Non-destructive Spent Fuel Characterization with Semiconducting Gallium Arsenide Neutron Imaging Arrays

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Douglas S. McGregor, Holly K. Gersch, Jeffrey D. Sanders
Semiconductor Materials and Radiological Technologies Laboratory, Department of Nuclear Engineering and Radiological Sciences, University of Michigan, Ann Arbor, MI 48109

John C. Lee, Mark D. Hammig, Michael R. Hartman, Yong Hong Yang
Department of Nuclear Engineering and Radiological Sciences, University of Michigan, Ann Arbor, MI 48109

Raymond T. Klann
Argonne National Laboratory, 9700 South Cass Avenue, Building 362, Room B-113, Argonne, IL 60439

Brian Van Der Elzen
Solid State Electronics Laboratory, 1301 Beal, University of Michigan, Ann Arbor, MI 48109

John T. Lindsay, Philip A. Simpson
Phoenix Memorial Laboratory, 2301 Bonisteel Blvd., University of Michigan, Ann Arbor, MI 48109

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Dr. Douglas S. McGregor, Director of the Semiconductor Materials and Radiological Technologies Laboratory at the University of Michigan, assumed subsequent project direction and conducted detector development and characterization.

Dr. John C. Lee, Chairman of the Nuclear Engineering and Radiological Sciences Department at the University of Michigan, conducted algorithm development for neutron flux deconvolution.

The Ford Nuclear Reactor, the 2 MW materials research reactor used to characterize the neutron detectors in the NEER project "Non-destructive Spent Fuel Characterization with Semiconducting Gallium Arsenide Neutron Imaging Arrays," Agreement Number DE-FG07-98ID13633.

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Abstract

High resistivity bulk grown GaAs has been used to produce thermal neutron imaging devices for use in neutron radiography and characterizing burnup in spent fuel. The basic scheme utilizes a portable Sb/Be source for monoenergetic (24 keV) neutron radiation source coupled to an Fe filter with a radiation hard ^{10}B -coated pixellated GaAs detector array as the primary neutron detector.

The coated neutron detectors have been tested for efficiency and radiation hardness in order to determine their fitness for the harsh environments imposed by spent fuel. Theoretical and experimental results are presented, showing detector radiation hardness, expected detection efficiency and the spatial resolution from such a scheme. A variety of advanced neutron detector designs have been explored, with experimental results achieving 13% thermal neutron detection efficiency while projecting the possibility of over 30% thermal neutron detection efficiency.

The bulk GaAs devices have been designed in various sizes and pixel densities ranging from 16 to 64 per device. The pixels are arranged in linear rows and can be connected in line to extend the total pixel count. ^{10}B layers ranging from 1 to 4 microns are applied to the front of the devices. The devices are operated at relatively low bias voltages ranging between 20 to 100 volts. Although very sensitive to thermal neutrons, the lower neutron absorption cross section for 24 keV neutrons significantly decreases the sensitivity of the devices.

The GaAs-based neutron tomography system intended for the assay of nuclear fuel has been developed and tested. Its main components include a 16 pixel, dual in-line array of ^{10}B coated GaAs neutron detectors, a step motor driven table capable of both linear and rotational motion, and custom data acquisition software capable of automated data collection. A filtered-backprojection based reconstruction algorithm was implemented to reconstruct tomographic slices. Results from thermal and epithermal neutron imaging experiments are presented and possible system improvements discussed.

An algorithm was developed to obtain the fuel composition through spectral deconvolution of the neutron flux emitted from the fuel. The MCNP Monte Carlo code was used to simulate fission neutron spectra, similar to those expected from an active assay of spent nuclear fuel, incident upon polyethylene moderators of varying thicknesses followed by an array of pixellated detectors. The spectral deconvolution algorithm was applied to the simulated detector response and showed qualitative agreement with the incident spectrum.

Nondestructive Spent Fuel Characterization with Semiconducting Gallium Arsenide Neutron Imaging Arrays

Chapter I: Introduction

Overall Project Goal: The goal of the NEER research project was to develop a safe, convenient, and semi-portable inspection system for characterizing fuel burnup in irradiated nuclear fuel elements. This project utilizes unique properties of antimony/beryllium photoneutron sources with the extensive radiation hardness of gallium arsenide semiconductor devices. Analytical deconvolution methods were developed to determine characteristics of spent fuel elements from a combination of passive and active neutron radiography data.

Task 1: Develop, characterize, and optimize a GaAs-based neutron imaging detector.

A primary goal has been to develop an optimized detector design for the GaAs semiconductor neutron devices, in particular, in the composition and structure of the neutron absorber film, and to determine the radiation hardness of the detector arrays. Much of the project involved the study, development, and characterization of the thin-film neutron detectors. In addition, lattice physics analysis has been performed to determine localized fuel burnup characteristics from tomographic neutron radiography data for irradiated fuel elements.

The GaAs imaging devices have two linear arrays designed such that they can be sliced to 8, 16, 32, 64, 128, and 256 pixel chips. Three pixel sizes were initially designed, which include sets of (30 μm x 300 μm) pixels, sets of (200 μm x 500 μm) pixels and sets of (500 μm x 1000 μm) pixels. For the duration of the project, the characterization was performed with the 500 μm x 1000 μm design. Arrays were mounted and linearly scanned to image test objects. A system to readout pixels in real time was constructed and has been successfully tested. Both linear and tomographic images of phantoms have been produced.

Theoretical calculations have been performed on several additional neutron/charged particle converters other than ^{10}B and several different physical designs. For front-coated devices, the maximum theoretical efficiency for ^{10}B films is achieved at a thickness of 2.4 μm , whereas the maximum efficiency for ^6LiF films is achieved at 26 μm of material with only a modest 10% efficiency increase relative to ^{10}B coated films. Manufacturing accurate film coatings that are greater than a few μm in thickness is difficult, hence ^{10}B coated devices are clearly easier to fabricate. Analysis of pure ^6Li films indicates that single front-coated devices can yield neutron detection efficiencies almost three times greater than that achievable with ^{10}B and ^6LiF films. Yet, such an increase in efficiency comes at the expense of depositing films that are 100 μm thick, which is even more difficult to achieve than that required of ^6LiF films. Additionally, pure ^6Li is very reactive and must be encapsulated before the device can be exposed to air. Hence, the handling ease of ^{10}B and ^6LiF far outweighs the gains realized by utilizing pure ^6Li for the neutron reactive film. From the calculations, GaAs detectors coated with pure ^{10}B for the neutron converter continues to be the main design for the neutron detector structure. A variety of advanced structures have been explored with success.

Numerous detector arrays are being studied to obtain a relatively wide statistical database. GaAs detector radiation hardness tests have been accelerated by irradiating the arrays at near-core and in-core irradiation positions in the 2.0-MW Ford Nuclear Reactor (FNR) at the University of Michigan. These positions have neutron fluxes ranging up to 10^{12} neutrons/ $\text{cm}^2\text{-s}$. Previous generations of these detectors have been continually irradiated in excess of 3000 hours in a special beam port at 1.7×10^6 neutrons/ $\text{cm}^2\text{-s}$ and 1.1 rad/hr gamma ray exposure without observable change in operating characteristics. The exposures were varied and leakage

currents, bias voltages, and pulse height responses were documented as a function of total gamma ray and neutron exposure. A 20-kCi ^{60}Co source was used for accelerated gamma ray hardness tests. Device performance has been documented at various cumulative gamma ray doses. The experiments have assisted in determining both the catastrophic gamma ray and neutron irradiation limits for the GaAs detectors.

The GaAs detectors were also investigated for neutron and gamma ray detection efficiency. A set of GaAs detector arrays were tested in a double copper crystal diffracted neutron beam to document neutron detection efficiency in the absence of a large gamma ray background. The detectors were also tested in a neutron beam port with both high gamma ray and neutron components in order to determine the ability of the detectors to discriminate between gamma ray signals and neutron signals.

Task 2: Develop methods to determine fuel characteristics from neutron tomography.

The basic approach used to determine the characteristics of spent nuclear fuel involves a combination of passive and active neutron interrogation techniques with GaAs detectors having neutron reactive films (described above). As the first step in studying an optimal way to determine spent fuel characteristics from neutron reaction rates measured by GaAs(^{10}B) detectors, fuel depletion calculations have been performed for typical FNR fuel elements with the WIMS collision probability code. Analysis of few-group macroscopic cross sections obtained for the entire fuel elements as a function of fuel burnup indicates that the absorption cross section depends weakly on fuel burnup. This suggests the desirability of determining, from detector signals, the fission cross section of the fuel element, which is naturally a sensitive function of fuel burnup.

The analytical approach requires the determination of the ^{244}Cm concentration of any fuel mass through measurements of its spontaneous fission (SF) rate, which is essentially proportional to the fuel burnup. Determination of the SF rate in passive tomography, as well as accounting for induced fission contributions to the GaAs(^{10}B) detector response in active tomography, would require creative detector arrangements, including the placement of an iron filter and/or a moderator film in front of the boron absorption layer. With the iron filter, 24-keV neutrons from the Sb/Be source penetrating the fuel element will be selectively detected. In contrast, with the moderator film, fission neutrons would interact first in the moderator layer before undergoing capture in the boron film to produce charged particles, which will eventually be detected in the semiconductor detector. The placement of different thicknesses of the moderator film may be considered transformations of the absorber reaction cross section so that the determination of the incident beam intensity and spectrum would amount to the unfolding of neutron flux spectra through multiple foil activations.

As part of the effort to study burnup characteristics of FNR fuel elements through WIMS calculations, inconsistencies in the few-group cross section edits of the WIMS code have been corrected. Since the latest version of the code, WIMS-D4M, reflects developments, often not well documented, at several different laboratories in the U. S. and Europe, considerable effort has been required, and will continue, to develop sufficient understanding of key segments of the code and to build in correct cross section collapsing schemes.

Task 3: Build a 24-keV Sb/Be source and neutron collimator.

An Sb/Be photoneutron source which uses the unique neutron resonance characteristics of iron has been developed. Sb/Be photoneutron sources emit monoenergetic neutrons of 24 keV. Also, ^{56}Fe has a neutron transmission resonance window at 24 keV which almost

perfectly matches the neutron emission energy of the Sb/Be source [10]. The combination of these unique properties enables a user to interrogate nuclear fuel with a monoenergetic neutron beam, which allows for easier mathematical modeling and cross section determination. In addition, using this Fe filter in front of the detector allows for a variety of probing measurements. Hence, thermal neutrons can be used for neutron imaging of various aspects of the material, while epithermal (24 keV) neutron imaging allows for the use of moderately penetrating neutron radiation. An Fe (or steel) shield placed between the detector and the object under investigation (as well as the Sb/Be source) reduces gamma ray background while barely affecting the neutron flux. Using multiple passive and active measurements can enhance the ability to image the fuel and determine the SF neutron signal, which will be critical to determining the fuel burnup. A collimator/filter was also constructed for the detector and source to remove unwanted scattered neutrons from the beam.

Chapter II: Detector Development

Section A: General Detector Concept

1. Introduction

Since neutrons hold no electronic charge, methods used to recognize neutron interactions within a detector generally rely on second order effects. Two very common neutron interactions that are used for a variety of thermal neutron detectors are the $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reaction and the $^6\text{Li}(\text{n},\alpha)^3\text{H}$ reaction [7-10] (see Figure 1).

The $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reaction leads to the following reaction products [7-10]:

	Reaction Q– Value
$^{10}\text{B} + {}^1_0\text{n} \rightarrow \begin{cases} {}^7\text{Li}(\text{at } 1.015\text{ MeV}) + \alpha(\text{at } 1.777\text{ MeV}), \\ {}^7\text{Li}^*(\text{at } 0.840\text{ MeV}) + \alpha(\text{at } 1.470\text{ MeV}), \end{cases}$	$2.792\text{ MeV}(\text{to ground state})$ $2.310\text{ MeV}(1\text{st excited state})$

which are released in opposite directions when thermal neutrons (0.0259 eV) are absorbed by ^{10}B . After absorption, 94% of the reactions leave the ^7Li ion in its first excited state, which rapidly de-excites to the ground state ($\sim 10^{-13}$ seconds) by releasing a 480 keV gamma ray. The remaining 6% of the reactions result in the ^7Li ion going directly to its ground state. The microscopic thermal neutron absorption cross section is 3840 barns. Additionally, the microscopic thermal neutron absorption cross section decreases with increasing neutron energy, with a dependence proportional to the inverse of the neutron velocity ($1/v$) over much of the energy range [5].

Figure 1: The two most commonly used neutron reactions for thermal neutron detection purposes are the $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reaction and the $^6\text{Li}(\text{n},\alpha)^3\text{H}$ reaction.

The $^6\text{Li}(\text{n},\alpha)^3\text{H}$ reaction leads to the following products:

	Reaction Q– Value
$^6\text{Li} + {}^1_0\text{n} \rightarrow {}^3\text{H}(\text{at } 2.73\text{ MeV}) + \alpha(\text{at } 2.05\text{ MeV}),$	4.78 MeV

which again are oppositely directed if the neutron energy is sufficiently small. The microscopic thermal neutron (0.0259 eV) absorption cross section is 940 barns. The thermal neutron

absorption cross section also demonstrates a $1/v$ dependence, except at a salient resonance above 100 keV, in which the absorption cross section surpasses that of ^{10}B for energies between approximately 150 keV to 300 keV [56]. Additional resonances characteristic to either isotope cause the absorption cross section to surpass one or the other as the neutron energy increases. Due to its higher absorption cross section, the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction leads to a generally higher reaction probability than the $^6\text{Li}(n,\alpha)^3\text{H}$ reaction for neutron energies below 100 keV. However, the higher energy reaction products emitted from the $^6\text{Li}(n,\alpha)^3\text{H}$ reaction lead to greater ease of detection than the particles emitted from the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction.

There are other neutron reactions that have been used in solid-form films for thermal neutron detectors, including the $^{157}\text{Gd}(n,\gamma)^{158}\text{Gd}$ reaction [11-15] and the $^{113}\text{Cd}(n,\gamma)^{114}\text{Cd}$ reaction [16-18], both of which have very large thermal neutron reaction cross sections. The $^{157}\text{Gd}(n,\gamma)^{158}\text{Gd}$ reaction leads to the emission of various low energy gamma rays and beta particles, most of which are emitted at energies below 220 keV. The appeal of using ^{157}Gd is due to its large thermal neutron cross section of 240,000 barns [19], but the resulting low energy conversion electron and gamma ray emissions can be difficult to distinguish from background gamma rays in a high radiation field. The ^{113}Cd reaction is intriguing as well, with its thermal neutron absorption cross section of 20,000 barns [20]. Discrete gamma ray emissions from the $^{113}\text{Cd}(n,\gamma)^{114}\text{Cd}$ reaction extend beyond 9 MeV, however the salient gamma ray emissions are at 558.6 keV and 651.3 keV [21-23], which can be discerned with a relatively high resolution gamma ray spectrometer. Unfortunately, any gamma ray spectrometer capable of detecting the $^{113}\text{Cd}(n,\gamma)^{114}\text{Cd}$ emissions will also detect any other background gamma rays, which can confuse the measurement.

The neutron detector design discussed in the present work consists of a solid-state detector (or semiconductor diode detector) coated with a solid form of neutron reactive material. Such a device has been often described as a “foil coated” neutron detector, although we adapt the more descriptive name of thin-film-coated neutron detector. Only neutron reactive coatings that liberate high-energy heavy particles, such as ^{10}B and ^6Li -based films, were used. The present analysis is also limited to thermal neutron detection, in which the reaction-product distribution is isotropic.

2. The Basic Concept

Thin film neutron detectors consist of semiconductor diodes, preferably with very thin contact layers, upon which a layer (or layers) of neutron reactive material has been deposited. The basic concept has been used by a variety of research groups [24-26]. As shown in Figure 2, neutrons absorbed in the neutron reactive film release charged particle reaction products in opposite directions, one of which may enter the semiconductor diode detector. Charged particles entering the detector lose their energy through Coulombic scattering, thereby creating a high-density plasma cloud of columnar ionization in the form of electron-hole pairs. The semiconductor diode detector is voltage biased to separate the electron-hole pairs and drift the charges to their respective contacts. The mobile charges each induce an image charge on the contacts as they move through the device, and the induced charge is integrated and measured by an external preamplifier and accompanying electronics.

It is important to note that the charged particles have discrete ranges within a material, and they lose energy as a function of the specific “stopping power” of the material through which they pass [27]. Charged particle energy loss is not linear with distance, and the energy loss for non-relativistic high-energy charged particles can generally be described by the Bethe-Bloch stopping power relation [34].

Figure 2: The fundamental approach to a thin-film-coated semiconductor neutron detector. The film thickness should not exceed the maximum range of the “long range” reaction product. The reaction products are emitted in opposite directions.

$$s = -\frac{1}{N} \frac{dE}{dx} = \frac{4\pi z^2 q^4}{m_o v^2} Z \ln \left(\frac{2m_o v^2}{J} \right), \quad (1)$$

where zq and v are the charge and the velocity of the charged particle, N is the atomic density, m_o is the rest mass of an electron, Z is the material atomic density, and J is an experimentally determined quantity describing the average excitation potential of a representative atom in the material. Although the average range is treated as constant, the actual range fluctuates from particle to particle due to deviations in the number of Coulombic scatters per unit distance and deviations in energy loss per collision. The fluctuation may exceed 5% of the total range, and the deviation is referred to as the “straggle” in the particle range. In the unique case of protons, alpha particles, and other relatively low mass heavy ions, the energy loss as a function of distance can be described by the Bragg ionization curve, in which [33]

$$I(r) = \int_0^{\infty} \frac{i(x-r)}{\beta\sqrt{\pi}} e^{-[(x-R)/\beta]^2} dx, \quad (2)$$

where x is the range of the individual particle, r is the distance that the particle has traveled in the material, $i(x-r)$ is the specific ionization along the path of the ionizing particle at point $(x-r)$ from the end of its path, R is the mean range and β is the straggling parameter. An interesting characteristic of the Bragg ionization curve for alpha (α) particles is the increase in columnar ionization per unit volume as the ion energy decreases. Hence, the highest density of electron-hole pairs are excited near the end of the range for α -particles. In the present treatment the *average effective range* (denoted L) is the distance through which a particle may travel *within the neutron reactive film* before its energy decreases below the set minimum detectable threshold, typically defined by the electronic lower-level-discriminator (*LLD*) setting. The term does not take into account additional energy losses from contact “dead regions.” L_{SR} and L_{LR} denote the average effective ranges for the short-range and long-range reaction products, respectively.

Neutrons may interact anywhere within the reactive film. From Figure 2, it is apparent that the reaction products will also lose energy as they move through the reactive film, thus limiting the energy transferred to the semiconductor detector. The finite specific energy loss in the reactive film limits the usable film thickness that can be deposited over the semiconductor device. The voltage signal measured is directly proportional to the number of electron-hole pairs created within the semiconductor. Reaction products that deposit most or all of their energy in the semiconductor volume will produce much larger voltage signals than those reaction products that lose most of their energy in the surface films and contacts.

The neutron flux transmitted through the film as a function of distance x can be described by [38]

$$I(x) = I_0 e^{-x\sigma_F N_F} = I_0 e^{-x\Sigma_F} \quad , \quad (3)$$

where I_0 is the initial neutron flux, N_F is the atomic density of the neutron reactive *isotope* in the film, σ_F is the microscopic thermal neutron absorption cross section of the film, and Σ_F is the film macroscopic thermal neutron absorption cross section. It is apparent that the fraction of neutrons absorbed in the film through distance x is

$$1 - \frac{I(x)}{I_0} = 1 - e^{-x\sigma_F N_F} = 1 - e^{-x\Sigma_F} \quad , \quad (4)$$

The neutron absorption probability per unit distance is described by [39]

$$p(x)dx = \Sigma_F e^{-x\Sigma_F} dx \quad . \quad (5)$$

The angular contribution to self-attenuation must also be addressed. Once a neutron is absorbed and the reaction products are emitted, the probability that a reaction product particle will enter the detector is determined by the solid angle subtending the surface within the average effective range (L) of the particle interaction. From Figure 3, a neutron interaction taking place at distance x from the semiconductor surface has a probability of entering the active volume as described by the fractional solid angle that subtends the detector,

$$p_p(x) = \frac{\Omega(x)}{4\pi} = \frac{2\pi}{4\pi} \left(1 - \frac{x}{L}\right) = 0.5 \left(1 - \frac{x}{L}\right), \quad x \leq L, \quad (6)$$

where the subscript p relates to the reaction product particle of interest. Since the reactions of interest in the present work release two different reaction products per event, the total probability of detecting a neutron reaction consists of adding the detection probabilities of both reaction products. Note that it is possible that only one reaction product may be able to reach the semiconductor active volume. Figure 4 demonstrates the variation in detection efficiency for different neutron interaction depths. Interactions occurring near the detector contact result in either particle entering the detector with high probability. As the neutron interaction distance increases, the solid angle of the short-range particle decreases more rapidly than that of the long-range particle, resulting in an overall decrease in detection sensitivity. At neutron interaction distances greater than L_{LR} , the long-range particle can no longer reach the detector and the neutron absorptions are undetected. Ultimately, it serves no purpose to increase the thickness of the neutron reactive thin film beyond L_{LR} , as will be shown, and actually works to decrease the overall neutron detection efficiency.

Figure 3: The solid angle formed by the reaction product ranges that subtends the detector at the interface predetermines the detection probability.

Figure 4: The solid angle subtending the detector at the interface is greatest for reactions occurring at the film/detector interface. Shown are cases where (A) either particle may enter the detector with high probability, (B) the short range product has diminished detection probability, and (C) the short range product will not be detected and the long range product has diminished detection probability.

Table 1: Calculated average L values and thermal neutron ΣL values for different reaction product particles and minimum energy detection thresholds in a ^{10}B film. The detector entrance dead region was excluded from the calculation, and should be accounted for in cases where dead region absorption is significant.

$^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ Reaction, in ^{10}B ($\Sigma = 500.00/\text{cm}$ at 0.0259 eV)								
Minimum Detection Threshold	1.777 MeV α particle		1.470 MeV α particle		1.015 MeV ^7Li ion		840 keV ^7Li ion	
	L (microns)	ΣL	L (microns)	ΣL	L (microns)	ΣL	L (microns)	ΣL
100 keV	4.0155	0.2008	3.1392	0.1570	1.5148	0.0757	1.2804	0.0640
200 keV	3.7482	0.1874	2.8722	0.1436	1.2527	0.0626	1.0180	0.0509
300 keV	3.5233	0.1762	2.6475	0.1324	1.0453	0.0523	0.8102	0.0405
400 keV	3.3146	0.1657	2.4390	0.1219	0.8667	0.0433	0.6315	0.0316
500 keV	3.1117	0.1556	2.2362	0.1118	0.7055	0.0353	0.4705	0.0235
600 keV	2.9086	0.1454	2.0330	0.1017	0.5548	0.0277	0.3199	0.0160
700 keV	2.7018	0.1351	1.8262	0.0913	0.4110	0.0205	0.1763	0.00881
800 keV	2.4894	0.1245	1.6140	0.0807	0.2720	0.0136	0.0374	0.0187

Table 2: Calculated average L values and thermal neutron ΣL values for different reaction product particles and minimum energy detection thresholds in ${}^6\text{LiF}$ and ${}^6\text{Li}$ films. The detector entrance dead region was excluded from the calculation, and should be accounted for in cases where dead region absorption is significant.

Minimum Detection Threshold	${}^6\text{Li}(n,\alpha){}^3\text{H}$ Reaction, in ${}^6\text{LiF}$ ($\Sigma = 57.51/\text{cm}$ at 0.0259 eV)				${}^6\text{Li}(n,\alpha){}^3\text{H}$ Reaction, in ${}^6\text{Li}$ ($\Sigma = 43.56/\text{cm}$ at 0.0259 eV)			
	2.730 MeV Triton L (microns)	ΣL	2.050 MeV α particle L (microns)	ΣL	2.730 MeV Triton L (microns)	ΣL	2.050 MeV α particle L (microns)	ΣL
100 keV	30.754	0.1769	5.3109	0.0305	132.29	0.5762	21.489	0.0936
200 keV	29.958	0.1723	4.9432	0.0284	129.38	0.5636	20.127	0.0877
300 keV	29.239	0.1682	4.6454	0.0267	126.77	0.5522	19.055	0.0830
400 keV	28.523	0.1640	4.3783	0.0252	124.14	0.5408	18.094	0.0788
500 keV	27.783	0.1598	4.2170	0.0238	121.36	0.5286	17.176	0.0748
600 keV	27.005	0.1553	3.8833	0.0223	118.35	0.5155	16.267	0.0709
700 keV	26.185	0.1506	3.6420	0.0209	115.08	0.5013	15.348	0.0669
800 keV	25.320	0.1456	3.4006	0.0196	111.57	0.4860	14.409	0.0628

3. Considerations Regarding ^{10}B , ^6LiF , and ^6Li Coatings

a. ^{10}B Coatings

As mentioned previously, the primary $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction (94%) results in the emission of a 1.47 MeV alpha particle and a 840 keV ^7Li ion in its first excited state, and the ground state reaction results in the emission of a 1.777 MeV alpha particle and a 1.015 MeV ^7Li ion. Comparison Bragg ionization curves for both particles within pure ^{10}B film are shown in Figure 5 [36]. Shown are the calculated range differences for either ion [36], in which the average range for a 0.840 MeV ^7Li ion in boron is 1.6 microns and the average range for a 1.47 MeV alpha particle is 3.6 microns.

Figure 5: $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction product energy loss in the ^{10}B film is described by the Bragg distribution.

Figure 6: Energy deposited into the detector is simply the original energy minus the energy lost through self-absorption. Shown is the energy remaining as a function of transit distance through boron.

The energy absorbed in the detector is simply the total energy minus the energy lost in both the boron film and the detector contact during transit. Assuming that energy loss in the detector contact is negligible, Figure 6 shows the energy retained by either charged particle as a function of interaction distance from the ^{10}B film-detector junction. At any reaction location within the ^{10}B film, the maximum detector entrance energy will be retained by either particle should it enter the detector in an orthogonal trajectory. Hence, if the interaction occurs in the ^{10}B film at a distance of 0.5 microns away from the detector, the maximum energy retained by the particle will be 430 keV, and the maximum energy retained by the alpha particle will be 1150 keV (see Figure 6). For the same interaction distance from the detector, the energy retained by the particle when it reaches the detector active region decreases as the entrance angle changes from orthogonal.

Given a predetermined minimum detection threshold (or *LLD* setting), the average effective range (*L*) for either particle can be determined. For instance, from Figure 6, an *LLD* setting of 300 keV gives L_{Li} as 0.810 microns and L_{α} as 2.648 microns. The microscopic thermal neutron absorption cross section (σ) for ^{10}B is 3840 barns and the atomic density is 1.3×10^{23} atoms/cm³. The resulting macroscopic absorption cross section (Σ) is 500/cm. At 300 keV, the ΣL product for 840 keV ^7Li ions is 0.0405 and the ΣL product for 1.47 MeV alpha

particles is 0.1324. Relatively high ΣL products are desirable for improved detection sensitivity. Table 1 lists the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction ΣL products for different detection thresholds.

b. ^6LiF Coatings

The $^6\text{Li}(n,\alpha)^3\text{H}$ reaction emits a 2.73 MeV triton (^3H) and a 2.05 MeV alpha particle upon the absorption of a thermal neutron. One popular form of ^6Li is the stable compound of ^6LiF . The molecular density of ^6LiF is 6.118×10^{22} molecules/cm³, hence the atomic density of ^6Li atoms within amounts to the same. The mass density of ^6LiF is 2.541 g/cm³. The microscopic thermal neutron cross section of ^6Li is 940 barns, resulting in a macroscopic thermal neutron cross section of 57.51/cm.

Figure 7 shows the Bragg ionization distributions for the 2.73 MeV triton and the 2.05 MeV alpha particles as they transit through a film of ^6LiF . It is clear that the ranges are significantly different for the two different reaction products. The average range for the 2.73 MeV triton in ^6LiF is 32.1 microns and the average range for the 2.05 MeV alpha particle in ^6LiF is 6.11 microns. Figure 8 shows the energy retained by either charged particle as a function of interaction distance from the ^6LiF film-detector junction. As with the ^{10}B film, the maximum detector entrance energy will be retained by either particle should it transit the ^6LiF film and enter the detector in an orthogonal trajectory. From Figure 8, an LLD setting of 300 keV gives L_α as 4.64 microns and L_H as 29.25 microns. At 300 keV, the ΣL product for 2.05 MeV α particles is 0.0267 and the ΣL product for 2.73 MeV tritons is 0.1682. Hence, the long-range particle (triton) for the $^6\text{Li}(n,\alpha)^3\text{H}$ reaction in ^6LiF gives a higher ΣL product than that realized by the long-range particle (α particle) for the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction in pure ^{10}B . Conversely, the short-range particle (α particle) for the $^6\text{Li}(n,\alpha)^3\text{H}$ reaction in ^6LiF gives a lower ΣL product than the short-range particle (^7Li ion) for the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction in pure ^{10}B . The similar ΣL values for the ^6LiF and the pure ^{10}B films indicate that the maximum achievable thermal neutron efficiencies when using these films will differ only slightly. Table 2 lists the $^6\text{Li}(n,\alpha)^3\text{H}$ reaction ΣL products for different detection thresholds and materials.

Figure 7: $^6\text{Li}(n,\alpha)^3\text{H}$ reaction product energy loss in the ^6LiF film is described by the Bragg distribution.

Figure 8: Energy deposited into the detector is simply the original energy minus the energy lost through self-absorption. Shown is the energy remaining as a function of transit distance through ^6LiF .

c. Pure ${}^6\text{Li}$ Coatings

Pure ${}^6\text{Li}$ can also be used as a neutron reactive coating, although its corrosive and reactive nature results in cumbersome handling procedures [37]. Pure ${}^6\text{Li}$ has a mass density of 0.463 g/cm^3 and an atomic density of $4.634 \times 10^{22} \text{ atoms/cm}^3$. The macroscopic thermal neutron absorption cross section is $43.56/\text{cm}$. The low atom density and the low mass density result in rather large reaction product ranges, as can be seen from the Bragg ionization distributions in Figure 9. Figure 10 shows the energy retained by either charged particle reaction product as a function of interaction distance from the ${}^6\text{Li}$ film-detector junction. Clearly, the values of L_{SR} and L_{LR} for the pure ${}^6\text{Li}$ film far surpass those calculated for the pure ${}^{10}\text{B}$ film and the ${}^6\text{LiF}$ film. Additionally, the resulting ΣL products are greater. For instance, at an LLD setting of 300 keV, the 2.73 MeV triton ΣL product is 0.5532 and the 2.05 MeV alpha particle ΣL product is 0.0828. These high ΣL values allow for, and require, rather thick reaction product films, with $L_\alpha = 19.05$ microns and $L_H = 126.77$ microns. Such thick films can be difficult to apply and prevent from chemically decomposing [37].

Figure 9: ${}^6\text{Li}(n,\alpha){}^3\text{H}$ reaction product energy loss in the pure ${}^6\text{Li}$ film is described by the Bragg distribution.

Figure 10: Energy deposited into the detector is simply the original energy minus the energy lost through self-absorption. Shown is the energy remaining as a function of transit distance through pure Li.

4. Devices Coated on a Single Surface

a. Front Side Irradiation – The General 3-Dimensional Surface

Although the semiconductor and film surfaces will typically be planar, theoretically they can take on any functional form. In order to determine the optimal configuration for a thin film neutron detector, it is instructive to calculate the detection efficiency for a general surface. For the sake of simplicity, the following assumptions are made:

- i. Assume that the macroscopic neutron absorption cross section (Σ) is independent of the incident neutron. This assumption implies that the incident neutrons are mono-energetic or that the interactions can be acceptably modeled with an average macroscopic cross section ($\Sigma = \Sigma_{\text{ave}} = \Sigma[E_{\text{ave}}]$).
- ii. Assume that the neutron undergoes at most one interaction.
- iii. Assume that only one ion is produced from the neutron interaction, noting that the following formulation can be easily extended to multiple reaction products.

- iv. Neglect charged particle energy straggling, so that the average effective range (L) of the ion reaction product is constant.
- v. Assume that the neutron reactive film thickness (D_F) is less than L .
- vi. Assume that the emission of the ion is uniformly distributed in angle. Such an assumption is reasonably accurate for thermal neutrons, but inaccurate for neutrons of appreciable energy.
- vii. Assume that the probability of neutron capture and the probability that an ion enters the active detector volume are independent.

Consider the general three-dimensional surface, S , of the form $z = f(x,y)$, as shown in Figure 11. An incoming neutron will produce a measurable detector signal if the following two events occur. First, the neutron must interact in the conversion layer and produce one or more charged particles. Second, the reaction product charged particles must enter the detector volume to excite electron-hole pairs. In the following section, the probability that a reaction product ion will enter the active volume, following a neutron capture at point (x', y', z') , is first derived. The probability of a neutron interacting within the converter film is then developed.

Figure 11: Three dimensional surface (S) with function $z = f(x,y)$ with a neutron interaction reaction products within range of intersecting the surface.

Deriving the ion impact probability amounts to calculating the fractional solid angle subtended by the surface that is within the average effective range (L) of the interaction point, as shown in Figure 11. It is advantageous to first find the point $(x_s, y_s, f(x_s, y_s))$ on surface S through which a normal vector to S passes and intersects the interaction point (x', y', z') . The normal vector to a surface, $f(x,y)$, is given by:

$$\hat{n} = \frac{1}{\sqrt{1 + f_x^2 + f_y^2}} \left[-f_x \hat{i} - f_y \hat{j} + \hat{k} \right] , \quad (7)$$

where f_x and f_y are the partial derivatives of f with respect to x and y . Consider the projection of the normal vector into the (z, x) -plane. The *magnitude* of the slope of the resulting projection

is given by $(-1/f_x)$. Recognizing that the line passes through the projection of the interaction point, (x', z') and the projection of the surface interaction point, $(x_s, f(x_s, y_s))$, the line equation can be used to write (x_s, y_s) in terms of (x', z') as:

$$f(x_s, y_s) = z' + \frac{x' - x_s}{f_x(x_s, y_s)} . \quad (8)$$

Similarly, the projection of the normal vector in the (z, y) -plane yields:

$$f(x_s, y_s) = z' + \frac{y' - y_s}{f_y(x_s, y_s)} . \quad (9)$$

Equations 8 and 9 may then be solved for the surface point (x_s, y_s) through which the unit normal passes. Note that depending on the form of S , Equations 8 and 9 may yield multiple solutions of (x_s, y_s) as illustrated in Figure 12.

Figure 12: More than one solution for the normal point is possible with curved surfaces.

In order to calculate the angle subtended by the surface, the point of intersection between the vector $\bar{L}$ (with base at (x', y', z')) and surface S must be found for all (x, y) . The calculations are simplified by forming a cylindrical polar reference frame, (r, ϕ, z) , centered at $(x_s, y_s, f(x_s, y_s))$ with the z -axis parallel to that in the base frame of (x, y, z) (see Figure 13). The intersection between $\bar{L}$ and the surface thus occurs at r_R , as given by Equation 10, in terms of derived (x_s, y_s) and known $(x', y', z', \text{ and } \phi)$ quantities,

$$L^2 = (x' - (x_s + r_R \cos \phi))^2 + (y' - (y_s + r_R \sin \phi))^2 + (z' - f(x_s + r_R \cos \phi, y_s + r_R \sin \phi))^2 . \quad (10)$$

The distances d and l can also be derived from the points defined in Figure 14, as shown in Equations 11 and 12:

$$d^2 = (x' - x_s)^2 + (y' - y_s)^2 + (z' - f(x_s, y_s))^2 , \quad (11)$$

and

$$l^2 = r_R^2 + (f(x_s + r_R \cos \phi, y_s + r_R \sin \phi) - f(x_s, y_s))^2 . \quad (12)$$

From the law of cosines, the desired angle α is therefore given by:

$$\cos \alpha = \frac{d^2 + L^2 - l^2}{2dL} . \quad (13)$$

If A_p is defined as the area of the converter film projected into the (x,y) -plane, then the detection efficiency can be written as

$$\varepsilon = \frac{1}{A_p} \iint_{A_p} dx dy \int_{f(x,y)}^{f(x,y)+D} dz \Sigma e^{-\Sigma(D+f(x,y)-z)} \left[\frac{\int_0^{2\pi} d\phi \int_0^\alpha d\theta \sin \theta}{4\pi} \right] . \quad (14a)$$

or

$$\varepsilon = \frac{1}{A_p} \iint_{A_p} dx dy \int_{f(x,y)}^{f(x,y)+D} dz \Sigma e^{-\Sigma(D+f(x,y)-z)} \left[\frac{\int_0^{2\pi} d\phi (1 - \cos \alpha)}{4\pi} \right] \quad (14b)$$

where

$$D = D_F \left[1 + f_x^2 + f_y^2 \right]^{1/2} . \quad (15)$$

Figure 13: Cylindrical coordinate system.

Figure 14: Relationship of the distances d and l .

The part of the integral inside the brackets in Equations 14a and 14b is independent of the neutron trajectory. However, the attenuation integral (in z) assumes that the neutron beam is parallel and aligned along the z -axis. Furthermore, the converter layer is assumed to conform to the shape of the underlying surface. The method described in this section as well as the analytical results can be used to calculate the theoretical efficiency for various converter film morphologies of interest. In the following sections, some simple configurations will be examined.

b. Front Side Irradiation - The Uniformly Coated Planar Case

A commonly used geometry involves the use of a planar semiconductor detector on which a neutron reactive film has been deposited. For such a device, the boundary conditions are:

- i. The neutron reactive films coating the device are uniform and conformal.
- ii. The device surface is void of morphological imperfections.
- iii. The device plane is situated perpendicular to a neutron beam.
- iv. The lateral dimensions of the detector are large compared to the maximum charge particle ranges within the reactive film.
- v. The source neutrons are of low enough energy such that kinetic energy effects are negligible. In other words, the reaction products are emitted opposite each other at an angle of 180° .

The coordinate system shown in Figure 15 is adapted for the following analysis. It is assumed that the contact thickness (D_I) is negligible. Equation 3 becomes

$$I(x) = I_0 e^{-\Sigma_F(D_F - x)}, \quad (16)$$

where D_F is the converter film thickness and I_o is the initial neutron beam intensity before entering the reactive film. Additionally, Equation 5 becomes

$$p(x)dx = \Sigma_F e^{-\Sigma_F(D_F-x)} dx . \quad (17)$$

Figure 15: Coordinate system for front irradiation of coated neutron detectors.

The sensitivity contribution for a reaction product particle can be found by integrating over the product of the interaction probability (Equation 16) and the fractional solid angle through film thickness D_F . The efficiency can equivalently be derived from the framework described in section 4 above. If the surface, S , is parameterized as $f(x, y) = 0$ for all x, y , then Equation 14b reduces to:

$$\begin{aligned} S_p(D_F) &= \frac{F_p}{4\pi I_o} \int_0^{D_F} I_o 2\pi \Sigma_F e^{-\Sigma_F(D_F-x)} \left(1 - \frac{x}{L}\right) dx \\ &= 0.5 F_p \left\{ \left(1 + \frac{1}{\Sigma_F L}\right) \left(1 - e^{-\Sigma_F D_F}\right) - \frac{D_F}{L} \right\} , \end{aligned} \quad (18a)$$

for $D_F \leq L$, and

$$\begin{aligned} S_p(D_F) &= \frac{F_p e^{-\Sigma_F(D_F-L)}}{4\pi I_o} \int_0^L I_o 2\pi \Sigma_F e^{-\Sigma_F(D_F-x)} \left(1 - \frac{x}{L}\right) dx \\ &= 0.5 F_p e^{-\Sigma_F(D_F-L)} \left\{ \left(1 + \frac{1}{\Sigma_F L}\right) \left(1 - e^{-\Sigma_F L}\right) - 1 \right\} , \end{aligned} \quad (18b)$$

for $D_F > L$, where F_p refers to the branching ratio of the reaction product emission. The total sensitivity can be found by adding all of the reaction product sensitivities, hence

$$S(D_F)|_{Total} = \sum_{p=1}^N S_p(D_F) \quad , \quad (19)$$

where N is the number of different possible reaction products. In the case of ^{10}B -based films N equals 4, whereas for ^6Li -based films N equals 2. Notice from Equation 18b that the value of S_p decreases as D_F becomes larger than the value of L . As a result, there will be an optimal neutron reactive film thickness for front irradiated detectors. Since the minimum particle detection threshold determines the average effective range (L), the optimal film thickness is also a function of the LLD setting.

The maximum neutron sensitivity from any particular charged particle reaction product can be found by differentiating Equation 18a with respect to the film thickness D_F ,

$$\frac{dS_p}{dD_F} = \frac{d}{dD_F} \left[0.5F_p \left\{ \left(1 + \frac{1}{\Sigma_F L} \right) (1 - e^{-\Sigma_F D_F}) - \frac{D_F}{L} \right\} \right] = 0.5F_p \left\{ \left(\Sigma_F + \frac{1}{L} \right) e^{-\Sigma_F D_F} - \frac{1}{L} \right\} \quad , \quad (20)$$

where the local maxima is found by setting Equation 20 to zero. For a particular charged particle reaction product p , the film thickness that allows for the maximum detection efficiency is

$$D_F|_{p_{\max}} = \frac{1}{\Sigma_F} \ln(\Sigma_F L + 1) \quad . \quad (21)$$

Figure 16 shows the value of $D_{F(p_{\max})}$ as a function of the $\Sigma_F L$ product. As shown, at small values of $\Sigma_F L$, the value of $D_{F(p_{\max})}$ is approximately equal to the value of L . When all reaction products are included, the optimal film thickness is determined by taking the derivative of Equation 19. The process becomes complicated by the need to select the appropriate equation, either Equation 18a or 18b, for each of the reaction products. For the sake of illustrating the behavior of the system, some appropriate assumptions can be made. In a two-product system, the long-range particle can be assumed to dominate. Hence, the derivative of Equation 18a is used for the long-range particle and the derivative of Equation 18b is used for the short-range particle. Continuing with such a theme, for multiple particle systems (as with $^{10}\text{B}(n,\alpha)^7\text{Li}$), we use Equation 18a for the longest-range particle, and Equation 18b for the remaining particles. It can therefore be shown that

$$D_F|_{\max} = \frac{1}{\Sigma_F} \ln \left[(\Sigma_F L_{LR} + 1) - \frac{L_{LR}}{F_{LR}} \sum_{p=2}^N \Psi_p \right] \quad , \quad (22)$$

where the subscript LR refers to the longest range reaction product, N is the total number of different reaction products, and

$$\Psi_p = F_p \Sigma_F e^{\Sigma_F L_p} \left[\left(1 + \frac{1}{\Sigma_F L_p} \right) (1 - e^{-\Sigma_F L_p}) - 1 \right] \quad , \quad (23)$$

which represents the remaining shorter range charged particle reaction products. For a two-particle system in which the reaction products differ in mass, Equations 22 and 23 are reliable. However, with a multiple-particle system, the branching ratios may dictate that a shorter-range particle actually dominates the maximum achievable sensitivity, in which case Equation 22 will not yield the correct optimal film thickness. In such cases, the optimal film thickness can be found by iteratively solving Equation 19 for the maximum calculated efficiency and corresponding film thickness. Regardless, the ΣL product remains an important parameter in determining the optimum neutron reactive film thickness.

Figure 16: Comparison of optimized ^{10}B thickness for front and back-irradiated detectors as a function of Σ and L .

It is important to note from Figure 15 that, for front side irradiation, the highest neutron interaction rate within the film is the farthest distance from the film-detector junction. Additionally, the lowest neutron flux and corresponding interaction rate is adjacent to the film-detector junction, and the rate decreases as the film thickness increases.

Overall, the front-coated device operates as a neutron counter. In order to discriminate against background gamma rays, the energy discriminator should be set high enough to reduce the probability of mistaking a gamma ray event for a neutron event. Figures 17 and 18 show the expected thermal neutron detection efficiency for ^{10}B , ^6LiF , and ^6Li films as a function of thickness (D_F) with an *LLD* setting of 300 keV. The ^{10}B film has the lowest efficiency, peaking at approximately 4% detection efficiency. However, the ^{10}B film thickness required to achieve 4% efficiency is only 2.4 microns, whereas comparable efficiencies for pure ^6Li and ^6LiF layers require 12 microns and 18 microns of material, respectively. Only a marginal gain in efficiency can be achieved with ^6LiF , peaking at approximately 4.4% thermal neutron detection efficiency at a film thickness of 26 microns. Pure ^6Li offers the highest achievable efficiency, peaking at 11.5% efficiency for a film thickness of 100 microns. Although pure ^6Li may appear attractive as a relatively efficient thermal neutron detection film, the fabrication, handling and packaging issues render detector production much more difficult than that required for ^{10}B or ^6LiF -coated devices. Additionally, other schemes can increase the efficiency of ^{10}B - or ^6LiF -coated devices.

Figure 17: The calculated thermal neutron detection efficiencies for front and back irradiation as a function of film thickness for ^{10}B - and ^6LiF -coated detectors.

Figure 18: The calculated thermal neutron detection efficiencies for front and back irradiation as a function of film thickness for pure ^6Li -coated detectors.

c. Backside Irradiation

To eliminate the dependence of the optimum film thickness on the *LLD* setting, the thin film coated neutron detector can be designed to operate via backside irradiation. The geometry of such a device is depicted in Figure 19. Notice that neutron absorption can take place in the detector before reaching the reactive film, and these losses must be accounted for by incorporating Equation 3 for each layer. Hence, the neutron intensity reaching the reactive film can be represented by

$$I|_{\text{interface}} = I_0 \prod_{i=1}^M e^{-\Sigma_i D_i}, \quad (24)$$

where Σ_i is the macroscopic cross section for each layer, D_i is the thickness of each layer, and M is the number of detector layers *excluding* the thermal neutron reactive film. Assuming orthogonal backside irradiation, the sensitivity contribution for each particle is determined by integrating the product of Equations 5 and 6 over the film thickness D_F . Hence, we have

$$\begin{aligned} S_p(D_F) &= \frac{F_p}{4\pi I_0} I_0 \prod_{i=1}^M e^{-\Sigma_i D_i} \int_0^{D_F} 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{x}{L}\right) dx \\ &= 0.5 F_p \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ \left(1 - \frac{1}{\Sigma_F L}\right) \left(1 - e^{-\Sigma_F D_F}\right) + \frac{D_F}{L} e^{-\Sigma_F D_F} \right\}, \quad D_F \leq L \end{aligned} \quad (25a)$$

and

$$\begin{aligned} S_p(D_F) &= \frac{F_p}{4\pi I_0} I_0 \prod_{i=1}^M e^{-\Sigma_i D_i} \int_0^L 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{x}{L}\right) dx \\ &= 0.5 F_p \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ \left(1 - \frac{1}{\Sigma_F L}\right) \left(1 - e^{-\Sigma_F L}\right) + e^{-\Sigma_F L} \right\}, \quad D_F > L \end{aligned} \quad (25b)$$

Figure 19: Coordinate system for back irradiation of coated neutron detectors.

Contrary to the front irradiation case, the lowest neutron interaction rate within the reactive film is the farthest distance from the film-detector junction, and the highest neutron interaction rate within the reactive film is adjacent to the film-detector junction (Figure 19). As a result, the backside irradiated devices can achieve slightly higher neutron sensitivities than front side irradiated devices (Figures 17 and 18). To use the backside irradiation scheme effectively, the semiconductor detector layers should be very thin and composed of materials with relatively small neutron interaction cross sections. Otherwise, significant neutron losses will occur before they can reach the reactive film, thereby undermining the advantage of backside irradiation. The total neutron sensitivity can be found by employing Equations 19 and 25.

d. Detector Angle and the Effects

i. The Simple Infinite Plane Case

Turning the device at an angle is a method that can be used to increase the thermal neutron detection efficiency per unit area for neutrons approaching from one direction. In the present analysis, it is assumed that the device has infinite dimensions in the Z (perpendicular to the page) and Y directions, thereby eliminating end effects. Referring to Figure 20, the device can be rotated through angle θ between 0 and 90° with respect to the direction of the intersecting neutrons, where a detector at 0 degrees is orthogonal to the neutron beam and 90° is parallel to the neutron beam.

The effective neutron absorbing film thickness is increased with increasing angle, while the charged particle transit distance required to reach the detector interface remains unchanged. In other words, the neutron absorption efficiency of the detector is increased while maintaining the same minimum detectable solid angle. The neutron detection efficiency for each charged particle reaction product is represented by:

Figure 20: The coordinate system used for determining angular effects on neutron detection efficiency.

$$S_p(D_F) = 0.5 F_p \left\{ \left(1 + \frac{\cos \theta}{\Sigma_F L} \right) \left(1 - e^{-\Sigma_F D_F / \cos \theta} \right) - \frac{D_F}{L} \right\} \quad \text{for } D_F \leq L \quad (26a)$$

and

$$S_p(D_F) = 0.5 F_p e^{-\Sigma_F (D_F - L) / \cos \theta} \left\{ \left(1 + \frac{\cos \theta}{\Sigma_F L} \right) \left(1 - e^{-\Sigma_F L / \cos \theta} \right) - 1 \right\} \quad \text{for } D_F > L. \quad (26b)$$

With the *LLD* set at 300 keV and $D = L_{LR}$, Figure 21 shows a comparison of the calculated sensitivities for a ^{10}B coated device as a function of angle. Notice how the efficiency increases dramatically at angles greater than 45° . The results would indicate that turning the device such that the intersection angle is almost 90° would allow for very high efficiencies, yet the result is a consequence of the infinite plane assumption. For small devices, the actual neutron detection efficiency will be significantly less due to both the “effective sensitive area” decreasing in size and charged particle end effect losses.

ii. The Short Plane Case

The effective sensitive area (A_{eff}) pertains to the actual projected area that the neutron beam intersects for a given device angle. For the single-side-coated neutron detection device, A_{eff} changes proportionally to $\cos \theta$, or rather

$$A_{eff} = A|_{\theta=0} \cos \theta + R_\perp D_F \sin \theta, \quad (27)$$

where $R_\perp$ is the film layer length parallel to the axis of rotation (see Figure 20). Since D_F is relatively thin, the actual sensitivity will decrease almost to zero as the device angle increases to 90° . Correcting for the diminishing apparent detector area, the corrected neutron detection efficiency is represented by:

$$S_p(D_F) = 0.5 \cos \theta F_p \left\{ \left(1 + \frac{\cos \theta}{\Sigma_F L} \right) \left(1 - e^{-\Sigma_F D_F / \cos \theta} \right) - \frac{D_F}{L} \right\} \text{ for } D_F \leq L \quad (28a)$$

and

$$S_p(D_F) = 0.5 \cos \theta F_p e^{-\Sigma_F (D_F - L) / \cos \theta} \left\{ \left(1 + \frac{\cos \theta}{\Sigma_F L} \right) \left(1 - e^{-\Sigma_F L / \cos \theta} \right) - 1 \right\} \text{ for } D_F > L \quad (28b)$$

The results are shown in Figure 21, in which it is clear that the efficiency actually does not increase, but diminishes at angles above 60°. The very important result also indicates that the sensitivity of thin film coated detectors is fairly constant over a large angular distribution, remaining almost constant for neutron sources at angles 60° or less from the detector front and back planes.

Figure 21: Although the infinite plane treatment appears to show efficiency as a strongly changing function with angle, in reality the thermal neutron detection efficiency is fairly constant for angles up to 60°. Efficiency diminishes at source to detector angles greater than 60°.

e. Efficiency as a Function of *LLD* Criteria

Thus far a 300 keV minimum detectable energy criteria has been imposed such that the *LLD* is set at 300 keV equivalent. As the case may be, the background radiation field, most likely gamma rays, will either allow for a higher or lower *LLD* setting. The observed efficiency for front or back irradiation devices will increase as the *LLD* is lowered, and will decrease as the *LLD* is raised. Obviously, if the *LLD* is lowered, then the minimum detectable energy criteria is less, which allows for longer effective ranges for both particles. Longer effective ranges allow for thicker films, hence the efficiency should increase. The opposite effect is true if the *LLD* is raised.

Since the effective range (*L*) is a function of the *LLD* setting, then the optimum film thickness will also change as a function of *LLD* setting. Using the *L* and ΣL values listed in Tables 1 and 2, Figure 22 shows the expected efficiencies for ^{10}B -coated devices as a function of *LLD* setting for front irradiation. As can be seen, simply reducing the *LLD* to lower values can increase the efficiency, yet the risk of including background radiation events, as from gamma rays, increases as the *LLD* is lowered [20]. Hence, the operator must find a useful

balance for the environment. Also, it is important to note that the optimal thickness changes as a function of LLD as well. Prior knowledge of the radiation environment will assist in design the optimum film thickness for the device.

Figure 22: Thermal neutron detection efficiency as a function of ^{10}B thickness and LLD setting.

Section C: Advanced Designs

1. Stacked Detectors

Stacking individual devices can increase the overall system detection efficiency, although the increase in efficiency does not accumulate linearly. The initial neutron flux will be attenuated by each of the neutron absorbing layers. As a result, the neutron flux decreases for subsequent detectors and the effect must be taken into account when calculating the overall neutron detection efficiency of the stack. For the case in which the neutron sensitive film is of the same thickness for all of the detectors in the stack, the total sensitivity to a reaction product particle can be expressed as

$$S_p(D_F) = 0.5 F_p \sum_{n=0}^{N-1} \left\{ e^{-n(\Sigma_F D_F)} \prod_{i=1}^M e^{-n(\Sigma_i D_i)} \right\} \left\{ \left(1 + \frac{1}{\Sigma_F L} \right) (1 - e^{-\Sigma_F D_F}) - \frac{D_F}{L} \right\} \quad \text{for } D_F \leq L, \quad (29a)$$

and

$$S_p(D_F) = 0.5 F_p e^{-\Sigma_F (D_F - L)} \sum_{n=0}^{N-1} \left\{ e^{-n(\Sigma_F D_F)} \prod_{i=1}^M e^{-n(\Sigma_i D_i)} \right\} \left\{ \left(1 + \frac{1}{\Sigma_F L} \right) (1 - e^{-\Sigma_F L}) - 1 \right\} \quad \text{for } D_F > L, \quad (29b)$$

where N is the number of detectors in the stack, M is the number of layers in each of the individual detectors *excluding* the neutron reactive film, n is the detector number, and i is the film number. For the case in which there is only one detector, the reader may verify that Equations 29a and 29b reduce to Equations 18a and 18b, respectively.

The authors first consider the case in which the semiconductor substrate has negligible neutron scattering and absorption. The optimum thickness for the individual neutron reactive films decreases as the number of detectors in the stack increases. Figures 23-25 show the calculated thermal neutron detection efficiencies for pure ^{10}B , ^6LiF , and pure ^6Li as a function of film thickness and the number of detectors in the stack. Only slight differences in efficiency are detectable for the ^{10}B and ^6LiF coated devices as shown in Figures 23 and 24. The calculations indicate that a stack of ^6LiF coated devices can achieve slightly higher efficiency than a stack of ^{10}B coated devices for a small number of detectors, whereas a stack of ^{10}B coated devices can achieve a slightly higher efficiency than a stack of ^6LiF coated devices for a relatively large number of detectors. Yet, the optimum thickness of material required in the ^6LiF film is over 10 times greater than that required of the ^{10}B film, indicating that detector fabrication will be easiest for ^{10}B coated devices. Figure 25 shows the calculated thermal neutron detection efficiency for a stack of ^6Li coated devices, in which it becomes clear that a stack of ^6Li devices provides much higher achievable neutron detection efficiencies than stacks of ^{10}B or ^6LiF devices. However, the hygroscopic and reactive nature of pure ^6Li requires special handling and processing techniques in order to achieve accurate and optimized film thicknesses.

In all cases shown in Figures 23-25, the optimum film thickness decreases as the number of detectors in the stack increases. Additionally, the change in efficiency as a function of film thickness increases as the number of detectors in the stack increases. Hence, film thickness accuracy becomes increasingly important as the stack number increases.

It should also be pointed out that the *increase* in efficiency diminishes as the number of detectors in the stack increases. The overall device design should incorporate a reasonable number of detectors compatible with the available detector power supply and output electronics. Additionally, increasing devices in the detector stack will increase the statistical fluctuations which ultimately contribute to system noise.

Operation of the detector stack affects the performance as well. Two straightforward methods of applying voltage across the detector stack and extracting the signal are the parallel method and the series method. Briefly explained, detectors operated in parallel are individually biased with one electrode connected to the pre-amplifier circuit. Detectors operated in series have the bias applied across the entire detector stack, which are connected in series, and only one of the devices in the series is connected to the pre-amplifier circuit. The parallel method allows for retention of the full induced charge signal, but the capacitance increases by the number of detectors in the stack. The series method reduces the capacitance, but the induced charge signal is reduced by the number of detectors in the stack.

Figure 23: Thermal neutron detection efficiency as a function of ^{10}B thickness and number of stacked detectors.

Figure 24: Thermal neutron detection efficiency as a function of ^6LiF thickness and number of stacked detectors.

Figure 25: Thermal neutron detection efficiency as a function of ^6LiF thickness and number of stacked detectors

Table 3: Corrected ${}^6\text{Li}(\text{n},\alpha){}^3\text{H}$ reaction product ranges for both ${}^6\text{LiF}$ films on ${}^{10}\text{B}$ and ${}^6\text{Li}$ Films on ${}^{10}\text{B}$. The minimum detection threshold used for the calculation was 300 keV.

${}^{10}\text{B}$ thickness (μm)	${}^6\text{LiF}$ atop ${}^{10}\text{B}$		${}^6\text{Li}$ atop ${}^{10}\text{B}$	
	2.730 MeV Triton <u>L (microns)</u>	2.050 MeV α particle <u>L (microns)</u>	2.730 MeV Triton <u>L (microns)</u>	2.050 MeV α particle <u>L (microns)</u>
0.0	29.2	4.65	127	19.1
0.5	29.1	4.57	126	17.0
1.0	29.0	4.43	124	15.4
1.5	28.9	4.35	122	13.7
2.0	28.8	4.34	120	12.0
2.5	28.8	4.30	118	10.3
3.0	28.8	4.25	116	8.40
3.5	28.8	4.23	113	6.80
4.0	28.8	4.23	112	5.05

Figure 26: The general concept of a “double-coated” design.

2. Double-Coated Designs

A method of increasing the overall efficiency of a device incorporates a double-layered film, such as with ${}^6\text{LiF}$ on ${}^{10}\text{B}$ or ${}^6\text{Li}$ on ${}^{10}\text{B}$. From Tables 1 and 2, it is clear that the ${}^{10}\text{B}(\text{n},\alpha){}^7\text{Li}$ reaction products have shorter ranges within ${}^{10}\text{B}$ material than do ${}^6\text{Li}(\text{n},\alpha){}^3\text{H}$ reaction products. Hence, it is possible to put down a coating of ${}^{10}\text{B}$ followed by a coating of either ${}^6\text{LiF}$ or ${}^6\text{Li}$. The charged particle reaction products emanating from ${}^{10}\text{B}(\text{n},\alpha){}^7\text{Li}$ reactions can reach the detector as before since the film is applied directly to the detector contact. In addition, the longer range charged particle products emanating from ${}^6\text{Li}(\text{n},\alpha){}^3\text{H}$ reactions can still reach detector even though they must transit the ${}^{10}\text{B}$ film as well. Since the ${}^{10}\text{B}$ film has a higher neutron interaction cross section than ${}^6\text{LiF}$ or ${}^6\text{Li}$ films, a net gain is realized. Hence, the short product range and high cross section material is deposited closest to the contact (denoted film layer 1) while the longest range and lowest cross section material is placed atop the first film where the upper film is denoted film layer 2. The opposite case renders no improvement, and in fact actually decreases efficiency.

Calculating the actual efficiency of the double film as a function of both film depths can be a bit complicated since the particle ranges and energy losses are not linearly coupled. Additionally, the effective range changes as a function of angle since the distance that the charged particle reaction product must traverse film layer 1 becomes a function of $D_{F1} \sec \theta$, where D_{F1} is the film layer 1 thickness and θ refers to the angle increasing from orthogonal. Hence, Monte-Carlo simulations should be used for the best results. However, given the thin applications of ${}^{10}\text{B}$ that are of practical use (0 to 2.5 μm), it is fortuitous to note that the maximum range, with a 300 keV *LLD* equivalent for 2.7 MeV tritons and 2.05 MeV alpha particles, is only slightly different for a pure ${}^6\text{LiF}$ film and a layered film of ${}^6\text{LiF}$ and 2.5 microns of ${}^{10}\text{B}$ [36]. Therefore sufficiently accurate results can be acquired by using an average effective range for the ${}^6\text{Li}(\text{n},\alpha){}^3\text{H}$ reaction products, which is only slightly different than for the pure ${}^6\text{LiF}$ film. Unfortunately, the effective range *does* change considerably for a system with a pure ${}^6\text{Li}$ film atop a ${}^{10}\text{B}$ film. As a result, the actual effective range, rather than an average, should be used when determining the optimum film thickness combination. Table 3 shows the calculated maximum ranges (300 keV equivalent) for the ${}^6\text{Li}(\text{n},\alpha){}^3\text{H}$ reaction products for both pure ${}^6\text{Li}$ films and ${}^6\text{LiF}$ films on various depths of ${}^{10}\text{B}$ films.

For frontal irradiation, the efficiency for film layer 1 is similar to Equations 18a and 18b, with the difference noted as being corrected for neutron attenuation in film layer 2:

$$S_p(D_{F1}) = \frac{F_p}{4\pi I_0} e^{-\Sigma_{F2}D_{F2}} \int_0^{D_{F1}} I_0 2\pi \Sigma_{F1} e^{-\Sigma_{F1}(D_{F1}-x)} \left(1 - \frac{x}{L_{F1}}\right) dx$$

$$= 0.5 F_p e^{-\Sigma_{F2}D_{F2}} \left\{ \left(1 + \frac{1}{\Sigma_{F1}L_{F1}}\right) \left(1 - e^{-\Sigma_{F1}D_{F1}}\right) - \frac{D_{F1}}{L_{F1}} \right\}, \quad (30a)$$

for $D_{F1} \leq L_{F1}$, and

$$S_p(D_{F1}) = \frac{F_p}{4\pi I_0} e^{-\Sigma_{F1}(D_{F1}-L_{F1})} e^{-\Sigma_{F2}D_{F2}} \int_0^{L_{F1}} I_0 2\pi \Sigma_{F1} e^{-\Sigma_{F1}(D_{F1}-x)} \left(1 - \frac{x}{L_{F1}}\right) dx$$

$$= 0.5 F_p e^{-\Sigma_{F2}D_{F2}} e^{-\Sigma_{F1}(D_{F1}-L_{F1})} \left\{ \left(1 + \frac{1}{\Sigma_{F1}L_{F1}}\right) \left(1 - e^{-\Sigma_{F1}L_{F1}}\right) - 1 \right\}, \quad (30b)$$

for $D_{F1} > L_{F1}$, where the argument $\Sigma_{F2}D_{F2}$ accounts for neutron absorption in film layer 2 before they reach film layer 1. Notice that the effective range is not altered since the products produced need only transit through film layer 1 to reach the detector interface. The efficiency contribution from film layer 2 must account for the change in effective range for reaction products as they transit from film layer 2 into film layer 1, and it must account for the reduction in the maximum solid angle detector intersection probability that is diminished by film layer 1. The efficiency contribution from film layer 2 is:

$$S_p(D_{F2}) = \frac{F_p}{4\pi I_0} \int_{D_{F1}}^{D_{F2}+D_{F1}} I_0 2\pi \Sigma_{F2} e^{-\Sigma_{F2}(D_{F2}-x)} \left(1 - \frac{x}{L_{D_{F1},D_{F2}}}\right) dx$$

$$= \frac{0.5 F_p}{\Sigma_{F2}L_{D_{F1},D_{F2}}} \left\{ \left(1 + \Sigma_{F2}L_{D_{F1},D_{F2}} - \Sigma_{F2}[D_{F1} + D_{F2}]\right) - \left(1 + \Sigma_{F2}L_{D_{F1},D_{F2}} - \Sigma_{F2}D_{F1}\right) \left(e^{\Sigma_{F2}(D_{F1}-D_{F2})}\right) \right\}, \quad (31a)$$

for $D_{F2} + D_{F1} \leq L_{D_{F1},D_{F2}}$, and

$$S_p(D_{F2}) = \frac{F_p}{4\pi I_0} e^{-\Sigma_{F2}(D_{F2}-L_{D_{F1},D_{F2}})} \int_{D_{F1}}^{L_{D_{F1},D_{F2}}} I_0 2\pi \Sigma_{F2} e^{-\Sigma_{F2}(D_{F2}-x)} \left(1 - \frac{x}{L_{D_{F1},D_{F2}}}\right) dx$$

$$= \frac{0.5 F_p e^{-\Sigma_{F2}(D_{F2}-L_{D_{F1},D_{F2}})}}{\Sigma_{F2}L_{D_{F1},D_{F2}}} \left\{ e^{\Sigma_{F2}(L_{D_{F1},D_{F2}}-D_{F2}+D_{F1})} - \left(1 + \Sigma_{F2}L_{D_{F1},D_{F2}} - D_{F1}\Sigma_{F2}\right) e^{\Sigma_{F2}(D_{F1}-D_{F2})} \right\}, \quad (31b)$$

for $D_{F2} + D_{F1} > L_{D_{F1},D_{F2}}$.

From Equations 30 through 31, the thermal neutron detection efficiency as a function of film thickness are shown in Figures 27-30. Figures 27 and 28 show the expected results from a ${}^6\text{LiF}/{}^{10}\text{B}$ system and Figures 29 and 30 show the expected results from a ${}^6\text{LiF}/{}^{10}\text{B}$ system, showing an optimum efficiency of 6.8% for the former and 12.86% for the latter.

Figure 27: The thermal neutron detection efficiency for a double-coated detector using ^6LiF atop ^{10}B .

Figure 28: Expanded view of ^6LiF and ^{10}B film depths for optimized thermal neutron detection efficiency, showing a optimized value of 6.8% detection efficiency when a 300 keV *LLD* criteria is used.

Figure 29: The thermal neutron detection efficiency for a double-coated detector using ^6Li atop ^{10}B .

Figure 30: Expanded view of ^6LiF and ^{10}B film depths for optimized thermal neutron detection efficiency, showing a optimized value of 12.86% detection efficiency when a 300 keV *LLD* criteria is used.

3. The Sandwich Design

“Sandwich” designs can assist in increasing the overall thermal neutron detection efficiency by either doubling the amount of neutron reactive material or doubling the sensitive region in contact with the neutron reactive material. Two configurations will be discussed in the following section; the double-sided film detector design and the double inward facing detector design.

a. Double-Sided Film (or Outward-Facing) Detector

The double-sided film detector consists of a semiconductor detector in which a neutron sensitive film is deposited on the front and backsides. Such a choice seems obvious, but does have drawbacks. The advantages include high neutron detection efficiency and a very low probability of reaction product coincidence counts (from a single neutron) during a measurement. Disadvantages include higher operating voltages and higher gamma ray background interference.

Figure 31 shows a simple depiction of the double-sided film detector design, in which neutron reactive films are deposited on both sides of the device. Semiconductor diodes rarely have similar contacts for both surfaces and the analysis should take this point into account. In general, neutrons orthogonally intersecting the front of the device undergo some absorption in the first film, experience further losses in the detector contacts and detector bulk, and again undergo absorption in the second film. Charged particles emitted from reactions in either film can be detected from the detector side adjacent to the film. For simplicity, assume that the neutron reactive films, both front and back, are of the same thickness.

Figure 31: The most basic “sandwich” design, in which a neutron reactive coating is placed upon both sides of a diode.

Consider the case in which that the detector thickness is greater than L_{LR} . The solution to the overall efficiency can be found by adding the appropriate mix of Equations 18 and 25, while accounting for neutron attenuation in the first film. Hence, for each charged particle reaction product,

$$S_p(D_F) = 0.5F_p \left[\left\{ \left(1 + \frac{1}{\Sigma_F L} \right) (1 - e^{-\Sigma_F D_F}) - \frac{D_F}{L} \right\} + e^{-\Sigma_F D_F} \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ \left(1 - \frac{1}{\Sigma_F L} \right) (1 - e^{-\Sigma_F D_F}) + \frac{D_F}{L} e^{-\Sigma_F D_F} \right\} \right],$$

$$D_F \leq L, \quad (32a)$$

and

$$S_p(D_F) = 0.5 F_p \left[e^{-\Sigma_F(D_F-L)} \left\{ \left(1 + \frac{1}{\Sigma_F L} \right) (1 - e^{-\Sigma_F L}) - 1 \right\} + e^{-\Sigma_F D_F} \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ \left(1 - \frac{1}{\Sigma_F L} \right) (1 - e^{-\Sigma_F L}) + e^{-\Sigma_F L} \right\} \right], \quad D_F > L, \quad (32b)$$

where the subscript i refers to the bulk detector layer and the contact layers, both front and back, and M refers to the total number of layers *excluding* the neutron reactive films. Notice that the back facing film terms (second portion) in Equations 32a and 32b are corrected for neutron attenuation through the front facing neutron reactive film.

The device will only work properly if the applied operating voltage extends the electric field entirely across the semiconductor detector. The entire detector bulk then becomes sensitive to charge particles entering the device from either side. Unfortunately, it also becomes more sensitive to background gamma rays as well. Minimizing the bulk detector thickness minimizes the required operating voltage and reduces background radiation noise. However, decreasing the overall bulk detector thickness increases the device capacitance, which produces higher electronic noise. Materials composed of relatively high atomic number (Z) materials increase the gamma ray absorption probability. Hence, a low Z material is preferred to reduce gamma ray background interference.

An alternative, but similar, method of producing the same effect is shown in Figure 32, in which two single coated devices are placed back-to-back and connected as if they were one device. The advantages includes:

- i. Simple fabrication with only one coating step.
- ii. Lower operating voltages, requiring only that needed to deplete a small region near the reactive film.

The main disadvantage is the increase in capacitance, which will double when using two identical devices.

b. Double Inward Facing Detectors

Another design that does not rely on the full depletion of the detectors, and can still be operated with modest operating voltages, is the double inward facing detector design, as shown in Figure 33. The design offers another method to increase the thermal neutron detection efficiency, but does have drawbacks.

If the neutron reactive film thickness is too thin, coincident charged particle emissions from a single neutron absorption event can be measured simultaneously by both detectors, thus giving rise to the erroneous recording of two neutron interaction events when only one actually occurred. The total film thickness separating the detectors must be greater than the sum of $L_{LR} + L_{SR}$ in order to confidently eliminate the possibility of recording coincident particles from a single neutron absorption. The most straightforward method for producing such a device is to simply fasten two front-coated devices together. With identical devices, the neutron sensitive film thickness for each individual detector should range between and $(L_{LR} + L_{SR})/2$ and L_{LR} . If the device is operated in pulse mode, anti-coincidence counting techniques can be used to reduce the probability of recording two counts for one neutron absorption event, thereby lifting the film thickness constraint

Figure 32: The back-to-back method of producing a sandwich detector.

Figure 33: The double-inward facing detector design.

The double inward facing detector configuration allows for the realization of coincident counts, in which both detectors can detect an ion from the very same neutron absorption. The problem can be remedied with anti-coincident counting electronics if using the devices individually, or most easily, by simply connecting the diodes together with one output. However, to determine the expected efficiency, the probability of a coincident event must be subtracted from the total added probabilities that individual reaction product ions will enter the semiconductor detectors. In other words:

$$S(D_F)|_{Total} = \sum_{p=1}^N S_p(D_F) - \sum_{i=1}^R [Q_{LR,SR}(D_F) + Q_{SR,LR}(D_F)] \quad , \quad (33)$$

where R is the number of reaction branches (assume two for ^{10}B and one for ^6Li) and Q is the integrated fractional probability that coincident counts occur with film thickness D_F . The subscripts of Q refer to which detector (front or back) that the long range and short range particles enter. For instance, $Q_{LR,SR}$ refers to coincidences in which a long-range particle enters the front detector and a coincident count from a short-range particle enters the back detector. Obviously, $Q_{SR,LR}$ refers to the opposite case. Equations 18a, 18b, 25a and 25b can be used together to determine the values of $S_p(D_F)$, although the integration boundary limits must be administered correctly. Solving for the integrals, the detection probability integrals for a particle entering the *front* detector are:

$$S_{pF}(D_F) = \frac{F_p}{4\pi I_0} I_0 \prod_{i=1}^M e^{-\Sigma_i D_i} \int_0^{2D_F} 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{x}{L}\right) dx$$

$$= 0.5 F_p \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ \left(1 - \frac{1}{\Sigma_F L}\right) \left(1 - e^{-2\Sigma_F D_F}\right) + \frac{2D_F}{L} e^{-2\Sigma_F D_F} \right\} \quad \text{for } L \geq 2D_F, \quad (34a)$$

$$S_{pF}(D_F) = \frac{F_p}{4\pi I_0} I_0 \prod_{i=1}^M e^{-\Sigma_i D_i} \int_0^L 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{x}{L}\right) dx$$

$$= 0.5 F_p \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ \left(1 - \frac{1}{\Sigma_F L}\right) \left(1 - e^{-\Sigma_F L}\right) + e^{-\Sigma_F L} \right\} \quad \text{for } L \leq 2D_F. \quad (34b)$$

The detection probability integrals for a particle entering the back detector are:

$$S_{pB}(D_F) = \frac{F_p}{4\pi I_0} \prod_{i=1}^M e^{-\Sigma_i D_i} \int_0^{2D_F} I_0 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{(2D_F - x)}{L}\right) dx$$

$$= 0.5 F_p \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ \left(1 + \frac{1}{\Sigma_F L}\right) \left(1 - e^{-2\Sigma_F D_F}\right) - \frac{2D_F}{L} \right\} \quad \text{for } L \geq 2D_F, \quad (35a)$$

$$S_{pB}(D_F) = \frac{F_p e^{-\Sigma_F(2D_F-L)}}{4\pi I_0} \prod_{i=1}^M e^{-\Sigma_i D_i} \int_{2D_F-L}^{2D_F} I_0 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{(2D_F - x)}{L}\right) dx$$

$$= 0.5 F_p e^{-\Sigma_F(2D_F-L)} \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ \left(1 + \frac{1}{\Sigma_F L}\right) \left(1 - e^{-\Sigma_F L}\right) - 1 \right\} \quad \text{for } L \leq 2D_F. \quad (35b)$$

For any thickness of D_F , the appropriate equations must be used for each reaction product particle and for each detector, front or back.

After adding the solutions from the appropriate forms of Equations 34a, 34b, 35a and 35b, the reader should understand that coincident events have been included. In other words, many events have been counted twice. The probability of coincident events must be subtracted from the solution shown in Equations 34 and 35. Coincident events occur when both particles, after being ejected in opposite directions, are within solid angles which intersect both the front and back detectors. The smallest intersecting solid angle for a single particle determines the probability, *not* the addition of solid angles for both particles, and *not* the larger of solid angles intersecting either of the detectors. The appropriate integration range can be found by setting

the solid angles for the opposing particles equal. Hence, from Equation 6, the largest ratio of x/L for L_{LR} or L_{SR} determines the smallest angle. With respect to range along the x -axis (see Figure 34), the integration limits when the short-range particle enters the front detector are then defined as:

$$0 \leq x \leq \frac{2D_F L_{SR}}{L_{LR} + L_{SR}} = \Theta, \quad (36a)$$

where the solid angle of the *long range particle* determines the coincidence probability, and

$$2D_F \geq x \geq \frac{2D_F L_{SR}}{L_{LR} + L_{SR}} = \Theta, \quad (36b)$$

where the solid angle of the *short range particle* determines the coincidence probability.

Figure 34: Diagram depicting coincidence counts when the short-range product enters the front device and the long-range product enters the back device.

The accompanying integrals become particle specific:

$$\begin{aligned} Q_{SR,LR}(D_F) &= \frac{F_p}{4\pi I_0} I_0 \prod_{i=1}^M e^{-\Sigma_i D_i} \left[\int_0^{\Theta} 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{(2D_F - x)}{L_{LR}} \right) dx \right. \\ &\quad \left. + \int_{\Theta}^{2D_F} 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{x}{L_{SR}} \right) dx \right] \\ &= 0.5 F_p \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ e^{-\Sigma_F \Theta} \left[\frac{2\Sigma_F D_F - L_{LR} \Sigma_F - \Sigma_F \Theta - 1}{L_{LR} \Sigma_F} \right] - \left[\frac{2\Sigma_F D_F - L_{LR} \Sigma_F - 1}{L_{LR} \Sigma_F} \right] \right. \\ &\quad \left. + e^{-2\Sigma_F D_F} \left[\frac{2\Sigma_F D_F - L_{SR} \Sigma_F + 1}{L_{SR} \Sigma_F} \right] + e^{-\Sigma_F \Theta} \left[\frac{L_{SR} \Sigma_F - \Theta \Sigma_F - 1}{L_{SR} \Sigma_F} \right] \right\} \\ &\quad , \quad \text{for } 2D_F \leq L_{SR}, \quad (37a) \end{aligned}$$

$$\begin{aligned}
Q_{SR,LR}(D_F) &= \frac{F_p}{4\pi I_0} I_0 \prod_{i=1}^M e^{-\Sigma_i D_i} \left[\int_0^{\Theta} I_0 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{(2D_F - x)}{L_{LR}} \right) dx \right. \\
&\quad \left. + \int_{\Theta}^{L_{SR}} 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{x}{L_{SR}} \right) dx \right] \\
&= 0.5 F_p \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ e^{-\Sigma_F \Theta} \left[\frac{2\Sigma_F D_F - \Sigma_F L_{LR} - \Theta \Sigma_{LR} - 1}{\Sigma_F L_{LR}} \right] - \left[\frac{2\Sigma_F D_F - \Sigma_F L_{LR} - 1}{\Sigma_F L_{LR}} \right] \right. \\
&\quad \left. + \left[\frac{e^{-\Sigma_F L_{SR}}}{\Sigma_F L_{SR}} \right] + e^{-\Sigma_F \Theta} \left[\frac{\Sigma_F L_{SR} - \Sigma_F \Theta - 1}{\Sigma_F L_{SR}} \right] \right\} \\
&\quad , \quad \text{for } L_{SR} \leq 2D_F \leq L_{LR} , \quad (37b)
\end{aligned}$$

$$\begin{aligned}
Q_{SR,LR}(D_F) &= \frac{F_p}{4\pi I_0} I_0 \prod_{i=1}^M e^{-\Sigma_i D_i} \left[\int_{2D_F - L_{LR}}^{\Theta} I_0 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{(2D_F - x)}{L_{LR}} \right) dx \right. \\
&\quad \left. + \int_{\Theta}^{L_{SR}} 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{x}{L_{SR}} \right) dx \right] \\
&= 0.5 F_p \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ e^{-\Sigma_F \Theta} \left[\frac{2\Sigma_F D_F - \Sigma_F L_{LR} - \Sigma_F \Theta - 1}{\Sigma_F L_{LR}} \right] + \left[\frac{e^{-\Sigma_F (2D_F - L_{LR})}}{\Sigma_F L_{LR}} \right] \right. \\
&\quad \left. + \left[\frac{e^{-\Sigma_F L_{SR}}}{\Sigma_F L_{SR}} \right] + e^{-\Sigma_F \Theta} \left[\frac{\Sigma_F L_{SR} - \Sigma_F \Theta - 1}{\Sigma_F L_{SR}} \right] \right\} \\
&\quad , \quad \text{for } L_{LR} \leq 2D_F \leq L_{SR} + L_{LR} , \quad (37c)
\end{aligned}$$

$$Q_{SR,LR}(D_F) = 0 \quad , \quad \text{for } L_{SR} + L_{LR} \leq 2D_F . \quad (37d)$$

Figure 35: Diagram depicting coincidence counts when the long-range product enters the front device and the short-range product enters the back device.

For the *long-range particle* intersecting the *front detector* (see Figure 35), the integration limits are:

$$0 \leq x \leq \frac{2D_F L_{LR}}{L_{LR} + L_{SR}} = \Gamma, \quad (38a)$$

where the solid angle of the *short range particle* determines the coincidence probability, and

$$2D_F \geq x \geq \frac{2D_F L_{LR}}{L_{LR} + L_{SR}} = \Gamma, \quad (38b)$$

where the solid angle of the *long range particle* determines the coincidence probability. The accompanying integrals become particle specific:

$$\begin{aligned} Q_{LR,SR}(D_F) &= \frac{F_p}{4\pi I_0} I_0 \prod_{i=1}^M e^{-\Sigma_i D_i} \left[\int_0^\Gamma I_0 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{(2D_F - x)}{L_{SR}} \right) dx \right. \\ &\quad \left. + \int_\Gamma^{2D_F} 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{x}{L_{LR}} \right) dx \right] \\ &= 0.5 F_p \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ e^{-\Sigma_F \Gamma} \left[\frac{2\Sigma_F D_F - \Sigma_F L_{SR} - \Sigma_F \Gamma - 1}{\Sigma_F L_{SR}} \right] - \left[\frac{2\Sigma_F D_F - \Sigma_F L_{SR} - 1}{\Sigma_F L_{SR}} \right] \right. \\ &\quad \left. + e^{-2\Sigma_F D_F} \left[\frac{2\Sigma_F D_F - \Sigma_F L_{LR} + 1}{\Sigma_F L_{LR}} \right] + e^{-\Sigma_F \Gamma} \left[\frac{\Sigma_F L_{LR} - \Sigma_F \Gamma - 1}{\Sigma_F L_{LR}} \right] \right\} \\ &, \quad \text{for } 2D_F \leq L_{SR}, \quad (39a) \end{aligned}$$

$$\begin{aligned}
Q_{LR,SR}(D_F) &= \frac{F_p}{4\pi I_0} I_0 \prod_{i=1}^M e^{-\Sigma_i D_i} \left[\int_{2D_F-L_{SR}}^{\Gamma} 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{x}{L_{SR}}\right) dx \right. \\
&\quad \left. + \int_{\Gamma}^{2D_F} I_0 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{(2D_F - x)}{L_{LR}}\right) dx \right] \\
&= 0.5 F_p \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ e^{-\Sigma_F \Gamma} \left[\frac{2\Sigma_F D_F - \Sigma_F L_{SR} - \Sigma_F \Gamma - 1}{\Sigma_F L_{SR}} \right] + \left[\frac{e^{-(2D_F-L_{SR})\Sigma_F}}{\Sigma_F L_{SR}} \right] \right. \\
&\quad \left. + e^{-2\Sigma_F D_F} \left[\frac{2\Sigma_F D_F - \Sigma_F L_{LR} + 1}{\Sigma_F L_{LR}} \right] + e^{-\Sigma_F \Gamma} \left[\frac{\Sigma_F L_{LR} - \Sigma_F \Gamma - 1}{\Sigma_F L_{LR}} \right] \right\} \\
&\quad , \quad \text{for } L_{SR} \leq 2D_F \leq L_{LR}, \quad (39b)
\end{aligned}$$

$$\begin{aligned}
Q_{LR,SR}(D_F) &= \frac{F_p}{4\pi I_0} I_0 \prod_{i=1}^M e^{-\Sigma_i D_i} \left[\int_{2D_F-L_{SR}}^{\Gamma} 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{x}{L_{LR}}\right) dx \right. \\
&\quad \left. + \int_{\Gamma}^{L_{LR}} I_0 2\pi \Sigma_F e^{-x \Sigma_F} \left(1 - \frac{(2D_F - x)}{L_{SR}}\right) dx \right] \\
&= 0.5 F_p \prod_{i=1}^M e^{-\Sigma_i D_i} \left\{ e^{-\Sigma_F \Gamma} \left[\frac{2\Sigma_F D_F - \Sigma_F L_{SR} - \Sigma_F \Gamma - 1}{\Sigma_F L_{SR}} \right] + \left[\frac{e^{-(2D_F-L_{SR})\Sigma_F}}{\Sigma_F L_{SR}} \right] \right. \\
&\quad \left. + \left[\frac{e^{-\Sigma_F L_{LR}}}{\Sigma_F L_{LR}} \right] + e^{-\Sigma_F \Gamma} \left[\frac{\Sigma_F L_{LR} - \Sigma_F \Gamma - 1}{\Sigma_F L_{LR}} \right] \right\} \\
&\quad , \quad \text{for } L_{LR} \leq 2D_F \leq L_{SR} + L_{LR}, \quad (39c)
\end{aligned}$$

$$Q_{LR,SR}(D_F) = 0 \quad , \quad \text{for } L_{SR} + L_{LR} \leq 2D_F. \quad (39d)$$

The possibility of coincident counts is eliminated for the condition $(L_{SR} + L_{LR}) \leq 2D_F$. The appropriate combinations of Equations 34 through 39 can describe the neutron detection efficiency of the device. Figures 36-38 show comparison efficiencies for sandwich devices, both double-outward and double-inward facing designs, coated with ^{10}B , ^6LiF , and pure ^6Li films. Both methods of producing higher efficiency are valid. Note that neutron absorption in the contacts and the detector bulk should be accounted for when designing the device, especially in cases where the contact and/or detector materials have relatively large neutron attenuation cross sections. The reader is directed to inspect the obvious improvement for the double-inward facing device structure over the double outward facing (front and back coated) device structure. Not only is there a slight increase in efficiency for the double-inward facing device design, but also the required film thickness per each device to optimize performance is less. For instance, in Figure 36, given the 300 keV *LLD* requirement, outward-facing ^{10}B -

coated devices need to have a film thickness of 2.4 microns each to achieve a maximum thermal neutron efficiency of 7.73%, whereas inward-facing devices need to have a film thickness of only 1.7 microns each to achieve a maximum efficiency of 8.06%. Figures 37 and 38 show similar improvements for ${}^6\text{LiF}$ films and pure ${}^6\text{Li}$ films. Hence, the double-inward sandwich design is best for thermal neutron detection optimization.

Attempts to utilize the coincidence feature for double inward facing devices has been reported [38]. Background radiation interference can be significantly decreased by including only those counts that enter the opposing devices in coincidence, which can be performed in a straightforward fashion with a commercial coincidence analyzer. The expected efficiency from such a system can be calculated from Equations 36-39, in which only the coincident counts contribute to the neutron detection.

Figure 39 shows the calculated thermal neutron detection efficiency for double inward facing devices operated in coincidence mode as a function of film thickness on *each* detector. Hence, the total converter thickness between the two inward facing detectors would be twice that shown on the graph x-axis. The reader is directed to notice that pure ${}^{10}\text{B}$ -based films can achieve higher neutron detection efficiencies than ${}^6\text{LiF}$ -based films, with pure ${}^6\text{Li}$ -based films yielding the highest overall neutron detection efficiencies. It should also be noticed that the maximum thermal neutron detection efficiency achievable is dramatically lower than previous cases, being only 1.87% for ${}^{10}\text{B}$ films, 1.29% for ${}^6\text{LiF}$ films, and 3.91% for ${}^6\text{Li}$ films.

Figure 36: Comparison thermal neutron detection efficiencies for ${}^{10}\text{B}$ -coated devices using the double-inward and the double-outward designs.

Figure 37: Comparison thermal neutron detection efficiencies for ${}^6\text{LiF}$ -coated devices using the double-inward and the double-outward designs.

Figure 38: Comparison thermal neutron detection efficiencies for ${}^6\text{Li}$ -coated devices using the double-inward and the double-outward designs.

To summarize, the double-coated device structures will yield efficiency improvements for thermal neutron detection. The double-coated (front and back) and the double-outward facing devices need not have coincident rejection electronics, yet have a lower maximum achievable thermal neutron efficiency. Also, double-coated outward facing devices require higher operating voltages since the device active region (or depletion region) must extend across the entire detector. The double-inward facing device structure requires lower operating voltages than double-sided devices, and the structure allows for higher thermal neutron detection efficiencies with thinner films than achievable with the double-outward facing design.

Figure 39: The expected thermal neutron detection efficiency as a function of film thickness for double inward devices operated in coincidence.

4. Via Hole Designs

Another method of improving the thermal neutron detection efficiency is by altering the morphology of the detector surface. Devices with cylindrical via holes or straight trenches filled with neutron reactive material will have better neutron detection efficiency than typical flat devices. For a flat device, the probability of a reaction product entering the detector decreases as reactions occur further away from the detector-film/contact interface. However, the same restrictions do not apply for interactions occurring within the via holes, and the probability of a reaction product entering the device is increased for neutron interactions within the via holes. Increased sensitivity is best if the via hole diameters are less than the summed total effective ranges of both reaction products. For the alpha particle and ^7Li emissions in boron, the summed effective ranges [24] equals $3.14\text{ }\mu\text{m}$.

Figure 40. Diagram of a single via hole with dimensions similar to the added ranges of the reaction products. The geometry increases the reaction product detection probability above that of flat surface designs since the products can enter the device from many directions. The via holes are etched so as to cover the entire surface of the detector.

The reaction products can be emitted in any direction over 4π steradians, and the probability that a reaction product will enter the detector is determined by the solid angle that intersects the detector [24]. Figure 40 depicts a sphere with radius equal to a reaction product particle effective range, showing that the solid angle can intersect the bottom and side of the via hole.

Figure 41 shows that there is some amount of sensitivity loss for neutron interactions adjacent to the semiconductor detector flat regions that are also near the via hole edges. The large and small circles represent the long range and short-range particle effective ranges, respectively. In Figures 41a and 41b, the dark gray areas represent angles that the reaction products can fully intersect the semiconductor detector and lose their remaining energy. The light gray areas indicate angles that the reaction products can intersect the semiconductor detector, but can also exit thereby depositing only a portion of their energy. The white areas represent angles where the reaction products would ordinarily be detected by a flat surface device, but will be entirely undetected with the via hole design.

Figure 41. Cross section side views of possible reaction product trajectories for via holes or well structures, in which the neutron interaction occurs above the flat device area and near a via hole edge. The dark gray areas represent angles over which particles enter the detector and lose the remainder of their energy, the light gray areas represent angles over which particle enters the detector and reemerges. The white areas represent angles where particles do not enter the detector, but ordinarily would for a flat surface device. The large and small circles represent the long range and short range reaction-product-effective-ranges, respectively.

Referring to Figures 42 and 43, shown are side and top views of via holes with several possibilities that allow for reaction products to enter the semiconductor detector. The via hole diameters are less than the sum values of the reaction product effective ranges. The gray areas show the angles through which the reaction products can fully intersect the semiconductor detector and lose their remaining energy. Figures 42A, 42B, and 42C show cases in which neutron interactions occur in the center of the via hole at different elevations. In Figures 42A and 42B, the reaction products would not be able to reach the detector for a traditional flat device due to the elevation distance, but since the interactions occurred within a via hole they may enter the device through the side-walls. Figure 42D shows a top view, which demonstrates that there is a good probability that a reaction product will enter the device for any neutron interaction occurring in the via hole center.

Figures 43A, 43B, and 43C show cases in which neutron interactions occur near the side-wall of the via hole at different elevations. At the elevations shown in Figures 43A and 41B, it is again observed that the reaction products would not be able to reach the detector for a

traditional flat device, but may enter the via hole device through the side-walls. Figure 43D shows a top view, which indicates that the probability of a reaction product entering the device is greater for the curved wall than for a flat wall.

Figure 42. Cross section (A, B, and C) and top (D) views of several possible reaction product trajectories for via hole or well structures. The diagrams show cases for neutron interactions at various elevations in the via hole centers. The gray areas represent angles over which particles enter the detector and lose the remainder of their energy. The large and small circles represent the long range and short range reaction-product-effective-ranges, respectively.

Figures 42C and 43C show cases in which the neutron interaction occurs near the via hole bottom and reaction products can enter the bottom of the via holes. At such an interaction elevation, the reaction products could reach the semiconductor detector interface of a traditional flat device, but within a via hole they can also enter the detector sidewalls, indicating a net gain in sensitivity. Sensitivity lost from interactions such as depicted in Figures 41A and 41B is countered by the increased sensitivity depicted in Figures 42C and 43C. The interaction possibilities illustrated in Figures 42 and 43 work to increase the overall detection efficiency. Hence, it is expected that the efficiency will continue to increase with increasing via hole depth.

There are other patterns that may be used to increase the neutron detection efficiency. One such configuration is a system of narrow straight trenches. Figure 44 depicts a top view comparison of straight trenches and via holes. Straight trenches can be designed to run the entire length of a device. The straight trench design allows for an increase in detection sensitivity for the same reasons previously explained. Figure 45 shows side and top views of trenches with two interaction possibilities that allow for reaction products to enter the semiconductor detector. As shown in Figures 45A1 and 45A2, only those particles emitted in the general direction towards a trench wall can be detected. Unfortunately, particles with trajectories generally directed parallel to the trench length will not enter the semiconductor detector and will not be detected, a case not found with the via hole design.

However, the filled trench areal density can be quite high. Using the criteria that the trenches are 3 μm wide, a 50% areal density is achieved with the trenches spaced 6 μm apart

center-to-center, and a 60% areal density is achieved if the trenches are spaced 5 μm apart center-to-center. From Figure 44, using the same criteria of 3 μm diameter holes, an areal density of only 22.67% is achieved with the via holes spaced 6 μm apart center-to-center (hexagonal pattern), and only 32.64% areal density is achieved if the via holes are spaced 5 μm apart center-to-center. As a result, the total area of etched regions filled with reactive material is much smaller for the circular via hole design than the long-trench design.

Although the trenches and via holes can be packed closer together, the detector mass available to absorb the particle energy becomes diminished. Using the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction as an example and assuming orthogonal trajectories, 3 μm of GaAs material can absorb all of the ^7Li ion energy, but only about 1.1 MeV of the alpha particle energy [30]. Furthermore, 2 μm of GaAs material can still absorb almost all of the ^7Li ion energy, but only about 700 keV of the alpha particle energy [30]. Research for the optimum design that will yield the highest possible detection efficiency is ongoing, including patterns mixing via holes with circular, square, rectangular, and ellipse shapes. Some via hole patterns can exceed 40% area coverage.

It is important to note that reaction products emitted from interactions occurring within a via hole have a much higher chance of being detected than interactions occurring within long trenches. Yet, the long-trench design allows for more area to be covered with depressions filled with neutron reactive material. It is expected that an optimum design will use both concepts together to render the best neutron detection sensitivity. Calculations using a simple geometrical model project that thermal-neutron detection efficiency can reach 10% with devices using 10 micron deep via holes with a surface area coverage of 40% (Figure 46), a 2.5 times increase in efficiency over the calculated 3.95% efficiency limit for flat devices [34, 35].

Figure 43. Cross section (A, B, and C) and top (D) views of several possible reaction product trajectories for via hole or well structures. The diagrams show cases for neutron interactions at various elevations at locations near the via hole side-walls. The gray areas represent angles over which particles enter the detector and lose the remainder of their energy. The large and small circles represent the long range and short range reaction-product-effective-ranges, respectively.

Figure 44. Top views of long-trench and via hole patterns for neutron detectors. The dark gray areas represent the reactive-film-filled depressions.

Figure 45. Cross section (A) and top (B) views of two possible reaction product trajectories for long-trench structures. The diagrams show cases for neutron interactions locations in the trench center and near the side-wall. The gray areas represent angles over which particles enter the detector and lose the remainder of their energy. The large and small circles represent the long range and short range reaction-product-effective-ranges, respectively.

Figure 46. Modeled efficiency for a device structure using a matrix of via holes covering 40% of the area. The via hole diameters are 3.0 μm .

5. Compound Designs

The compound design utilizes all of the detection efficiency enhancing features discussed in this section. Another advantage of the pocked design is its flexibility with compound designs. Integrating the efficiency enhancing features described in sections A, B, and C results in a remarkable device capable of exceeding 35% thermal neutron detection efficiency for a device only 1 mm thick! Figure 47 shows the basic concept. A semiconductor substrate, such as Si, SiC, or GaAs, is etched such that miniature holes only 3.5 microns in diameter cover the device. Additionally, larger via holes 100 microns in diameter cover the device, and the smaller holes are within the larger holes. The device is coated with a conductive layer, easily accomplished with sputter deposition, over which a 2-micron deep first layer of ^{10}B is deposited. ^{10}B material is subsequently deposited into the small holes with ultrasonic vibration. Afterwards, a thicker layer of pure ^6Li is deposited over the device such that it fills the larger holes. Two devices are then pressed together such that they face each other. The end result is a remarkably efficient and compact neutron detector promising over 35% thermal neutron detection efficiency for a device no thicker than 1 mm.

Figure 47: Conceptual cross section view of the proposed integrated sandwich/double-layered/pocked neutron detector.

Section C: Device Fabrication

1. General Device Fabrication

Figure 48: The basic structure of a SI GaAs Schottky barrier device.

The Schottky contact provides a blocking barrier that reduces reverse leakage currents

Figure 49: The general band structure of a Schottky barrier diode.

Bulk GaAs Schottky barrier detectors have been shown to have unusual electric field distributions. The electric field is confined near the Schottky contact, in which the magnitude is fairly constant between 10^4 and 2×10^4 V/cm. The phenomenon is referred to as the “truncated effect field” (TEF) effect [10]. Further in the bulk the electric field rapidly reduces to negligible levels. The electric field region width increases linearly with applied voltage across the diode with an increase of approximately 1 volt per micron. Hence, with only 10 volts applied reverse bias, the high electric field region will extend beyond the range of the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction product ranges. The remaining inactive portion of the diode is insensitive to gamma ray background, thus allowing for self-discrimination of gamma rays. Additionally, the diode capacitance, since they are manufactured from the semi-insulating material, is a function of the device width and not the active region width. Hence, the capacitance can be kept relatively low for the very thin active region device. Figure 48 shows the general device configuration for a SI bulk GaAs neutron detector. Figure 49 shows the energy band structure for a Schottky barrier diode. The energy barrier assists in reducing contact leakage current.

Schottky barrier bulk GaAs diodes were manufactured for experimental comparisons of theoretical values. A variety of other semiconductor substrates can be used, such as SiC, Si, or diamond, yet bulk GaAs devices are very easy to manufacture and are actually less costly than devices fabricated from zone-refined Si. The GaAs diodes were manufactured in the following manner. Bulk semi-insulating (SI) GaAs wafers were acquired from American Xtal Technologies (AXT). The back surfaces were lapped at 30 rpm with 3 micron calcined aluminum oxide powder/ de-ionized water solution over an optically flat glass plate until 100 microns of GaAs material was removed. Afterwards, the backsides were polished with 0.3 micron calcined aluminum oxide powder mixed in a sodium hypochlorite solution (trade name “Chemlox” [11]) over a chemically resistant polishing cloth (“Chemcloth” [12]) at 65 rpm for 10 minutes. A final polish was performed with a methyl alcohol (250 ml): glycerol (250 ml): bromine (5 ml) solution for 10 minutes over another Chemcloth polishing pad at 70 rpm. The wafers were cleaned in a series of solvents and etched in a H_2SO_4 (1 ml): H_2O_2 (1 ml): de-ionized water (320 ml) solution for 5 minutes followed by a 2 minute etch in a HCl (200 ml):

de-ionized water (200 ml) solution. The wafers were then cleaned in a de-ionized water cascade and blown dry with N_2 .

Two methods were used to produce low resistivity *n*-type contacts on the backsides of the devices. In the more traditional manner, a layered combination of Ni, Au, and Ge was evaporated onto the polished and etched GaAs surfaces. The evaporated contacts were subsequently sintered for 1 minute at 400°C in argon to produce a low-resistivity ("ohmic") contact [41].

With a lower temperature procedure, the backsides were implanted at an angle of 7° from normal with ^{29}Si ions at energies of 50 keV, 100 keV, and 180 keV with doses amounting to 4×10^{13} ions per cm^2 , 5×10^{13} ions per cm^2 and 6×10^{13} ions per cm^2 , respectively. The implants were activated with a rapid thermal anneal in Ar for 30 seconds at 800°C. Afterwards, a stacked layer of Ge (500 Å): Pd (1300 Å) was vacuum evaporated over the backsides, followed by a low temperature anneal of 250°C in N_2 for 30 minutes. Vacuum evaporation of a stacked layer of Ti (150 Å): Au (700 Å) completed backside processing of the devices.

Front side processing of the devices included lapping and polishing of the samples. The initial lapping with 3 micron calcined aluminum oxide powder was used to thin the wafers to 250 microns total thickness. Next, the wafers were polished using the 0.3 micron calcined aluminum oxide powder mixed in a sodium hypochlorite solution followed by the methyl alcohol (250 ml): glycerol (250 ml): bromine (5 ml) solution as previously described. Again, the wafers were cleaned in a series of solvents and etched in a H_2SO_4 (1 ml): H_2O_2 (1 ml): de-ionized water (320 ml) solution for 5 minutes followed by a 2 minute etch in a HCl (200 ml): de-ionized water (200 ml) solution. Care was taken to coat the backsides with photoresist to prevent the acid solution from attacking the previously processed surfaces. Afterwards, the wafers were cleaned in a de-ionized water cascade and blown dry with N_2 .

Circular defined patterns of 3.5 mm diameter were patterned onto the surface with photoresist. Photoresist residue in the patterns was removed (or "de-scumed") in an oxygen plasma for 2 minutes. A final etch in the circular patterns was performed with the H_2SO_4 (1 ml): H_2O_2 (1 ml): de-ionized water (320 ml) solution (5 minutes again) followed by the HCl (200 ml): de-ionized water (200 ml) solution (2 minutes). The wafers were washed in a de-ionized water cascade and blown dry with N_2 . A stacked layer of Ti (150 Å): Au (450 Å) was evaporated over the wafer and lifted off in acetone. Circular photoresist defined patterns of 3 mm diameter were centered and patterned over the Ti/Au metal Schottky contacts. The photoresist patterns were "de-scumed" in an oxygen plasma for 2 minutes. The wafers were fastened to a chilled plate in a vacuum evaporator and 98% ^{10}B was vacuum evaporated over the Schottky contacts. Each wafer sample received a different ^{10}B deposition thickness, ranging from 1000 Å up to 2 microns. After evaporation, the residue was lifted off in acetone, leaving behind a layer of ^{10}B fastened in intimate contact with the front Schottky contact.

The ^{10}B film applied onto the diode blocking contacts by an electron beam evaporator ensures accuracy in the film thickness. Furthermore, precise features for small, single pad detectors and pixel detectors are easily achieved with a lift-off process when using evaporation techniques [42]. Achieving the correct ^{10}B film thickness is critical for optimum performance, and undershooting or overshooting the proper thickness will result in decreased neutron detection efficiency, as shown in previous sections. Unfortunately, achieving the optimum ^{10}B thickness of 2.4 μm through evaporation methods has been elusive until recently. For flat surface devices, the ^{10}B film becomes stressed in evaporated layers greater than approximately 1 μm , which causes cracking and peeling (see Figures 50 and 51). A variety of multiple layering schemes have been attempted with little success. Films thicker than 1 μm that did adhere often peeled later, especially noticeable with temperature changes. To help prevent peeling, devices were coated immediately after the ^{10}B evaporation step with Humiseal 1A33

or 1B73. Yet, coating with Humiseal for ^{10}B films greater than $1.3\text{ }\mu\text{m}$ was seldom effective, and peeling would still occur for over 90% of the devices while the coating cured. Consequently, ^{10}B coatings were generally held at $1\text{ }\mu\text{m}$ or less, thereby limiting the thermal neutron detection efficiency to only 2.9%. Recently, a new method of surface preparation has allowed for the deposition of films up to $4\text{ }\mu\text{m}$ thick with repeatable and very reliable results. The films no longer peel even without the Humiseal application.

The individual devices were cleaved from the GaAs wafers, and fastened with Ag based epoxy to 1 mm thick aluminum oxide mounts. The mounts were designed with a 3.8 mm hole in the center, around which a Ti ($300\text{ }\text{\AA}$)/ Au ($3000\text{ }\text{\AA}$) contact had been evaporated. The devices were mounted with the backside attached to the alumina substrate holder, and the front side facing out. The hole in the center of the alumina mount was designed to reduce neutron absorption or scattering effects that might alter the measurement accuracy. The detectors were then mounted into aluminum light tight boxes to perform the neutron measurements.

Figure 50. An example of boron film cracking from thin-film stress. Samples generally began showing signs of boron film cracking at a thickness greater than $1.0\text{ }\mu\text{m}$.

Figure 51. An example of boron film peeling off of a detector contact after the completed process.

2. Via Hole Device Fabrication

The devices were built from vertical gradient freeze-grown semi-insulating (SI) GaAs. The SI GaAs wafers were lapped and polished to remove $100\text{ }\mu\text{m}$ of material as described in the literature [30,41]. The wafers were then cleaned and etched to remove residual lapping and polish damage [30,41]. A layered combination of Ni, Au, and Ge was evaporated onto the polished and etched GaAs surfaces. The evaporated contacts were subsequently sintered for 1 minute at 400°C in argon to produce a low-resistivity (“ohmic”) contact [41].

The front surfaces were then lapped and polished until the GaAs wafer thickness was only $200\text{ }\mu\text{m}$. The front surface was then cleaned and etched. A matrix of open circles $3.5\text{ }\mu\text{m}$ in diameter and spaced $9.5\text{ }\mu\text{m}$ apart from center-to-center was patterned with photoresist over the entire wafer surface [43]. A short oxygen plasma clean (2 minutes) was used to remove the residual photoresist film from the hole patterns.

Via holes were then etched into the GaAs wafers in a reactive ion etching (RIE) system with a plasma composed of Ar and BCl_3 gases (Figures 52 and 53). Various via hole depths were fabricated; $1.7\text{ }\mu\text{m}$, $4.0\text{ }\mu\text{m}$, and $5.0\text{ }\mu\text{m}$. Afterwards, the photoresist was removed in the

RIE with an oxygen plasma. For comparison, control samples were also fabricated without undergoing the RIE etch.

The devices were then photoresist-patterned with 3.5 mm diameter open circles covering the surface, spaced apart by 7 mm from center-to-center. A sputtering system was used to coat the wafer with 200 Å of Ti followed by 1000 Å of Au. Direct current (DC) sputtering was used to ensure the formation of a conformal metal coating over and into the via holes. Lift-off in acetone was used to remove excess Ti/Au material. Smaller photoresist circular patterns 3 mm in diameter were centered over the 3.5 mm metal contacts.

Evaporating boron radiates a tremendous amount of heat over the time period required to deposit several microns of material. Excessive substrate heating will cause the photoresist to soften and reflow, which can cause the pattern to smear and can also cause difficulty with the liftoff procedure. To prevent excessive heating, the photoresist patterned samples were attached to a cooling plate inside the evaporator and cooled with chilled water during the ^{10}B evaporation. ^{10}B layers of different thicknesses were deposited, ranging from 1 µm up to 4 µm (Figures 54 -56). Lift-off was used to remove the excess ^{10}B material from the surface.

The devices were cleaved into 7 mm x 7mm squares and mounted with Ag epoxy upon custom Al_2O_3 substrate plates for testing. The substrate plates have Au patterns to which the detectors were bonded for electrical connection.

Before the devices were coated with conductive contacts, the depth of the etched via holes were determined by profilometer measurements and confirmed with a scanning electron microscope (SEM). The etch rate was determined to be 85 Å per minute. Figures 52 and 53 show the resulting matrix of 1.7 µm deep via holes in which straight sidewalls and smooth bottoms are apparent. The smooth surface is important for the manufacture of reliable Schottky blocking contacts.

Figure 54 shows a top view of the same via holes after depositing 4 µm of ^{10}B . The SEM photograph clearly shows that the film did not peel, nor did the other devices in the batch. Indentations are apparent from the top view, and the via hole openings are narrower, which is also shown in the cross section view of Figure 55. The via holes in Figure 55 are filled with conical-shaped boron plugs, which are a consequence of the slowly narrowing opening with increased boron deposition. Also visible in Figure 55 are the conductive contacts on the sides and bottoms of the via holes. This important result demonstrates that evaporation is a method by which the holes can be filled with neutron reactive material.

Figure 52. SEM photograph showing the via hole matrix used in the present work. The substrate is SI GaAs and the via holes are 3.5 µm in diameter, spaced at 9.5 µm center-to-center, and are 1.7 µm deep.

Figure 53. SEM photograph showing the bottoms of the via holes used in the present work. The hole bottoms are smooth, an important result for reliable Schottky contacts.

Figure 1 is a 5x5 grid of 25 micrographs showing the evolution of a polymer film surface under UV irradiation. The images show a progression from a smooth surface to one with small, circular features, and finally to a surface with larger, more complex structures. The grid is labeled with "0 min" at the top left and "240 min" at the bottom right. A scale bar at the bottom right indicates 100 μm .

The image shows a mechanical assembly, possibly a valve or actuator, with a handle and a shaft. The component is mounted on a dark, possibly metallic, surface. The lighting is dramatic, highlighting the metallic surfaces and the mechanical details.

Figure 1 consists of a 5x5 grid of scanning electron micrographs (SEI) showing the surface morphology of the polymer electrolyte membrane. The columns represent different temperatures: 25, 35, 45, 55, and 65 °C. The rows represent different times: 0, 1, 2, 3, and 4 hours. The images show a transition from a relatively smooth surface at 25 °C to a highly textured, porous surface at 65 °C, with the texture becoming more pronounced over time. A scale bar at the bottom indicates 10 μm.

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3. Pixel Detectors

The main pixel detector design incorporated for the present work is a dual in-line array of 0.5 mm x 1.0 mm rectangular Schottky diodes fabricated from bulk semi-insulating (SI) GaAs material. The devices were fabricated in the same manner as described in the section C.1. with the exception that the detectors consisted of long dual in-line arrays. Each individual pad is coated with ^{10}B and has its own small wire bond pad for electrical connection (see Figure 58). A 50-micron gap separates each pixel from its immediate neighbor, hence removing the problem of cross-talk. The pixel design allows for several small detectors to be operated simultaneously so as to gather a digitized image of neutron interactions in the device plane. The linear arrays were designed to extend completely across the GaAs wafers. Hence, the linear array can be sliced to any length thereby allowing for arrays ranging from 8 pixels up to 256 pixels in a single device.

Figure 58: A depiction of a 32-pixel dual in-line array, the typical size used in the present work.

The pixel devices were Ag-epoxied to common chip carriers and wire bonded to the pin-outs (see Figure 59). The chip carriers selected plug into a custom socket produced for the project. The socket was hard-wired into an aluminum box where 16 individual miniaturized preamplifiers were wired to a selected side of the socket (see Figure 60). The aluminum box serves several purposes:

- i. Aluminum has a very low interaction cross section for neutrons, hence will not interfere with the neutron imaging measurements.
- ii. The light impenetrable aluminum box prevents photoelectric noise.
- iii. The conductive sealed aluminum box shields from electromagnetic noise.

The Cremat model CR-101D was the preamplifier type chosen for the preliminary work. The devices have low noise and have a large enough dynamic range to accommodate the large signals expected from the coated neutron detectors. The preamplifiers were arranged in two stacks next to the GaAs array chip, and each preamplifier was capacitively coupled a pixel through the socket. The preamplifier stacks power inputs were connected in parallel, hence only *one power supply*, capable of supplying ± 9 volts, was required to operate the both stacks of preamplifiers. Output signals were extracted through LIMO connectors to standard NIM electronics (described elsewhere).

Figure 59: A 32-pixel dual in-line ^{10}B -coated bulk GaAs Schottky barrier array fastened and packaged in the chip carrier.

Figure 60: The 32-pixel dual in-line ^{10}B -coated bulk GaAs Schottky barrier array located in the aluminum testing box with the accompanying electronics.

4. Self-biased Detectors

The “self-biased” radiation detector design incorporates a thin high purity semiconductor region, ranging from 1 to 6 microns thick, grown upon a bulk substrate. Placed atop the high purity region a blocking contact that is either a thin semiconductor or a Schottky contact. The internal potential formed by the potential difference in the work function of the blocking contact and the high purity region provides enough voltage to operate the device. In the present work, the self-biased detectors were manufactured from n-type GaAs substrates. High purity v-type GaAs epitaxial layers were grown on the GaAs substrates, over which blocking contacts were fabricated. Due to the thinness of the epitaxial layers, only ^{10}B -coated devices were manufactured, primarily because $^6\text{Li}(n,\alpha)^3\text{H}$ reaction products (from ^6LiF and ^6Li layers) would transit well beyond the 6 micron layer and deposit most of their energy within the substrate (insensitive) region.

Self-biased GaAs Neutron Detectors

Figure 61. The basic construction of a ^{10}B -coated self-biased high-purity epitaxial GaAs neutron detector. Neutrons interact in the ^{10}B film, thereby releasing an alpha particle and a ^7Li ion in opposite directions. Only one particle from the interaction can enter the detector. The built-in potential at the contact/v-type GaAs interface supplies enough voltage to operate the device.

Two basic diode structures were investigated, both of which incorporated high-purity undoped GaAs epitaxial layers grown onto *n*-type GaAs substrates. The undoped GaAs regions had background impurity concentrations of $n = 3 \times 10^{14}/\text{cm}^3$, yielding *v*-type unintentionally-doped GaAs material. Two blocking contact adaptations were applied to the high-purity *v*-type GaAs regions: 2000 Å thick *p*+ GaAs blocking contacts and 200 Å thick Schottky blocking contacts. Six variations of these structures were fabricated (see Table 4). The *p*+ contact and high-purity *v*-type epitaxial GaAs films were grown by low-pressure metalorganic chemical vapor deposition. The *v*-type GaAs active layers ranged between 1 micron and 5 microns in thickness. The device sensitive areas were 3mm x 3mm, each coated with a 1.5mm diameter film of 98% enriched high-purity ^{10}B . Two variations were coated with 6500 Å of ^{10}B , one with 7200 Å of ^{10}B , and three with 5000 Å of ^{10}B . The detector elements were bonded to common T0-5 headers and tested for thermal neutron sensitivity in custom designed, light impenetrable aluminum boxes. The difference between the work functions of the blocking contacts (*p*+ GaAs layers or Schottky contacts) and the *v*-type high-purity GaAs layers produces an internal potential difference at the film interface [44]. The “built-in” potential produces an internal electric field. The internal electric field is strong enough to deplete a good portion of the *v*-type GaAs region and transport a considerable fraction of the charge carriers excited by the charged particle across the active *v*-type region. Therefore, external voltage bias is not required to operate the devices. Voltage need only be applied in order to operate the supporting readout electronics. The basic *p-n* junction approximation [44] indicates that the built-in potential should deplete at least 1.5 microns of the *v*-type region. The Schottky barrier devices demonstrated slightly higher reverse leakage currents than the *p*+ contact devices, an expected result due to the lower barrier height generally formed by Schottky contacts as compared to *p-i-n* structures. Furthermore, the built-in potential of the *p-i-n* structure is higher than the Schottky barrier structure due to the larger difference in work functions between the *p*+ region and the *v*-type GaAs region, resulting in a higher internal electric field.

Table 4: High-Purity GaAs Diodes Investigated as Neutron Detectors

Sample Number	Diode Type	Boron-10 Thickness	Blocking Contact	<i>v</i> -type GaAs ($n = 3 \times 10^{14}/\text{cm}^3$)	Intermediate Layer	Substrate
3241	<i>p-i-n</i>	5000 Å	<i>p</i> + GaAs ($2 \times 10^{19}/\text{cm}^3$) 2000 Å	5 µm thick	None	<i>n</i> -type GaAs
3242	<i>p-i-n</i>	6500 Å	<i>p</i> + GaAs ($2 \times 10^{19}/\text{cm}^3$) 2000 Å	2 µm thick	None	<i>n</i> -type GaAs
3243	<i>p-i-n</i>	7200 Å	<i>p</i> + GaAs ($2 \times 10^{19}/\text{cm}^3$) 2000 Å	5 µm thick	None	<i>n</i> -type GaAs
3246	<i>p-i-n</i>	5000 Å	<i>p</i> + GaAs ($2 \times 10^{19}/\text{cm}^3$) 2000 Å	1 µm thick	None	<i>n</i> -type GaAs
3247	<i>p-i-n</i>	5000 Å	<i>p</i> + GaAs ($2 \times 10^{19}/\text{cm}^3$) 2000 Å	2 µm thick	<i>n</i> -type $\text{Al}_x\text{Ga}_{1-x}\text{As}$ ($5 \times 10^{15}/\text{cm}^3$) 2500 Å	<i>n</i> -type GaAs
3278	Schottky Diode	6500 Å	Au Schottky Contact 200 Å	5 µm thick	None	<i>n</i> -type GaAs

Energy self-absorption from ions crossing the 2000 Å p^+ layer will ultimately reduce the amount of energy deposited directly in the detector. Assuming that electron-hole pairs excited in the p^+ region do *not* contribute to the measured output signal, calculations indicate that energy lost from 840 keV ^7Li ions crossing the p^+ region will range between 40 and 100 keV depending upon the particle trajectory and $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction location [16]. Additionally, the average energy lost from 1.47 MeV alpha particles crossing the p^+ region will be 74 keV. Yet, such a view is somewhat pessimistic in that a significant portion of excited electrons can diffuse from the p^+ region into the v -type region whereupon they will be collected and contribute to the output signal. Regardless, the thin 200 Å Schottky barriers should have much less charge loss in the contact region than the p^+ contact devices, so that a direct comparison can be made between the two styles of devices.

5. Complex Detectors: Double-Coated, Sandwich, and Compound

Double-coated detectors were fabricated from both ^6LiF and ^6Li laid upon ^{10}B films. The devices were fabricated as described in section C.1. with only 1 micron of ^{10}B upon the detector. Afterwards, polished samples of ^6LiF were attached to the front of the devices with epoxy. An overcoat of Humiseal was applied to prevent possible delamination.

Devices fabricated with pure ^6Li required special handling [37]. Pure ^6Li , kept in mineral oil, was placed along with a hot plate, a custom stainless steel press, and hollow brass punch, wax paper, and xylene in a plastic glove box. Argon was plumbed into the glove box and allowed to flow for 15 minutes. A piece of ^6Li was removed from its container and a small slice removed with a razor blade, followed by cleaning (to remove the mineral oil) in xylene. The slice was placed between two small squares of wax paper and then into the stainless steel press calibrated to press films to 100 microns thick. After pressing, small circles 3 mm in diameter were punched from the flattened material and placed upon the ^{10}B films of the prepared detectors. The devices were then transferred to a hot plate and more wax paper was placed over them. A small weight was placed over the samples to keep the ^6Li circles flat. The hot plate was warmed to 180°C so that the ^6Li attached itself to the ^{10}B film. Afterwards, a piece of Kapton tape was applied to the detector to cover the ^6Li disc, followed by Humiseal. Unfortunately, even with the sealing precautions, the ^6Li still decomposed within two weeks, yet measurements were made that provided meaningful detection efficiency data.

Sandwich and compound devices were fabricated as indicated above with the following differences. Output Au wires were first attached to all devices. Simple sandwich designs were fabricated by attaching a bare detector, facing inward, atop a coated detector with insulating epoxy. The output wires were attached together, anode to anode and cathode to cathode. Compound designs were fabricated by placing thin polished samples (20 microns) of ^6LiF between two inward-facing ^{10}B -coated via-hole devices. The samples were fastened together with insulating epoxy and the output wires attached as previously described.

Section D: Characterization

Detector characterization was conducted primarily within the Phoenix Memorial Laboratory using either the Ford Nuclear Reactor or the 20 kCi ^{60}Co gamma ray irradiator. Two neutron beam ports were used for detector characterization, beam port 'A' and beam port 'E.' Figure 62 shows the general arrangement for beam ports 'A' and 'E.' Neutrons from the reactor are thermalized in the D_2O tank. Beam port 'A' points into the D_2O tank, but in such a fashion that it does not point at the reactor. As a result, the gamma ray background component is decreased. The neutron beam is then diffracted twice to further remove it from background gamma rays. Although the thermal neutron beam emerging from the double diffractometer in beam 'A' has a low gamma ray background component, it also has a low neutron flux, being approximately $3 \times 10^4 \text{ n/cm}^2\text{-s}$. Beam port 'E' points directly at a corner of the Ford Nuclear Reactor through the D_2O tank (see Figure 62). Beam port 'E' has a higher background gamma ray component, but it also has a much higher thermal neutron flux of $2 \times 10^6 \text{ n/cm}^2\text{-s}$. As a result, beam port 'A' is used to determine detector thermal neutron efficiency and spectral performance, while beam port 'E' is used to determine gamma ray discrimination and for thermal neutron imaging.

Figure 62. The experimental configuration for testing the thermal neutron detectors. The aluminum chimney box ensured consistent alignment within the neutron beam.

1. Spectral Performance and Counting Efficiency

The mounted devices were installed into aluminum testing boxes designed to reduce radiofrequency (RF) and photoelectric noise [31-45]. The boxes have two cylindrical, close-

ended, hollow chimneys opposite each other on the front and on the back. The chimney box design enables repeatable indexing of the detectors by ensuring a standardized and fixed location in the thermal neutron beam. Each detector was mounted in order that the Schottky contact was centered relative to the hollow chimneys and furthermore that the detector plane was centered within the chimney box. The closed, light impenetrable box was inserted into a HDPE collar that fit snugly into the receiving end of the neutron beam port collimator block. The detector was connected through the aluminum box to an Ortec 142A preamplifier. The signals were shaped in an amplifier and recorded on a multichannel analyzer. One-hour duration measurements were performed, and the dead time was adjusted and maintained at 2%.

The Ford Nuclear Reactor, a 2 MW research reactor, supplied the source of neutrons. The neutron detection efficiency measurements were performed with a beam port ('A'-port) that did not point towards the reactor; this configuration helped to reduce γ -ray background. Neutrons were moderated through a D₂O tank, and the thermalized neutron beam was doubly diffracted through two copper plates to further separate the neutrons from γ -ray background [31,35]. The doubly Bragg diffracted neutron beam was then collimated through a HDPE tube to the aluminum chimney box at testing Position 2 in Figure 62. Before any measurements began, a fission chamber was used to measure the neutron flux emerging from the A-port which gave a value of approximately $\Phi = 2.7 \times 10^4$ n/cm²-s. Due to control rod movements over the period in which the GaAs neutron detectors were tested, it is expected that the neutron flux value also changed with time. Adjacent beam port experiments also contributed to changes in neutron background. The changes in the neutron flux may amount to a few percent (2-3%), hence the neutron flux value quoted represents only an average measurement.

Figure 63: Spectra taken with ¹⁰B-coated SI bulk GaAs detectors of neutron reactions. The data shows the detector response for frontal exposure to a doubly- diffracted thermal neutron beam ($\Phi = 2.7 \times 10^4$ n/cm²-s). The spectral changes as a function of ¹⁰B thickness are apparent. Energy peak resolution is best for thin ¹⁰B films, whereas efficiency is highest for the thickest ¹⁰B films.

Detectors with various ¹⁰B film thicknesses were tested for both reaction product spectral changes and neutron counting efficiency. Both sides of the detector were tested facing the neutron beam: the Schottky contact side was first tested followed by the ohmic contact side. Although the ohmic contact side did not have a ¹⁰B coating on it, thermal neutrons pass through the GaAs detector and still become absorbed in the ¹⁰B film [29,46]. The reaction product emissions were detected as before. The electronics remained unchanged throughout the testing, and all measurements were conducted for a “live time” of one hour. For the Schottky contact side irradiation, Figure 63 shows pulse height spectra from ¹⁰B(n, α)⁷Li reactions for ¹⁰B films

ranging from 1750 Å to 1.84 μm. It is clear that the number of counts under the curves increase with increasing ^{10}B thickness. The reaction products' energy peaks become less distinct with increasing ^{10}B thickness, the expected result of reaction-product energy self-absorption within the ^{10}B film. To illustrate, two distinct energy peaks (840 keV ^7Li ion and 1.47 MeV α particle) are present for the 1750 Å coated device, yet neither energy peak is discernable for the 1.84 μm coated device. By summing the number of counts in the channels exceeding 300 keV, the efficiency was determined for several devices as a function of ^{10}B -film thickness for both obverse and reverse irradiations (see Figure 64). Also shown is the theoretical thermal neutron efficiency curve for an *LLD* setting of 300 keV, the curve having been calculated from Equations 18 and 19. Figure 64 shows good agreement with the developed theory and the measurements. A SI bulk GaAs device (200 μm thick) coated with a 5000 Å layer of ^{10}B was used to observe spectra and the count rate as a function of reverse bias voltage (see Figures 65 and 66). From Figure 65, spectral changes were apparent, in that the charge collection (and resulting charge induction) increased with the bias voltage. However, as shown in Figure 66, due to the TEF effect observed with SI bulk GaAs, the neutron count rate changes only slightly above 50 V bias, thus showing only a few percent increase in count rate when the bias is increased from 50 V to 130 V. Since the devices are operated as counters and not spectrometers, charge collection efficiency is less important than γ -ray rejection. An increase in bias voltage will increase the detector active region, and will result in more γ -ray background interference. As a consequence, the devices were generally operated with only 50 volts bias for the thermal neutron measurements.

Figure 64: Thermal neutron detection efficiency for ^{10}B -coated SI bulk GaAs detectors as a function of ^{10}B thickness. The ^{10}B -coated GaAs detectors were irradiated with a doubly-diffracted thermal neutron beam ($\Phi = 2.7 \times 10^4 \text{ n/cm}^2\text{-s}$). The *LLD* was set at 300 keV equivalent.

Figure 65: Spectral changes as a function of reverse bias voltage for a ^{10}B -coated SI bulk GaAs detector. Although the spectra appearance changes dramatically due to improved charge collection, the total number of counts under each spectrum changes minimally after an applied reverse bias of 50 volts.

Figure 67 shows comparison thermal neutron reaction spectra from two detectors, one coated with ^{10}B and the other coated with ^6LiF . The ^{10}B -coated device (not optimized) has a measured efficiency of 3.0%, whereas the ^6LiF -coated device (optimized) has an efficiency of 4.6%. Although the efficiencies are similar, it is important to note that the energy distribution for the ^6LiF -coated device is much higher than the ^{10}B -coated device. If operated in a mixed gamma ray field, the ^6LiF -coated device can have the *LLD* set higher than the ^{10}B -coated device while losing only a small portion of the spectrum (and detection efficiency), thereby allowing for better gamma-ray discrimination. The benefit becomes clear in a later section of the present report.

Figure 66: Count rate as a function of reverse bias voltage for a ^{10}B -coated SI bulk GaAs detector. Due to improving charge collection efficiency, the count rate continued to improve with increasing bias voltage. However, the improvement became minimal for bias voltages above 50 V, showing only a 7.4% increase in count rate for a bias voltage of 130 V.

Figure 67: Comparison neutron reaction product spectra from ^{10}B -coated and ^6LiF -coated detectors. The ^{10}B -coated device has only $1.1\ \mu\text{m}$ of material whereas the ^6LiF -coated device has $35\ \mu\text{m}$ of material. Gamma-ray discrimination with the *LLD* is more effective for the ^6LiF -coated detector.

The double-inward-facing design yields the highest efficiency for a sandwich design. It does not rely on the full depletion of the detectors and can still be operated with modest operating voltages. If the neutron-reactive film thickness between the devices is thinner than the summed reaction product ranges, both detectors can simultaneously measure coincident charged particle emissions from a single neutron absorption event. Erroneous coincident recordings, in which counts occur in both devices when only one neutron interaction actually occurred, can be avoided by simply connecting the output of both devices to a common input preamplifier. Figure 68 shows comparison neutron reaction spectra between a single-coated device (count-time corrected) and a double-inward-facing device, both having ^6LiF as the film. As with the front irradiated single-coated detectors, there is an optimum film thickness to achieve the maximum thermal neutron detection efficiency.

Figure 68. Shown is a comparison of neutron reaction product spectra from a single-coated device (count time corrected) and a double-inward facing sandwich device. ^6LiF layers are used as the converters.

Another method of increasing the overall efficiency of a device incorporates a double-layered film, such as with ${}^6\text{LiF}$ on ${}^{10}\text{B}$ or pure ${}^6\text{Li}$ on ${}^{10}\text{B}$. ${}^{10}\text{B}(\text{n},\alpha){}^7\text{Li}$ reaction products have shorter ranges within ${}^{10}\text{B}$ material than do ${}^6\text{Li}(\text{n},\alpha){}^3\text{H}$ reaction products. Hence, it is possible to increase the device neutron detection efficiency by applying a coat of ${}^{10}\text{B}$ followed by a coat of either ${}^6\text{LiF}$ or ${}^6\text{Li}$. The charged-particle reaction products emanating from ${}^{10}\text{B}(\text{n},\alpha){}^7\text{Li}$ reactions can reach the detector as before since the film is applied directly to the detector contact. In addition, the longer range charged particle products emanating from ${}^6\text{Li}(\text{n},\alpha){}^3\text{H}$ reactions can still reach detector even though they must transit through the ${}^{10}\text{B}$ film as well. However, energy loss occurs, thereby shifting the detected energy spectrum. Since the ${}^{10}\text{B}$ film has a higher neutron interaction cross section than ${}^6\text{LiF}$ or ${}^6\text{Li}$ films, a net gain in efficiency is realized. Hence, the short range and high cross section material is deposited closest to the contact while the long range and low cross section material is placed atop the first film. The opposite case renders no improvement and in fact actually decreases efficiency. Figure 69 shows comparisons between devices with single coatings of ${}^{10}\text{B}$ and ${}^6\text{LiF}$, and double coatings of ${}^{10}\text{B}/{}^6\text{LiF}$ and ${}^{10}\text{B}/{}^6\text{Li}$.

Figure 69. Comparison neutron reaction product spectra from a ${}^{10}\text{B}$ single-coated device, a ${}^6\text{LiF}$ single-coated device, and two double-coated devices using ${}^{10}\text{B}/{}^6\text{LiF}$ and ${}^{10}\text{B}/{}^6\text{Li}$ systems.

2. Operating Performance

a. The Self-biased Design

Various detectors from all six diode configurations were tested in a beam port from a thermal neutron reactor (Ford Nuclear Reactor at the University of Michigan). The neutron beam was doubly diffracted through copper plates and collimated through a combined Benelex beam stop and polyethylene block. All detectors were tested under zero bias conditions. The devices were installed into aluminum testing boxes designed to reduce radiofrequency (RF) and photoelectric noise [31,45]. The boxes were also designed with cylindrical, close-ended, hollow chimneys on them, enabling repeatable indexing of the detector into the same location of the thermal neutron beam. Each detector was mounted such that it fit into the hollow chimney. Afterwards, the chimney was inserted into a polyethylene collar that fit snugly into the receiving end of a beam port polyethylene-collimated block. The detector was connected through the aluminum box to an Ortec 142A preamplifier. The signals were shaped in an amplifier and recorded on a multichannel analyzer. One-hour duration measurements were performed. Dead time was adjusted such that it was maintained at 2%.

Figures 70 through 75 show representative spectra from the various GaAs diode samples. Although all devices worked, certain unique characteristics can be understood from the various

spectra. A reaction product spectrum from sample diode 3241A is shown in Figure 70; the four major charged particle reaction products of the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction are clearly apparent. Some energy self-absorption occurs in the thin 5000 Å ^{10}B film. The 5 µm thick v-type GaAs layer is thick enough to absorb the full energy of the higher energy alpha-particle emissions. The device “built-in” potential is high enough to deplete the *n*-type GaAs region such that most or all of the energy from the reaction products can be absorbed. It is also apparent that the spectrum is constituted primarily from charged-particle reaction products and not background gamma rays. The device is not designed to be a spectrometer for charged-particle reaction products, but instead a simple low-powered neutron counter. However, the reaction product spectrum clearly shows that the device is (a) detecting neutron reaction product species and (b) has a sufficiently high gamma-ray-to-neutron-rejection ratio.

Figure 70. Pulse height spectrum from sample diode 3241A of thermal neutron reactions in ^{10}B . All of the charged-particle reaction product energies are apparent. No voltage was applied to the device during the measurement.

Figure 71. Pulse height spectrum from sample diode 3242A of thermal neutron reactions in ^{10}B . None of the charged-particle reaction product energies are discernable. No voltage was applied to the device during the measurement. The artifact peak appears as a result of incomplete energy deposition in the device active region.

Sample diode 3241C was reverse biased up to 31 volts before breaking down, but only a slight difference in the pulse height spectrum was observed, within which the maximum channel number for the 1.777 MeV ^7Li ion increased from approximately 830 to 850. The minor improvement in performance indicates that v-type region was fully active in charge collection from the built-in potential. Average neutron detection efficiency was 1.6%.

Figure 71 shows a spectrum from sample diode 3242A, a device similar to sample 3241A, but with only a 2 µm thick v-type GaAs layer. Sample 3242A also has 6500 Å of deposited ^{10}B rather than 5000 Å. None of the reaction product energies are discernable, indicating that the active region of the device is not absorbing the full energies of the charged particles. Average neutron detection efficiency was 2.3%.

Figure 72 shows a spectrum from sample diode 3243A, and although the diode construction was similar to 3241A, 7200 Å of ^{10}B was deposited over it. The negative slope in the device energy spectrum is indicative of energy self absorption, in which the combined effect of the *p*⁺ layer and the ^{10}B film have reduced and dispersed the pulse height spectrum to a greater extent than detectors from sample 3241. Some improvement is observed when the diode

is reverse biased, yet the distinctive shape remains. Average neutron detection efficiency was 2.62%, comparing well to the expected value from Figure 17.

Figure 73 shows a spectrum from sample diode 3246A. Within a very thin v-type active region (only 1 μm), the spectrum shows artifact peaks, although not quite as distinctly as the diodes from sample 3242. The clear separation of the 840 keV ^7Li peak is expected with the thin ^{10}B layer. Surprisingly, the maximum energy deposition is not very much less than seen for samples 3241 and 3243, denoting that much of the charge excited external to the active region is being recorded. Hence, a considerable fraction of charge carriers are diffusing from both the p^+ contact layer and the n -type substrate layer into the active region. Average neutron detection efficiency was 1.7%.

Figure 74 shows a spectrum from sample diode 3247A. The diode has a similar construction to sample 3242, except that a 2500 Å thick n -type $\text{Al}_x\text{Ga}_{1-x}\text{As}$ layer was grown between the v-type active region and the n -type substrate. The n -type $\text{Al}_x\text{Ga}_{1-x}\text{As}$ layer was added in an attempt to confine electron-hole pairs to the active region of the detector by means of a small bandgap step. Here we see that the performance was not very different than 3243A. The dispersion of the 1.47 MeV alpha-particle peak is most likely a compound effect from both energy self-absorption in the ^{10}B layer and p^+ contact layer and also energy loss from transmission of particles into the n -type substrate. Again, the clear observation of the major charged-particle energy peaks indicates that diffusing charges from the p^+ region and the n -type $\text{Al}_x\text{Ga}_{1-x}\text{As}$ layer are participating in charge collection. Average neutron detection efficiency was 1.9%, comparing well to the expected value shown in Figure 17.

Figure 75 shows a spectrum from sample diode 3278A, which is a Schottky barrier device. All devices from sample 3278 were electronically noisier than the p^+ contact devices. Additionally, the devices did not withstand any appreciable reverse-bias voltage before breaking down. The absence of the 2000 Å p^+ layer reduces energy self-absorption, which is apparent since the only other devices with energy resolution showing all four reaction product energies are from sample 3241 (sample 3241 had the same thickness of active region as samples from 3278 with the ^{10}B layer reduced by 1500 Å). In other words, the combined total thickness of the contact and converter layer were very similar, with only a difference of approximately 500 Å between sample 3241 and 3278. Average neutron detection efficiency was 2.2% for 3278.

Self-biased devices from all six samples demonstrated counting efficiency similar to the expected efficiency, indicating that the v-type regions need not be very thick to operate as viable, low-power neutron counters. The best spectral performance was observed from sample 3241, and the best counting efficiency from sample 3243. The worst spectral performance was from sample 3242, although the counting efficiency matched almost exactly to theoretical values. The Schottky barrier devices (3278) did perform well, although they demonstrated higher electronic noise and could not hold any appreciable reverse voltage.

The self-biased design offers a straightforward method to produce low-cost, lightweight, compact and low-power neutron detectors for remote deployment. Future devices will have thicker ^{10}B films to increase neutron detection efficiency closer to the theoretical maximum indicated in Figure 17. Stacking the devices, as well as other previously discussed designs, can further increase thermal neutron detection efficiency. Various detectors from the six samples have been irradiated with fast and thermal neutrons as well as gamma rays in order to determine their radiation hardness and the limiting radiation doses through which the different configurations may survive.

Figure 72. Pulse height spectrum from sample diode 3243A of thermal neutron reactions in ^{10}B . All of the charged-particle reaction product energies are apparent. No voltage was applied to the device during the measurement.

Figure 73. Pulse height spectrum from sample diode 3246A of thermal neutron reactions in ^{10}B . All of the charged-particle reaction product energies are apparent. No voltage was applied to the device during the measurement. The satellite artifact peak appears as a result of incomplete energy deposition in the device active region.

Figure 74. Pulse height spectrum from sample diode 3247A of thermal neutron reactions in ^{10}B . All of the charged-particle reaction product energies are apparent. No voltage was applied to the device during the measurement.

Figure 75. Pulse height spectrum from sample diode 3278A of thermal neutron reactions in ^{10}B . All of the charged-particle reaction product energies are apparent. No voltage was applied to the device during the measurement. More energy is retained due to the thin Schottky contact, but the device clearly has higher electronic noise.

b. Via Hole Devices

The finished devices were tested in a double-diffracted thermal neutron beam at the Ford Nuclear Reactor at Position 2 in Figure 62. Each device was installed in a custom aluminum housing (or “chimney box”) that allowed for repeatable indexing into the same location of the neutron beam [11, 44]. The double-diffracted beam had a calibrated flux of $3.3 \times 10^4 \text{ n/cm}^2\text{-s}$, the value used to determine the efficiency for each device.

Figure 76 shows a comparison between the via-hole-covered device with a 4 μm ^{10}B coating and a flat device with a 1 μm ^{10}B coating. Operated as counters, the thermal neutron detection efficiency of the via-hole-covered device is 3.9% compared to only 2.9% for the flat device, demonstrating the advantage of using the thicker film. However, as described earlier, the maximum achievable single film value for front irradiated ^{10}B -coated devices is only 3.95% [24-29], which occurs at a film thickness of 2.4 μm .

As previously explained, ^{10}B films thicker than 2.4 μm cause neutron attenuation that decreases the detection efficiency, and a device coated with 4 μm of ^{10}B should be able to achieve only 3.67% efficiency [46]. Hence, the observed efficiency is higher than expected but easily explained as an additional benefit of the via-hole-covered detector surface. The ^{10}B material within the via holes allows for reaction products to enter the sides of the cavity, thereby increasing the detection probability above that of a flat device. From Figure 55, it is apparent that a portion of the cavity is not filled with ^{10}B , indicating the neutron detection efficiency could actually be increased further. The ^{10}B powder-filling method was used as an alternative technique to fill the via holes (Figure 57).

Figure 76. Comparison $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reaction product spectra for a flat device with a 1 μm ^{10}B coating and a via hole covered device with a 4 μm ^{10}B coating. The via holes on the etched device are 1.7 μm deep, 3.5 μm in diameter, and spaced 9.5 μm apart center-to-center.

Figure 77 shows a comparison between devices with no via holes and with 5.0 μm deep via holes. All devices were simultaneously coated with 1.1 μm of evaporated ^{10}B . The via holes in the etched devices were subsequently filled with ^{10}B powder. The devices were operated with a reverse bias of 100 volts and a gain of 300. A commercial Ortec 142A preamplifier was used to measure the induced charge.

The bare device had a measured thermal-neutron detection efficiency of 3.05%, matching well with theory [24-29]. The devices with 5.0 μm deep via holes had 3.34% efficiency, an efficiency increase of 10% over the flat device. The total counts (and count rate) are statistically beyond 3σ ($+2\sigma$ would increase the efficiency to only 3.06%). It could be argued that the via holes have increased the detector surface area, thereby allowing for more boron to contact the surface. Yet, such a claim is not supported by Figure 55, where it is clear that the boron along the

walls is either absent or very thin. The observed efficiency increase is not adequately explained to result solely from increased surface area imposed by the via holes. In other words, the via holes cover 10.6% of the device area, which if filled entirely, would give a flat device value of only 3.6% efficiency for those regions. Correcting the remaining areas such that the 3.05% efficiency is represented by 89.4% coverage, the result would be an increase up to only 3.11% efficiency, not 3.34%. The increase to 3.34% is due to an increased detection probability of neutron interactions within the via holes. Given the experimentally observed trend, devices with 5.0 μm deep via holes and 40% coverage will yield 4.28% thermal neutron detection efficiency. Etching 10 μm deep holes and filling with ^{10}B will increase the efficiency to 6.03%.

Ultimately, it is the compound device with via holes that should yield the highest efficiency. Shown in Figure 78 are comparison spectra from the simple ^{10}B -coated device with 3.9% efficiency and a compound device with 13% efficiency. The compound device has two inward facing 5 micron deep via holes filled with ^{10}B powder, between which is a 20- μm film of ^6LiF . The compound device parameters are *not* optimized, yet the efficiency tripled over that of the simple ^{10}B -coated device. A higher density of via holes along with optimized ^{10}B and ^6LiF layers can increase the thermal neutron detection efficiency above 20%. In such a case, larger via holes (30 μm diameter) are used for the ^6LiF , and smaller via holes (3.1 μm diameter) spaced *in and out* of the larger via holes are used for the ^{10}B . A ^{10}B layer is applied first followed by the ^6LiF layer. Two such devices are then fastened together in the double-inward configuration. A system using pure ^6Li instead of ^6LiF can increase the thermal neutron detection efficiency above 30%.

Figure 77. Comparison spectra of $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction products from a flat device and from a device with 5.0 μm deep via holes. The via holes, 3.5 μm in diameter and covering 10.6% of the surface area, were filled with ^{10}B powder. The evaporated ^{10}B coating was 1.1 μm thick for all devices.

Figure 78. Comparison neutron reaction product spectra from a via hole covered (10.6%) ^{10}B single-coated device and a compound device that uses both ^{10}B -filled via holes (10.6% coverage) and a ^6LiF film combined in a double-coated and double-inward sandwich design.

c. Gamma ray discrimination

Several ^{10}B -coated devices were subsequently placed in the mixed γ -ray and neutron field of a beam port (E-port) that pointed directly at the nuclear reactor core through the D_2O moderator pool (position 3 in Figure 62). The measured γ -ray component was 1.1 R/hr and the neutron flux was approximately $\Phi = 2 \times 10^6 \text{ n/cm}^2 \cdot \text{s}$. Tests were conducted to determine the neutron/ γ -ray (n/γ) discrimination ratio as described. Various thicknesses of Pb and Cd were placed between the GaAs detector and the beam port. Lead (Pb) is used to effectively absorb γ rays while permitting a majority of the neutrons to pass through. Cadmium (Cd) is a highly effective absorber of neutrons for energies below 0.5 eV [56], and thin sheets of the material (in this case, 5 mm or less) do not significantly reduce the γ -ray background. Prompt γ -ray emissions occur from neutron absorption in Cd [18, 19] and some of the γ rays emitted from the Cd will enter the GaAs detector. The γ -ray exposure decreases inversely with the distance squared ($1/d^2$), and the γ -ray background contamination from (n, γ) reactions in the Cd sheet can be significantly reduced by simply placing the Cd sheets closer to the beam port opening and farther away from the GaAs detector. Otherwise, the γ -ray background would increase and an erroneous measurement of the γ -ray interference emanating from the reactor core would result. For the experiment, a 2 mm Cd sheet was placed in the beam at a distance of 2.5 m away from the detectors, and an additional 3 mm Cd sheet was placed in the beam 1.25 m away from the detectors. Lead (Pb) bricks 5 cm thick, when used, were placed in the neutron beam 10 cm away from the detectors.

The experiment was first conducted by measuring various detectors with the bare beam of mixed γ rays and neutrons. The Cd sheets (5 mm) were placed in the mixed beam to remove the thermal neutron component. Since most of the thermal neutrons have been removed, the resulting spectra are primarily due to γ -ray background from the beam. One can observe the neutron-induced component of the pulse height spectrum by subtracting the Cd attenuated neutron beam spectrum from the bare neutron beam spectrum. The n/γ ratio becomes

$$\frac{n}{\gamma} \approx \frac{\text{Bare Beam Counts} - \text{Cd Shielded Beam Counts}}{\text{Cd Shielded Beam Counts}} \approx \frac{(n + \gamma) - \gamma}{\gamma} . \quad (40)$$

Placing a 5 cm thick Pb brick in the path of the beam effectively reduces the γ -ray component while allowing much of the thermal neutron component to still pass through. The effectiveness of altering the n/γ ratio by Pb shielding was then determined as described. The pulse height spectrum was measured with a 5-cm Pb brick in the path. Afterward, both the 5 mm of Cd sheet and the 5 cm Pb brick are placed in the mixed beam, which yields the remaining γ -ray component. Subtracting the spectra resulting from the combination of Cd and Pb shielding from the Pb-shielded spectra yields the filtered (remaining) neutron component. The neutron component *also* reduces, but not quite as much as the γ -ray component. The new n/γ ratio can be found:

$$\left. \frac{n}{\gamma} \right|_{5\text{cmPb}} \approx \frac{5\text{ cm Pb Shielded Beam Counts} - (\text{Cd} + 5\text{ cm Pb Shielded Beam Counts})}{\text{Cd} + 5\text{ cm Pb Shielded Beam Counts}} . \quad (41)$$

Figure 79 shows spectra taken with a GaAs diode coated with 8750 Å of ^{10}B . The bulk SI GaAs diode was biased at 50 V for all measurements. Both the ^7Li -ion distribution and the α -particle peak are apparent, although γ -ray interference distorts and obscures the ^7Li ion distribution. The 5 mm Cd shielding effectively removes the neutron-induced portion of the spectrum, resulting in the γ -ray induced spectrum. From Figure 79, it is apparent that Pb shielding reduces both the neutron interaction rate and the γ -ray interaction rate. However the γ -ray spectrum is reduced by a factor of approximately 6.4 while the neutron spectrum is reduced by a factor of approximately 2.2 (an increase of 2.9 for the n/γ ratio). Figure 80 shows a comparison of the bare beam spectrum, the spectrum of the Cd-attenuated beam, and the subtracted spectrum. The expected $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction product spectrum is clearly shown by subtracting the Cd-shielded spectrum from the bare beam spectrum. Figure 81 shows a comparison of spectra from 5 cm Pb shielding, combined 5 μm Cd and 5 cm Pb shielding, and the resulting subtracted spectrum. The expected $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction product spectrum is observable in the 5 cm Pb-shielded spectrum and is clearly shown by subtracting the Cd-shielded spectrum from the Pb-shielded spectrum.

The peak channel number for the 1.47 MeV α particle was calibrated relative to the channel number of the spectra shown in Figures 80 and 81. The n/γ ratio was determined as a function of the *LLD* setting with Equations 40 and 41. The results are shown in Figure 82. The γ -ray field of 1.1 R/hr causes background counts to be registered, yet from Figures 80 and 81, the γ -ray spectrum is quite different from the neutron-induced spectrum. The n/γ ratio is only 2.47 for the bare beam condition at a *LLD* setting of 300 keV. As the *LLD* is increased, fewer γ -ray

and thermal neutron induced counts are recorded. It is important to note that the γ -ray distribution is composed primarily of Compton scattering events, hence it is a negatively decreasing exponential, whereas the neutron-induced spectrum is a charged-particle spectral distribution. As a result, the number of γ -ray counts excluded from the measurement dramatically decreases as the *LLD* is increased, however the number of thermal neutron-induced counts gradually decreases.

Figure 82 shows that the n/γ ratio has a maximum in the distribution, and for this detector the maximum was $n/\gamma = 585$ at a *LLD* setting of approximately 1500 keV. By shielding the device with 5 cm of Pb, the maximum n/γ ratio is increased to 798 at a *LLD* of 1500 keV. The high *LLD* setting is due to the energy spread in the 1.47 MeV α -particle peak, within which the thermal neutron-induced counts begin to diminish rapidly beyond this energy. On the other hand, there is a significant disadvantage to setting the *LLD* at such a high point. Figure 83 shows the measured neutron detection efficiency as a function of *LLD* setting and demonstrates two very important points. First, the thermal neutron detection efficiency diminishes with increasing *LLD* [24]. Second, Pb shielding helps to increase the n/γ ratio but works to *decrease* the overall neutron detection efficiency. The measurement system can be optimized by selecting a n/γ rejection ratio and then adjusting the *LLD* so that the highest possible neutron detection efficiency is achieved. Shielding with Pb should be used primarily when a n/γ rejection ratio greater than that achievable with a bare beam is required.

Figure 79: Comparison spectra of a mixed γ -ray (1.1 R/hr) and thermal neutron beam ($\Phi = 2 \times 10^6$ n/cm²-s). The spectra were taken with a ¹⁰B-coated (8750 Å) SI bulk GaAs detector. Shown are spectra with no shielding (bare beam), 5 mm of Cd, 5 cm of Pb, and combined 5 mm of Cd and 5 cm of Pb. The main features of the ¹⁰B(n, α)⁷Li reaction products are apparent in the Pb-filtered and unfiltered spectra but are not observed for the Cd-shielded spectra. Thermal neutrons and γ -rays are both attenuated by the Pb shielding, however the figure shows that a higher percentage of γ -rays are removed than neutrons.

Figure 80: Comparison spectra of a mixed γ -ray (1.1 R/ hr) and thermal neutron beam ($\Phi = 2 \times 10^6$ n/cm²-s). The spectra were taken with a ¹⁰B-coated (8750 Å) SI bulk GaAs detector. Shown are spectra with no shielding (bare beam), 5 mm of Cd, and the subtracted spectrum revealing the neutron-induced counts from the bare beam.

Figure 81: Comparison spectra of a mixed γ -ray (1.1 R/ hr) and thermal neutron beam ($\Phi = 2 \times 10^6$ n/cm²-s). The spectra were taken with a ¹⁰B-coated (8750 Å) SI bulk GaAs detector. Shown are spectra with 5 cm of Pb shielding, combined 5 mm of Cd and 5 cm of Pb shielding, and the subtracted spectrum revealing the neutron-induced counts remaining from the Pb shielding.

Figure 82: Experimentally determined n/γ ratios for a ^{10}B -coated (8750 Å), 200 μm thick SI bulk GaAs Schottky barrier detector. The ratio becomes very large as the *LLD* setting is increased. The figure shows that Pb shielding can increase the n/γ ratio significantly.

Figure 83: Experimentally determined thermal neutron detection efficiencies for a ^{10}B -coated (8750 Å), 200- μm thick SI bulk GaAs Schottky barrier detector. The neutron detection efficiency decreases as the *LLD* setting is increased. Neutron attenuation in the lead (Pb) shielding reduces the neutron detection efficiency of the system.

3. Radiation Hardness Tests

a. Diode Structures

As previously shown, high-purity, undoped GaAs epitaxial layers grown onto *n*-type GaAs substrates are used as “self-biased” neutron detectors. The self-biased concept was also demonstrated successfully with Si diodes, in conjunction with LiF converters rather than ^{10}B films [26]. The $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction products have very short ranges in semiconductor solids, shorter than $^6\text{Li}(n,\alpha)^3\text{H}$ reaction products, thereby only requiring very thin active regions for coated diode structures.

A variety of devices have been investigated which have been separated in to two collections, shown in Tables 4 and 5. Some of the Schottky-type detector diodes operated slightly better when low reverse voltages were applied, most notably type ‘M’ (Table 5, Spire 92). The Schottky barrier devices demonstrated expectedly higher reverse leakage currents than the *p*+ contact devices due to the lower barrier height generally formed by Schottky contacts in comparison to *p-i-n* structures. Furthermore, the previously described difference in the work functions between the *p*+ and the *v*-type GaAs region results in a greater built-in potential for the *p-i-n* structure than the Schottky barrier structure and an accordingly greater internal electric field.

The assorted detector diodes are being studied for the potential application to the fields of remote neutron sensing and thermal neutron imaging. In both cases, it is beneficial to eliminate the operating bias requirement. Mainly, remote sensing requires low power consumption as is accomplished with the self-biased configuration. Remote neutron detection, if the devices are to be used in hostile environments, may require that the devices withstand extreme radiation environments. Also, for the devices to become viable thermal neutron beam monitors they must be able to withstand the harsh environment produced from a variety of neutron sources, such as a nuclear reactor, a solid source, or a generator. In order to ascertain their radiation hardness, the devices of this study have been exposed to a variety of radiations, consisting of systematic and incremental doses of gamma rays, fast and thermal neutrons, and reaction product charged particles from $^{10}\text{B}(n,\alpha)^7\text{Li}$ reactions.

For comparison purposes, bulk SI GaAs devices were fabricated as well. The GaAs devices were lapped and etched to thicknesses between 200 microns and 250 microns and were fabricated as Schottky barrier diodes [30,39]. As with the epitaxial devices, the bulk GaAs detectors were also coated with films of 98% enriched high-purity ^{10}B . The SI Bulk GaAs detectors have much lower built-in potentials than the epitaxial devices and therefore required 100 volts reverse bias for operation. The operating reverse bias appears higher than quoted in literature elsewhere [39], yet the high reverse voltage was required because of the resistance divider circuit within the preamplifier (Ortec 142A) used for the measurements. The actual voltage applied across the SI bulk GaAs diodes ranged only from 10-15 volts.

b. Experimental Arrangement

A variety of detectors were exposed in the moderator pool near the FNR to accelerate radiation damage and determine the radiation hardness of each detector configuration. The assorted sample detector configurations are delineated in Tables 4 and 5. As seen in both tables, two detector *collections* were studied. A set of control sample detectors from each collection was *not* irradiated so as to serve as a reference.

The first collection of detectors was irradiated in the moderator pool of the 2 MW Ford Nuclear Reactor. The ^{10}B -coated devices were exposed to gamma rays, thermal and fast

neutrons, and to the charged particles produced from the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction, and no attempt was made with the first collection to separate the damage effects from the different radiation types. The irradiated detectors of Collection One consisted of six different sample detector styles fabricated on *n*-type GaAs substrates and divided into two basic diode types (each with its associated blocking contact). Five of the sample detector types were *p-i-n* diodes and one was a Schottky diode. Specific characteristics that distinguish the sample detectors include variable thickness of the applied 98% ^{10}B film layer and of the *v*-type epitaxial GaAs region (Table 4). The six sample detector styles were further divided into four sets, defined by the four total neutron fluences to which the epitaxial detectors were exposed. Sample detector sets were irradiated at fixed grid locations near the core of the aforementioned 2 MW research reactor so as to ensure conditions of identical reactor fuel configuration. Additionally, the fast-to-thermal neutron flux was kept as similar as possible for each increment in neutron fluence. The sample sets were placed in moisture-proof aluminum irradiation cartridges and were fastened to an internal aluminum fixture so that irradiation would be spatially consistent in reference to the reactor flux. Gold foils and iron wires were located within each cartridge to monitor the total neutron exposure and later measured to confirm the total slow and fast neutron fluences received. Total neutron fluences ranged between $7.0 \times 10^{11} \text{ n/cm}^2$ and $7.63 \times 10^{13} \text{ n/cm}^2$ with incremental factors of approximately ten (Table 6). In addition to thermal and fast neutrons, the detectors experienced radiation damage from gamma rays and charged-particle reaction products from $^{10}\text{B}(n,\alpha)^7\text{Li}$ reactions. It was observed from Collection One that catastrophic damage occurred to all sample detectors for neutron fluences exceeding 10^{13} n/cm^2 . Of the configurations, device samples from 3278 (Schottky barrier devices) demonstrated failure at the lowest neutron fluences. Unfortunately, since the detectors were simultaneously exposed to gamma rays, slow and fast neutrons, and resultant $^{10}\text{B}(n,\alpha)^7\text{Li}$ charged particles in the reactor pool, determining the type of radiation that contributed most to device failure was not possible.

Table 5 - Collection Two of Various Detector Diodes Investigated as Neutron Detectors.

Sample Letter (Number)	Diode Type	Boron-10 Film Thickness	Blocking Contact	<i>v</i> -type GaAs ($n = 3 \times 10^{14} / \text{cm}^3$)	Intermediate Layer	Substrate
A	Schottky Diode	5000 Å	Ti/Au Schottky 550 Å	none	None	SI Bulk GaAs
E (Spire 76)	GaAs <i>p-i-n</i> Diode	5000 Å	<i>p+</i> GaAs ($2 \times 10^{19} / \text{cm}^3$) 1000 Å	5 µm thick	None	<i>n</i> -type GaAs
I (Spire 88)	GaAs Schottky Diode	5000 Å	Au Schottky Contact 200 Å	5 µm thick	None	<i>n</i> -type GaAs
K (Spire 89)	GaAs Schottky Diode	5000 Å	Au Schottky Contact 200 Å	5 µm thick	None	<i>n</i> -type GaAs
M (Spire 92)	GaAs Schottky Diode	5000 Å	Au Schottky Contact 200 Å	5 µm thick	<i>n</i> -type $\text{Al}_x\text{Ga}_{1-x}\text{As}$ ($5 \times 10^{15} / \text{cm}^3$) 2500 Å	<i>n</i> -type GaAs

An experiment was then designed to determine which form of radiation caused the catastrophic damage. The second collection of detectors in Table 5 was irradiated under altered experimental parameters in order to isolate the type of radiation predominantly responsible for detector damage and failure. The irradiated detectors of Collection Two consisted of five sample

detector styles divided into four diode types (each with its associated blocking contact). The specific characteristics that primarily distinguish the sample detectors include dissimilar substrates and either the presence or the absence of the v-type epitaxial GaAs active region. All sample detectors were further classified into three sets, defined by the ^{10}B layer. Sample detectors in Sets One and Three were coated with 5000 angstroms of ^{10}B , while sample detectors in Set Two were uncoated. Numerous representative detectors of each sample detector cited in Table 5 were present in all three sets. Additionally, each set was subdivided into five subsets. The subsets were defined by the five total neutron fluences to which all sets of the sample detectors were exposed (Table 7).

Numerous representative detectors from each sample detector cited in Table 5 and from each of the three sets were present in all six subsets. Irradiation procedures were performed as those previously described for all the detectors in Collection One, and gold foils were placed with each sample subset to monitor the total neutron fluence. Sets One and Two were irradiated *together* in the reactor moderator pool so that each subset experienced a tenfold increase in neutron fluence over the previous subset. Since Sets One and Two were irradiated in the same reactor pool locations at the same time, the neutron fluences experienced by each are identical. The third detector set was irradiated in the Phoenix Memorial Laboratory ^{60}Co Irradiator Facility so that the calculated gamma ray doses correspond with the doses received by detector Sets One and Two whilst in the reactor pool. The uncoated samples of Set Two were later coated with 5000 angstroms of 98% ^{10}B after the reactor pool irradiations were complete, thereby allowing for a direct comparison in device performance to the other two sets. As a consequence of manipulating research parameters, *Set One* was subjected to gamma-ray, neutron, and charged-particle damage, *Set Two* only to gamma-ray and neutron damage, and *Set Three* only to gamma ray damage. In summary, the ^{10}B -coated devices underwent damage from gamma rays, thermal and fast neutrons, and charged particles resulting from $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reactions. The uncoated devices, however, experienced damage only from gamma rays and neutrons; the uncoated devices did not undergo damage from $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reaction charged particles since they did not have ^{10}B coatings.

The neutron-produced, radioactive reaction products (mainly activated Au from the contacts) were allowed to decay to minimal levels before the detectors were tested. Afterward, the detectors were sealed in light impenetrable, self-aligning aluminum boxes [31-45] for testing that shielded them from radiofrequency and photoelectric noise. The boxes were also designed to incorporate cylindrical, close-ended, hollow chimneys on the obverse and reverse, enabling repeatable indexing of the detector's location in the thermal neutron beam. The devices were tested by placing them in a diffracted thermalized neutron beam in a port at the 2 MW Ford Nuclear Reactor (Figure 62) [31-45]. The diffracted beam substantially reduced the gamma-ray background during the measurements, hence the pulses observed were primarily from $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reactions in the detector films.

Sample detectors from Collection One were placed in a doubly diffracted thermal neutron beam and all detectors were operated under zero bias conditions (Position 2 in Figure 62). Thermal neutron measurements employing a fission chamber yielded an estimated flux of 2×10^4 n/cm²-s. One-hour measurements were conducted, and the $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reaction product spectrum was recorded for each sample detector irradiated with the specified fluences.

Sample detectors from Collection Two were placed in a singly diffracted thermal neutron beam for fifteen-minute measurements (Position 1 in Figure 62). A non-irradiated control detector from each of the sample detector sets and subsets was first tested. For all control detectors, the operating voltage and gain were adjusted and documented such that the 1.47 MeV

alpha particle peak centroids lie at channel number 1000. For all subsequent sample detectors, the voltage and gain were re-adjusted to the values documented from the control sample detectors corresponding to the same type and set. Allowing for some hysteresis, the procedure permitted a direct comparison of device degradation between the detector diode types. The spectra from each sample detector diode type and collection were recorded and the changes notated. The peak shift and the total number of counts in the peaks (both 840 keV ^7Li ion and 1.47 MeV alpha particle) were documented and compared, giving a standard of measure for detector degradation.

Table 6- Measured Neutron Fluences to Which the First Collection of Detectors was Subjected.

Sample Set	Thermal n Fluence (n/cm ²)	Fast n Fluence (n/cm ²)	Total n Fluence (n/cm ²)
1	none	none	none
2	4.58×10^{11}	2.43×10^{11}	7.01×10^{11}
3	4.98×10^{12}	2.65×10^{12}	7.63×10^{12}
4	4.98×10^{13}	2.65×10^{13}	7.63×10^{13}

Table 7 - Measured Gamma-Ray Doses and Neutron Fluences to Which the Second Collection of Detectors was Subjected.

Set 1 (^{10}B -coated) and Set 2 (uncoated)				
Subset	Thermal n Fluence (n/cm ²)	Fast n Fluence (n/cm ²)	Total n Fluence (n/cm ²)	γ - ray Exposure (Rads)
1	None	None	None	None
2	1.0×10^{11}	5.1×10^{10}	1.51×10^{11}	2.99×10^4
3	1.1×10^{12}	5.4×10^{11}	1.6×10^{12}	2.39×10^5
4	9.3×10^{12}	4.6×10^{12}	$1.4 \times 10^{13} *$	2.24×10^6
5	8.9×10^{13}	4.4×10^{13}	1.3×10^{14}	1.89×10^7
Set 3 (^{10}B -coated)				
1	None	None	None	None
2	None	None	None	2.41×10^4
3	None	None	None	2.44×10^5
4	None	None	None	2.27×10^6
5	None	None	None	1.85×10^7

*Sample detectors M received a measured neutron fluence of 1.6×10^{13} n/cm².

c. Results

i. Collection One

The damage is distinctively noticeable for the Schottky barrier diodes at neutron fluences of $7 \times 10^{11} \text{ n/cm}^2$. However the *p-i-n* diode shows little change for the same neutron fluence. Figure 84 shows the pulse height spectra for *p-i-n* diode detectors of sample 3246, and Figure 85 shows pulse height spectra for Schottky diodes of sample 3278.

The neutron-induced count rate provides another comparative gauge for results. The normalized neutron-induced counts for all the sample detectors as a function of neutron fluence appear in Figure 86. Sample detector diodes 3246 and 3247 show the least operational degradation, while the other devices show total catastrophic failure (no pulses) above a neutron fluence of $7.6 \times 10^{13} \text{ n/cm}^2$. This data is based on measured and tabulated fast and thermal neutron fluences in Table 6. The distinctive difference between diode styles 3246, 3247, and the remainder diodes is the v-type high-purity GaAs region thickness. Diode style 3246 has a $1 \mu\text{m}$ thick v-type high purity region, diode style 3247 has a $2 \mu\text{m}$ thick v-type high purity region, and the remainder (3241, 3243 and 3278) all have $5 \mu\text{m}$ thick v-type high purity regions. Overall, the *p-i-n* diodes performed similarly. Diode type 3278 was the only Schottky contact device type, which has a lower blocking barrier height and resultant built-in potential than the *p-i-n* diode structures. It is very possible, due to the lower internal voltage, that displacement damage sites caused charge carrier trapping more effectively in the Schottky barrier devices than the *p-i-n* devices, which may explain their observed failure at lower neutron fluences than observed for the *p-i-n* devices.

The information acquired from the Collection One identified the irradiation region of interest, in which damage was observed to occur between thermal neutron fluences of 10^{12} n/cm^2 - 10^{14} n/cm^2 . Since the devices experienced radiation damage from gamma rays, neutrons, and charged particle reaction products at the same time, it became important to identify the form of radiation most responsible for the observed device failure. The irradiation procedure for samples from Collection Two was designed to separate the damage effects of the different forms of radiation.

Figure 84. Pulse height spectra as a function of neutron fluence from sample detector diode configuration 3246 of thermal neutron reactions in ^{10}B . The diodes were operated under zero bias conditions.

Figure 85. Pulse height spectra as a function of neutron fluence from sample detector diode configuration 3278 of thermal neutron reactions in ^{10}B . The diodes were operated under zero bias conditions.

Figure 86. Normalized neutron-induced counts for various detector types as a function of neutron fluence. Degradation is less prominent for detector configurations 3246 and 3247, which consist of the thinnest epitaxial GaAs layers of the *p-i-n* diodes.

ii. Collection Two

Gamma-Ray Irradiation

Set Three consisted of five distinct sample detectors of one detector diode type each (Table 5). Each detector had 5000 Å of ^{10}B film on the blocking contact, and each detector was mounted with silver epoxy on alumina (Al_2O_3) substrate plates, the latter employed as a mount for the detectors in the aluminum chimney test boxes. All five of the sample detectors were placed into basic, laboratory-grade plastic wafer holders and inserted into the central region of a 20 kCi ^{60}Co gamma-ray chamber. The devices were completely surrounded by the ^{60}Co source. Irradiations ranged from 2.4×10^4 Rad to 1.8×10^7 Rad (calibrated with a Reuter-Stokes model RS-C4-1606-207 ionization chamber), and the dose increments were intentionally matched to the gamma-ray doses experienced by the detectors irradiated in the reactor pool. Following each gamma-ray dose, the devices were tested in the diffracted neutron beam, after which they were reinserted into the ^{60}Co chamber to receive a tenfold increase in gamma-ray exposure before the next incremented test measurement.

Figure 87a shows the results for sample detectors 'E' (Spire 76), in which the neutron reaction product spectra are plotted according to total gamma-ray dosage. Shown are the changes in spectra as a function of gamma-ray doses received in a ^{60}Co irradiator chamber. It is important to note that there is very little change in the spectra for the detectors, which indicates that the gamma-ray irradiations had minimal effect on performance. Figure 88a show the results for sample detectors 'M' as biased at -40 volts and Figure 89a shows that results for sample detectors 'A' as biased at -100 volts. Both sample types 'M' and 'A' seem unaffected by the gamma radiation. All other detectors represented in Table 5 experienced similar results, in which the pulse height spectrum showed very little change for the different doses of gamma rays. The detectors operate as counters, therefore stability in the total spectral neutron count rate is most important. Figure 90 shows the total neutron spectral counts as normalized to the control detector of the same diode type. Very little deviation in the count rate may be distinguished. Some slight manual errors in bias and gain settings may have contributed to the differences noted. Regardless, the spectral deviations and count rate differences are minimal. For this reason it is unlikely that gamma rays are responsible for the catastrophic failure of the devices.

Slow and Fast Neutron Irradiation

Sample detector Set Two consisted of five subsets in which samples of each diode type (Table 5) were within the subsets. Hence, unlike sample Set Three where the same detectors were repeatedly irradiated, *different* detectors from the same fabrication batch were irradiated for each radiation increment. The procedure was found necessary to eliminate neutron activation of the sample mounts. Each subset received a different neutron fluence that was monitored and recorded by the Au foils included with the detectors in the irradiation cartridges. For high neutron fluences, Fe wires were included with the Au foils as well. The total slow and fast neutron fluences ranged from 1.5×10^{11} n/cm² to 1.3×10^{14} n/cm² (Table 7). Hence, the devices were exposed to both gamma ray and neutron damage, but not to $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reaction product damage. Following irradiation, a 5000 Å film of ^{10}B was deposited on the detector blocking contacts to observe the neutron detection performance.

Figure 87. Pulse height spectra for detector diodes of type E as a function of (a) gamma-ray dose, (b) thermal and fast neutron fluence on bare detectors, and (c) thermal and fast neutron fluence on ^{10}B -coated detectors. The diodes were operated under zero bias conditions. Devices with ^{10}B -coating before neutron irradiation suffered damage from charged particle reaction products.

Figure 88. Pulse height spectra for detector diodes of type M as a function of (a) gamma-ray dose, (b) thermal and fast neutron fluence on bare detectors, and (c) thermal and fast neutron fluence on ^{10}B -coated detectors. The diodes were operated at -40 volts reverse bias. Devices with ^{10}B -coating before neutron irradiation suffered damage from charged particle reaction products.

The gamma ray doses for Set Two were similar to those doses experienced by sample Set Three in the ^{60}Co gamma-ray chamber (Refer to Table 7). Consequently, the detectors of sample Set Two were exposed to the same gamma-ray increments as sample Set Three and were also exposed to the neutron fluences described in the preceding paragraph and shown in Table 7. As discussed earlier, they were coated with ^{10}B *after* irradiation, thereby eliminating the damage contribution from the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction products. Figure 87b shows spectra for detector diodes 'E' from Set Two plotted according to the total neutron fluence received. Similar to sample Set Three, the spectra do not reflect appreciable spectral changes with increasing neutron fluence. All other detectors from Set Two demonstrated similar results in which very little difference was noted in the reaction product spectra for the various radiation exposures. For

instance, Figure 88b shows the results from detector diodes of type 'M' (at a bias of -40 volts), showing little difference for the various spectra.

Figure 89. Pulse height spectra for detector diodes of type A as a function of (a) gamma-ray dose and (b) thermal and fast neutron fluence on ^{10}B -coated detectors. The diodes were operated at -100 volts reverse bias. Devices with ^{10}B -coating before neutron irradiation suffered damage from charged particle reaction products.

The slight spectral changes observed in Figures 87b and 88b are most likely due to characteristic differences between the separate diodes and trivial manual error in the bias and gain settings. Some small degree of change perhaps may be due to neutron damage, and Figure 91 indicates that there is a decreasing trend in the neutron count rate with increasing neutron fluence. However, the devices continue to work well even at fluences in the 10^{14} n/cm^2 range. Regardless, the spectral deviations and count rate differences are minimal and are clearly unlike the changes observed in Figures 84-86 for Collection One. It is therefore unlikely that neutron exposure, for both fast and thermal neutrons, is responsible for the observed catastrophic failure of the devices in Collection One at neutron fluences near 10^{13} n/cm^2 .

$^{10}\text{B}(n,\alpha)^7\text{Li}$ Reaction Product Irradiation

Sample detector Set One was similar to sample Set Two, in that both sets were irradiated together in the reactor moderator pool and therefore received identical neutron fluences. The main difference between sample Set One and Set Two was that the detector diodes in sample Set One were coated with a 5000 Å film of ^{10}B before neutron irradiation. As a result, detectors from sample Set One received damage from gamma rays, fast and thermal neutrons, and the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction products.

Figures 87c, 88c, and 89b show the thermal neutron reaction product spectra for sample detector diode types 'E', 'M', and 'A' as plotted according to total neutron fluence. The effect of damage at fluences of $1.4 \times 10^{13} \text{ n/cm}^2$ and greater is unmistakable. Since the uncoated devices did not manifest such damage, it becomes evident that the ^{10}B film is responsible for the change. It is deduced that the charged-particle emissions from the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reactions are causing the observed damage effects. From previous work, a 5000 Å thick ^{10}B film yields a detection efficiency of about 2%, mainly because only 2% of the intersecting neutrons cause charged-particle reaction products to enter the detector [21, 22]. Since the onset of catastrophic failure is observed at neutron fluences near $1.4 \times 10^{13} \text{ n/cm}^2$, one can deduce that it is actually the exposure

to approximately 2×10^{11} charged-particle reaction products (alpha particles and ${}^7\text{Li}$ ions) per square centimeter that causes the catastrophic damage to the devices. Also, since neutron detection efficiency increases with ${}^{10}\text{B}$ film thickness, which results in more charge particles entering the device, the neutron fluence catastrophic failure limit will *increase* for thinner ${}^{10}\text{B}$ films and it will *decrease* with for thicker ${}^{10}\text{B}$ films, an important fact to consider.

Figure 90. Changes in the normalized total neutron-induced counts for various detector types as a function of gamma-ray dose.

Figure 91. Changes in the normalized total neutron-induced counts for detector types E and M as a function of neutron fluence. The devices were bare (uncoated) during the irradiation procedure.

Figure 92. Normalized total neutron-induced counts for all detector diode types in Collection Two as a function of neutron fluence. Type A detectors (SI bulk GaAs) show the greatest radiation hardness, most likely a result of the higher operating voltage. It is deduced that the $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reaction products caused the degradation shown.

Figure 93. Spectral centroid location for alpha-particle reaction products (from $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reactions) as a function of neutron fluence. It is deduced that the $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reaction products caused the degradation shown.

Figure 92 shows a count rate comparison of all diode types from Table 5. Bulk SI GaAs Schottky barrier devices (type 'A') appear to withstand the highest fluences. Although the added radiation hardness of bulk SI GaAs may be due to the high density of surface states and deep

level defects generally found with the material, the extended performance is most likely due to the external bias applied to the devices (-100 volts). Nevertheless, they too underwent catastrophic damage at neutron fluences above $1.4 \times 10^{13} \text{ n/cm}^2$. High-purity epitaxial GaAs *p-i-n* diode devices and high-purity epitaxial GaAs Schottky barrier devices performed similarly, and they failed at the lowest neutron fluences.

Figure 93 shows the 1.47 MeV peak centroid position as a function of neutron fluence. All devices showed considerable changes at $1.4 \times 10^{13} \text{ n/cm}^2$, however detector diode 'M' (an epitaxial Schottky diode) unexpectedly demonstrated the least amount of change. It should be noted that diode type 'M' devices required -40 volts reverse bias to operate, which (as with the bulk GaAs devices) most likely helped to increase their tolerance level. Type 'M' devices also demonstrated the highest leakage currents of all tested devices.

4. Observations

The self-biased design offers a straightforward method to produce low-cost, lightweight, compact, and low-power thermal neutron detectors for remote deployment. The design also can be pixelated for neutron imaging devices. Nonetheless, the environment near a thermal or fast neutron source is generally very hostile for semiconductor detectors. The irradiation results clearly indicate that the detectors are substantially radiation resistant to gamma rays, thermal neutrons, and fast neutrons. However, the devices undergo catastrophic failure when exposed to charged-particle reaction products from $^{10}\text{B}(n,\alpha)^7\text{Li}$ reactions. With 2% detection efficiency for 5000 Å ^{10}B films [24,29,31], it presently appears that it is the exposure to approximately 10^{13} n/cm^2 , which correlates to 2×10^{11} charged-particle reaction products (alpha particles and ^7Li ions) per square centimeter, that causes failure to the devices. These findings are consistent with previous work with both low energy charged particle irradiation and gamma ray irradiation (^{60}Co) of GaAs substrates [47-50].

The damage produced by the charged particle reaction products most likely introduces electron and hole traps into the active region, thereby reducing charge carrier mean free drift times [51-53] and compromising the built-in potential for self-biasing. Reduction in charge carrier mean free drift times and internal bias voltage both contribute to reduced charge collection efficiency, which would manifest itself as a steady reduction in observed pulse height [51-53], as observed. Such a hypothesis is supported by the fact that the devices that were operated with external bias withstood the highest neutron fluences before catastrophic failure. Since the detectors operate by sensing the 1.47 MeV alpha particles and the 840 keV ^7Li ions, it is mandatory that they withstand damage inflicted by these reaction products. At present the cause of damage has not been investigated, although it is suspected that the semiconductor contact and surface region is becoming highly non-crystalline due to atom displacement and void formation. Such a conclusion seems logical since the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction products have very short ranges in GaAs, those being 2.1 μm for 840 keV ^7Li ions and 4.2 μm for 1.47 MeV alpha particles [24]. However, the precise nature of the charged-particle damage from the irradiated, self-biased ^{10}B -coated high-purity epitaxial GaAs neutron detectors has, as yet, not been thoroughly investigated. Future studies will be directed toward understanding the exact mechanism of detector diode device failure and toward advanced designs that will supercede the present limitations.

Chapter III: Neutron Flux Deconvolution

As a key part of the project to determine the composition of spent fuel using an array of pixellated GaAs(^{10}B) detectors, an algorithm was developed to obtain the fuel composition through spectral deconvolution of the neutron flux emitted from the fuel. The MCNP Monte Carlo code was used to simulate fission neutron spectra, similar to those expected from an active assay of spent nuclear fuel, incident upon polyethylene moderators of varying thicknesses followed by an array of pixellated detectors. The spectral deconvolution algorithm was applied to the simulated detector response and showed qualitative agreement with the incident spectrum [54]. Toward the end of the project, effort was also made to test the validity of the flux deconvolution algorithm using a PuBe source at the Phoenix Memorial Laboratory of the University of Michigan, but no verifiable results were obtained due to the limitation of time available for the project.

Characterization of spent nuclear fuel is an important technological challenge to ensure proper storage and disposal of the irradiated fuel. One type of nondestructive technique to assess the fissile and fertile contents of spent fuel uses passive and active neutron interrogations [55]. Using recently developed GaAs(^{10}B) semiconductor detector arrays [56], it is proposed to determine the fuel composition through spectral deconvolution of the neutron flux emitted from the spent fuel. Presented are results of Monte Carlo simulations of the detector responses, illustrating the feasibility of the spectral unfolding technique for spent fuel characterization.

Considered is a shielded array of pixellated GaAs(^{10}B) detectors, as illustrated in Figure 94, where a collimated beam of neutrons emerging from the spent fuel is to be assayed. For a detector located at radius r from the axis of the neutron beam, $\phi(E)$, passing through a moderating film and incident upon the detector array, the detector reaction rate registered on the ^{10}B film of absorption cross section $\Sigma(E)$ is:

$$R(r) = \int_0^\infty dE' \int_0^\infty dE \phi(E') K(r, E' \rightarrow E) \Sigma(E). \quad (42)$$

Here $K(r, E' \rightarrow E)$ is a kernel representing the neutron flux of energy E emerging from a moderating film and arriving at the detector due to a unit flux of energy E' incident on the moderating film. For a collimated beam, $K(r, E' \rightarrow E)$ may be interpreted [57] as the point spread function (PSF) on the detector plane. To obtain the flux $\phi(E)$ of neutrons emerging from the fuel element is found with:

$$A_i(r, E') = \int_0^\infty dE K_i(r, E' \rightarrow E) \Sigma(E), \quad (43)$$

representing the detector reaction rate associated with a unit flux of energy E' incident on the i^{th} moderating film and rewrite Equation 42:

$$R_i(r) = \int_0^\infty dE' \phi(E') A_i(r, E'). \quad (44)$$

In terms of Equations 43 and 44, $A_i(r, E')$ may now be interpreted as the detector cross section modulated by the moderating film PSF. Using a multigroup structure, $j = 1, 2, \dots, N$, Equation 44 may be discretized:

$$R_i(r) = \sum_{j=1}^N A_{ij}(r) \phi_j, \text{ for the } i^{\text{th}} \text{ moderating film,} \quad (45)$$

or equivalently as a matrix equation $\mathbf{R} = \mathbf{A}\Phi$. Using a 33 energy-group structure and representing polyethylene moderating films with thicknesses of 40, 60, 80, 100, and 120 mm followed by a 2mm film of cadmium, continuous-energy Monte Carlo simulations of the detector geometry shown in Figure 94 have been performed with the MCNP code [58] to determine $\mathbf{A}$. MCNP simulations have been also performed for an evaporation fission neutron spectrum similar to that expected from the assay of spent nuclear fuel. Since the matrix $\mathbf{A}$ is often nearly singular and not square, a regularized truncated singular value decomposition method [59-60] to recast Equation 45 is employed:

$$(\mathbf{A}^T \mathbf{A} + \lambda \mathbf{S}^T \mathbf{S}) \Phi = \mathbf{A}^T \mathbf{R}, \quad (46)$$

where $\mathbf{S}$ is the Hessian matrix and λ is a regularization parameter.

With $\lambda = 1.5 \times 10^{-9}$ and singular values retained within a factor of 2×10^{-6} of the largest singular value, Φ is determined by applying Equation 46 to the MCNP simulations. The particular value of λ was chosen to minimize the 2-norm of the residual vector $\rho = \Phi - \Phi^*$, where Φ is the unfolded flux and Φ^* is the reference fission spectrum in the 33 group structure. The comparison between Φ and Φ^* in Figure 95 demonstrates the feasibility of gaining spectral information from the assay of spent fuel and indicates reasonably good overall agreement. Some visible deviations are, however, noted for $E < 100$ keV, together with negative flux values for low flux regions for $E > 7$ MeV.

Effort is underway to refine the modulated cross section matrix $\mathbf{A}$ through alternate energy-group structures so that errors in the unfolded spectrum in Figure 95 may be reduced. This study assists with the development of a robust unfolding algorithm essential for the deconvolution of complex, composite spectra emerging from spent fuel.

Figure 94. Simulated detector geometry with a rotational symmetry about the centerline. The detector is modeled with a 2- μm enriched ^{10}B coating on a GaAs device of 198- μm thickness.

Figure 95. Comparison of the unfolded neutron spectrum Φ (bar chart) and reference fission spectrum Φ^* (solid line).

Chapter IV: Neutron Images

Section A: Introduction

While many non-destructive techniques are presently used to evaluate spent nuclear fuel (SNF), few offer a direct method to determine the spatial mapping of the material within the fuel. Knowledge of the internal distribution of nuclear materials in SNF can lead to increased safety and better fuel burn-up economics. Recently, neutron computed tomography (CT) techniques have been implemented as a method to determine the spatial distribution of nuclear material. One of the limiting factors in the development of these systems, however, has been detector response in the harsh radiation environment. Further, since SNF cannot be easily relocated, a portable system that can rapidly determine the spatial distribution of nuclear materials is desired.

Radiation hard neutron CT systems using gas detectors have been developed for the non-destructive analysis of SNF, but the limited spatial resolution of gas detectors restricts their applicability. Neutron CT systems based on CCD detectors have higher spatial resolution than gas-based systems, but the neutron conversion screen and mirror system currently used limits system portability. Solid state GaAs neutron detectors have been shown to be both radiation hard [45] as well as gamma ray insensitive [29,55], both advantageous properties in the SNF environment. Further, arrays of these detectors are relatively easy to assemble and the resulting detection system is much smaller than current CCD-based systems.

A neutron CT system based on a solid state GaAs neutron detecting array has been developed and tested. Presented in the following is a description of the system developed and resulting reconstructed images [61].

Section B: Equipment and Methods

The neutron CT system developed consists of a GaAs neutron detector, sample platform capable of linear and rotary movement, computerized control and data acquisition system, and a filtered-backprojection CT reconstruction algorithm. A collimated thermal neutron beam was provided by E-port of the Ford Nuclear Reactor (FNR), and epithermal neutrons were provided by a Sb/Be photoneutron source. Transmission radiography of several phantoms was performed at multiple angles and used to reconstruct tomographic images.

1. GaAs Neutron Detectors

The detector developed for this system is composed of a 16x2 pixel array of ^{10}B -coated GaAs Schottky barrier detectors [29,55]. Each pixel has dimensions of 1 mm by 0.5 mm with 50 μm of insulation between the pixels to prevent cross talk. The detector is mounted on a 64-pin chip holder and housed in a light tight Al box to provide shielding from RF and visible light interference (Figures 59 and 60).

The thickness of the ^{10}B film used for this detector is 1 μm , resulting in a measured thermal neutron detection efficiency of 2.8%, matching well with theory [29]. Custom preamplifier circuits capable of measuring the energies deposited in the device from the $^{10}\text{B}(\text{n},\alpha)^9\text{Li}$ reaction were developed and mounted within the Al box. The current detector only uses one row of the array, creating an 8.8 mm x 1 mm active area.

The design of the detector allows for discrimination between the signals produced from background gamma rays and those from neutrons by setting the lower level discriminator (*LLD*) at 300 keV for each pixel [55]. Radiation hardness studies previously mentioned have revealed that catastrophic damage occurs at fluence above 10^{13} n/cm² [11]. Pixel size of the current detector was chosen for ease of manufacturing and is ultimately limited by the range of charge particles in the neutron-absorbing medium. For boron-coated devices this is less than 5 μ m [20,55]. Detector matrices with higher efficiency conversion layers and smaller pixels are currently being investigated.

The output from each channel was amplified using a Canberra 2022 amplifier and lower level discrimination performed using an Ortec 423 single channel analyzer (SCA). The *LLD* setting was determined by the system noise and gamma ray contribution from the beam. As shown elsewhere [29], setting the *LLD* to approximately 300 keV discriminates out much of the gamma-ray background. An Ortec Labmaster PC card was used as a multichannel counter, enabling automated data collection.

2. Sample Platform

The object to be imaged was placed on a sample platform with step motor controlled linear and rotational axes, both controlled via the serial port of a personal computer. For samples larger than the detector active area, the linear step motor was used to move the platform along the face of the detector. Future system improvements will include the addition of vertical platform motion to allow image acquisition of large objects

3. Neutron sources

Two neutron sources were used for the CT experiments presented, a thermal neutron beam and an epithermal Sb/Be photoneutron source. E-port of the 2 MW FNR was the source of the thermal neutron beam, which is aligned with the corner of the FNR D₂O tank (see Figure 62, position 3). The resulting thermal neutron flux is 2×10^6 n/cm²-s with a gamma-ray exposure of 1.1 R/hour. The E-port has a length over distance (L/D) ratio of 55.2 and a diameter of 12.7 cm. The Sb/Be source is composed of an Sb cylinder surrounded by a Be sleeve and produces neutrons through a photoneutron reaction, resulting in a neutron distribution centered on 23 keV as well as a significant gamma-ray component [20].

The Sb/Be neutron source was chosen for its portability and neutron energy, which closely match the transmission peak of iron [3,4]. Future development will involve investigating the effects of placing an iron filter between the source and detector to filter background radiation from the SNF and scattered epithermal neutrons.

A collimator containing lead and borated polyethylene was placed between the sources and the phantom, as shown in Figure 96. This collimator limited the beam size to the active area of the detector, but was unpixelated. Experiments were also performed with a collimator placed between the sample and detector in order to assess the effects of neutron scattering within the sample.

Figure 96. Neutron CT experimental setup with thermal neutron beam. A custom holder was developed to hold the Sb/Be photoneutron source.

4. Data Correction and Reconstruction

A set of transmission projection data was collected for each sample at multiple angles. Raw data from objects imaged in the thermal beam were corrected for fluctuations in reactor power and the profile of the beam by normalizing each profile to a profile taken without the object. Due to the decay of the Sb source, epithermal beam images were corrected to account for both the beam profile and the decay of the source. Before performing the reconstruction, the data was scaled to a range from 0 to 255 and arranged into a sinogram.

A filtered-backprojection reconstruction algorithm utilizing several Fourier filters was used to map the spatial neutron attenuation recorded in the sinogram. A Shepp & Logan filter was used for all reconstructions.

Section C: Results and Discussion

Several phantoms were developed to assess the quality of the system. The first sample investigated was an aluminum block containing several water columns of different diameter, as shown in Figure 97. This sample was placed in the thermal beam as close to the detector as possible. Since the object was much larger than the active area of the detector, the sample was walked across the face of the detector in increments of 6.6 mm. Figure 98 shows the neutron intensity along the sample. The steep intensity profile at the edges of the holes is one pixel wide, indicating that the pixel size is currently the major limit on spatial resolution.

The epithermal Sb/Be source was also used to image the aluminum block with water columns. At 23 keV the total macroscopic cross section of water is H_2O nearly the same as that of Al. As a result, images of the sample in the epithermal beam did not reveal the water columns.

Figure 97. Schematic of Al block containing H₂O columns of various diameters used to determine spatial resolution.

Figure 98. Thermal neutron intensity profile of Al block sample with water columns of various diameters. Resolution appears to be limited by pixel size.

Step wedge samples of Cu, Al, steel and polyethylene were imaged to characterize the ability of the system to differentiate the attenuation coefficients of the materials. These samples were placed as close to the detector as possible and again walked across the detector face in 6.6 mm increments. Except for the polyethylene sample, all six steps were easily discernable a neutron intensity plot (see Figures 99 through 102).

Figure 99: Polyethylene step wedge dimensions and results.

Figure 100: Copper step wedge dimensions and results.

Figure 101: Aluminum step wedge dimensions and results.

Figure 102: Steel step wedge dimensions and results.

As shown in Figure 103, a plot of the measured vs. accepted macroscopic cross section shows a non-linear relationship, making material determination difficult using the current setup. The disagreement between the measured cross section and accepted values is mainly due to scatter within the sample. Placing a pixilated collimator between the sample and detector is one method of decreasing this effect, while active measurements of the scattering effects can allow correction for the total scatter present, as shown in the literature [52]. The attachment of a pixilated collimator to the detector face is currently being investigated.

Figure 103. Plot of measured vs. accepted macroscopic cross section for Al, Cu and steel, demonstrating the non-linear response of the current setup.

Figure 104 shows the schematic drawing of a cylindrical sample placed in the thermal beam to characterize the tomographic system. In order to limit the effects of scatter within the sample, a non-pixelated Pb collimator was placed between the sample and the detector. The sample was rotated through 180 degrees in increments of 3 degrees, and 10-minute counts collected.

Figure 104. Schematic of Al cylinder sample containing water columns of different diameters.

Performing a filtered-backprojection reconstruction with a Shepp & Logan filter results in the image shown in Figure 105. An intensity threshold was applied the reconstructed image based upon the neutron transmission intensity profile of the sample. A threshold level of 50% of the maximum water column intensity was chosen, corresponding to a gray scale intensity of 115. As can be seen in Figure 106, all three water columns are clearly visible. The measured area of all three water columns is 0.78 cm^2 , approximately 80% of that expected, indicating a decreased resolution to that found in the radiographic line image shown in Figure 98. This resolution decrease is expected due to the convolution used during the reconstruction process, which adds additional noise to each pixel. Experiments to determine the tomographic resolution of the system are currently being performed.

Figure 105. Reconstructed image of Al cylinder sample with a Shepp & Logan filter applied. Each pixel is $0.25 \text{ mm} \times 0.25 \text{ mm}$.

Figure 106. Reconstructed image of Al cylinder sample in thermal beam with a threshold value of 115. All three water columns are easily discernable and the measured area 80% of that expected.

The epithermal Sb/Be source was also used to image a cylindrical steel sample, shown in Figure 107. The sample was rotated through 180 degrees in increments of 5 degrees and 45 minute counts collected. Increasing the count time was necessitated by the decreased cross section of boron at 23 keV, lowering the detection efficiency from 2.8% to less than .001% [55]. The experimental setup was the same as used for the thermal reconstruction except the collimator between the detector and sample was not present. As shown in Figure 108, the reconstructed tomographic image was extremely blurred, even with the noise filter applied. It is suspected that this is due to high-energy gamma-ray contamination and low neutron count rates. Despite the relative gamma-ray insensitivity of the GaAs array [29,55], the 240,000:1 gamma to neutron ratio from the Sb./Be source [19] and low epithermal neutron detection efficiency decrease neutron count rates to below that of the background gamma-rays. Improved Sb/Be source designs and investigations into higher efficiency neutron conversion layers have been developed in order to decrease this affect. Figure 109 shows the reconstructed image with a threshold value of 150, arbitrarily shown to highlight the edges of the steel annulus.

Figure 107. Schematic drawing of steel cylinder sample imaged using the Sb/Be epithermal neutron source.

Figure 108. Reconstructed image of steel cylinder sample using a Shepp & Logan filter.

Figure 109. Reconstructed image of steel cylinder sample in an epithermal neutron beam with a threshold value of 150.

Figure 110. The experimental arrangement located within the hot-cell. The detector box (from Figure 60) is located on the right within the lead and cinder block shielding.

Figure 111. Another view of the detector box as located within the hot-cell.

Figure 112. The NIM electronics, SCA, amplifiers and computers used to operate the pixel detector were located outside of the hot-cell.

Chapter V: Conclusions

^{10}B -coated GaAs devices have been demonstrated as viable compact thermal neutron detectors. They can be used under low power conditions and can withstand thermal neutron fluences up to 10^{13} n/cm^2 before noticeable degradation begins to occur. The devices are relatively inexpensive to produce and are reliable. Additionally, pixelated arrays can be developed and integrated onto GaAs wafers (or silicon) due to the maturity of the technology.

Different substrates and neutron-reactive materials are now under investigation. Si and SiC are being investigated as alternative semiconductor substrates for the devices, and ^6LiF and 98% pure ^6Li are being studied as alternative neutron-reactive films. New via hole patterns are under development that will increase the surface area coverage, thereby increasing the observed thermal neutron detection efficiency.

Future devices will incorporate the use of Li-based films to reduce gamma ray background effects. The films, either ^6LiF or pure ^6Li , will be tailored to optimize film thickness. Encapsulation techniques to prevent Li decomposition are being explored. Also, low gamma ray absorbing substrates will be investigated as alternatives to GaAs. At present, SiC is under investigation as a possible alternative substrate, mainly because of its low average atomic number, its thermal tolerance, and its radiation hardness.

A variety of simple methods are available to increase the thermal neutron detection efficiency of thin-film-coated semiconductor neutron detectors. Individually the methods offer respectable efficiency increases. Yet, if used together, dramatic increases in thermal neutron detection efficiency can be realized. Efforts are now underway to optimize systems using $^{10}\text{B}/^6\text{LiF}$ films. Systems using $^{10}\text{B}/^6\text{Li}$ are also under investigation with an emphasis on preserving the ^6Li layer and preventing its decomposition. Neutron imaging arrays using the improved technology are also under investigation.

A neutron CT system has been developed based on a ^{10}B coated GaAs detector array capable of withstanding 10^{13} n/cm^2 . This detector system has been used to obtain radiographic and neutron CT images from a thermal neutron beam located in the FNR and an epithermal Sb/Be neutron source. The data acquisition and control system allows for automated data collection. Thermal tomographic slices of aluminum cylinders containing water columns with diameters of 53 mm have been reconstructed. Epithermal tomographic experiments with a steel annulus have also been performed, though low count rates and the possibility of photon contamination are currently being addressed. Improved source and shielding designs are being implemented to reduce the gamma ray background component from the Sb/Be photoneutron source.

Thermal radiographic images of an aluminum block indicate that the system resolution is currently limited by the size of the array pixels, which can be reduced to $5 \mu\text{m}$ if needed. The results from thermal radiography experiments with step wedges also indicate that the determination of material attenuation coefficients can be improved by placing a collimator between the detector and sample. Investigations on large area two-dimensional detector arrays with both improved detection efficiency and pixelated collimators are currently underway.

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Appendix I. List of published papers resulting from the project.

Refereed Publications:

1. D.S. McGregor, R.T. Klann, H.K. Gersch, E. Ariesanti, J.D. Sanders, and B. VanDerElzen, "New Surface Morphology for Low Stress Thin-Film-Coated Thermal Neutron Detectors," IEEE Trans. Nuclear Science (2001) submitted.
2. J.D. Sanders, J.T. Lindsay, and D.S. McGregor, "Development of a GaAs-Based Neutron Tomography System for the Assay of Nuclear Fuel," IEEE Trans. Nuclear Science (2001) submitted.
3. H.K. Gersch, D.S. McGregor, and P.A. Simpson, "A Study of the Effect of Incremental Gamma-Ray Doses and Incremental Neutron Fluences Upon the Performance of Self-Biased ^{10}B -Coated High-Purity Epitaxial GaAs Thermal Neutron Detectors," Nuclear Instruments and Methods (2001) in press.
4. D.S. McGregor, H.K. Gersch, J.D. Sanders, R.T. Klann, and J.T. Lindsay, "Thin-Film-Coated Detectors for Neutron Detection," Journal of the Korean Association for Radiation Protection, 26 (2001) pp. 167-175.
5. Michael R. Hartman and John C. Lee, "Spent Nuclear Fuel Characterization through Neutron Flux Deconvolution," Transactions of the American Nuclear Society, 84 (2001) pp. 105-107.
6. D.S. McGregor, R.T. Klann, H.K. Gersch, and Y-H. Yang, "Thin-Film-Coated Bulk GaAs Detectors for Thermal and Fast Neutron Measurements," Nuclear Instruments and Methods, A466 (2001) pp. 126-141.
7. D.S. McGregor, S.M. Vernon, H.K. Gersch, S.M. Markham, S.J. Wojtczuk and D.K. Wehe, "Self-Biased Boron-10 Coated High Purity Epitaxial GaAs Thermal Neutron Detectors," IEEE Trans. Nuclear Science, 47 (2000) pp. 1364-1370.

Conference Proceedings:

1. D.S. McGregor, H.K. Gersch, J.D. Sanders, and R.T. Klann, "Designs for Thin-Film-Coated Semiconductor Neutron Detectors," Conference Record of the IEEE Nuclear Science Symposium, San Diego, California, Nov. 4-9, 2001.
2. D.S. McGregor, R.T. Klann, H.K. Gersch, E. Ariesanti, J.D. Sanders, and B. VanDerElzen, "New Surface Morphology for Low Stress Thin-Film-Coated Thermal Neutron Detectors," Conference Record of the IEEE Nuclear Science Symposium, San Diego, California, Nov. 4-9, 2001.
3. J.D. Sanders, J.T. Lindsay, and D.S. McGregor, "Development of a GaAs-Based Neutron Tomography System for the Assay of Nuclear Fuel," Conference Record of the IEEE Nuclear Science Symposium, San Diego, California, Nov. 4-9, 2001.

4. D.S. McGregor, H.K. Gersch, J.D. Sanders, R.T. Klann, and J.T. Lindsay, "Thin-Film-Coated Detectors for Neutron Detection," Conference Record of the First iTRS International Symposium On Radiation Safety and Detection Technology, Seoul, Korea, July 18-19, 2001.
5. R.T. Klann, C. L. Fink, D. S. McGregor, and H. K. Gersch, "Development of Semiconductor Detectors for Fast Neutron Radiography," Conference Record of the 15th Int. Conf. on Applications of Accelerators in Research and Industry, November, 2000.
6. H.K. Gersch, D.S. McGregor, and P.A. Simpson, "A Study of the Effect of Incremental Gamma-Ray Doses and Incremental Neutron Fluences Upon the Performance of Self-Biased ^{10}B -Coated High-Purity Epitaxial GaAs Thermal Neutron Detectors," Conference Record of the IEEE Nuclear Science Symposium, Lyon, France, Oct. 15-20, 2000.
7. D.S. McGregor, J.T. Lindsay, Y-H. Yang, and J.C. Lee, "Bulk GaAs-Based Neutron Detectors for Spent Fuel Analysis," Conference Record of ICONE-8, 8th International Conference on Nuclear Engineering, April 2-6, 2000, Baltimore, MD USA.
8. R.T. Klann and D.S. McGregor, "Development of Coated GaAs Neutron Detectors," Conference Record of ICONE-8, 8th International Conference on Nuclear Engineering, April 2-6, 2000, Baltimore, MD USA.
9. D.S. McGregor, S.M. Vernon, H.K. Gersch and D.K. Wehe, "Self-Biased Boron-10 Coated High Purity Epitaxial GaAs Thermal Neutron Detectors," Conference Record of IEEE Nuclear Science Symposium, Seattle, Washington, Oct. 25-29, 1999.

Patents filed:

1. Patent filed – "Coated GaAs Devices for Neutron Detection," R.T. Klann and D.S. McGregor, September 2000.
2. Patent Filed – "Pocked Surface Neutron Detector," D.S. McGregor and R.T. Klann, June 2001.