Improving Cadmium Capture-Gated Neutron Detection Techniques

Andrew K Hoffman

A senior thesis submitted to the faculty of
Brigham Young University
in partial fulfillment of the requirements for the degree of

Bachelor of Science

Dr. Lawrence B Rees, Advisor

Department of Physics and Astronomy

Brigham Young University

August 2012

ABSTRACT

Improving Cadmium Capture-Gated Neutron Detection Techniques

Andrew K Hoffman
Department of Physics and Astronomy
Bachelor of Science

We have improved on our original design for a cadmium capture-gated neutron detector by designing a "wedge" detector. This new design allowed for more consistent pulses for the entire face of the attached photomultiplier tube. Testing at the Ohio University accelerator facility showed efficiency of this new detector lies around 12%. Testing was also performed with bismuth as a gamma shield to reduce the number of accidental events recorded. Three inches of bismuth proved to reduce the amount of accidental pulses recorded by the detector by a factor of 10.

Keywords: neutron, detection, shielding, radiation

ACKNOWLEDGMENTS

Special thanks to Dr. Rees, Dr. Czirr, and John Ellsworth for their patience, teaching, training, and wisdom without which I could not have accomplished any of my research. Thanks to Adam Wallace and Nathan Hoggan who mentored me in the lab for countless hours. Thanks to my high school physics and calculus teachers Tony Wayne and Carla Hunt who inspired me to become who I am today. Very special thanks to my mother who always believed in me and my scientific aspirations without whose support I could never have accomplished any of my educational goals.

Parts of this work were funded by NNSA Grant no. DE-FG52-10NA29655 and DHS Award no. 2010-DN-077-ARI039-02.

Contents

Chapter 1

Background

1.1 Introduction

Since the discovery of nuclear fission in 1938 our knowledge of nuclear physics has vastly increased. With an increased knowledge of nuclear physics came a variety of applications including power operations, nuclear weapons, and nuclear medicine. These applications also brought concerns of radiation exposure problems and the threat of nuclear terrorism.

To deal with these problems scientists have been developing various radiation detectors for decades. One of the difficulties in radiation detection is being able to discriminate between types of radiation. In some applications, different types of radiation produce differently shaped pulses. Although pulse shape discrimination can be extremely useful in many applications, being able to build a detector that is sensitive to only one type of radiation source would be extremely useful.

In this paper I will discuss our efforts to build an efficient neutron detector that is acceptably gamma insensitive. The applications for such a detector could include use as portal monitors to detect smuggled nuclear weapons at border checkpoints, radiation exposure monitoring, and a wide variety of experiments that could help us better understand the strong force between nuclei.

1.2 Compton Scattering

Compton scattering is crucially important to radiation detectors that use scintillating materials. In Compton scattering, gamma rays produce light in a scintillator by colliding elastically with electrons within the material. When such a collision occurs, the electron gains kinetic energy and the gamma photon scatters with a much longer wavelength. The electron then transfers its energy to activator ions within the scintillator to produce light. The type of scintillating material determines the energy of the photons released.

1.3 Proton Recoil

Because most detectors are inefficient at detecting energetic neutrons, it is important to slow down a neutron before it reaches the capture material. This process is called moderation. Most moderators use materials high in hydrogen content. This is because of hydrogen's large scattering cross section and because its mass is comparable to that of a neutron. Most designs for neutron detectors use some form of plastic moderator which gains its high hydrogen content from the long hydrocarbon chains in the material. When a neutron enters the material, it collides elastically with hydrogen and loses a significant amount or all of its kinetic energy. This proton will then transfer energy to the scintillating component of the detector releasing a photon (similar to Compton scattered electrons). The type of scintillating material determines the energy of the photons released.

1.4 Neutron Cross Sections

Neutron cross sections describe the probability that a neutron will interact with a nucleus. [1] Cross sections are typically measured in barns (10^{-28} m^2) and are defined as:

$$\frac{\# \, of \ interactions}{\# \, of \ incident \ neutrons \times \# \, of \ nuclei \ per \ unit \ area}$$

For neutron detection it is important to consider both the scattering and the absorption cross sections. The scattering cross section describes the probability of a neutron being scattered from a particular nucleus. The absorption cross section describes the probability of a neutron being absorbed by a nucleus. Ignoring the effects of nuclear structure, low energy cross sections usually decrease with an inverse relationship to the velocity of the neutron ($\sigma \propto 1/v$). There are also resonant peaks which occur at certain neutron energies.

Most neutron detectors use some type of material which has a high absorption cross section because this is the easiest way to detect an interaction with the neutron. Table 1.1 illustrates this by showing the absorption cross sections of Helium-3, Lithium-6, Boron, Cadmium, and Gadolinium (all common materials used in neutron spectrometry).

Scattering cross sections are also important in determining materials which supplement the detector. In order to slow down the neutrons so that they can be captured, a moderator is required which will allow the neutrons to collide several times, imparting some of its kinetic energy upon each collision. In addition to being relatively the same mass as neutrons, hydrogen has a very high scattering cross section ($\approx$ 82 barns) allowing it to slow down neutrons effectively. Most detectors will use some type of plastic moderator which is effective due to the high hydrogen content of the hydrocarbons contained within the material. One way of making a detector less sensitive to gamma radiation is to employ shielding. While considering gamma shielding it is important to find materials with small absorption cross sections to reduce the number of neutrons lost within the shielding. Lead and Bismuth are both excellent gamma shields because of low absorption cross sections but high densities.

Material	Absorption Cross Section in Barns
He-3	5333
Li-6	940
B	767
Cd	2520
Gd	49700
Pb	0.171
BI	0.0338

Table 1.1 Absorption cross sections of nuclei used in detectors and in shielding. [2]

1.5 Gamma Insensitivity Criteria for Department of Homeland Security Applications

For use in radiation portal monitoring, the U.S. Department of Homeland Security has defined criteria for gamma insensitivity of a neutron detector. Gamma insensitivity is important to distinguish an actual neutron source from background radiation such as radium in building materials or an individual that has been received medical gamma treatments. The criteria were created to ensure that a strong gamma source will not set off an alarm.

The first requirement is that the detector must have an absolute neutron efficiency of at least 2.5 $\frac{cps}{ng\ of\ ^{252}Cf}$from a source located 2 m from the detector and encased in 1 inch of polyethylene shielding. The second requirement is the intrinsic gamma efficiency should be less than or equal to 10^{-6} while being irradiated with a source emitting at 10 $\frac{mR}{h}$. This is defined as the number of gamma pulses recorded per number of radiation quanta that reach the detector in the presence of only a gamma source. The third requirement is that the "gamma absolute rejection ratio for neutrons" (GARRn) must be within .9 and 1.1 for an exposure of $10\frac{mR}{h}$. The GARRn is the absolute neutron detection efficiency in the presence of both a neutron and gamma source divided by the absolute efficiency in the presence of only a neutron source:

$$GARRn = \frac{\varepsilon_{abs\ \gamma n}}{\varepsilon_{abs\ n}}$$

To use our detectors for this application, we are attempting to reduce gamma sensitivity while still maintaining an acceptable neutron detection efficiency [3].

1.6 MCNP Computations

To test detector designs without actually building a detector, we use the computer code MCNP (Monte Carlo Neutral Particle) developed by Los Alamos National Laboratory. The detector is described by cells bounded by planes and cylinders. Each cell is also described by a material card giving its chemical makeup. A neutron source is defined with a certain geometry and by numbers of neurons emitted in specified energy bins. Every neutron then has a probability of interacting with each cell it comes into contact with, and all the interactions are tallied. The number of neutrons absorbed by a particular cell can then be added to determine the efficiency of the detector.

Chapter 2

Current Neutron Detector Designs

2.1 He-3 Detector

The helium-3 detector is the current design on which the U.S. Department of Homeland Security has focused. Its advantages are in the high thermal cross section of He-3 and its ability to be completely gamma insensitive. When a neutron is absorbed by He-3 it produces protium and tritium as well as releasing a monoenergetic gamma: ^{3}He(n,^{1}H)^{3}H*. The tritium and protium are then detected in a gas proportional detector. The main disadvantage of He-3 is the actual shortage of the isotope itself, primarily being produced in nuclear reactors and disassembled nuclear weapons. [4]

2.2 BF_3 Detector

The BF_3 detector is similar to the He-3 detectors. Since boron is not naturally gaseous, it is converted to a gaseous boron tri-fluoride. Typically it is enriched with B-10 in order to increase efficiency. The B-10 absorbs a neutron releasing an alpha and becoming Li-7 which then de-excites by releasing a gamma: ^{10}B(n,α)^{7}Li*. The alpha particle is then detected by a gas proportional tube. The advantages of the BF3 detectors are that they are cheap to make and can use the same electronics as current He-3 detector units as well as being even less gamma sensitive than He-3 tubes. This comes at a steep price; however, because they are much less efficient than He-3 due to their smaller cross section as well as being toxic and corrosive. They also require larger tubes to be effective. [4]

2.3 Boron-Lined Detector

The boron lined detectors are based on the same reaction as the BF_3 detector, only the boron is not in a gas form. It relies on several gas proportional tubes lined with boron-10. The main advantages are that the tubes can be modified with any gas configuration and the boron is not toxic and corrosive in this form. Its disadvantages come from the instability of pulses generated because some energy is lost while traveling through the tube itself. It also tends be more gamma sensitive and requires several tubes in order to have an effective amount of boron to interact with the neutrons. [4]

2.4 Li-6 Glass Detector

The lithium-6 glass detector is a detector designed with a neutron interacting material (Li-6) built into a glass scintillator. A neutron is captured by the Li-6 which then splits apart into tritium and an alpha: ^{6}Li(n,^{3}H)^{4}He. These charged particles then scintillate within the glass producing a light pulse that can be detected by a photomultiplier tube. The advantage of the Li-6 detector is that is can be relatively small and does not require pressurized gas tubes. Its disadvantages come in that it is fairly gamma sensitive, has a lower absorption cross section (reducing its efficiency), and the glass itself is expensive to manufacture. Our research group is currently working on designs and analysis in order to increase the efficiency of this detector while lowering its gamma sensitivity. [5]

2.5 Cadmium Capture-gated Detector

The cadmium capture-gated detector relies upon two different interactions with a neutron. The first interaction is an inelastic collision between a neutron and a proton within a scintillator material. The scattered proton will then cause a light pulse to be produced by the scintillator causing an initial pulse in the detector. The second interaction occurs within a thin sheet of cadmium foil, in which a neutron is captured in a cadmium nucleus which then decays from an excited state by releasing several gammas: ^{113}Cd(n,γ)^{114}Cd.These gammas then Compton scatter within the scintillator material causing a second pulse in the detector.

Our initial design for the cadmium detector was with slabs of EJ-200 scintillator, aluminum mylar, and cadmium foil. The slabs were then arranged across a PMT as seen in figure 2.1. In between each slab of scintillator is cadmium foil 0.1 mm thick placed within two sheets of aluminized mylar. The mylar is used to increase the internal reflection within the slabs in order to increase optical efficiency. We then looked for two correlated pulses from the PMT that could represent recoil and capture events from a single neutron. [6]

Although the initial cadmium slab detector did detect neutrons, the efficiency was lower than what we had hoped. One of the problems we proposed was due to the way the light travels within the slabs. Each slab is placed in a different location on the phototube meaning that the central slabs let light travel to the middle of the PMT while the outer slabs only carried light to the outer part of the PMT. This affects both the pulse size and the timing of the detection since the outside of the PMT produces a smaller pulse and has a slower response time than the center.

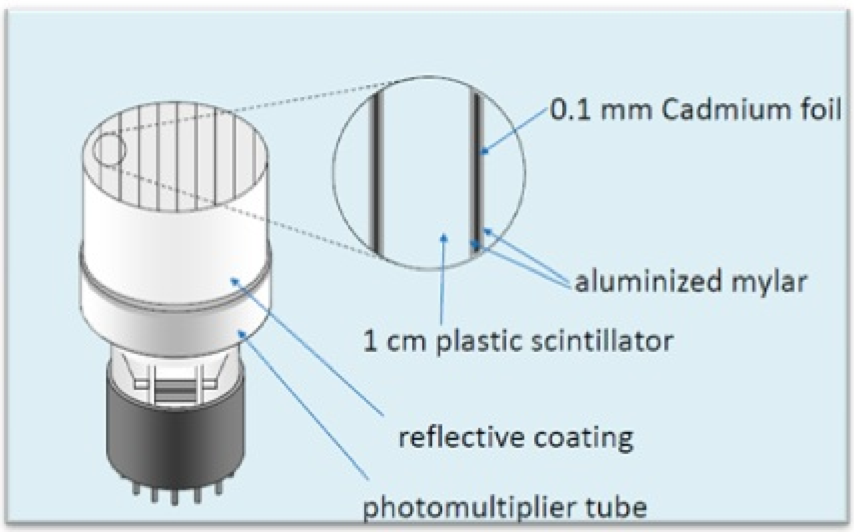

Figure 2.1 Cadmium Capture-gated "Slab" Detector

Chapter 3

Wedge Detector

3.1 Construction

In order to improve the optics of the cadmium detector, we proposed a design using scintillator wedges instead of slabs. This would help evenly distribute the light reaching the PMT. The detector is built with 21 wedges of EJ-200 scintillator with cadmium foil .1 mm thick in between the wedges. Each cadmium foil is covered by a thin sheet of aluminum mylar on both sides.

The detector is built by taking wedge pieces six inches tall with a 2.5 inch base and .788 inches in height (see figure 3.1). Each wedge is stood up and a layer of 0.1 mm cadmium foil is inserted between two pieces of aluminized mylar and placed on one side of the wedge. This mylar cadmium layer is then sandwiched between another slab and the process is repeated for all 21 plastic wedges. The wedges are then surrounded by a sheet of aluminized mylar which is taped to keep it in place. This sheet of mylar surrounds the entire detector and will also cover an additional three inches above and below the detector. After the mylar is in place and taped securely, the wedges are lined up by hand to create a flush top surface.

A hose clamp is then placed on the outside and tightened until all the wedges are securely. The

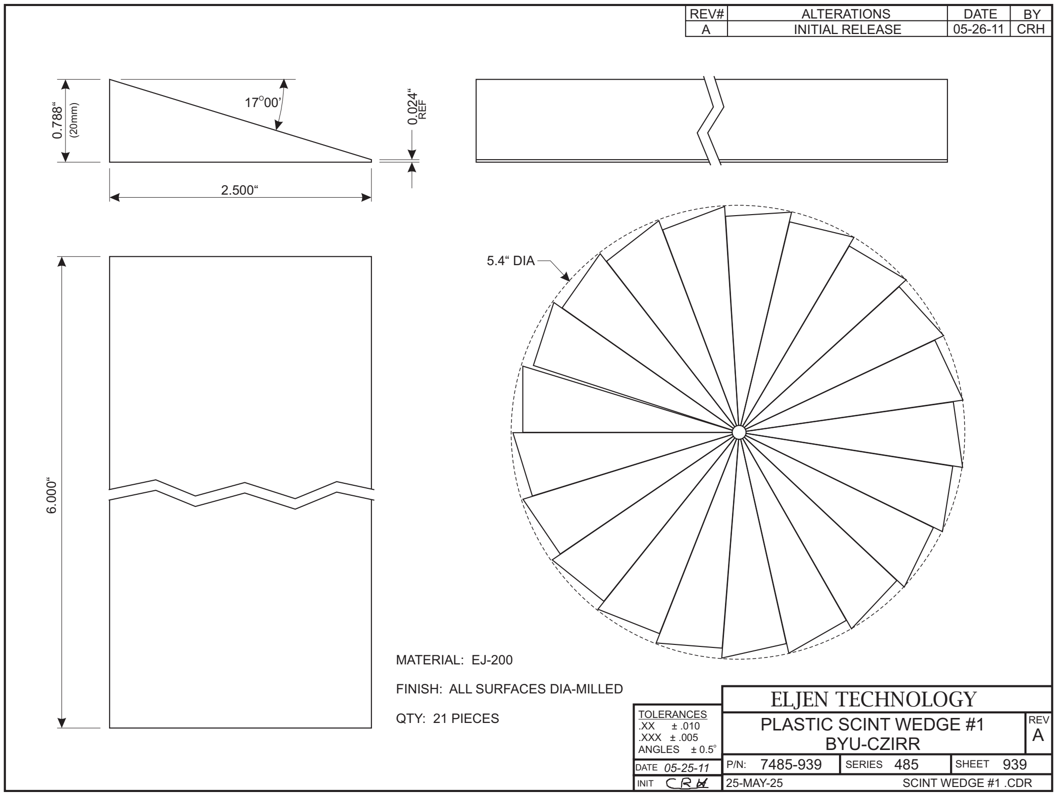

Figure 3.1 Machining specifications for the wedge detector

excess three inches of mylar on the bottom are then folded over (usually cut into an "orange peel" like pattern) and taped securely across the bottom of the detector to decrease light loss within the detector. The entire detector is then covered in about two layers of duct tape to ensure there are no light leaks. A thin layer of silicon grease is then applied to the top of the detector as well as the surface of the PMT. The PMT is then placed on the top of the detector, and twisted several times to remove the air bubbles in the grease to increase the optical clarity. After the phototube is in place, the three inches of mylar on the top of the detector is then taped flush to the phototube using electrical tape. Once again several layers are used to ensure there are no light leaks.

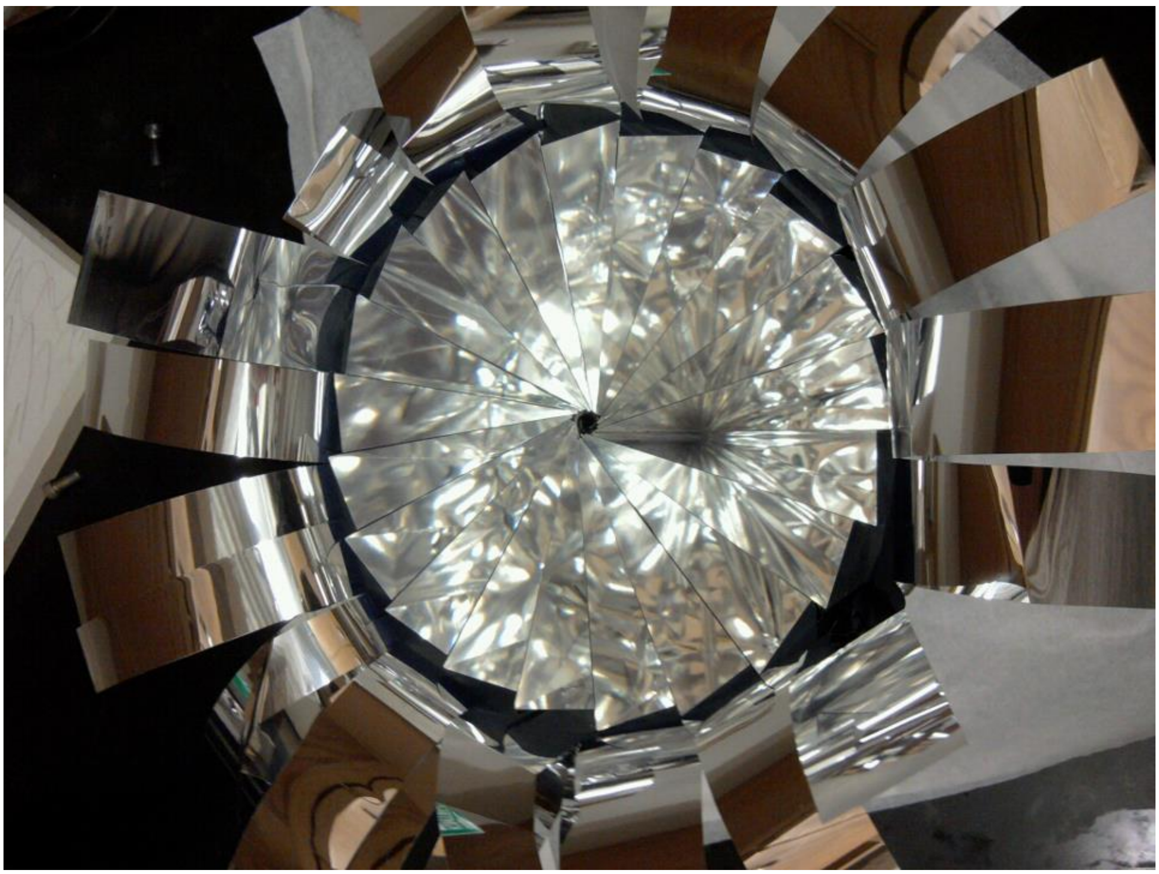

Figure 3.2 Final construction of wedge detector

3.2 Experiment Design

3.2.1 University of Ohio Accelerator

To test the efficiency of our detector at a range of energies, we took it to Ohio University's Edwards Accelerator Laboratory. Their 4.5 MV tandem accelerator sends beams into a highly calibrated tunnel. A deuterium beam from the accelerator strikes an aluminum target. The ^{27}Al will then absorb the deuteron, but is unstable with the extra neutron. The neutron is then emitted along with a gamma from the absorbed kinetic energy: ^{27}Al(^{2}H,n)^{28}Si*. The energy spectrum of neutrons in this beam is very well known, thus allowing calibration of the detector efficiency as a function of neutron energy.

The detector was placed in the center of the beam 9.729 meters away from the beam source. A Hamamatsu photomultiplier tube was used. The signals from the photomultiplier tube were sent through an amplifier and then into a Caen digitizer. Channel 0 recorded a pulse that matched the beam time and channel 1 recorded the event data. A 25 mV threshold was used to trigger an event. 16 ns before and after the pulse were recorded to find a correlating pulse. Due to technical issues and a limited amount of time for multiple detectors only one acceptable ninety minute run was obtained used the "wedge" detector.

Figure 3.3 Setup of the detector on the scissor lift

3.2.2 Scissor Lift

To reduce room return reaching the detector, we set the source and detector on a scissor lift 22' high. An ADIT PMT was used. To reduce light leaks the detector was placed in an aluminum can. A ^{252}Cf source was placed 12.75 in away from the detector. Signals from the tube were run through an ORTEC amplifier running at 2x10 gain. A timing filter was set to only record events with two pulses. A threshold was set to trigger an event at 40 mV and a second threshold to record a second pulse within the event was set to 20 mV. A sixty minute data run was taken, as well as a thirty minute background run.

3.2.3 Bi/Pb Shielding

Due to the high gamma sensitivity of the detector an experiment was proposed to test the possible use of a gamma shield to reduce the risk of a false alarm due to gamma accidentals. Since both lead and bismuth are fairly dense metals with low neutron absorption cross sections they would be able to shield gammas without drastically reducing the number of neutrons reaching the detector. Lead is denser than bismuth but has a higher absorption cross section (see table 1.1) and contains radioactive isotopes. MCNP results confirmed that lead absorbs significantly more low energy neutrons than bismuth.

Three runs were taken to determine the usefulness of a bismuth gamma shield. The first run was with 3 inches of bismuth shielding with a ^{252}Cf source placed 12.75 in away from the detector. The second run was with a ^{60}Co source placed 12.75 in above the bare detector. The third run was taken with the same ^{60}Co source above the detector with 3 inches of bismuth shielding. All runs were taken for one hour. Another one hour background run was taken to ensure accuracy of the background subtraction. The same amplifier, gain, and digitizer settings were used as with the bare detector ^{252}Cf run.

3.3 Data Analysis

3.3.1 University of Ohio Accelerator

The data taken from the Ohio accelerator facility was analyzed by taking each event and loading it into a Matlab array. The pulses were then inverted and the program searched for two peaks above our defined threshold of 2 mV. An early problem while analyzing the data was due to afterpulsing. Many of the large pulses included after pulses above our threshold. In order to compensate for this after each pulse the threshold was raised to 5 mV for 600 ns. This ensured that real double pulses were not mistaken for triples or more.

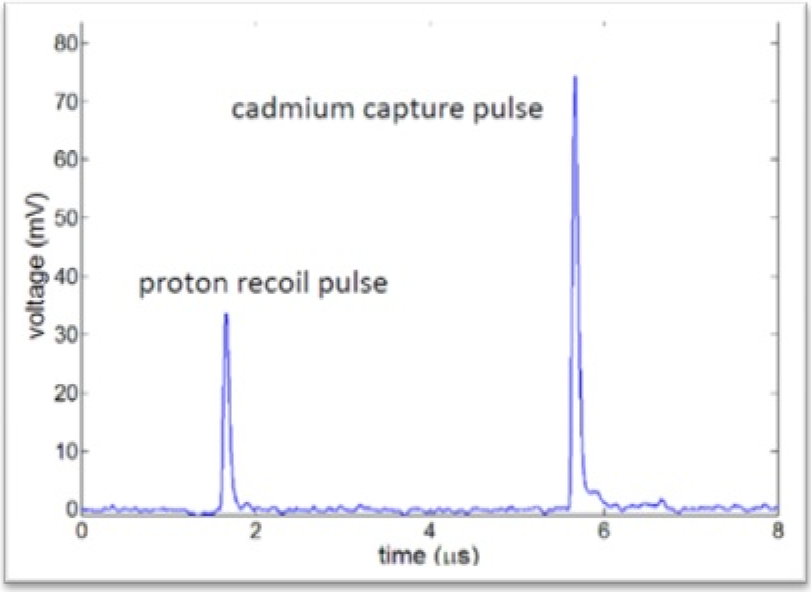

Figure 3.4 Example of a double pulse neutron event.

We then took the time difference between the beam pulse time zero and the first pulse using constant fraction discrimination (Figure 3.5). This time combined with the distance of 9.729 m from the target to the detector gave us a relativistic velocity for the neutron to be converted into kinetic energy (Figure 3.6). Our data was then divided up into 250 energy bins and a histogram was recorded with the number of double pulses in each energy bin. Background was found by looking at double events far beyond the neutron window and subtracting this from the entire histogram.

Ohio University provided us with neutron flux data in $\frac{neutrons}{MeV \times \mu C \times sR}$ (Figure 3.7). Using the solid angle of the detector we converted our $\frac{neutrons}{MeV}$ data to match the Ohio flux data. The data taken from the detector was then divided by the Ohio flux data in order to give us an intrinsic efficiency plot. The overall intrinsic efficiency of the detector was $\approx$12%. Although the result was similar to MCNP expectations for low energies (see Figure 2.8) the expected efficiency should have dropped off with an increase in neutron energy. We currently do not have an explanation for these results.

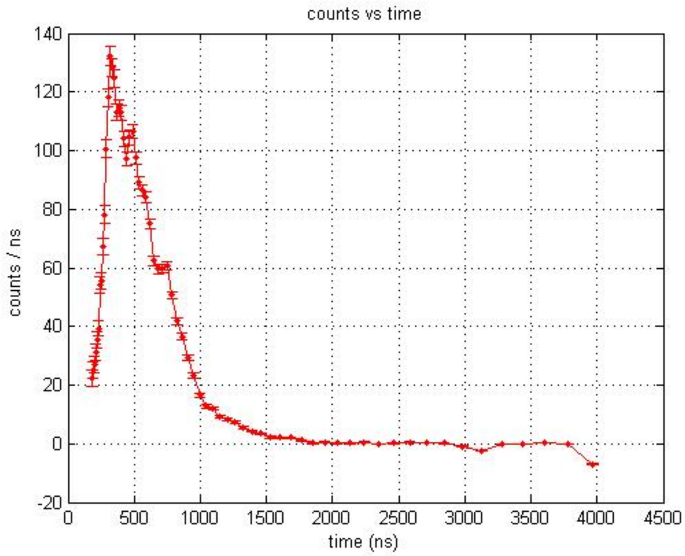

Figure 3.5 Plot of neutrons detected vs time

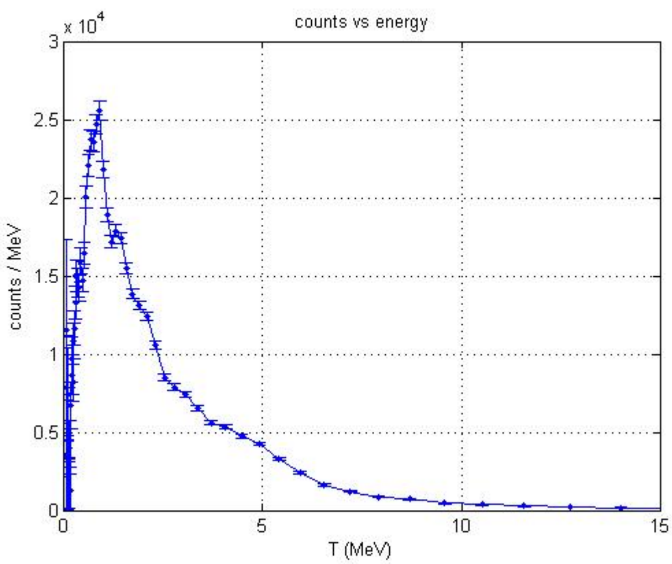

Figure 3.6 Plot of neutrons detected vs energy

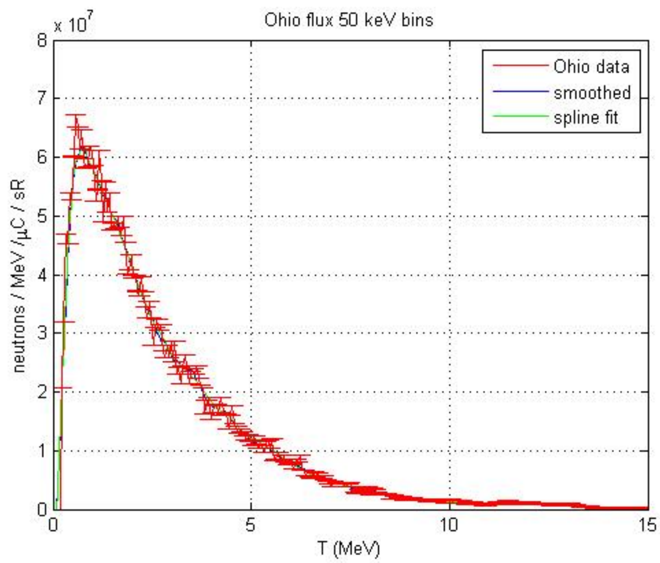

Figure 3.7 Flux data given to us by Ohio University

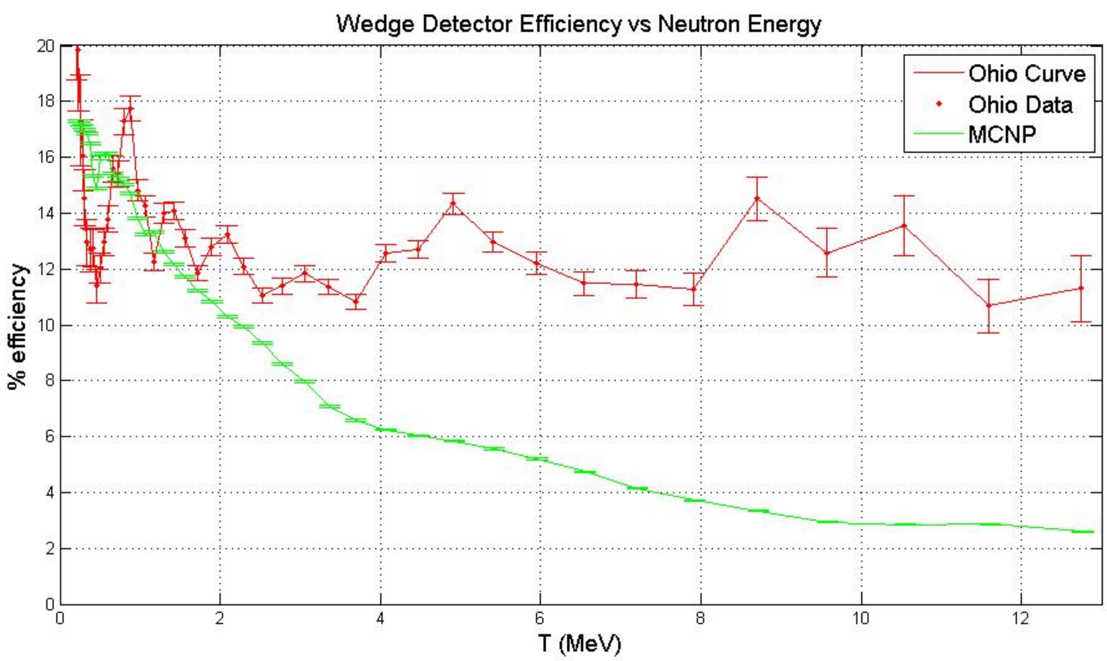

Figure 3.8 Wedge detector efficiency vs Energy. Includes both MCNP predictions and data from Ohio.

3.3.2 Scissor Lift

For data taken on the scissor lift, an event was only recorded in the case of a double pulse. Although there were a number of events counted, there were some problems in the recorded data. Actual data analysis of individual events has yet to be done and will be neccessary in order to gain a better understanding of the detector efficiency. The actual number of double pulse events recorded minus background subtraction was 183553. This divided by 3600 gives 50.99 $\frac{neutrons\, detected}{s}$. Our ^{252}Cf source emitted at a rate of 3.7435x10^4 $\frac{n}{s}$. In order to find the number of neutrons hitting the detector the number of neutrons emmited by the source is multiplied by the fractional solid angle:

$$\frac{Detector\ Surface\ Area}{4\pi d^2}$$

The number of neutrons detected per second can then be divided by this number of source neutrons which hit the detecter to give an intrinsic efficiency. The rough estimate for the intrinsic efficiency of the detector for a source without shielding was calculated at 13.4%. This efficiency could be inaccurate due to a large amount of accidental double pulses. The accidental double pulses could be due to either a large pulse with a large amount of afterpulsing or two gammas hitting the detector within a 32 μs period.

In order to find an accurate measurement for the detector efficiency it will be neccessary to eliminate accidentals by eliminating afterpulsing and double gamma pulses.

3.3.3 Bi Shielding

After placing 3 inches of bismuth shielding over the detector the number of double pulses recorded dropped by about a factor of two. 52945 total double pulse events were recorded after background subtraction. Using the same solid angle and timing calculation as the bare detector scissor lift calculations a total intrinsic efficiency with a ^{252}Cf source placed 12.75 in away was 3.87%. Although it appears that the bismuth shielding adversly affected the efficiency of the detector, no conclusions can be made until all events are analyzed individually to remove all accidental triggers.

The runs taken with the ^{60}Co source had more promising results. The source used was calculated to emit 531800 $\frac{\gamma}{s}$. Without shielding the detector counted 253956 accidental double events in one hour.The same solid angle and timing calculations were used without a gamma shield on the detector and resulted in 1.31% of the gammas that hit the detector giving an accidental double. When the shielding was added the number of doubles recorded dropped to 24121 in one hour. This corresponds to .124% of the gammas reaching the detector giving an accidental double.

A false alarm would only be set after the pulse rate exceeded some threshold. Thus, the use of bismuth shielding would allow for a much lower threshold. Careful adjustment of detection parameters will improve the detectors performance considerably.

Although no data can be concluded about the efficiency drop due to the bismuth shielding, the shielding proved to drop the number of accidental gammas by a factor of 10. This could serve as a valuable implementation for neutron detectors with high gamma sensitivity. Although the weight and cost of bismuth are high the results show that a minimal amount (3 inches) can have an impact up to a factor of 10.

3.3.4 Future Work and Improvements

The major concern with the detector is discerning accidental double pulses from actual neutron events. Although the data collected from Ohio University was able to be post processed, a detecter used as a portal monitor would need to analyze data in real time. After a pulse triggers the detector, a computer would need to determine if the following pulses crossing threshold are a result of afterpulsing and should be thrown out or if they are ligitimate events. Also the timing between pulses needs to be recorded and could determine if the two events are correlated or not.

Other concerns for the detector lie within the optics of the detector. There are concerns that in the middle of the detector photons are lost due to the small amount of scintillator compared to the amount of cadmium. A possible solution would be to machine out the center of the detector and insert a scintillating rod.

The data taken on the scissor lift is still to be analyzed and should be able to provide insight into the actual efficiency of the detector in a portal monitoring situation.

Bibliography

[1] U. D. of Energy, "Nuclear Physics and Reactor Theory Volume 1," DOE Fundamentals Handbook DOE-HDBK-1019/1-93 (1993).

[2] V. F. Sears, "Neutron Scattering Lengths and Cross Sections," Neutron News 3 (1992).

[3] R. Kouzes, J. Ely, A. Lintereur, and D. Stephens, "Neutron Detector Gamma Insensitivity Criteria," Pacific Northwest National Laboratory Report PNNL-18903 (2009).

[4] R. V. Ginhoven, R. Kouzes, and D. Stephens, "Alternative Neutron Detector Technologies for Homeland Security," Pacific Northwest National Laboratory Report PNNL-18471 (2009).

[5] A. Wallace, "Lithium-Glass Neutron Detection," B.S. Thesis (2011).

[6] N. Hogan, "Characterization of a Cadmium Capture-Gated Neutron Spectrometer," B.S. Thesis (2011).