



AmBeSim

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Theoretical
Framework

Neutron
Diffusion

Set-Up

Experimental

Computational
Conclusion

Moderation of AmBe neutrons in water: a computational approach with Geant4

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^{241}Am - ^9Be Neutron source

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Long half life and stable flux over a 10 - 15 year working life

Plethora of uses:

- Metrology
- Education environment
- Neutron Activation Analysis for identification of unknown materials
- Calibration (dosimeters and detectors)
- Industrial (e.g. well logging via $^1\text{H}(n, \gamma)^2\text{H}$)



Reaction of Interest

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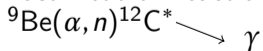
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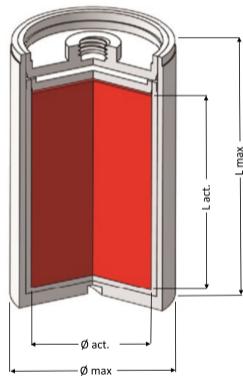
Mixture of AmO_2 and ^9Be powder
Stainless-steel casing, > 99% ^{241}Am
 ^{241}Am α emission:

Energy (keV)	Intensity (%)
5388	1.66
5442.80	13.1
5485.56	84.8
5511.5	0.225
5544.5	0.37

Fast Neutron reaction: Q value: 5.702 MeV



^{12}C can be either in ground, 1st, 2nd (Hoyle) excited state



Source drawing, AmBe mixture (red) encased in steel [Raims Ltd]



Reactions of interest

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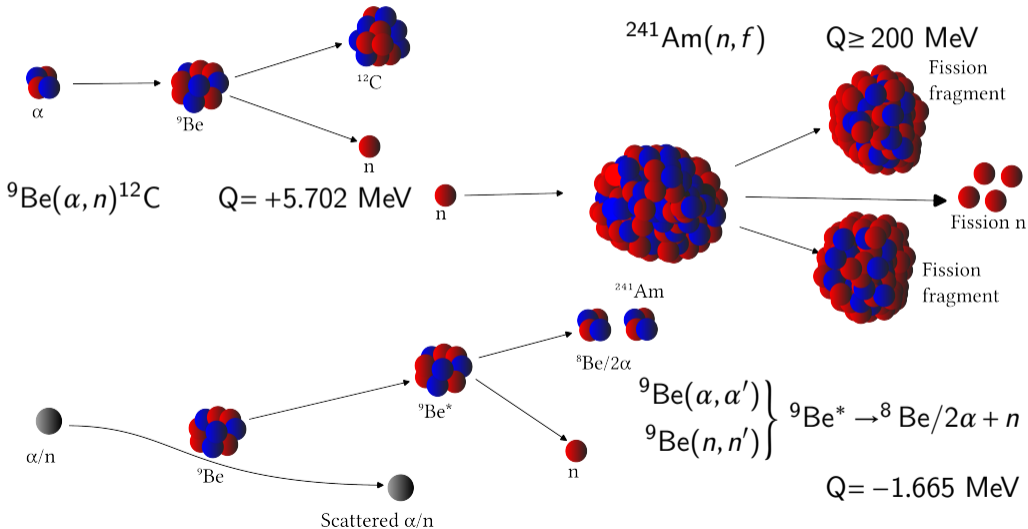
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Experimental

Computational

Conclusion





Model comparison

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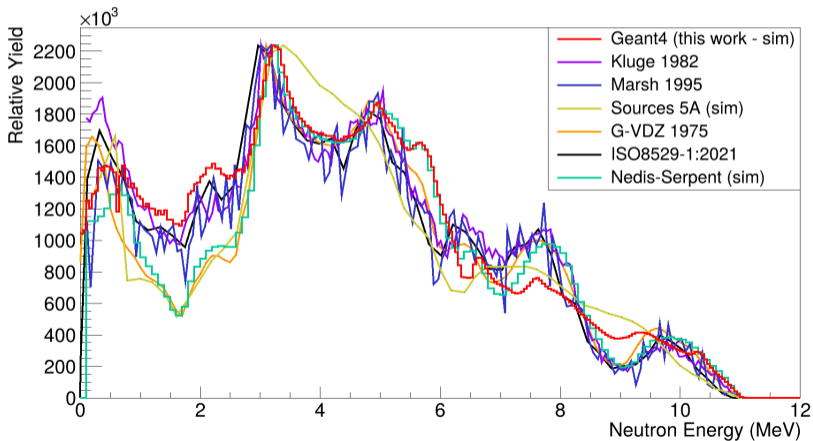
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Experimental

Computational

Conclusion

Comparison with AmBe standards





Neutron diffusion

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Set-Up

Experimental

Computational
Conclusion

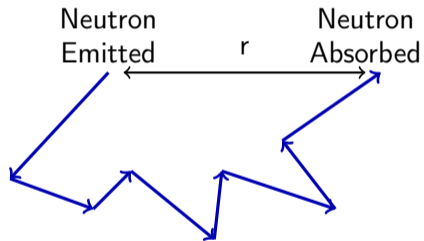
Neutrons lose (or gain) energy via collisions
Neutron moderation

$$L^2 = \frac{1}{6} \overline{r^2} \approx \frac{D^2}{\Sigma_{a,g} + \Sigma_{g \rightarrow h}}$$

Diffusion equation:

$$\text{flux transport} - \text{absorption} - \text{down-scattering} + \text{up-scattering} = -\text{source}$$

Assume N Energy groups: from $g=1$ (most energetic) down to $g=N$



[Lamarsh-Baratta 2001]



Two group model

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Computational

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Diffusion equation for point source with two group model: fast and thermal

$$\Phi_f(r) = \frac{S}{4\pi r D_f r} \left(e^{-r/\sqrt{\tau_T}} \right)$$

$$\Phi_T(r) = \frac{S L_T^2}{4\pi r D_f (L_T^2 - \tau_T)} \left(e^{-r/L_T} - e^{-r/\sqrt{\tau_T}} \right)$$

- $\tau_T = \frac{D_f}{\Sigma_f} \rightarrow$ (Fast) Fermi/neutron age
- $L_t^2 = \frac{D_t}{\Sigma_{a,t}} \rightarrow$ Thermal diffusion length



Experimental Setup: Water Bath

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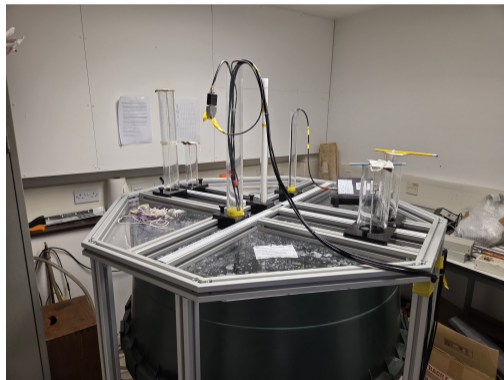
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Set-Up

Experimental

Computational
Conclusion

- AmBe source at centre of 1 m tall, 1 m diameter water tank
- Acquisitions radially from the source in 5 cm steps (from ≈ 10 cm to ≈ 45 cm)
- Run simulations and experimental acquisitions with a plethora of detectors: ^3He , BF_3 , CLLBC and $\text{LiI}(\text{Eu})$
- Acquisitions with and without cadmium detector shield (cut 0–1.14 eV neutrons)





Experimental results

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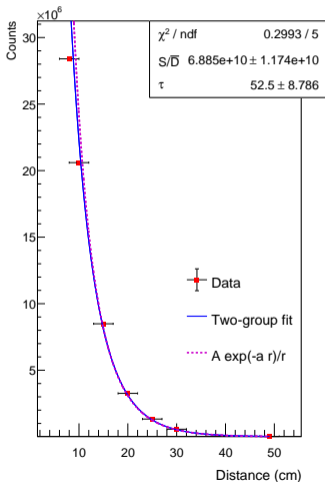
Set-Up

Experimental

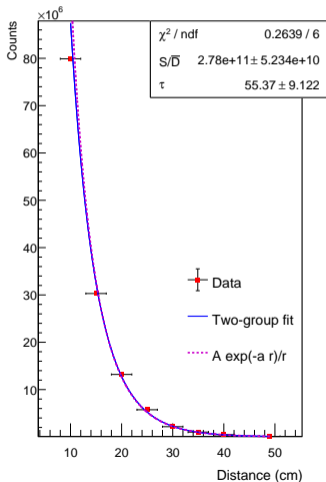
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Conclusion

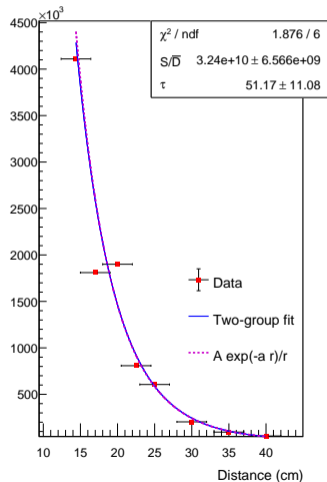
BF3_diff.dat



He3_diff.dat



CLLBC_diff.dat





Computational approach

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Set-Up

Experimental

Computational

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Developed AmBe simulation (published and peer-reviewed)

Simulate detectors or **Direct approach** — Validate neutron moderation:

- Track from start/creation point in medium to thermalisation point (1.44 eV indium resonance)
- Simulate batch of mono-energetic point-like neutron sources
- MFP to 1.44 eV (L^2) compared with literature data (Goldstein et al. 1961)

For inelastic collisions, tracks get split: $n1 \rightarrow n1 + n2$

Treat $n1$ neutron as single track

Consider $n2$ as created at original $n1$ creation point, count MFP from there (can also ignore if issues handling)



Simulation neutron moderation validation

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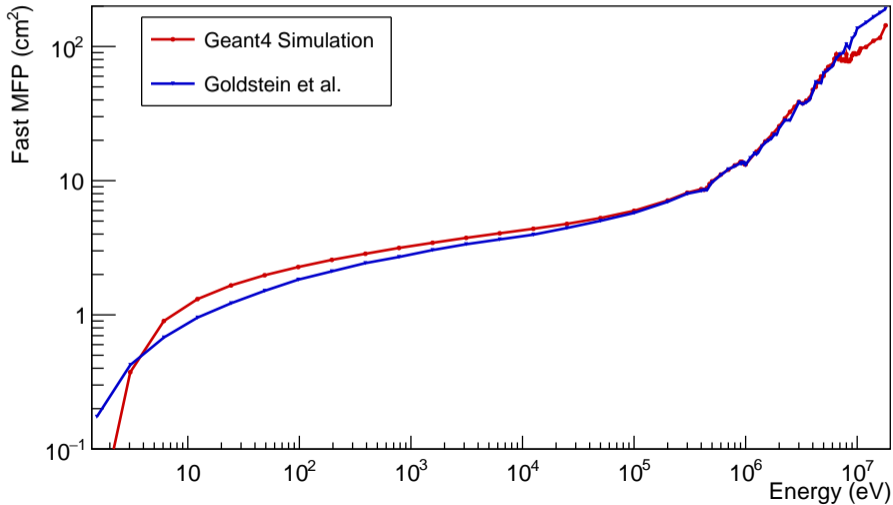
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Diffusion

Set-Up

Experimental

Computational

Conclusion





Simulation neutron moderation validation

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Framework

Neutron
Diffusion

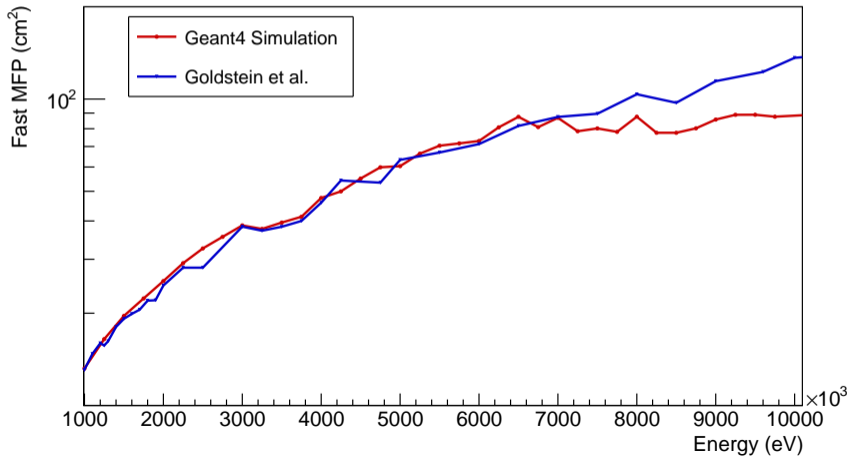
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Experimental

Computational

Conclusion

Zooming into 1—10 MeV Region





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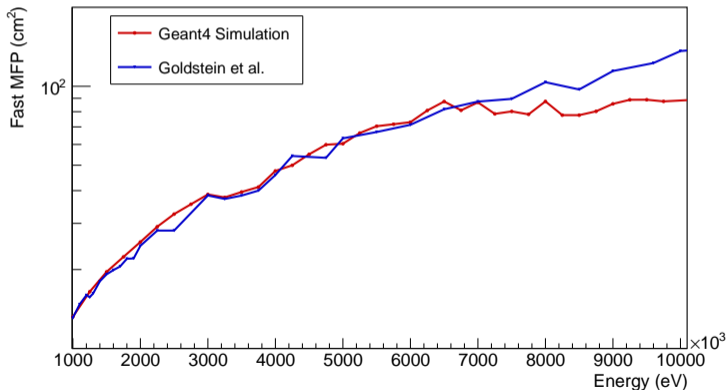
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Experimental

Computational

Conclusion

Zooming into 1—10 MeV Region



- AmBe source yields $56.93 \pm 0.02 \text{ cm}^2$, compatible with mean of 4.21MeV
- agrees with Goldstein
- experimental fits are all $< 1\sigma$ of this



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Neutron
Diffusion

Set-Up

Experimental

Computational

Conclusion

- Adopted validated AmBe neutron Geant4 simulation - Open Source simulation with complete physics
- Verified neutron moderation accuracy of Geant4 simulations using JEFF-3.3
- Matched experimental values to simulated MFP
- Simulated detectors results WIP

Thank you for listening



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Backup Slides

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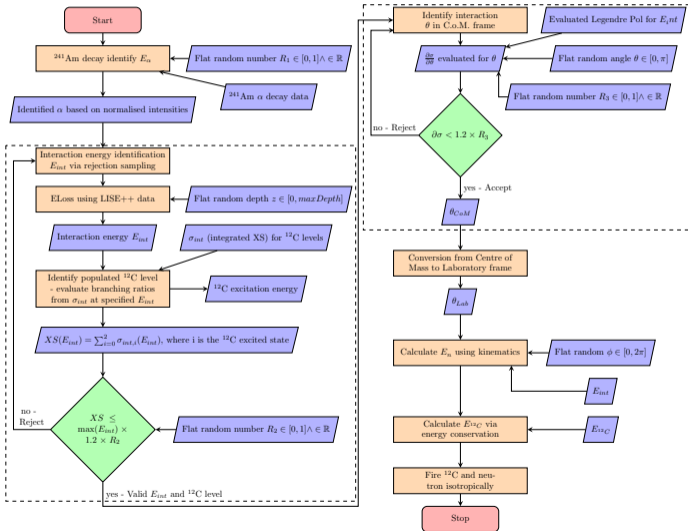


Primary Generator flowchart

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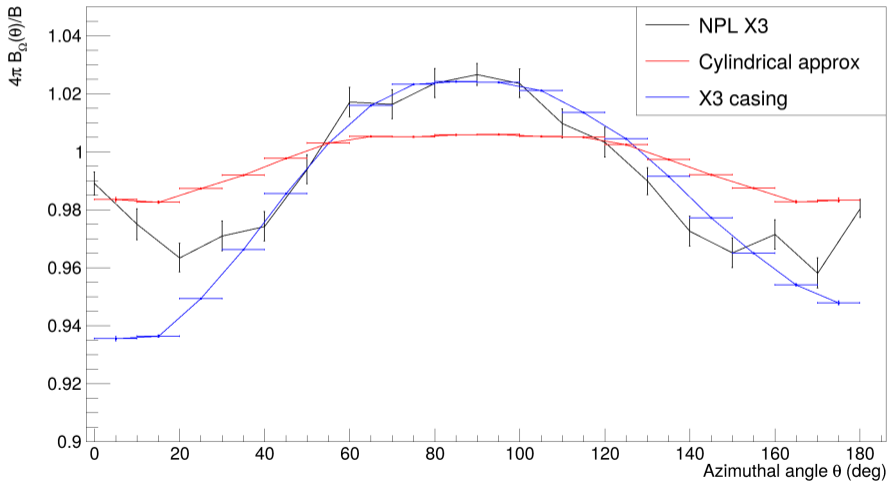


Azimuthal emission

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Finite Cylinder case: Fast Flux

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Interior ($r < R_0$): $\widetilde{\phi}_f = A_f(k) I_0(\kappa_f r) + P_f(k)$

Exterior ($r > R_0$): $\widetilde{\phi}_f = B_f(k) K_0(\kappa_f r)$

where the particular solution (constant in r) is:

$$P_f(k) = \frac{S_f \sin(kH_0)}{k \kappa_f^2}$$

Matching at $r = R_0$. Requiring continuity of $\widetilde{\phi}_f$ and $d\widetilde{\phi}_f/dr$ and applying the Wronskian identity $I_0(x)K_1(x) + I_1(x)K_0(x) = 1/x$:

$$B_f(k) = \frac{S_f R_0 \sin(kH_0) I_1(\kappa_f R_0)}{k \kappa_f}$$

$$A_f(k) = -\frac{S_f R_0 \sin(kH_0) K_1(\kappa_f R_0)}{k \kappa_f}$$



Finite Cylinder case: Thermal Flux

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$$\phi_t(r, z) = \frac{2}{\pi} \int_0^\infty \tilde{\phi}_t(r, k) \cos(kz) dk$$

with the auxiliary wave-numbers:

$$\kappa_f = \sqrt{k^2 + \frac{1}{\tau}}, \quad \lambda_t = \sqrt{k^2 + \frac{1}{L_t^2}}, \quad \Delta = \frac{1}{L_t^2} - \frac{1}{\tau}$$

Region $r \geq R_0$

$$\tilde{\phi}_t(r, k) = \frac{R_0 \sin(kH_0)}{k} \left[\frac{S_t^{\text{eff}} I_1(\lambda_t R_0) K_0(\lambda_t r)}{D_t \lambda_t} + \frac{\Sigma_f S_f I_1(\kappa_f R_0) K_0(\kappa_f r)}{D_t \Delta \kappa_f} \right]$$

The first term (with $K_0(\lambda_t r)$) decays with L_t (thermal) and arises from the effective thermal source.

The second term (with $K_0(\kappa_f r)$) decays with $\sqrt{\tau}$ (fast) and represents fast neutrons that thermalise in the moderator.



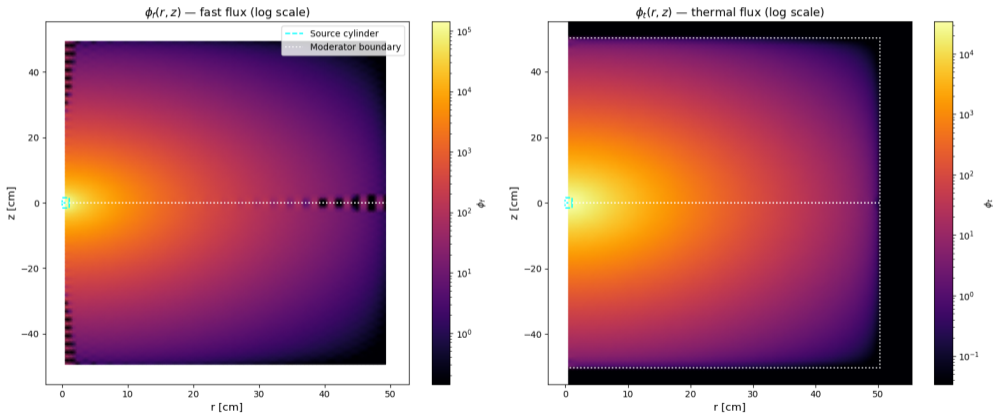
Finite Cylinder case: Flux Plot

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Fast vs thermal flux — finite cylindrical moderator





Equivalent Dose - Preliminary

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Calculated dose for outgoing γ and neutrons from the water bath and verified against experimental
Sampling over 0.2 s spectrum

Particle	Experimental [$\mu\text{Sv/h}$]	Simulated [$\mu\text{Sv/h}$]
γ	1.54	8.05
n	0.8	1.68

Notes:

- Neutrons measured with Nuclear Enterprises NM-2 dose monitor (BF_3)
- Gammas measured with dose monitor calibrated in the 59-1332 keV range