Reactor (ESFR), with the aim of finding the most loaded fuel rod.