Nuclear Reactor Theory Project #1
Group #3

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***Abstract*—**

I. Introduction & Background

Proving the capabilities and safety of a reactor design requires effective modeling of the neutron flux in the core (expressed in equation 1). For real cores, however, this is impossible, and must be first simplified, then discretized to provide the solution for a representative mesh.

For this project we have analyzed a simplified, monoenergetic, non-multiplying medium in one dimension. The flux originates from a single source at $x = 0$ with a strength of $S = 1 \times 10^8\,\mathrm{s}^{-1}$. These assumptions simplify the transport equation to that presented in equation 2.

In the following sections, we will first describe the terms in equation 2, then provide both an analytical and a discrete solution. We will also provide an analysis of the accuracy of the analysis as a function of the number of nodes. Finally, we will analzye the solution of different coordinate systems on our solution.

$$\frac{\partial n}{\partial t} + v\hat{\Omega}\cdot\nabla n + v\Sigma_t n\left(\mathbf{r}, E', \hat{\Omega}, t\right) = \int_{4\pi} d\hat{\Omega}' \int_0^\infty dE' v' \Sigma_s\left(E' \to E, \hat{\Omega}' \to \hat{\Omega}\right) n\left(\mathbf{r}, E', \hat{\Omega}', t\right) + s\left(\mathbf{r}, E, \hat{\Omega}, t\right) \quad (1)$$

$$-D_m \frac{\mathrm{d}^2\phi}{\mathrm{d}x^2} + \Sigma_a^m \phi = \begin{cases} S & (x = 0) \\ 0 & (x > 0) \end{cases} \quad (2)$$

II. Methodology

Equation 2 is a simplified description of neutron diffusion through a finite medium, similar to a point source travelling through a shielding material to a detector. I

III. Results

The results go here

IV. Conclusions

The conclusions go here

Material	$\Sigma_{tr}(\mathrm{cm}^{-1})$	$\Sigma_a(\mathrm{cm}^{-1})$	$\nu\Sigma_f(\mathrm{cm}^{-1})$	*Relative Absorption*
H	1.79×10^{-2}	8.08×10^{-3}	0	0.053
O	7.16×10^{-3}	4.90×10^{-6}	0	0
Zr	2.91×10^{-3}	7.01×10^{-4}	0	0.005
Fe	9.46×10^{-4}	3.99×10^{-3}	0	0.026
^{235}U	3.08×10^{-4}	9.24×10^{-2}	0.145	0.602
^{238}U	6.95×10^{-3}	1.39×10^{-2}	1.20×10^{-2}	0.091
^{10}B	8.77×10^{-6}	3.41×10^{-2}	0	0.223
	3.62×10^{-2}	0.1532	0.1570	1.000

TABLE I
MACROSCOPIC CROSS SECTIONS