Modeling Cd-based Neutron Detector Configurations

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ABSTRACT

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With the global supply of He-3 in short supply and an increased need for neutron detectors, Cd-based detectors provide an attractive alternative. I have done research on various possible designs using Monte Carlo methods to optimize detectors for efficient neutron capture.By examining Cd-based detectors with the highest neutron capture rate will be better able to distinguish neutrons and gamma rays in order decrease the possibility of false neutron detection.

Keywords: MCNP,Monte Carlo Methods,Input Builder,crosstalk

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Chapter 1

Introduction

1.1 Applications of Neutron Detectors

By far the most-used application of neutron detectors is in national security. Detectors already in use help to detect such threats as weapons of mass destruction, dirty bombs, and nuclear materials which may be used in the construction of such devices. These are used in detecting such items in the United States and abroad. Along with gamma ray detectors, these provide a vital level of security, lending increased ability to find the most potentially hazardous materials that terrorist organizations may attempt to smuggle through these locations [1].

The danger of nuclear weapons falling into the hands of terrorist has been acknowledged as the greatest threat to global security in our time. Especially since the 9/11 attack on the Nation there has been an increased need to monitor sea ports, airports, and border crossings for nuclear materials such as uranium and plutonium.

1.2 Need for new kinds of Neutron Detectors

Neutron detectors made with He-3 are currently used in radiation monitors at ports, airports, and border crossings. These, along with He-3 detectors in use for various other application, represents a large fraction of the neutron detectors currently in use.

Besides neutron detection, He-3 is also important in other fields. Among those, cryogenics below 1K, magnetic resonance imaging of lungs, and ring lasers, important in missile guidance and space navigational systems, require the use of He-3–there is no known substitute [2].

With such great need for neutron detectors, the global He-3 supply has become greatly reduced, while demand has likewise skyrocketed. Considering the additional need for He-3 in other application, a replacement needs to be found. Because neutrons do not carry charge, a detector must include a material which can produce a charged particle when interacting with neutrons. There is only a small number of such materials known at this time [3].

To replace the He-3 neutron detectors, we are exploring the potential of Cd-based neutron detectors. In combination with a plastic scintillator it is possible to detect the presence of low-energy neutrons, such as would be produced from a uranium or plutonium source. By finding a configuration with an optimal rate of neutron capture, we hope to be able to build a detector which can perform nearly as well as the current He-3 detectors, if not better.

1.3 Cd-based Neutron Detectors Mechanisms

All the Cd-based neutron detectors we are researching share some commonalities. They all consist of a head where neutrons coming in can interact to produce photons, and a photomultiplier tube (PMT) which detects photons and produces a signal which can be read by a computer. We employ a signal digitizer which converts each pulse into a digital string with voltage sampled at 4 nanosecond intervals. This head is made of sheets of cadmium and plastic scintillator, a plastic which is used to create photons when radiation passes through it. These photons are then detected in the PMT.

When a neutron first enters a Cd-based neutron detector it interacts with the hydrogen atoms in the plastic scintillator. As the neutron interacts with protons in the hydrogen atoms it loses speed. The energy lost to the hydrogen is then converted into light which can be detected by the PMT. Eventually, the neutron is slowed to a point where the capture cross section in Cd-113 is high.

When a neutron is captured in the Cd-113, the atom becomes Cd-114 and enters an excited state. Shortly after the capture, the newly formed Cd-114 atom deexcites, creating a second gamma pulse. These gammas interact with the scintillator, producing a second light pulse which can be detected by the PMT. By observing this double pulse one can infer that a neutron was captured. By improving the configuration for optimal neutron capture rate we will be able to increase our ability to detect signals from weak neutron sources. Since the proton recoil pulse and the cadmium capture pulse are correlated in time, we can determine the rate of accidental events produced by two gammas that happen to enter the detector in quick succession.

1.4 Short Introduction to Monte Carlo Methods

Monte Carlo methods refer to a set of methods of using random-number generation to create statistical data. These methods represent a physical system in which a random variable is representative of some physical constant. Especially in programs like the Monte Carlo Neutral Particle program (MCNP) where the physical property generated is inherently random and well documented, using Monte Carlo methods is useful. MCNP includes these methods to produce statistical data which can be used to determine relative efficiencies of different detector configurations [4].

```
3wedge.txt - Notepad
File Edit Format View Help
Generic MCNP Title
C     Cells
1     0   1
2     0    -1 (2:-4:5)
3     1  -8.65 -2 3 4 -5 6 -7 8
4     1  -8.65 -2 3 4 -5 7 -8 9
5     1  -8.65 -2 3 4 -5 -6 -10 11
6     2  -1.032 -2 3 4 -5 7 8
7     2  -1.032 -2 3 4 -5 -9 10
8     2  -1.032 -2 3 4 -5 -11 -6
9     1  -8.65 -3 4 -5

C     Surfaces
1     so  1000.0
2     cz  6.35
3     cz  0.01
4     pz  0.0
5     pz  15.24
6     py  -0.0050
7     py  0.0050
8     p  1.7320508075688783 1.0 0.0 0.010000000000000005
9     p  1.7320508075688783 1.0 0.0 -0.010000000000000005
10    p  -1.7320508075688754 1.0 0.0 0.009999999999999992
11    p  -1.7320508075688754 1.0 0.0 -0.009999999999999992

mode  n                        $  Neutron Mode
m1     48000.     1.0
m2     1001.      0.523      6012.       0.474
imp:n  0 1 7r    $  Importances (0 beyond outer sphere)
tmp1   2.55e-008        8r    $  Room Temperature (eV)
mt2    benz.60tsdef  POS=0.0 0.0 -100.0 CEL=2 ERG=D2 WGT=1 TME=0    $  Source information
C     Source info goes here!
si2    H 0.015 0.035 0.055 0.075 0.095 0.115 0.135 0.165 0.195
       0.225 0.255 0.305 0.355 0.405 0.455 0.505 0.555 0.605
       0.655 0.705 0.755 0.805 0.855 0.905 0.955 1.05 1.15
       1.25 1.35 1.45 1.55 1.65 1.75 1.85 1.95 2.15
       2.35 2.55 2.75 2.95 3.25 3.55 3.85 4.15 4.45
       4.75 5.05 5.55 6.05 6.55 7.05 7.55 8.05 8.55
       9.05 9.55 10.05 10.55 11.05 11.55 12.05 12.55
       13.05 13.55 14.05 14.6 15.9 16.9 17.9 19.1 20.0
sp2    D 0 1.98e-3 2.64e-3 3.13e-3 3.54e-3
       3.88e-3 4.18e-3 6.75e-3 7.25e-3 7.67e-3
       8.04e-3 1.41e-2 1.48e-2 1.54e-2 1.58e-2
       1.61e-2 1.64e-2 1.66e-2 1.67e-2 1.68e-2
       1.68e-2 1.68e-2 1.68e-2 1.67e-2 1.66e-2
       3.12e-2 3.23e-2 3.15e-2 3.07e-2 2.98e-2
       2.88e-2 2.77e-2 2.67e-2 2.56e-2 2.46e-2
       4.60e-2 4.20e-2 3.81e-2 3.44e-2 3.10e-2
       4.07e-2 3.44e-2 2.89e-2 2.42e-2 2.02e-2
```

Figure 1.1 A file for a Cd-based detector with three wedges of plastic. Note the separation into sections for cells, surfaces, and everything else.

1.5 Overview of MCNP

The Monte Carlo Neutral Particle program (MCNP) was originally developed at Los Alamos National Laboratory (LANL). MCNP is based on the Fortran programming language, and uses lines of up to 80 characters to describe the geometry, materials, and other parameters used in its calculations. Materials may be described either by the atomic composition, or by using built-in information which MCNP includes for some materials.

Files read by MCNP are structured into three sections. In trying to understand how these work, it is useful to start with the second section before trying to understand the first and third sections. In the second section are numbered surfaces, three-dimensional objects such as planes, cylinders, and spheres. These are used as boundaries for the object being designed. In the first section the cells are defined. A cell is a volume defined by intersections and unions of the space on one side or the other of one or more surfaces. A cube could be defined as the space to the right of one plane and to the left of another plane to the right of the first, etc. Each cell is homogenous. The third section defines everything else. This includes what materials are included, the temperatures of the cells, information about the radioactive source, and instructions on what data to take.

Among its many applications, MCNP may be used to approximate neutron capture probabilities. By adjusting the parameters in the third section of the file, the result can be made to correspond roughly with the actual probability of capture. This is useful for comparing the capture rates between different geometries.

1.6 Reasons for Using Computational Methods

Although experiment is the most accurate method for determining the effectiveness of a neutron detector design, there are good reasons for using computational methods as well. One advantage of computational methods is that one can compare many potential designs without using as many resources: just a computer with the proper software is required. Especially for expensive materials, computational methods allow for better management of funds. By running files on many similar, competing models, the best choice becomes more obvious and less the product of guesswork.

A second advantage to computational methods is their speed. While this may not always hold for simpler experiments, as the complexity increases the difference becomes more important. A bug in a program may be difficult to find, but it is often a much simpler thing to fix than a problem discovered after setting up a physical experiment. Additionally, in experiments with neutron detectors the experiment often must run for hours, whereas results can be obtained and analyzed in as little as 30 minutes using MCNP.

1.7 Purpose of the Research

In trying to find the most efficient design for capture, I first explore possible layouts. When starting my work, the Cd-based detector used by our group was made of rectangular prism slabs of plastic of varying lengths such as to roughly make a cylinder with sheets of Cd between. Thus it was natural to include this configuration as one of the designs for simulations (with the modification of using slabs in essence rounded off into a perfect cylinder for the purpose of simplifying the geometry of the build). The alternative to this layout is similarly made in a cylinder of the same proportions, but with wedges of plastic in place of slabs, the hope of this being that such a configuration would prove superior optically by assuring a more evenly distributed illumination of the photomultiplier tube used in the actual detector. Thus one aim of my project has been determining if one configuration or the other varies significantly in neutron capture.

In addition to the effects of the configuration of a single detector, I also study the effects of crosstalk, in which neutrons enter one detector, leave, and are then captured in another detector. In order to understand this, I am running simulations on a combination of detectors as well as single

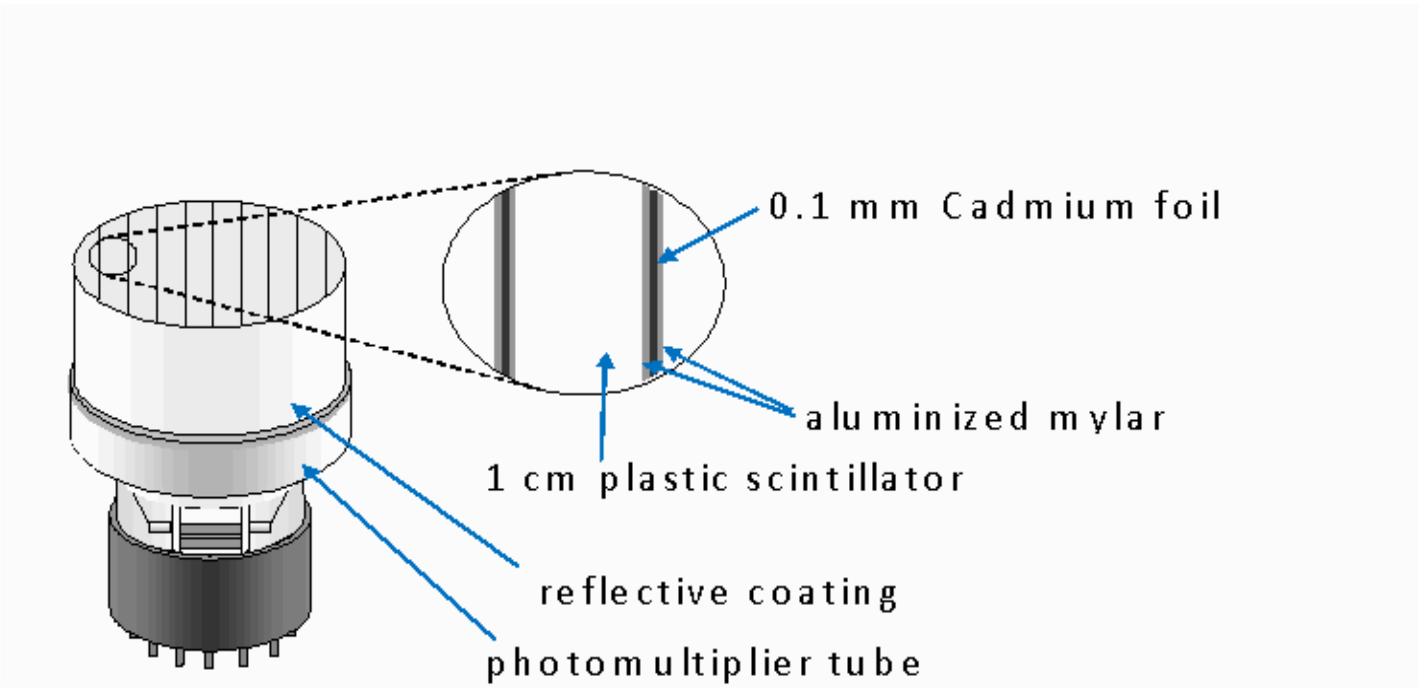

Figure 1.2 The configuration for a basic slab detector, as worked on by Christopher Busselberg and Nathaniel Hogan.

detectors. Through these simulations I will be able to determine which configurations provide the highest neutron capture rate for a fixed amount of material.

1.8 Previous Work on Similar Problems

The effort to optimize Cd-based neutron detectors using the Monte Carlo Neutral Particle Transport Code (MCNP) had already begun when I began my research. Previous to my work, Christopher Busselberg studied the effect of varying the thickness of Cd layers in detectors consisting of rectangular prisms of plastic scintillator separated by a layer of Cd. This effort helped to show that varying the thickness of Cd had virtually no effect on neutron capture, meaning that a thinner layer could be used to save on resources [5]. With this knowledge, I have used the same Cd thickness in my own simulations.

While I am still working to optimize the design, actual Cd-based detectors have already been built. Through his research Nathaniel Hogan has previously constructed and taken measurements on a detector based on the slab detector designed by Christopher Busselberg. These results support the possibility of using Cd-based detectors to replace current He-3 detectors [6].

Chapter 2

Methods

2.1 MCNP File Creation and Analysis

The Monte Carlo Neutral Particle Transport Code (MCNP) has been developed by Los Alamos National Lab as a program which, given a properly designed input code, simulates neutron capture. These files consist of information separated into lines, refered to as cards. These cards are then grouped into three sections, categorizing the information contained in each section's cards. Comments may be included at the end of any card or on their own card, but otherwise all MCNP files are made in the same way.

The first section always begins with a card for the title, followed by cards describing the cells. Cells are the three-dimensional pieces which define the entire universe that MCNP tries to replicate. In order for the file to function, everything must be inside a cell: there needs to be included a cell with infinite volume, representing the space outside the area of interest where neutron capture takes place (at least, where it is no longer significant to the detector), as well as the geometry of the detector and space immediately around it. Any neutron passing this area is forever lost to the program, representing all neutrons not captured by the detector. So long as MCNP determines that a neutron remains outside of this infinite cell, it can react with the cells around it.

The second section is composed of cards defining surfaces, boundaries which separate cells from one another. These must be formatted so that they can be interpreted by MCNP as planes, cylinders, spheres, and other geometric shapes.

The final section contains the cards telling MCNP how to handle the information provided. In my research, this includes defining which materials are involved, which cells in which to record capture events, capture information for relevant materials (in case of my research this means cadmium), and directives regarding when to finish analysis.

As long as the file is put together properly, MCNP will read an input file and create an ouput file. This output file includes a probability of capture for any cells which the input specified to be monitored. This can then be compared with the probabilities of other detector designs to determine relevant efficiencies between different systems.

2.2 Development of the InputBuilder Program

When using MCNP it quickly becomes apparent that creating a working file is more difficult than necessary for our work on detectors. Indeed, the layout of the file is unintuitive, a far cry from many of today's higher-level programming languages. Finding errors can be a long and difficult process, especially if one is unfamiliar with the manner in which MCNP reports errors. If there is some space defined by more or less than one cell, even if overlapping cells are made of the same material, the file will not run and it can be difficult to determine where the problem lies as one must parse through a large, cluttered text file to find the surface and cell where a particle will be reported lost. Even after finding this information, it is still possible that the cell in question is properly defined, but that the error lies with the geometry of a different cell.

In addition to the difficulty in making files for MCNP, for my research the work consists of making many similar geometries. Optimizing detector designs is repetitive, every file varying only slightly from the others. Furthermore, facing the problem of examining multiple detectors this problem was greatly magnified.

Considering both of these problems, I decided to create the InputBuilder program. The InputBuilder provides a graphical interface for creating MCNP files. Rather than entering line after line of information, the user enters information in a more organized, logical manner. The program changes the focus from the cells and surfaces of an MCNP file to pieces and containers, homogeneous and heterogeneous geometries which are built only once, then saved to be added into future files. These pieces and containers can even be translated and rotated as needed. Further, given that our work focuses on variations of two specific geometries (slab and wedge detectors), options are given to generate these as containers by simply including the number of slabs or wedges, thickness of the Cd, the dimensions of the cylindrical detector, and the default orientation of the detector. After making one of these containers, the user need only select which materials to use (again here the user enters a material once to be recalled at any future point), the location of the radioactive source, and the radius of a bounding sphere containing all the cells of interest.

The InputBuilder program has been designed in such a way that many of the mundane tasks which required simple yet nevertheless easy-to-overlook changes to the code are taken care of. This in turn reduces the number of attempts required to create a working file. Furthermore, as the designing of slab and wedge detectors has remained consistent in the field, special attention has been given to automate the task of creating these two basic designs. Instead of having to calculate the various planes and cylinders, the program only requires that the user enter the number of slabs or wedges, the dimensions of the cylindrical detector, and the thickness of the sheets separating them.

In addition to these benefits for developing single detectors, the individually designed detectors can be copied and combined; e.g. a user can make a detector with 20 wedges, then by simply calculating where each detector should be centered create a file to simulate a seven-fold detector of seven cylinders each having 20 wedges. All this means that a much greater number of files can be created in a much shorter time, comparing many similar designs in a relatively short period.

Chapter 3

Data and Analysis

3.1 Data Generated

As shown in Fig. 3.1, the data generated show capture probabilities ranging from 6% and 31%. At low numbers of slabs or wedges the data show that efficiency increases appreciably as the number of elements increasy, while after around 20 slabs or wedges the increase in capture rate diminishes, quickly flatlining. Also as the number of slabs and wedges increases, the probabilities become closer between the configurations. This pattern is also shown in Fig. 3.2, detailing two sixfold and a sevenfold configuration with slab detectors.

3.2 Comparison and Analysis

From the data generated, an optimum number of slabs or wedges becomes apparent (see Fig 3.1). In both the case of slabs and of wedges, the probability of capture increases until around 20. Increasing the number of slabs or wedges from this brings no significant increase in neutron capture. This trend holds both for the single detectors and the fourfold configurations for which experimental data were taken.

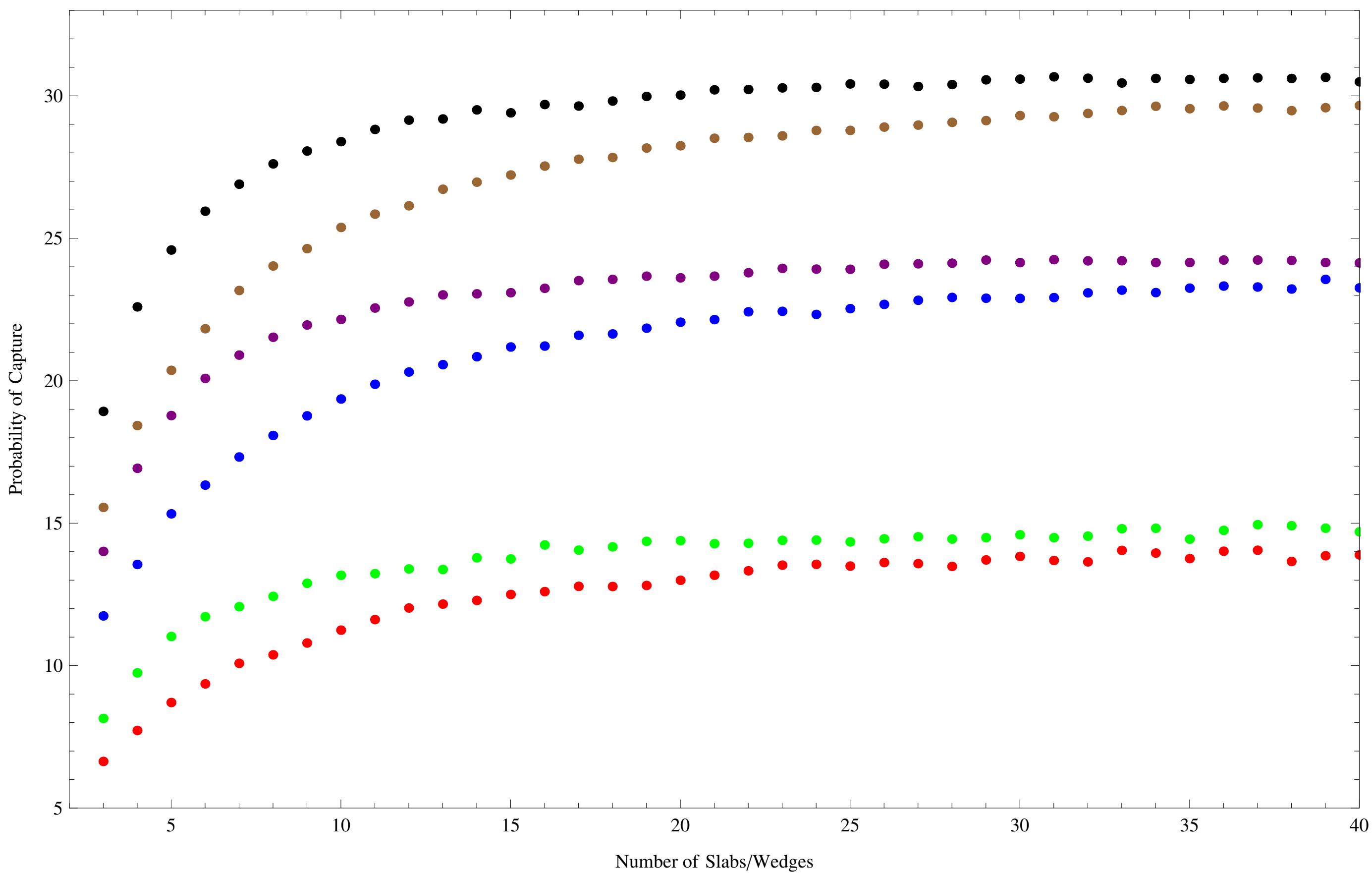

Figure 3.1 Probability of neutron capture for varying number of slabs and wedges in a single detector, in one detector in a fourfold configuration, and in one detector in a sevenfold configuration. The fourfold slab (purple) and wedge (blue) detectors have significantly better capture than the single slab (green) and wedge (red) detectors, and the sevenfold slab (black) and wedge (brown) provide better capture still.

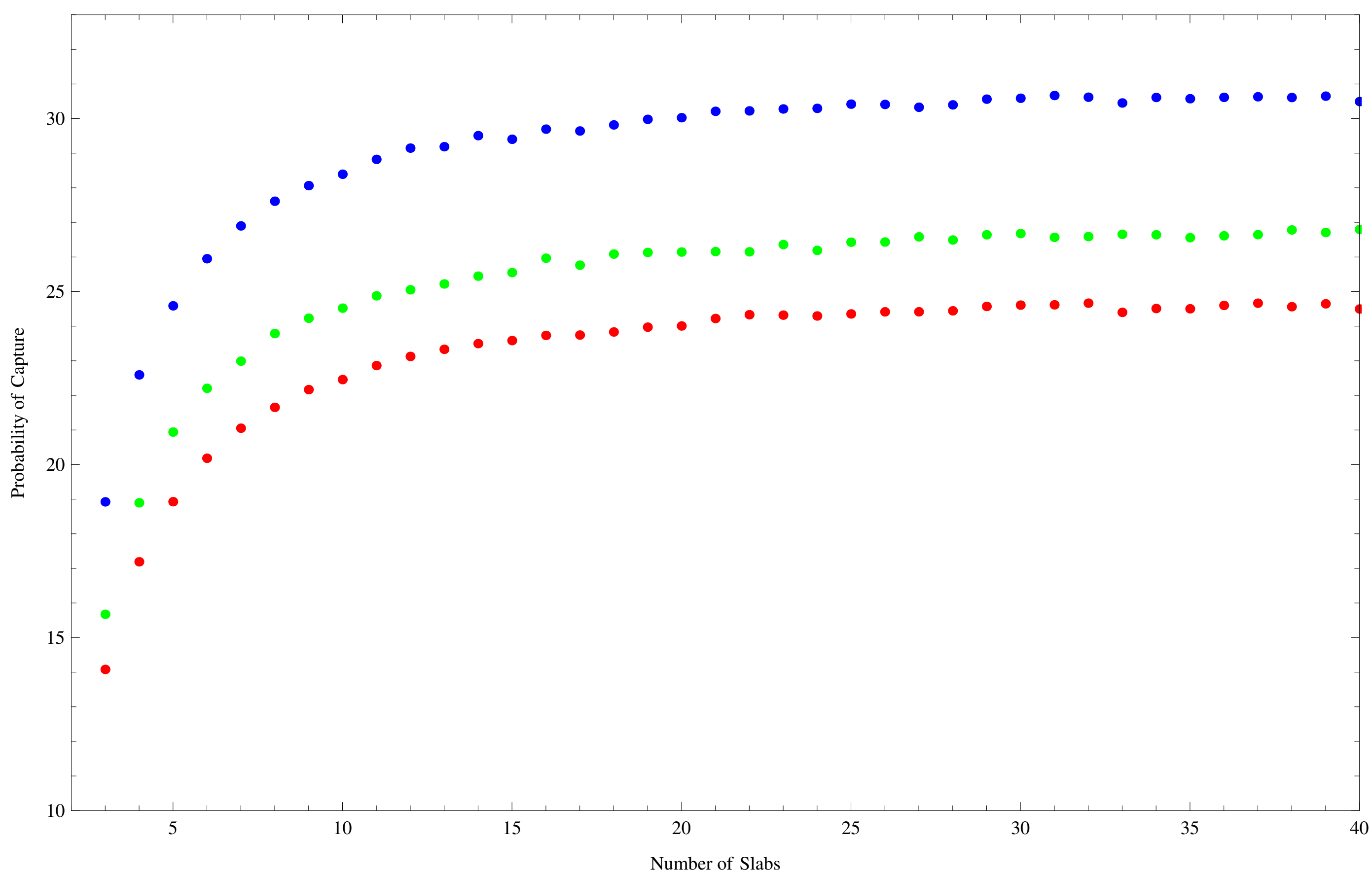

Figure 3.2 A plot comparing the relative capture rates per detector in three configurations: a ring of six slab detectors (red), six slab detectors with one central detector and five spaced equally around the center detector (green), and a group of seven detectors, in the same configuration as the ring of six detectors but with an additional detector in the center (blue).

The data also show the relative efficiency of slabs versus wedges in capturing neutrons. Here the difference begins at nearly 2.5%, but the discrepency decreases as the number of slabs and wedges increases. Comparing a detector with 20 wedges and a detector with 20 slabs the difference decreases to less than 1.5% capture. As we anticipate the wedge to have better optical properties when used with a photomultiplier tube, this difference isn't enough to make the use of slab detectors necessarily preferable.

While the difference between the slab and wedge detector is negligible, the advantages of using a fourfold design have a greater impact. When comparing the results for a single 20 wedge detector versus one with four detectors, the fourfold detector has an advantage of over 9% capture rate. The sevenfold detector has an even higher rate than the fourfold design, increasing the capture rate another 6% over the fourfold detector, effectively doubling the capture rate for a single detector. Apparently the effect of crosstalk, in which a neutron entering one detector leaves and enters another, has a large impact on the average capture rate for a single detector.

In Fig. 3.2 three slab detector designs are compared to study the effects of positioning on capture rates. Here two sets of six detectors are arranged in a ring of six detectors and a group of five detectors surrounding a single detector. Simulations predicted a capture rate increase of between 1.5 and 2.5% in capture rate for the more closely grouped configuration of five detectors surrounding a single detector. Capture rate increased dramatically for a configuration similar to the sixfold ring design with a seventh detector added at the center of the ring, suggesting that crosstalk plays an important role in a detector's ability to capture neutrons.

3.3 Significance for Similar Research

As shown perviously in Sec. 3.2, Cd-based detectors can have their design optimized for greater neutron capture. This fact can therefore be used in further attempts to build and improve neutron detectors based on Cd capture. By using optimized designs the ability to discriminate between neutrons and gamma rays is expected to increase, therefore providing hope for a viable alternative to current He-3 detectors.

Perhaps the most significant result for future work with Cd-based neutron detectors is the effect of crosstalk. The data show that having more than one detector nearby greatly increases the probability of neutron capture in a detector. This result can be applied in similar research as a way of increasing capture efficiency of any detector, thereby increasing the chances of discriminating neutron and gamma pulses. This greater sensitivity could also be used for detection of materials which emit only a small amount of neutrons over a period of time.

3.4 Conclusion

As shown in Fig. 3.1, the configuration of Cd in a neutron detector has a significant impact on the detector's ability to capture and therefore detect neutrons. Given the current need for neutron detectors, both in global security and other fields, it is critical that designs be made in as efficient a manner as possible.

There is stilll more to be explored in the search for more efficient neutron detectors. By using the InputBuilder program I have designed in conjunction with MCNP, the way forward has been prepared for testing a variety of designs not explored in my work. This will in turn lead to even better neutron detectors to be developed in the future.

By implementing the findings I have discussed here, the solution to the He-3 problem may finally be coming to light. By building more efficient detectors out of Cd and other materials, it seems very likely that a replacement for the He-3 detector may soon be found.

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