FUSION CAT

Validating NEUTRO: a deterministic neutron transport solver for fusion applications

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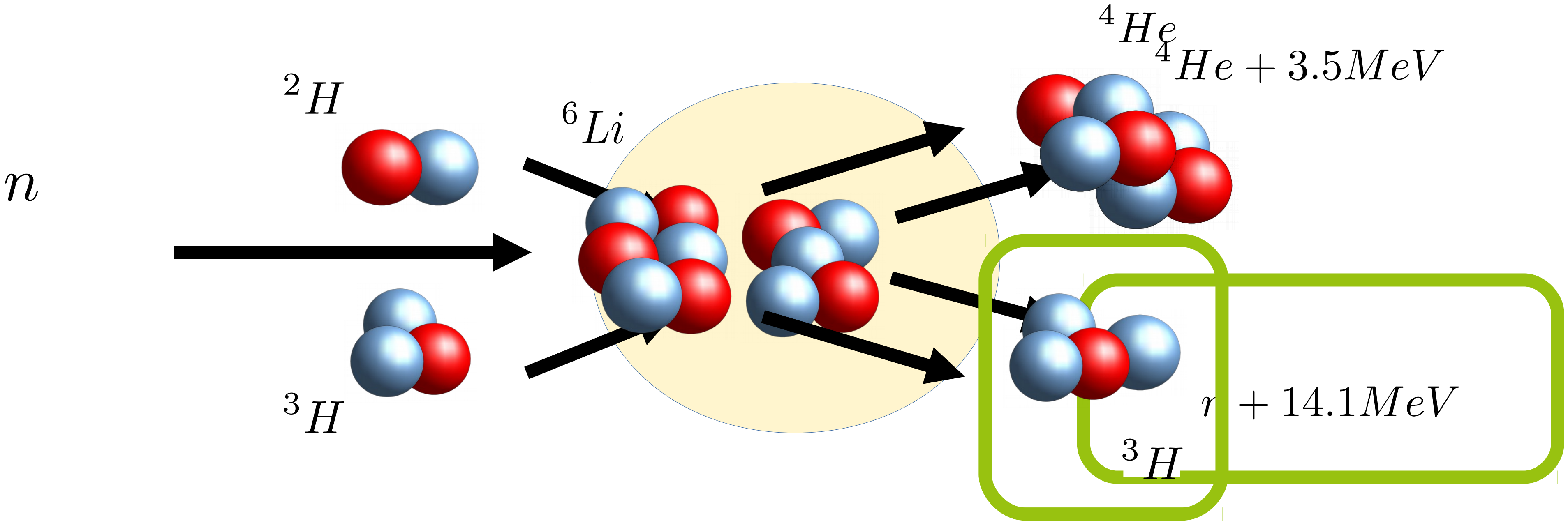
The FusionCAT project (001-P-001722) has been 50% co-financed with €1.960.963,66 bu the European Fund for Regional Development of the European Union within the framework of the 2014-2020 ERDF Operational Program of Catalonia, with the support of the Generalitat of Catalonia.

Outline

- Background
- NEUTRO
- Results
- HPC scalability
- Ongoing and future work
- Conclusions

Background

- Nuclear Fusion: D-T reaction produces high-energy neutrons.
- Tritium production (Breeding Blanket)



Key to achieve fuel self-sufficiency!

Background

- Nuclear Fusion: D-T reaction produces high-energy neutrons
- Tritium production (Breeding Blanket)

NEUTRO

Neutron transport solver coupled within Alya multiphysics framework to study:

- Shielding.
- Tritium production (Breeding Blanket).
- Heat
- Damage

- Deterministic
  - Less computationally intensive than statistical methods.
  - Global analysis.
  - Complements other approaches such as Monte Carlo.

NEUTRO

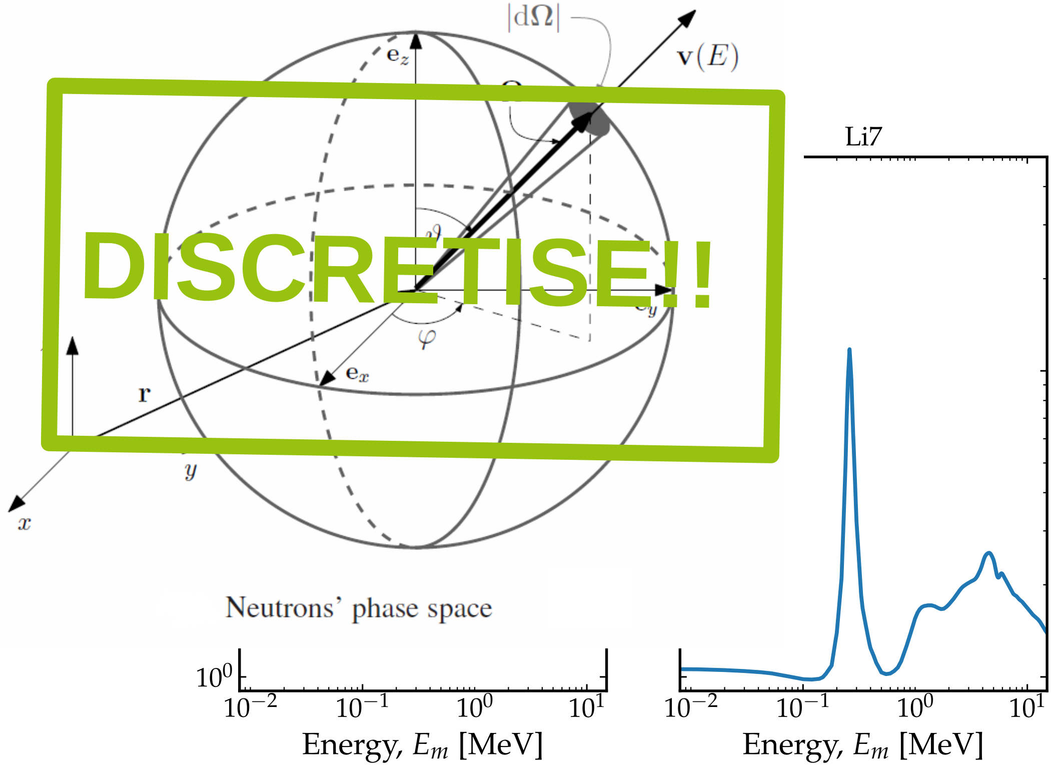
- Aims to solve the stationary Boltzmann neutron transport equation:

$$\underbrace{\underbrace{\widehat{\boldsymbol{\Omega}}\cdot\nabla\varphi}_{\text{Leakage}}+\underbrace{\Sigma_t(\boldsymbol{r},E,\widehat{\boldsymbol{\Omega}})\varphi}_{\text{Collision}}}_{\text{Loss terms}} = \underbrace{\underbrace{\int_0^\infty \int_{4\pi} \Sigma_s(\boldsymbol{r},E' \to E,\widehat{\boldsymbol{\Omega}}' \to \widehat{\boldsymbol{\Omega}})\varphi d\widehat{\boldsymbol{\Omega}}' dE'}_{\text{In-scattering}}+\underbrace{s(\boldsymbol{r},E,\widehat{\boldsymbol{\Omega}})}_{\text{Source}}}_{\text{Gain terms}}$$

NEUTRO

$$\widehat{\boldsymbol{\Omega}}\cdot\nabla\varphi+\Sigma_t(\boldsymbol{r},E,\widehat{\boldsymbol{\Omega}})\varphi=\int_0^\infty\int_{4\pi}\Sigma_s(\boldsymbol{r},E'\to E,\widehat{\boldsymbol{\Omega}}'\to\widehat{\boldsymbol{\Omega}})\varphi d\widehat{\boldsymbol{\Omega}}'dE'+s(\boldsymbol{r},E,\widehat{\boldsymbol{\Omega}})$$

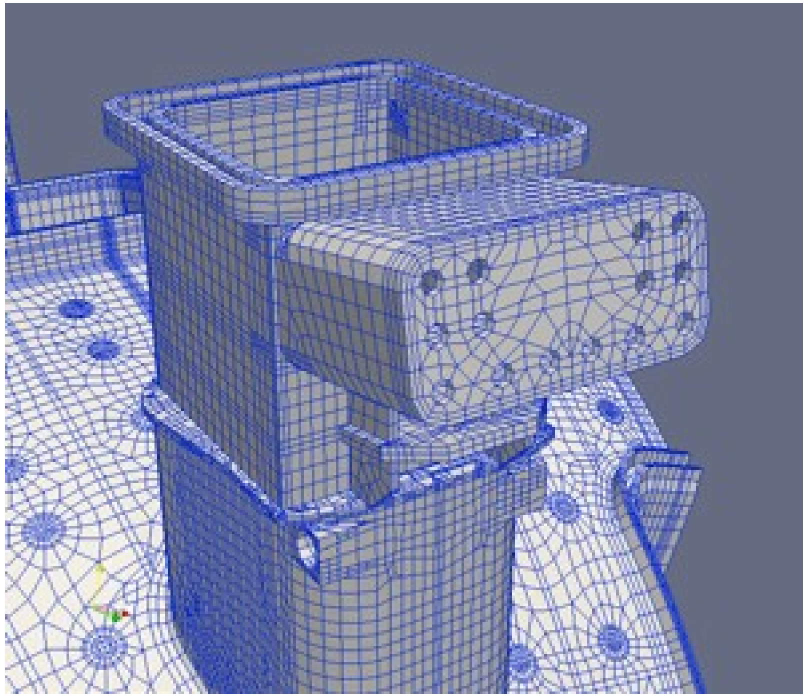
- Where:
  - $\widehat{\boldsymbol{\Omega}}$: neutron's direction of flight.
  - E : neutron's energy.
  - $\boldsymbol{r}$: spatial variable.
  - φ : distribution function of neutrons in $\boldsymbol{r}, E, \widehat{\boldsymbol{\Omega}}$.
  - Σ_t : total macroscopic cross-section.
  - Σ_s : scattering macroscopic cross-section.
  - s : external neutron source.



DISCRETISE!!

Neutrons' phase space

NEUTRO: Finite Element Method



- Spatial variable, $\boldsymbol{r}$, is discretised using FEM.
- Spatial meshes of a fusion reactor can easily reach $\sim 10^9$ elements.
- NEUTRO software is include within the **Alya** ecosystem.

Alya

- Solves partial differential equations (PDEs) in non-structured meshes.
- Designed for supercomputers (MareNostrum).
- Excellent scalability proven up to 100k processors.
- Implements parallel frameworks at different levels without user active intervention.
- Solves coupled multiphysics problems.

NEUTRO: Multigroup Approximation

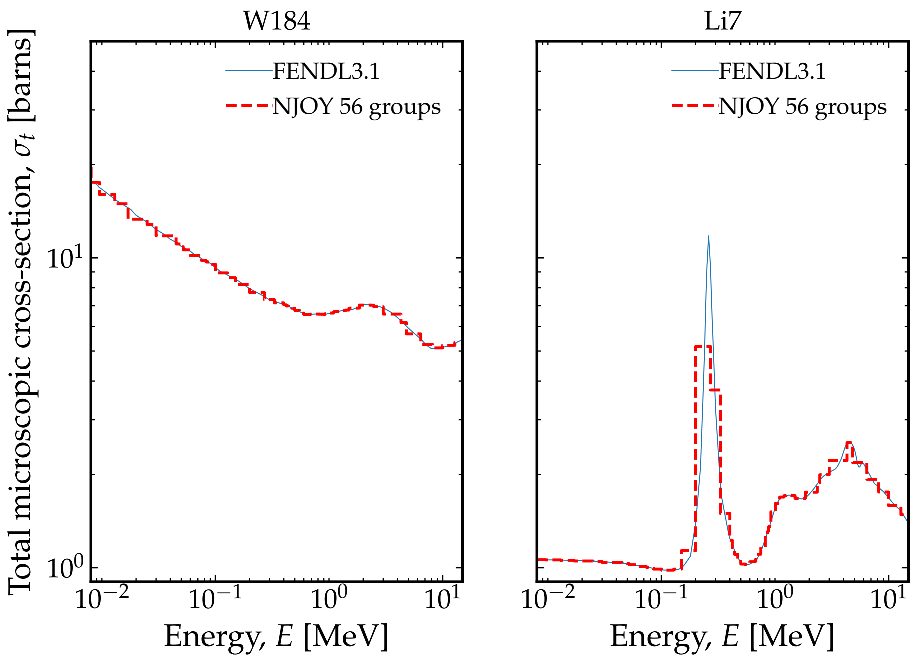
- Discretise the energy continuum into G energy groups: $E_1, E_2, \ldots, E_{G-1}, E_G$
- Properties of neutrons and materials remain constant within each interval of energy.

$$\widehat{\boldsymbol{\Omega}}\cdot\nabla\varphi+\Sigma_t(\boldsymbol{r},E,\widehat{\boldsymbol{\Omega}})\varphi = \int_0^\infty \int_{4\pi} \Sigma_s(\boldsymbol{r},E' \to$$

Evaluated Nuclear Data Files (ENDF)

NJOY

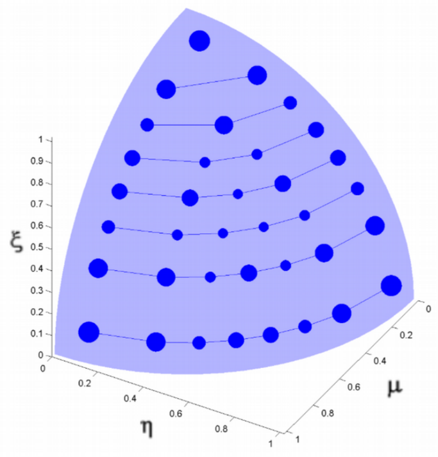
$$_g + \Sigma_{t\,g}(\boldsymbol{r},\widehat{\boldsymbol{\Omega}})\varphi_g = \sum_{g'=0}^{G} \int_{4\pi} \Sigma_{s\,g',g}($$



NEUTRO: Angular discretisation

- Discrete Ordinates Methods (S_N): equal-level set
- Divide neutron's phase space in quadrature points with adecuate weights.

$$\widehat{\boldsymbol{\Omega}} \cdot \nabla \varphi_g + \Sigma_{t\,g}(\boldsymbol{r}, \widehat{\boldsymbol{\Omega}})\varphi_g = \sum_{g'=0}^{G} \boxed{\int_{4\pi}} \Sigma_{s\,g',g}(\boldsymbol{r}, \widehat{\boldsymbol{\Omega}}' \to \widehat{\boldsymbol{\Omega}})\varphi_g \boxed{d\widehat{\boldsymbol{\Omega}}'} + s_g(\boldsymbol{r}, \widehat{\boldsymbol{\Omega}})$$



$$\Omega_m \cdot \nabla \varphi_g + \Sigma_{t\,gm}(\boldsymbol{r})\varphi_{gm} = \sum_{g'=0}^{G} \boxed{\sum_{m'=0}^{M}} \Sigma_{s\,g',gm',m}(\boldsymbol{r})\varphi_{g'm'} + s_{gm}(\boldsymbol{r})$$

NEUTRO: Material anisotropy

- Introduced in Σ_s using real-based spherical harmonics.
- $L > 0 \rightarrow$ angular anisotropy.

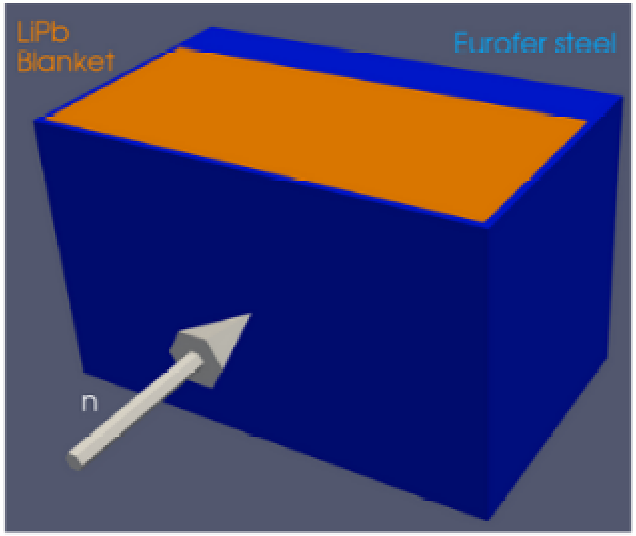
$$\Omega_m \cdot \nabla\varphi_g + \Sigma_{t\,gm}(\boldsymbol{r})\varphi_{gm} = \underbrace{\sum_{g'=0}^{G}\sum_{m'=0}^{M}\Sigma_{s\,g',gm',m}(\boldsymbol{r})}\varphi_{g'm'} + s_{gm}(\boldsymbol{r})$$

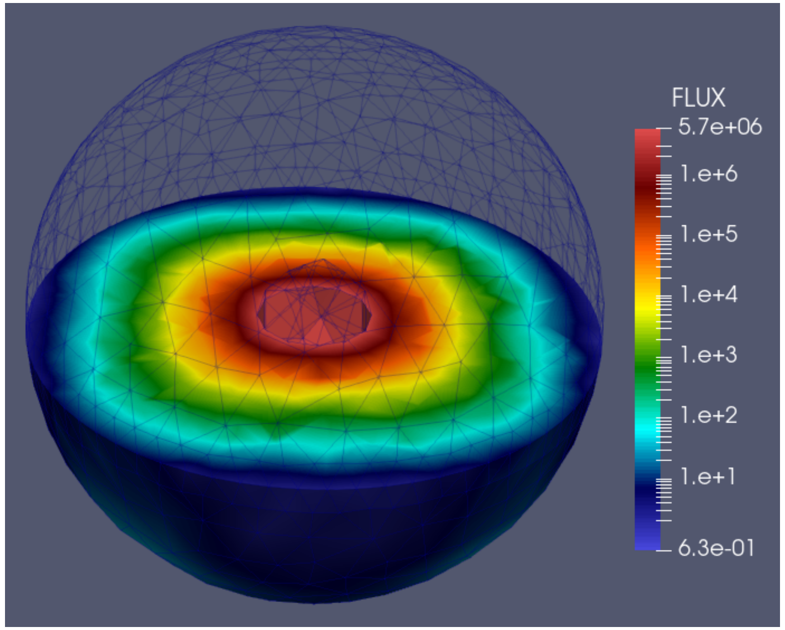
$$\Big\downarrow$$

$$\sum_{g'=1}^{G}\sum_{m'=1}^{M}\sum_{l=0}^{L}\sum_{ml=-l}^{l}\frac{4\pi}{2l+1}Y_l^{ml}(\Omega_m)Y_l^{*ml}(\Omega_{m'})\Sigma_{s_{mlg',gm',m}}(\boldsymbol{r})\varphi_{g',m'}(\boldsymbol{r})$$

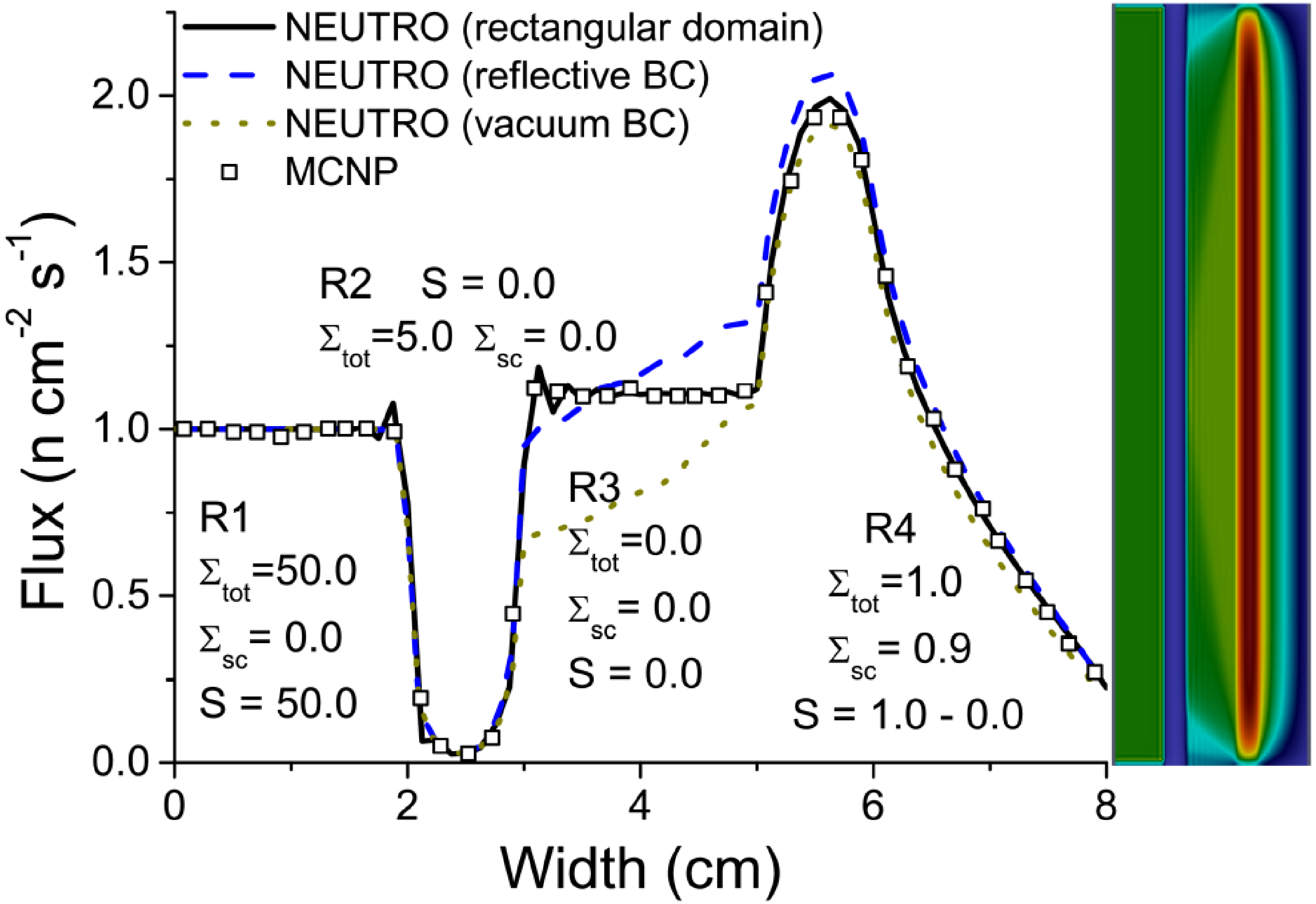
NEUTRO: Present Status

- 3D domains with unstructured meshes.
- Stationary simulations.
- Anisotropic scattering.
- Isotropic neutron source at boundaries.
- Multiple heterogeneous materials.
- Nuclear neutron-material heating.

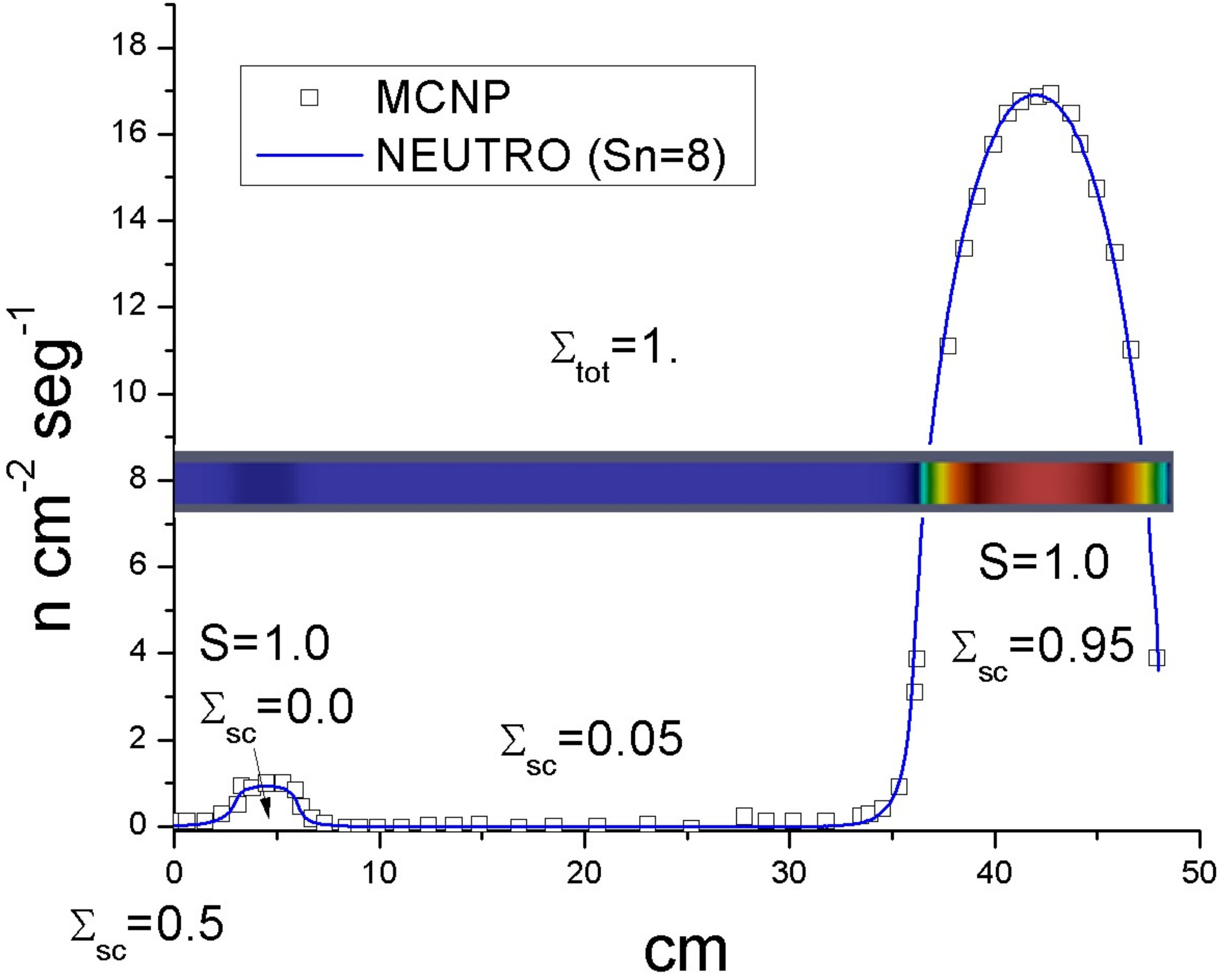




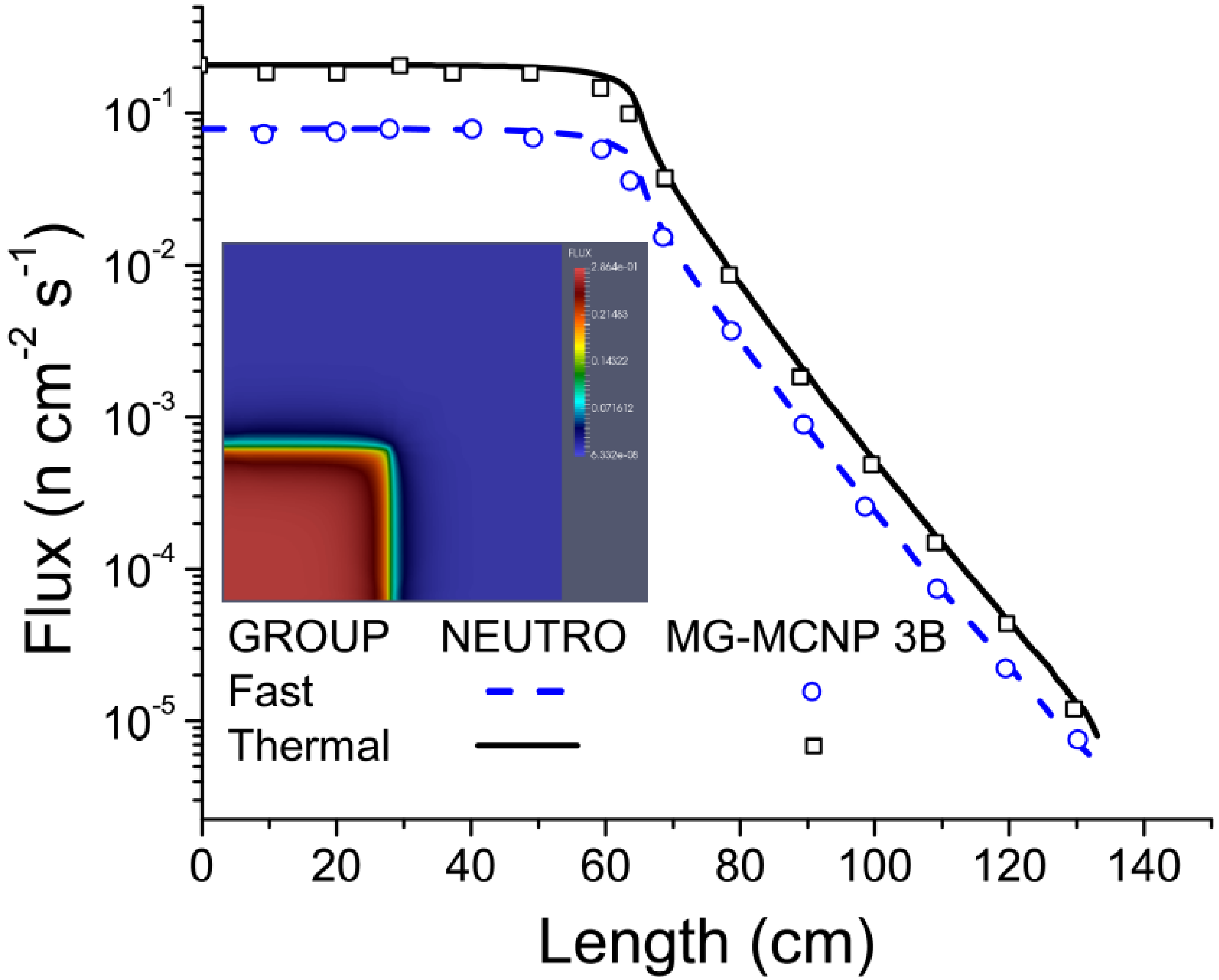
Results: Reed test



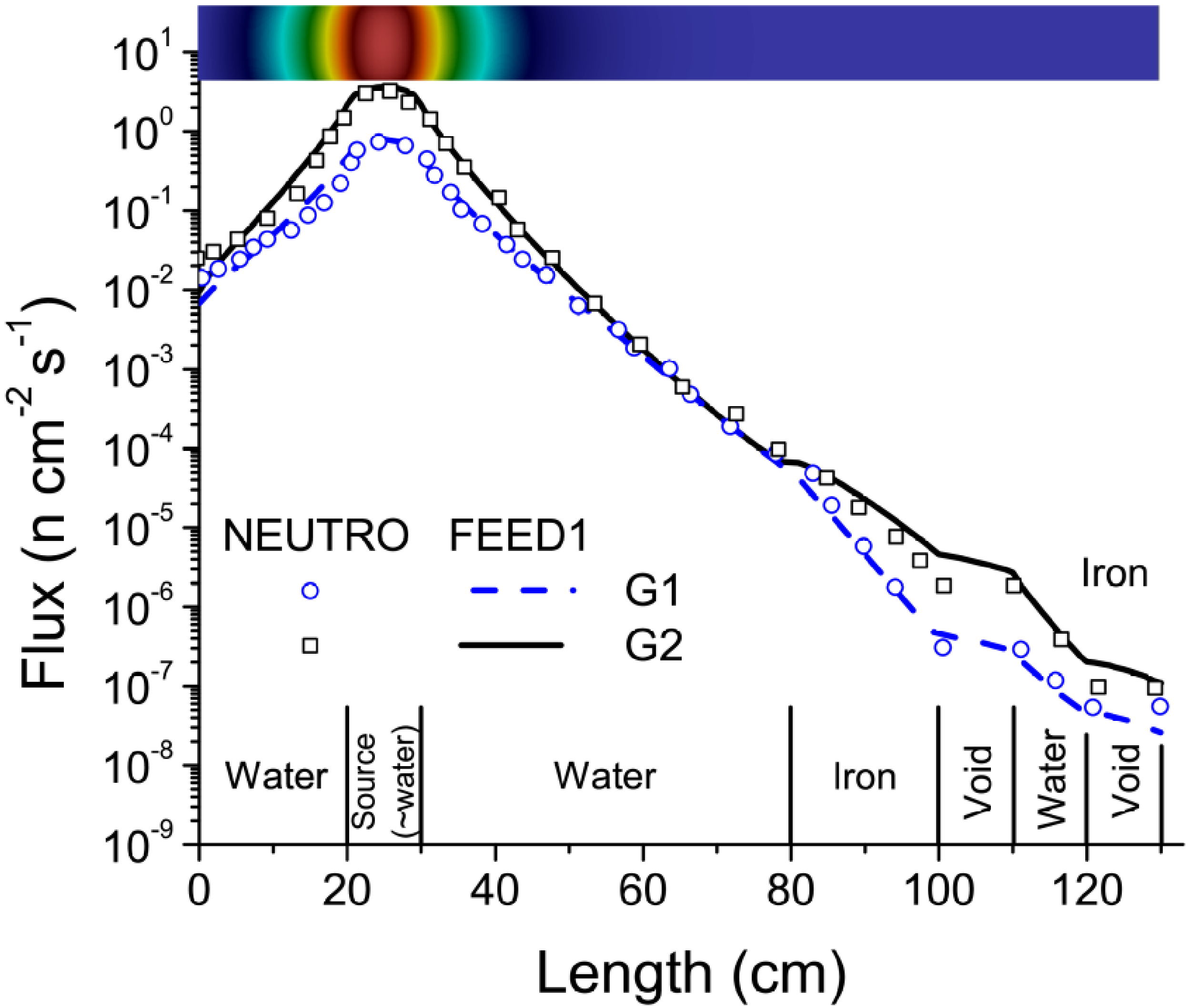
Results: Alcouffe test



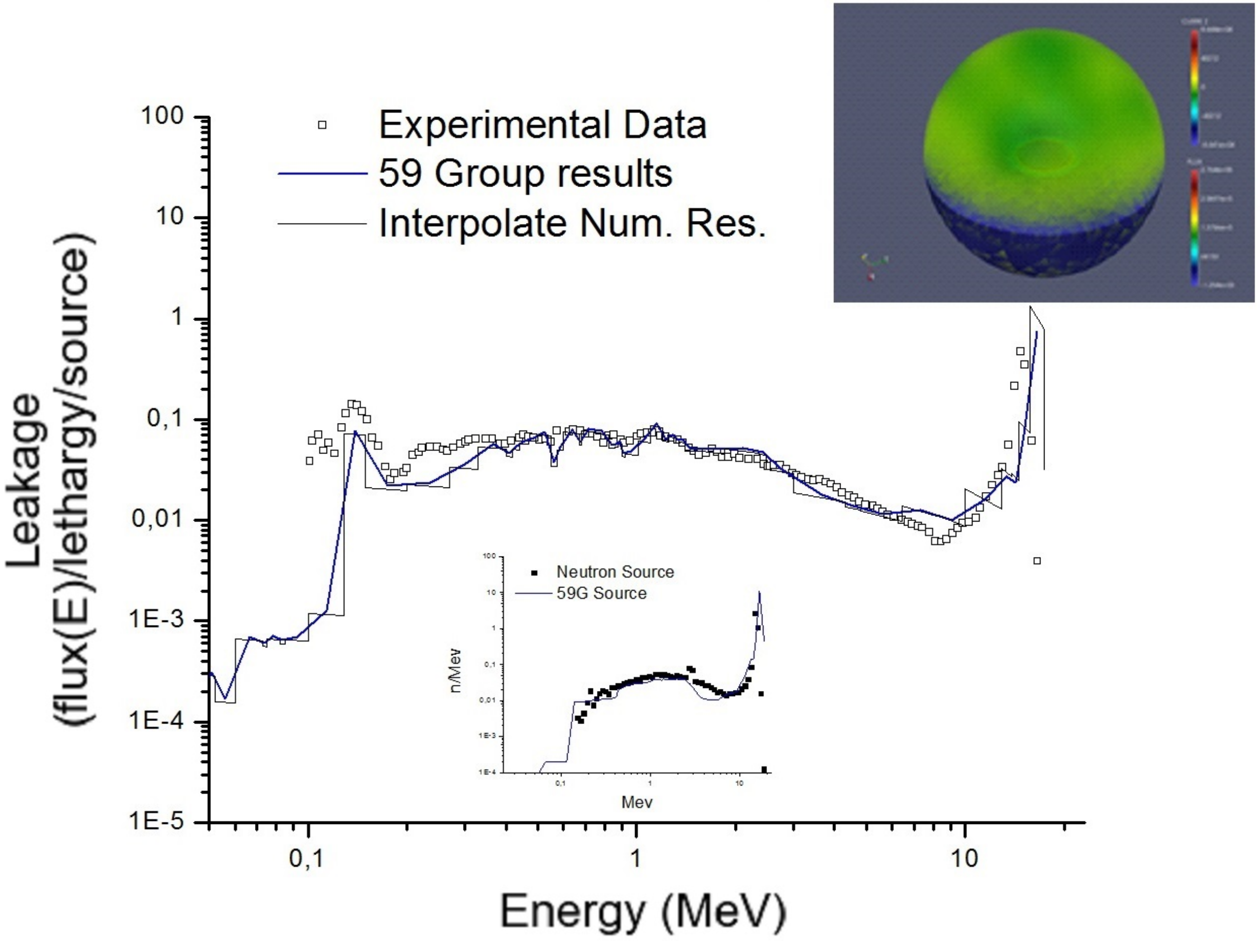
Results: Deep penetration case

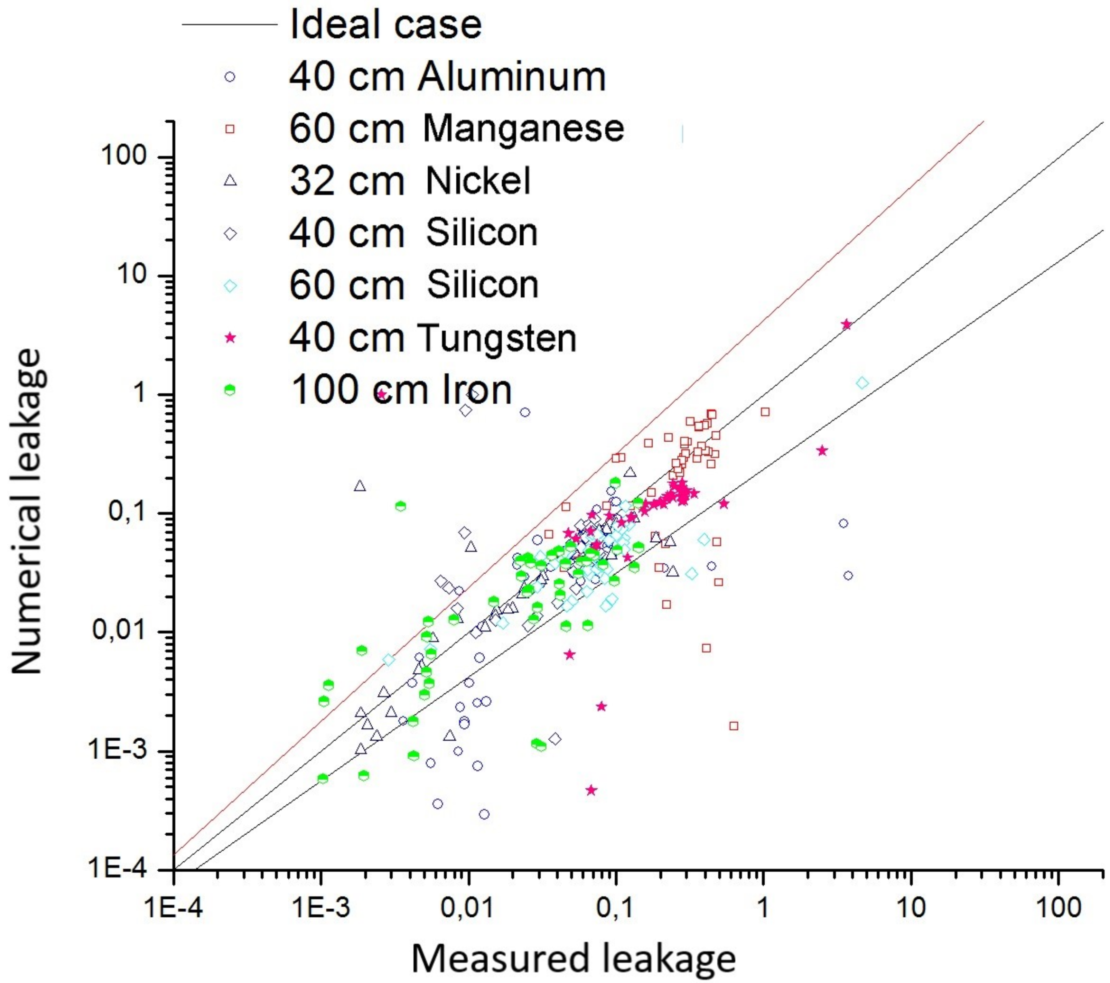


Results: Shielding

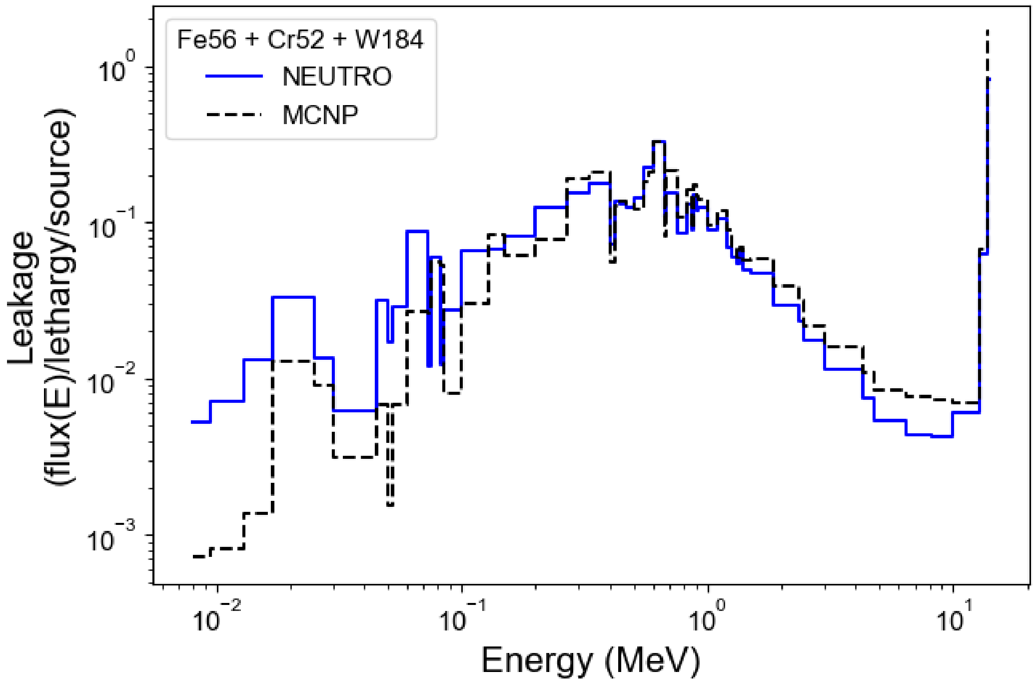


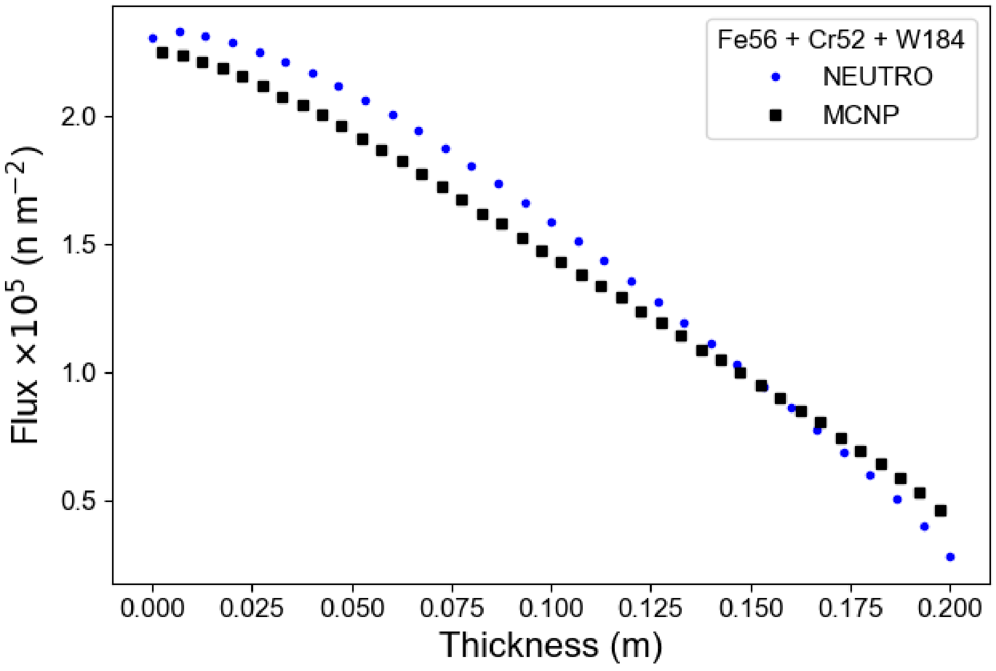
Results: SINBAD testing





Results: Comparing NEUTRO with MNCP



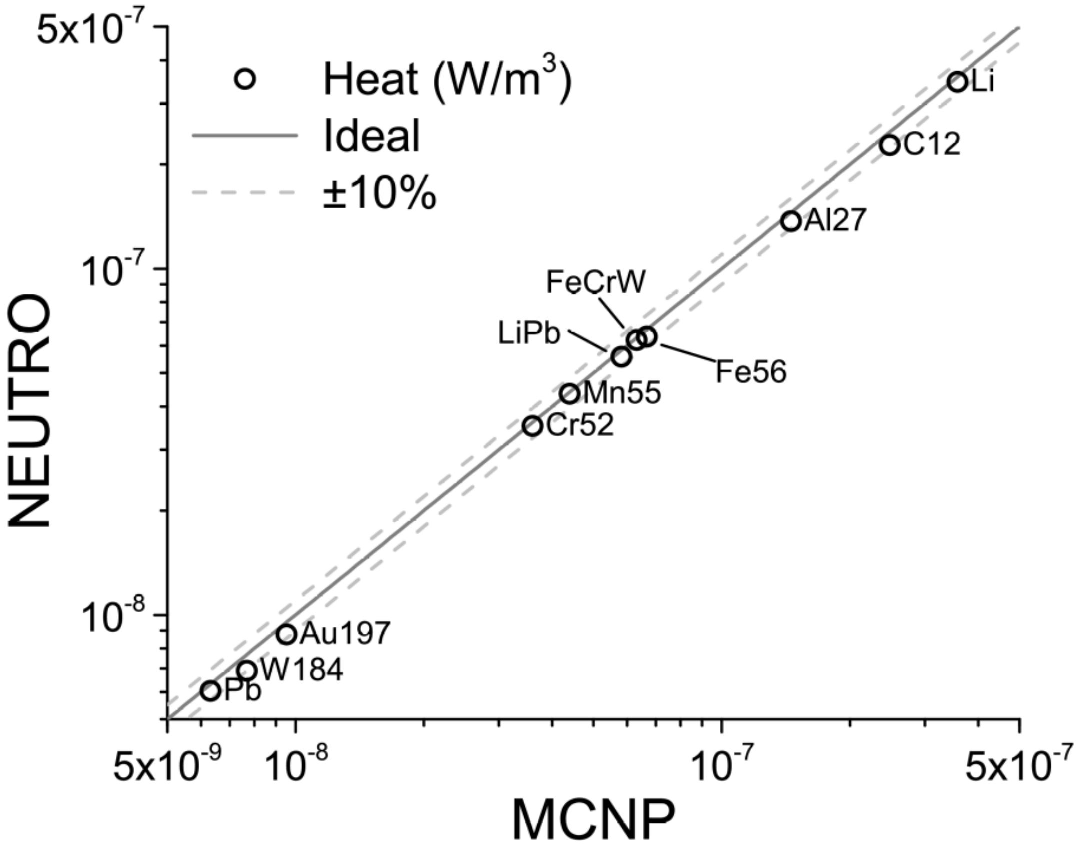


Results: Comparing NEUTRO with MCNP

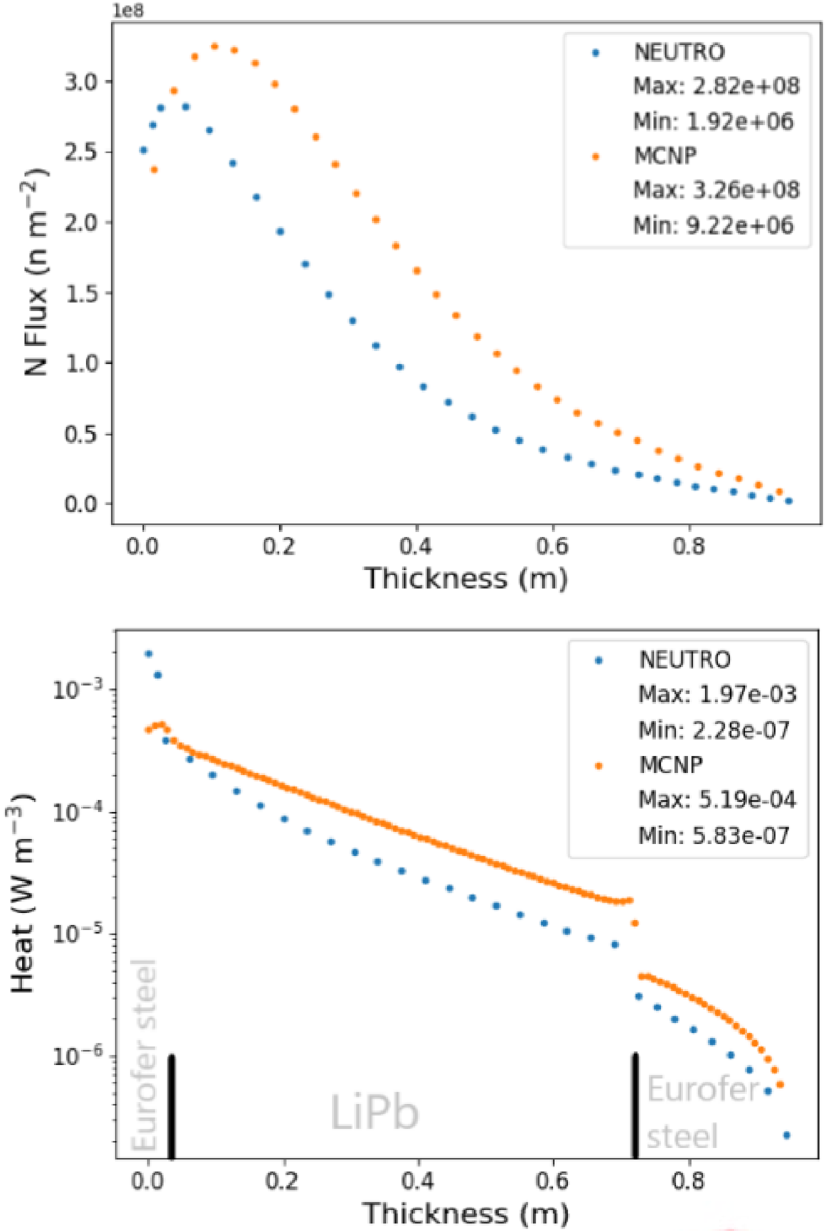
- Include neutron heating:
  - Neutron-material interaction.
  - Photon-material interaction.
- Heating is directly porportional to local neutron flux:

$$H(E) = \varphi(E) \cdot k(E)$$

where $k(E) \equiv$ KERMA factor.

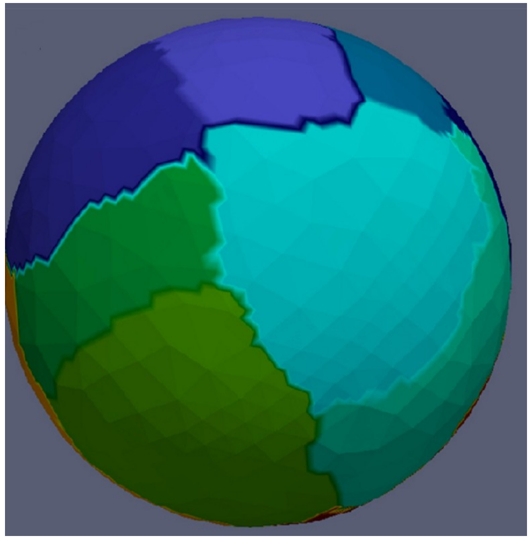


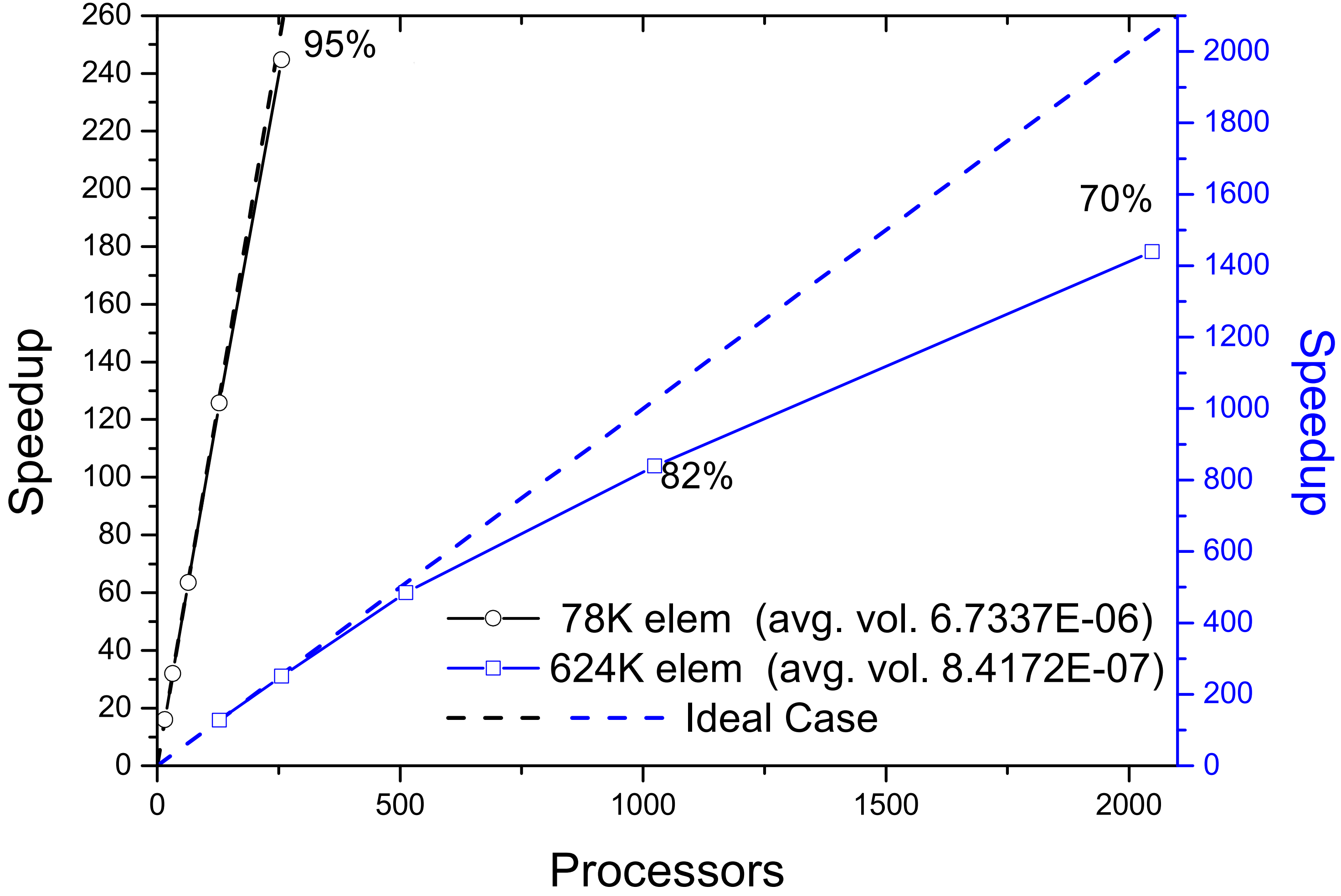
Results: DEMO blanket



HPC scalability

- Linear tetrahedral mesh
- Sn12 quadrature
- 59 energy groups





Ongoing and future work

- Calculate tritium production from neutron interaction with breeding blankets.
- Further validation of heating due to neutron radiation interaction with materials.
- Couple NEUTRO with other Alya modules and test validation cases solving multiphysics behaviour (thermal + mechanical + neutronics).

Conclusions

- Neutron transport code developed within Alya multiph
- Goal: a tool for contributing to reactor design studies.
- Handle composite and alloy materials in 3D domains with unstructured meshes.
- Tested against 4 classical tests.
- Validations against 7 SINBAD experimental benchmark.
- Heat due to neutron radiation interaction has been integrated in NEUTRO.
- Compared NEUTRO against standard neutronics code.
- Code scalability has been proved showing a nearly ideal behaviour up to 512 processors.
- Expand code's features as a next step.

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Thank you for your attention

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ICMAB

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IREC Institut de Recerca en Energia de Catalunya Catalonia Institute for Energy Research

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CNEA Comisión Nacional de Energía Atómica

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