Neutron Moderation Analysis for a Fusion-Based Neutron Source

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INTRODUCTION

Neutrons are valuable particles that have several applications ranging from material activation, testing, neutron scattering experiments, neutron spin experiments, and even radioisotope production. Currently, most neutron sources are in test reactors such as the TRIGA at several universities, the NBSR at the NIST Center for Neutron Research, fast pulse reactors such as the IBR-2 at the Frank Laboratory of Neutron Physics, and high flux reactors such as ILL in France.

To attempt minimizing the costs, improving the accessibility, and increasing the mobility of conducting neutron research, a student group is designing and building a fusion-based neutron source that works like a neutron generator, but can provide access to a sustainable flux of slow neutrons for the applications discussed earlier. Slow neutrons are classified as neutrons that have an energy of ≈1 eV – 10 eV.

The fusion-based neutron source uses deuterium $({}_{1}^{2}H)$ gas to induce fusion, this would provide the following fusion outcomes (additional outcomes may occur as well):

Outcome 1: ${}_{1}^{2}H + {}_{1}^{2}H \rightarrow {}_{2}^{3}He + {}_{0}^{1}n + 3.27\ MeV$

Outcome 2: ${}_{1}^{2}H + {}_{1}^{2}H \rightarrow {}_{1}^{3}H + {}_{1}^{1}H + 4.03\ MeV$

Outcome 3: ${}_{1}^{3}H + {}_{1}^{2}H \rightarrow {}_{2}^{4}He + {}_{0}^{1}n + 17.6\ MeV$

This illustrates that Tritium $({}_{1}^{3}H)$ can be produced in the source. The D-D reactions produce approximately 2.5 MeV neutrons, and the D-T reactions produce approximately 14 MeV neutrons. Either way, the produced neutron is a fast neutron, and needs to be slowed down.

Slowing down neutrons is necessary to allow for activation of materials, and testing of neutron absorption cross sections.

To choose the best moderator for the fusion-based neutron source, a moderation analysis study was conducted using MCNP6 [1], hereafter referred to as MCNP, which is a particle transport code package that allows for neutron transport simulations such as this one. The purpose of the study is to evaluate several moderator alternatives using a simplified moderation analysis method, and come-up with an appropriate moderation setup for the neutron source. The method used, and the results from this study were explained in this paper. Neutron attenuation coefficients for some materials were also derived in the analysis.

METHODS

MCNP was used to simulate a scenario that would allow for neutrons to be slowed down using multiple moderator candidates at multiple initial neutron energies and multiple moderator thicknesses. Each moderator was simulated from 1 to 15 cm of thickness to create a relation that can help estimate the amount of moderator required as a function of desired neutron energy. The initial energy of the neutron source was set at 2.5 MeV for one batch of simulations, and the other batch had 14 MeV as the initial energy of the neutron source. Each batch consists of the same simulation setup, but with different moderator thicknesses (1 to 15 cm). 15 cm was arbitrarily chosen as the maximum thickness for the MCNP simulations, and it was declared enough to generate functions about the moderation behavior of the different moderators using 15 reference points (1 to 15 cm). The process of the partially automated simulation was illustrated in the diagram below.

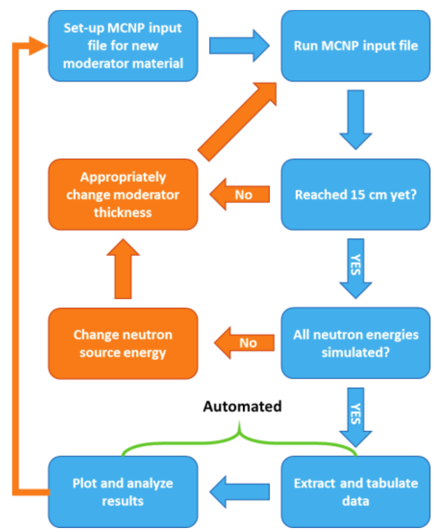

Fig. 1. Moderation Analysis Process.

Moderation Materials Selection

Four different moderation material candidates were analyzed:

- Light-Water (H_2O)
- Ordinary Concrete
- Lead
- Hydrogen gas

The materials were chosen due to their availability for the students developing the neutron source. H_2O was selected because it is a common moderator in LWRs, and it is readily available. Lead and Concrete were selected because they were both candidates for additional photon shielding for the neutron source, and repurposing them for both uses, that is photon shielding and neutron moderation, was an explorable possibility. Even though lead is known to be an ineffective moderator, since it will be used for photon shielding in the neutron source setup, it would make sense to explore its neutron moderation capabilities. Finally, hydrogen was selected due its large scattering cross-section, making it a strong moderation candidate.

Corresponding to the four materials, four sets of studies were conducted, each study included a set of results generated by varying the different study parameters.

Study Parameters

The parameters for the study included the moderator thicknesses and the neutron source energy. The thickness of the moderator was varied from 1 to 15 cm to better observe the behavior of the neutrons, and be able to generate a best-fit curve that allows for the estimation of the moderator's neutron attenuation profile.

In the MCNP study, the neutron source was set as a point source located at the origin of the geometry setup. Its energy varied according to the simulated fusion reaction. For D-D reactions, a 2.5 MeV neutron source was indicated; and for D-T reactions, a 14 MeV neutron source was indicated. Each simulation case included a total of 100,000 neutrons.

Geometry Setup

The geometry of the actual neutron source differs from the MCNP geometry setup, which was simplified for the sake of computational speed.

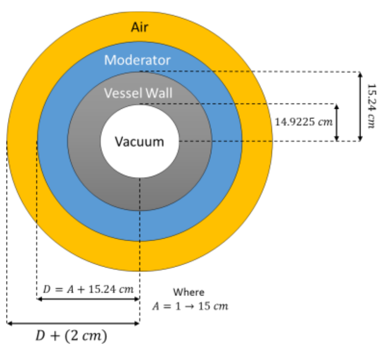

Fig. 2. MCNP Geometry Setup.

The MCNP geometry consists of spheres within spheres. Each sphere represents the following:

1. Fusion chamber (vacuum where neutrons are emitted)
2. Stainless steel spherical vessel outer wall
3. Moderator outer wall
4. Simulation boundary (air between 3rd and 4th spheres)

In this setup, the 3rd and 4th wall expand by 1 cm each time the moderator thickness increased by 1 cm. The dimensions in the figure seem strange because they are derived from the real vessel's dimensions in the fusion-based neutron source, which are in inches. This explains the wonky numbers dimensions.

Computation and Analysis

Collecting Attenuation Coefficients

The MCNP input file was outfitted to compute many results for safety and moderation purposes through the use of different tallies [2]. Amongst the results of importance to the moderation analysis are the following:

- F1 tally: The number of neutrons crossing the moderator's outer wall $(\boldsymbol{n})$
- *F1 tally: The total energy of the neutrons crossing the moderator's outer wall $(\boldsymbol{E_n})$

After all the simulations ran, the average neutron energy $(\overline{\boldsymbol{E_n}})$ was equated as follows:

$$\overline{\boldsymbol{E_n}} = \frac{\boldsymbol{E_n}}{\boldsymbol{n}} \tag{1}$$

Using a custom Python script, the results were then tabulated, plotted, and fitted to an approximated exponential function with the following format:

$$\boldsymbol{E_f} = \boldsymbol{E_0}\, \boldsymbol{e^{-\mu_n x}} \tag{2}$$

Where the final moderated neutron energy $(\boldsymbol{E_f})$ is represented as a function of the initial neutron energy $(\boldsymbol{E_0})$, the moderator thickness $(\boldsymbol{x})$, and the neutron attenuation coefficient $(\boldsymbol{\mu_n})$. This equation is familiar as it represents radioactive decay, and the results of this analysis yield that it is applicable in neutron moderation analysis, and they help create a rule-of-thumbs to estimate neutrons' behaviors in different moderators.

Benchmarking

To verify that the attenuation coefficients are valid and suitable, a benchmark was conducted by comparing theoretical calculations using the attenuation coefficients with an MCNP simulation.

The theoretical calculations were conducted by designing a multi-layered moderation section, and calculating the final energy of a neutron crossing the moderator using equation 2, which utilized the attenuation coefficients from the exponential fits of the data collected from the initial MCNP simulations.

These theoretical calculations were then compared to an MCNP simulation that contained the multi-layered moderation section designed earlier, and tallies for the neutrons count and energies crossing the moderator. Using equation 1, the average neutron energy is calculated, and compared with the theoretical calculations.

RESULTS & DISCUSSION

Moderator Materials Analysis

The four pre-selected moderator materials were analyzed to estimate attenuation coefficients for each of them. The plots show ceiling energies in the 100 eV range due to the fact that they represent the moderated, output neutrons starting at 1 cm of the respective moderator.

Light-Water (H_2O) Analysis

The results for the H_2O analysis yielded exponential decay functions that utilize decay equation discussed in the methods section of this paper.

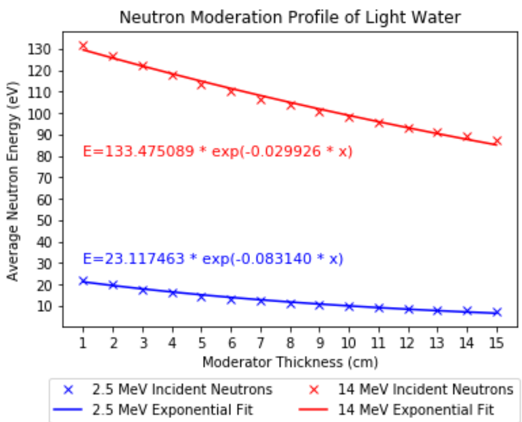

Fig. 3. H_2O Analysis Results

The H_2O analysis results above yielded the μ_n of 0.029 cm^{-1} and 0.077 cm^{-1} for the initial neutron energies of 14 MeV and 2.5 MeV respectively.

Ordinary Concrete Analysis

Similar to the H_2O results, ordinary concrete results were successfully fitted to exponential decay functions as shown below.

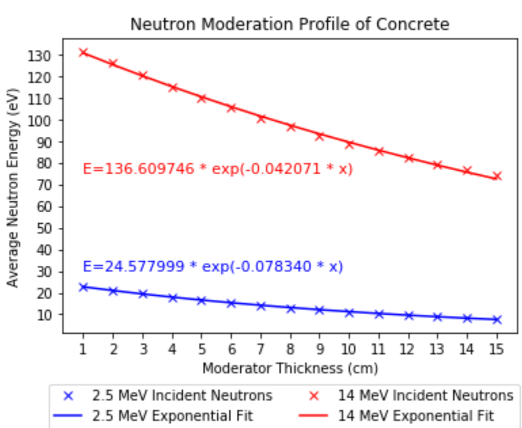

Fig. 4. Ordinary Concrete Analysis Results

The concrete results above yielded the μ_n of 0.042 cm^{-1} and 0.076 cm^{-1} for the initial neutron energies of 14 MeV and 2.5 MeV respectively.

Lead Analysis

The results for lead were also fitting to an exponential decay function, but it is interesting to note how much more effective lead is at decreasing the neutron energies when the initial neutron energy is as high as 14 MeV. It shows a steep decline, and then it settles at a certain energy.

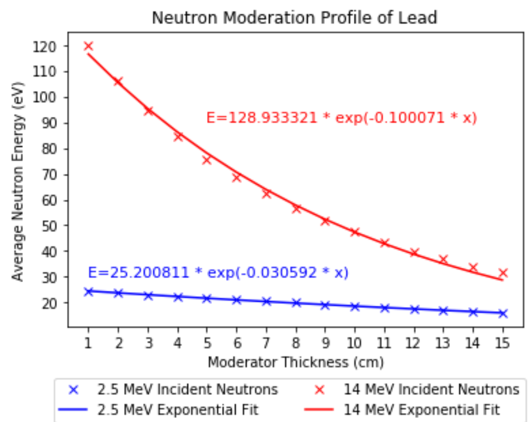

Fig. 5. Lead Analysis Results

The lead analysis results above yielded the μ_n of 0.095 cm^{-1} and 0.031 cm^{-1} for the initial neutron energies of 14 MeV and 2.5 MeV respectively.

Hydrogen Gas Analysis

Hydrogen does not cause the decay of the neutrons to occur at a rate as steep as H_2O, but it comes close to it. It is interesting to note that the first 1 cm of the moderator is most effective, as the neutron energy then settles at an energy equilibrium.

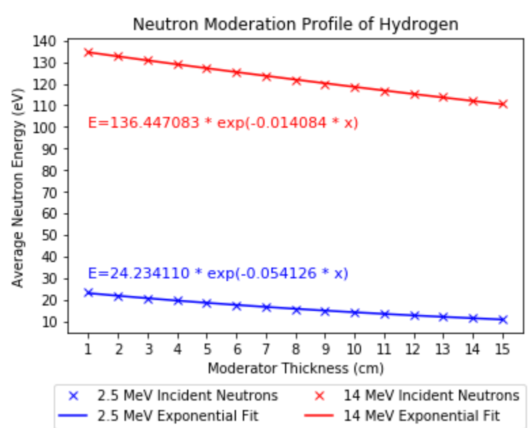

Fig. 6. Hydrogen Gas Analysis Results

The hydrogen analysis results above yielded the μ_n of 0.014 cm^{-1} and 0.053 cm^{-1} for the initial neutron energies of 14 MeV and 2.5 MeV respectively.

Attenuation Coefficients

With the attenuation coefficients compiled from the exponential fits, shown in table 1, a moderator optimization process to obtain any desired neutron energy.

Table I. Calculated Attenuation Coefficients

E_0	Material	$\mu\ (cm^{-1})$
14 MeV	H_2O	0.029926
	Concrete	0.042071
	Lead	0.100071
	Hydrogen Gas	0.014084
2.5 MeV	H_2O	0.08314
	Concrete	0.07834
	Lead	0.030592
	Hydrogen Gas	0.054126

Benchmark: Moderation Section Design

In this benchmark, a moderation section was designed to obtain slow neutrons. The moderation section, shown in the figure below, consists of 3 regions including hydrogen gas sandwiched between two layers of water. Physically, each of the region would be housed in a separate container of irrelevant material. This moderation section would be the blue moderator section in figure 2.

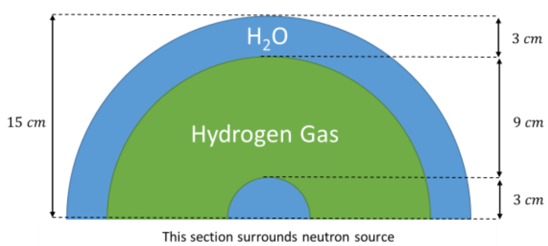

Fig. 7. Moderation Section: "Hydrogen Sandwich"

The results for the benchmark are found in table 2. The comparison shows a good accuracy for the method, with a discrepancy of approximately 6% between the calculated neutron energy and the MCNP simulation calculation.

This discrepancy is likely because of some inaccuracy in the exponential curve fitting performed by the Python script.

Table II. Moderated Neutron Energy Comparisons

Calculation	MCNP	Discrepancy
8.624563095 eV	9.163547889 eV	5.881835303 %

CONCLUSION

Based on the MCNP analysis, describing the neutron moderation profile of different moderators can be done using exponential decay functions. Such functions allowed for the determination of neutron attenuation coefficients that depend on the initial neutron energy. As the energy decreased, the attenuation coefficients increased, showing an inverse relationship between the change in neutron energy, and the neutron attenuation coefficients. The only anomaly in the results was the lead analysis. Lead's neutron attenuation coefficients were not inversely related to the initial neutron energy, as the attenuation coefficients decreased with the decrease of neutron energy. This is likely due to an unknown error in the simulation.

With the existing data, it was possible to design a multi-layered moderator using the estimated attenuation coefficients, providing a desired neutron energy (less than 10 eV). The results from the calculations were plausible as they were only about 6% off of the MCNP values.

It is evident that a material becomes less effective as a neutron loses its energy, as it illustrates a thermalizing process of the neutron. This study concludes with an appropriate design of the moderator for the neutron source.

This method will be further developed with more studies and materials being explored in the future.

REFERENCES

1. T. Goorley, et al., "Initial MCNP6 Release Overview", Nuclear Technology, **180**, 298-315 (2012).
2. J. K. Shultis, R. E. Faw, "An MCNP Primer", Dept. of Mechanical and Nuclear Engineering, Kansas State University, Manhattan, KS 66506 (2011).