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Development of a Process for Extracting Angular Flux from an MCNP Output

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Abstract

The MCNP code is one of the most widely used codes in the United States for conducting neutron transport simulations. It is a very thorough and reliable program and can provide one with a variety of information related to a given situation. However, despite its wide usage and decades of improvements by the development team at Los Alamos National Laboratories, it is still unable to perform certain tasks. Among these tasks is the ability to tell angular flux by itself. All this despite the fact that MCNP must store a given particle's position, velocity, energy, etc. until said particle is terminated. So, to circumvent this shortcoming, a method has been developed to extract the necessary information from MCNP to calculate the angular flux. Using a little known input parameter in MCNP combined with a self coded script in Python, one can efficiently end up with the angular flux incident upon a given location in the simulation and can have it sorted by however many bins they desire. While this method is not absolute and does contain its fair share of drawbacks, it is preferable considering that one only needs to use MCNP and does not have to dedicate their time or funds to learning a separate code.

Introduction

Angular flux can be a difficult quantity to measure. As it is a measurement of flux that is moving into a given steradian, it would be very hard to measure this experimentally. For this, one would need several detectors oriented in such a way that they would accept a given orientation, yet would be shielded from flux coming from any other direction. This would prove to be not only very challenging, but tedious as well, as to measure several components of the angular flux in the same area, the experiment would have to be repeated as many times as the number of steradian divisions one hopes to get. Because of these reasons, angular flux is not found experimentally much, if at all. Instead, most people who are looking for angular flux would have to rely upon computer simulations to tell them what the values are for a certain group of steradians in a given location that they are interested in. This is more feasible as a Monte Carlo simulation would keep track of a particle's complete history from its birth to the moment it dies. However, there is no option, tally, or parameter available to directly receive the angular flux of a simulation output directly in MCNP. The closest anyone has been able to come to doing so was using MCNP in tandem with another program to find the angular flux they are looking for. Yet this can be inefficient as one would then need to write two separate simulations in two separate programs in order to obtain said data. Not only that, but if either of the programs did not work the way they were intended, it would not proceed to give the correct data. Thus, it is more efficient to only run one simulation to minimize chances of failure as well as to simply take the given output data and filter it to leave what is desired.

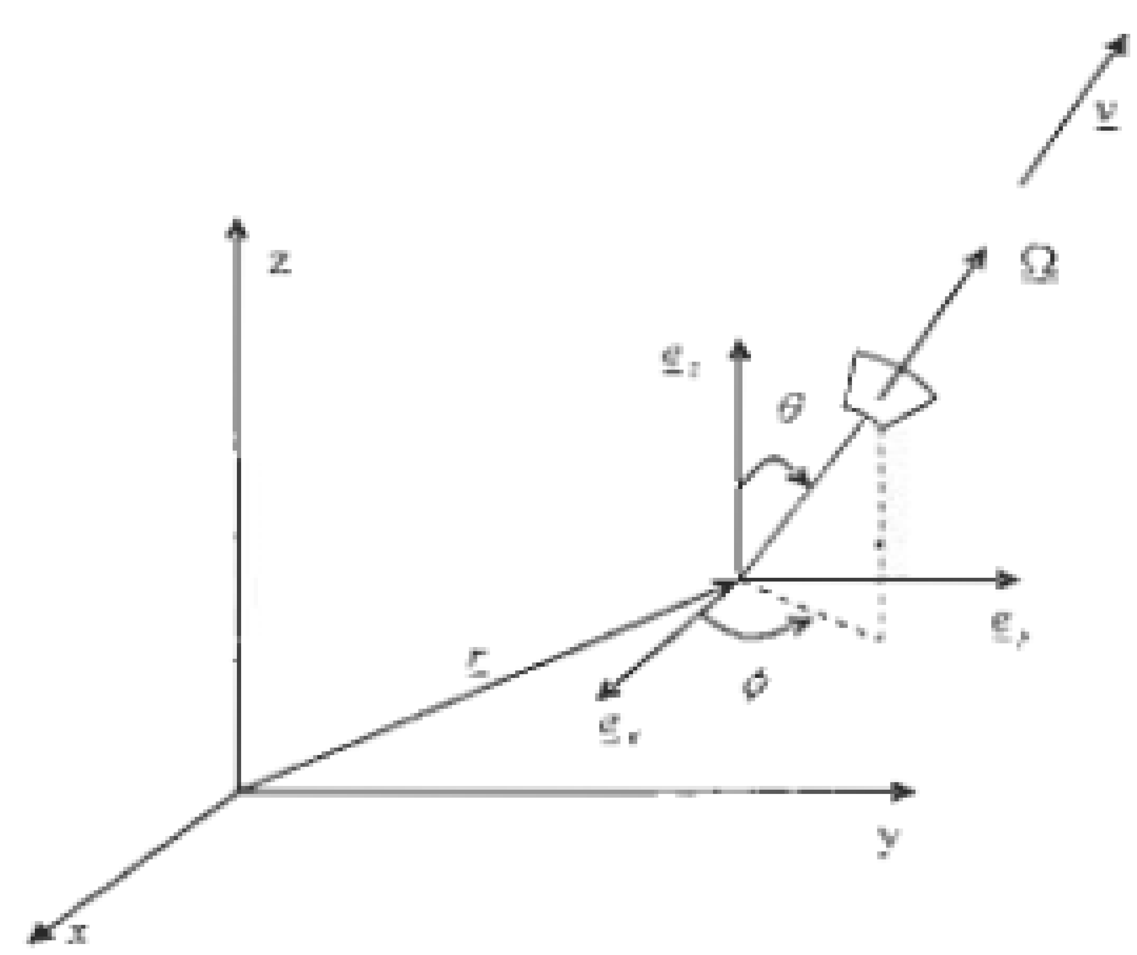

Figure 1. Characterization of position and direction of a particle

Methods

Angular fluxes are directly concerned with the direction that the particles are moving in. The position of the particle is not of any concern. All that angular distribution of the flux conveys is how many particles are headed in a specific direction. Using this, one can predict where the particles will end up and what might receive a higher dose of said particles. Because of this, only the vector components of the velocity need to be known. To further reduce the required knowledge, if the system one is looking at has cylindrical or azimuthal symmetry, then one can easily assume a symmetry in the flux about a given region if said region contributes to the symmetry. If such parameters are met, then one can assume symmetry in regards to the flux and only need to worry about angles with respect to the axis of symmetry. With this assumption, one can reduce the amount of data needed to a mere one piece of information per particle.

Recognizing the Problem: MCNP is very good about outputting a wide range of data. It can give fluxes through a surface or volume, equivalent doses, calculate response functions, energy deposition, and separate all of the above in accordance to the energy of a particle. More so, it will even do such things as tell you the number of times a certain reaction had occurred during the simulation, total number of particles that ended up existing in the simulation, and even total collisions without even being asked to do so. Yet, it commonly only separates things in terms of energy. This despite the fact that as a Monte Carlo simulation, MCNP must keep information such as the particle's position, velocity, and time it has been alive stored somewhere at least temporarily.

PTRAC: There is a built in function in MCNP called PTRAC, short for Particle Tracker, which is able to output the information about the particle that is commonly lost including the particle's position, direction, and time it has been alive. Using this data, one can get a much better picture of the flux profile. Using a simple python code, one can sort the data and take what is needed for whatever they wish to calculate into as many bins as they desire. Angular flux for example can be easily calculated . One can even use the energy output from PTRAC to find energy dependent angular flux. Using the same program, one can also get a graph of said output as shown in Figure 2.

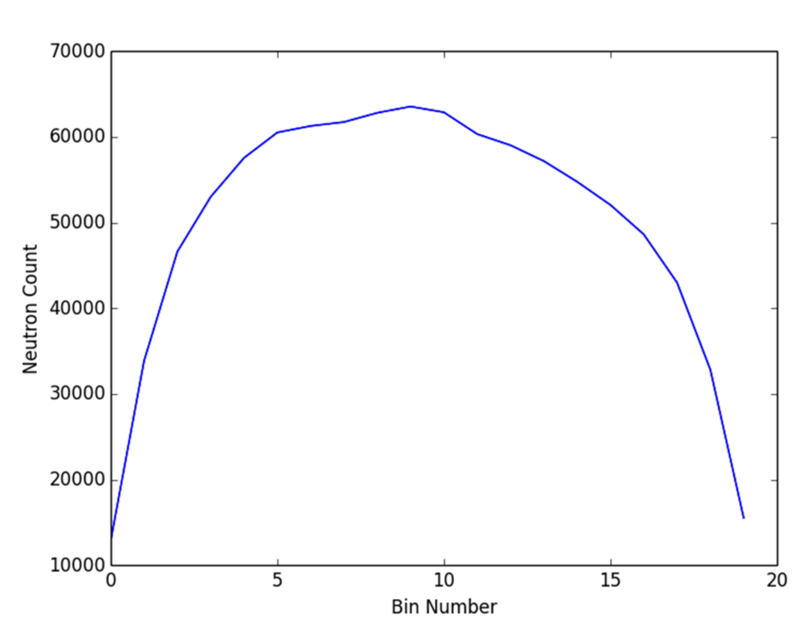

Figure 2. Plot of reactor angular flux by steradian.

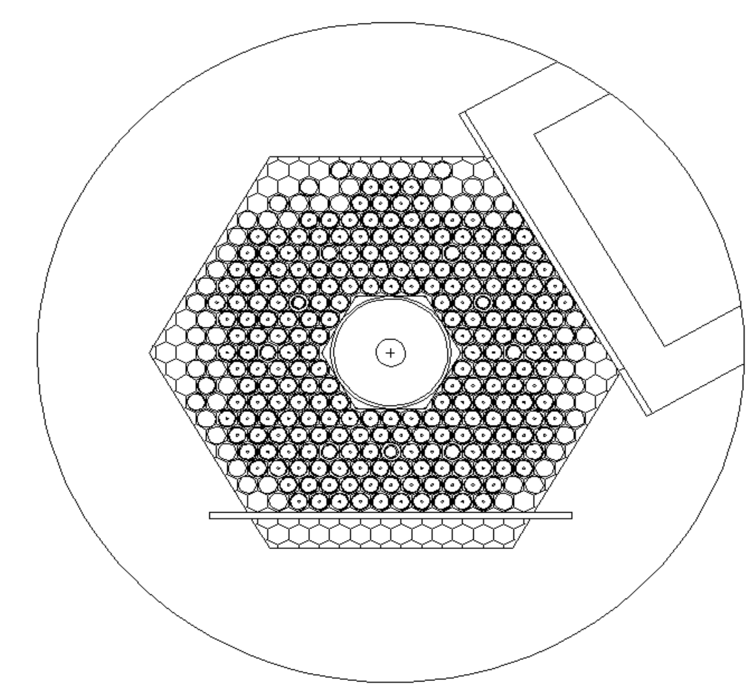

Figure 3. Depiction of the reactor modeled with the sphere of interest in the middle.

Results and Future Work

While this method of combining a basic script with the PTRAC function from MCNP works and paints an accurate picture of angular flux and can be used to gain other information, there is one major drawback: the process is rather time consuming. For whatever reason, MCNP does not allow for parallel processing while using PTRAC. Because of this, the simulation can only be run on one core which can take anywhere from days to weeks to complete depending on the complexity of the simulation geometry and the number of particles desired for the data output. Not only that, but the method only allows the sampling of one particle type and one sampling cell per simulation. Hence, one would need to do extra simulations to get data for say photons and neutrons that are incident on a surface. Despite these shortcomings, the method indeed works well. It can be used to test assumptions about isotropic irradiation based on geometrical symmetry to a given surface for experiments done in reactors in the future. Also, the program can be expanded to give plots of where exactly the particles cross a surface, what energy they enter with, and when they enter the cell. These can be used to double check results for previous experiments as well as help set up those to be done in the future. In the end, this is a method that not only can produce current results, but can be expanded upon in the future to produce even more.

Acknowledgements

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U.S. DEPARTMENT OF ENERGY

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Neutron Multiplication and Attenuation Through Beryllium Slabs

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Abstract

When it comes to producing neutrons, having a source can be somewhat costly. Therefore, if there are materials which can multiply the neutrons produced from a source, then the source does not have to be so powerful in the end, thus saving cost. However, the neutrons produced through multiplication may have different energies than what is desired. Not only that, but a particle multiplier could end up reducing the number of particles that ends up passing through it if it becomes too thick to where absorption and scattering effects lead to total loss outweighing total production in the multiplier. Therefore, it becomes necessary to run simulations for several situations pertaining to the slabs and materials. This is important to know for experiments dealing with material irradiation where the orientation and distance of the detector from the source as well as the slab thickness matters. With this information as a base, one can have a benchmark of what their experimental results should look like.

Introduction

Neutron multiplication can be done using materials that are capable of undergoing (n,2n) reactions. While there are a few materials that are capable of undergoing such reactions, the material of interest here is Beryllium. This is likely due to the fact that at most of the energies under which it can undergo an (n,2n) reaction, it has a fairly constant cross section as seen in Figure 1. It is important to note, however that the reaction in Beryllium is not a true (n,2n) reaction. Once a neutron hits the Beryllium-9 atom, it briefly transforms into the very unstable Beryllium-10. From there, it decays almost immediately into two alpha particles and two neutrons. Because this process is so fast and Beryllium-10 is so short lived, the reaction is simply classified as an (n,2n) reaction. It is also important to note that (n,2n) reactions typically occur only at high neutron energies as can be seen in Figure 1. Hence, in order to have Neutron multiplication occur in the first place, once needs a very high energy neutron source. Neutrons produced from a D-T reaction would suffice as they come off at around 14MeV, which is more than enough to trigger the (n,2n) reaction in Beryllium. With the extra neutrons produced in the Beryllium, any material on the other side of it should receive more radiation. The only caveat here being that if the Beryllium between the source and the irradiated material is too thick, then the other processes in Beryllium such as scattering and absorption would dominate and the resulting dose to the intended material on the other side would end up decreasing. So the issue to tackle is to find a slab thick enough so that a good amount of neutron multiplication occurs, but not so thick that shielding dominates.

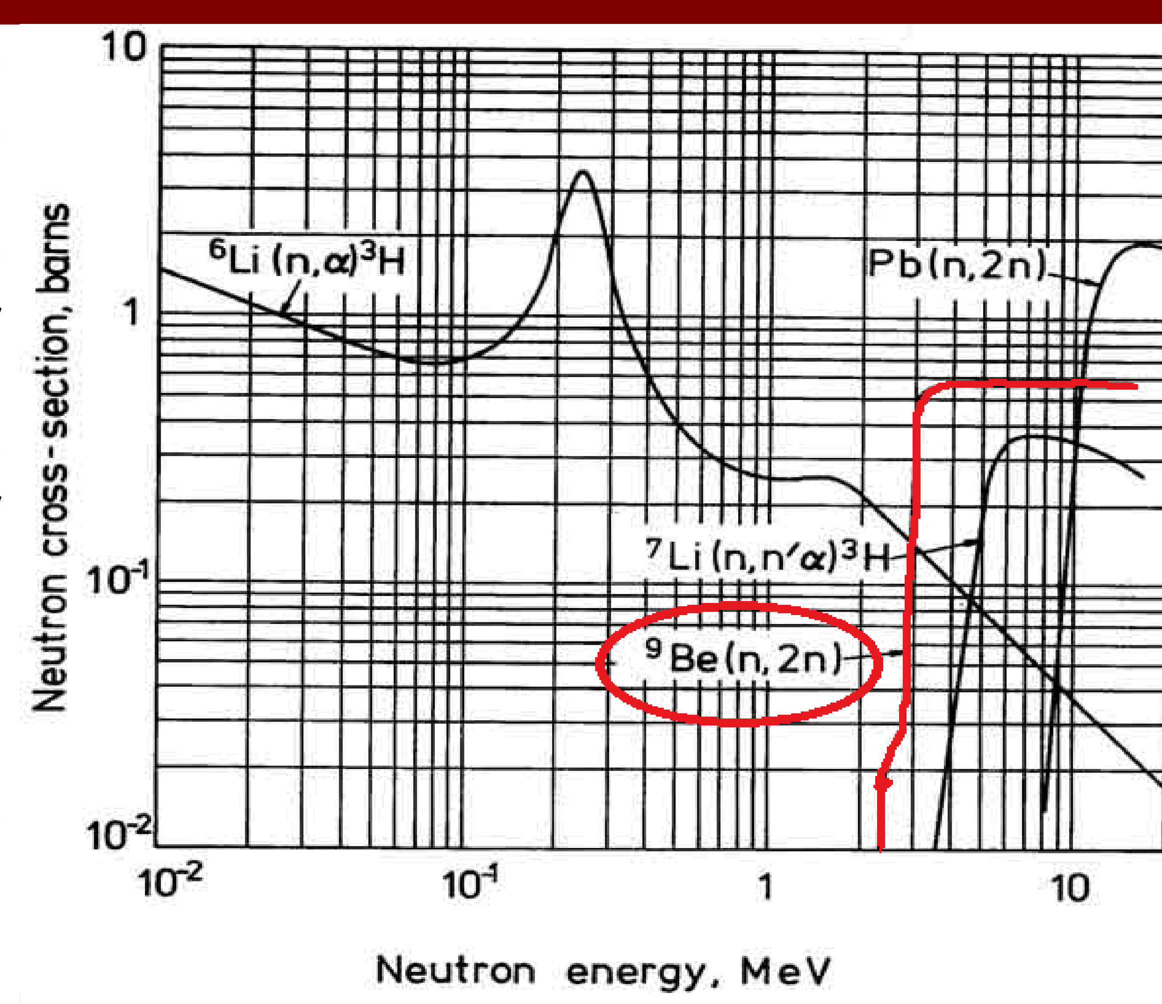

Figure 1. Cross Sections of Beryllium's (n,2n) reactions as a function of incident neutron energy.

Methods

To find the right slab thickness, MCNP was used in order to run neutron transport simulations. Nine very thin, cylindrical detectors were arranged in a cross pattern a set distance away from the source and placed behind a slab as shown in Figures 2 and 3. The detectors were split into three categories based on their distance from the middle of the slab. The one in the center, the four on the outer most points of the cross, and the four in between. In the groups containing four separate detectors, the average of their data was taken when making final calculations. In addition to a tally of raw, energy dependent flux, multipliers were input in the simulation to give an equivalent flux of 1MeV neutrons. The simulations were then carried out while varying thicknesses of the slab from one to ten centimeters. The results from these were compared to a control run of the detectors without a slab in front of them to find the neutron multiplication that had occurred. In addition, three sets of these runs were implemented with varying distances of the detectors from the source with the distances being ten, twenty, and thirty centimeters away. Once the runs were done, graphs were compiled to show how both flux and 1MeV equivalent flux varied in each detector group as a function of slab thickness, an example of which is given in Figure 4. In addition, using the data from the MCNP output, an energy spectrum was compiled for fluxes in each thickness for each distance from the source. Figure 5 gives an example of an energy spectrum for a given distance with all slab thicknesses superimposed on each other.

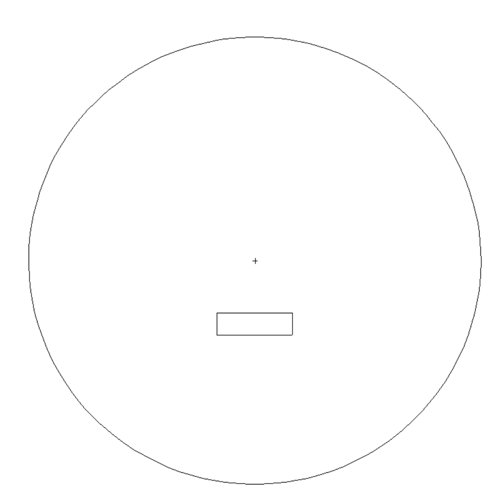

Figure 2. Diagram of the slab in relation to the source (shown as point in the middle).

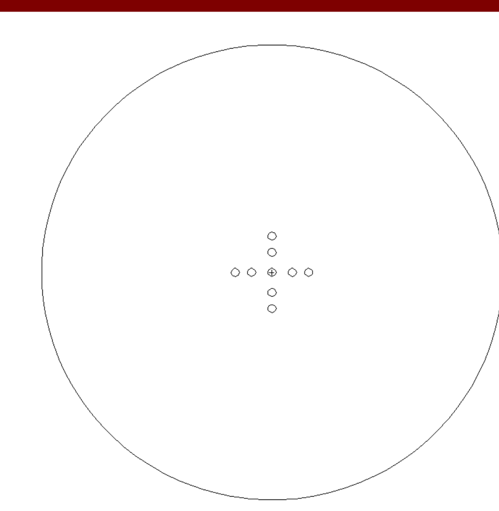

Figure 3. Depiction of the cross shaped array of detectors used.

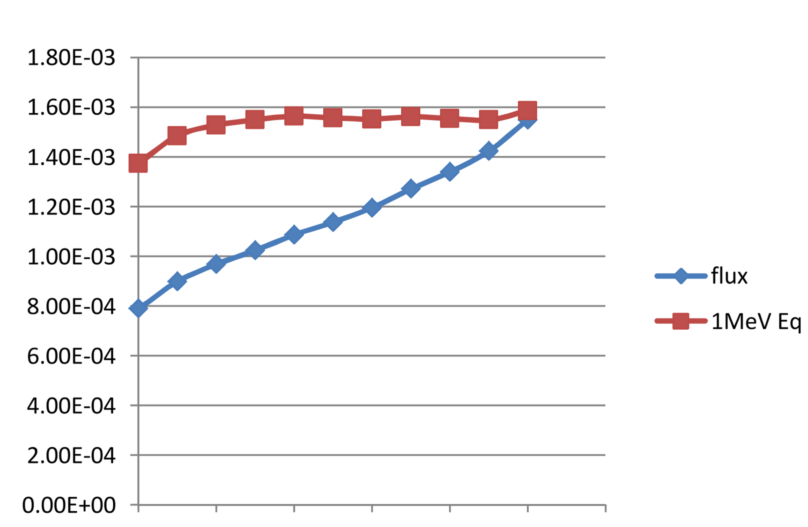

Figure 4. Plot of raw flux (blue) and 1MeV equivalent flux (red) as a function of slab thickness for detectors 10cm away.

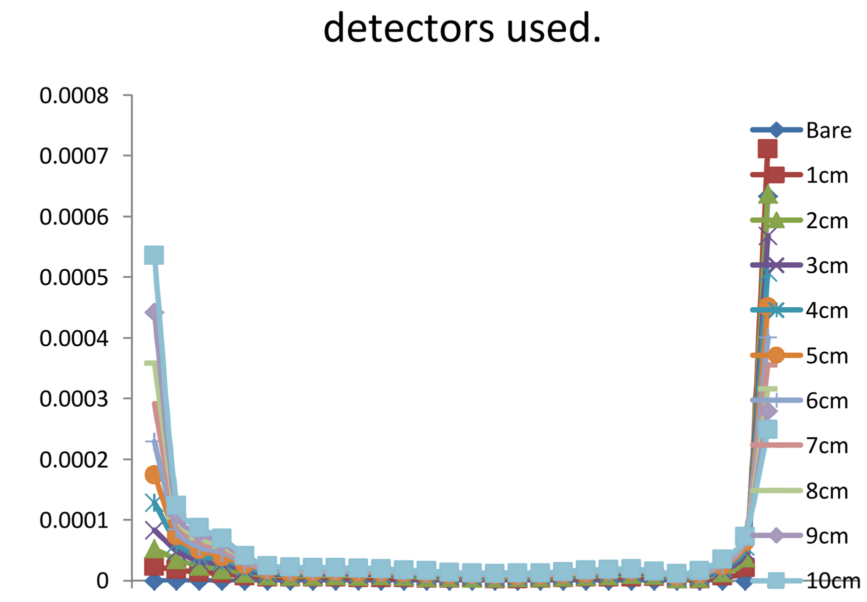

Figure 5. Plot of the fluxes as a function of energy with each thickness superimposed upon the graph.

Results and Future Work

The series of runs for neutron multiplication in Beryllium ended up being largely successful. At the maximum, raw neutron flux achieved a multiplication factor of 1.96 while the 1MeV equivalent flux had a maximum multiplication factor of 1.15. Hopefully, this will prove to be results enough for those in charge to use D-T shots in the Z-Machine instead of the currently used D-D shots. In addition, analysis of the energy spectrum of neutrons shoed that for most thicknesses, the majority of the flux came from 14MeV neutrons, meaning most passed straight through the slab. Neutrons of energies up to 2MeV also made a notable contribution while neutrons of energies between 2 and 14MeV were largely negligible. Perhaps one of the most important things to take away from the simulations was the differences between the detectors in the cross formation. Each section responded differently to changes in thickness and yielded different raw fluxes, 1MeV equivalent fluxes, and flux multiplications as a result. This data can be analyzed and used to create a geometry for the Beryllium shield that would transform a uniform radial flux from the point source into a flux that is uniform along a Cartesian direction. Doing so would allow for experiments to be done that would irradiate, say a piece of circuitry, uniformly to see how each part reacts to an equal incident neutron flux.

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