

The area radiation gamma and neutron origin telemetry system

By

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Abstract

In today's global security environment, the opportunities to alert to, search for, and identify radioactive material have increased as both China and Russia grow their nuclear stockpiles and aggressive posture, unrest in the middle east continues, and US borders endure an influx of migrants. Traditional mobile radiation detector systems are equipped with He-3 gas detectors that are often bulky, expensive, and offer limited information to the user. The need for low-cost, high-efficiency, mobile detection systems sensitive to thermal neutrons and gamma-rays persists. Single-sided MSNDs were developed as an early compact, low-power, inexpensive, mass-producible, and rugged solution offering moderate intrinsic thermal-neutron detection efficiency ($\sim 30\%$). While the single-sided MSNDs could be stacked to increase the neutron detection efficiency to up to 42%, aligning the devices was unreliable. Dual Sided-MSNDs were then developed and coupled with superior LiF backfill densities and increased trench depth etching processes on 1.5-mm thick silicon wafers, to achieve record-breaking intrinsic thermal-neutron detection efficiencies of $69.3 \pm 1.5\%$. In the present work, an alternative doping scheme known as the pvp-n-dot for DS-MSNDs was implemented. The pvp-n-dot doping profile was designed to bias across the p-type doped trenches and an n-type doped ring on the perimeter of the trenches. The top side of the device is read out through the traditional center bond pad, while the back side of the device is read out through the bottom contact. The devices produced with the pvp-n-dot doping scheme achieved an intrinsic thermal neutron detection efficiency of $64\% \pm 2.5\%$ and on average, achieved an intrinsic thermal neutron detection efficiency of approximately 24%. The GRR was determined to be $5.69\text{E-}05$ at a gamma-ray dose rate of 10mR/hr.

To accompany the thermal neutron detectors in the mobile radiation detection system known as the ARGANOT (Area Radiation Gamma and Neutron Origin Telemetry), $\text{SrI}_2(\text{Eu})$ and later $\text{CsI}(\text{Tl})$ scintillator using SiPMs were used to alert to gamma-rays and supply the user with gamma-ray spectrums. To ensure good energy resolution of the spectrum regardless of environmental conditions, a temperature-correcting algorithm was implemented for each module to maintain the energy calibration. Using Gamma-ray Detector Response and Analysis Software (GADRAS), directionality, source identification, and activity estimation were implemented.

The ARGANOT underwent extensive testing as a single module and as a full system worn in a garment on and off a phantom body. Results indicate that the ARGANOT optimal performance is best achieved when worn by a user, but it can alert to sources when operating in a standalone configuration or when flown on an Unmanned Aerial Vehicle.

The cost-effective ARGANOT system presents many benefits to a user such as its high efficiency, low profile, and discrete appearance; the ability to operate as a stand-alone module in a passive monitoring situation or as a directionality capable vest; and the ability to instantly provide the user information about the environment through tools such as GADRAS.

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Abbreviations

2D – Two Dimensional

3D – Three Dimensional

App – Application

ARGANOT– Area Radiation Gamma And Neutron Origin Telemetry

B – Boron

CAN – Controller Area Network

CLLB – Cs₂LiLaBr₆:Ce

CLLBC – Cs₂LiLa(Br,Cl)₆:Ce

CLYC – Cs₂LiYCl₆:Ce

CV – Capacitance vs Voltage

DFPR – Dry-Film Photoresist

DS-MSND – Dual-Sided Microstructured Semiconductor Neutron Detector

GaAs – Gallium Arsenide

GADRAS – Gamma-ray detector Response and Analysis Software

GRR – Gamma-ray Rejection Ratio

GARR_n – Gamma-Ray Absolute Rejection Ratio for neutrons

Gd – Gadolinium

HeRep –³He-Replacement

ICRP – International Commission on Radiation Protection

ICRU – International Commission on Radiation Units and Measurements

IV – Current vs Voltage

Li – Lithium

LiF – Lithium Fluoride

LLD – Lower Level Discriminator

MND – Modular Neutron Detector

MRD – Modular Radiation Detector

MSND – Microstructured Semiconductor Neutron Detectors

P – Phosphorus

PMT – Photomultiplier Tube

PNNL – Pacific Northwest National Laboratory

Si – Silicon

SiC – Silicon Carbide

SiPM – Silicon Photomultiplier

SNM – Special Nuclear Material

SOD – Spin-on Dopant

SrI₂(Eu)– Strontium Iodide

SWAP – Size, Weight, and Power

TCAD – Technology Computer-Aided Design

VLSI – Very-Large Scale Integration

WDD – Wearable Detector Device

WND – Wearable Neutron Detector

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Summary of Contributions to DS-MSND Technology, Science, and Instrumentation

The following list is a brief summary of the significant contributions that advanced the state of the art for DS-MSND based technology detailed within this dissertation.

- Further developed the *n*-ring style doping scheme of DS-MSND devices for increased charge collection.
- Compared the spectral performance of various boron doping recipes and used IV curve data as evidence to conclude that the wafers had been doped into a degenerate state.
- Developed temperature and agitation-controlled KOH anisotropic wet etching method to achieve uniform dual sided silicon features as deep as 700 μm while maintaining structural integrity.
- Developed packaging methods for DS-MSND devices and implemented them into a mobile radiation detection system.
- Developed perhaps the world's smallest most discrete gamma-ray and neutron detection capable modules that provide the user with competitive intrinsic efficiency, alerting, gamma-ray spectroscopy, GADRAS source identification and activity estimation, and source locating abilities.
- Characterized the ARGANOT system and individual module's neutron response in terms of angle and distance for both bare and moderated Cf-252 and AmBe neutron sources.
- Characterization of the ARGANOT system and single module's efficiency on and off the user's body for moderated and bare Cf-252 and AmBe neutron sources.
- Conducted aerial characterization of a single ARGANOT module when flown on board a DGI (Da-Jiang Innovations) drone to aid in the search of a 0.8 mCi ^{137}Cs gamma-ray source.
- Demonstrated the lifespan and performance degradation of MSND based devices when exposed to a thermal neutron flux spanning 1^7 n cm^{-2} up to 1^{12} n cm^{-2} and fast neutron fluxes as high as 1^{10} n cm^{-2} .

- Performed simulations to optimize the MSND dimensions when backfilled with paraylene for fast neutron detection to achieve the greatest fast neutron detection intrinsic efficiency.

1 The Nuclear Domain



Statue of Aristotle by Cipri Adolf Bermann, 1915 [1].

“We make war so that we may live in peace” [2].

~ Aristotle

1.1 Organization of Thesis

Chapter one will introduce radiation detection and the geopolitical motivations. The following two chapters will examine the discovery of both the gamma-ray and neutron and methods that have evolved to alert, quantify, and identify radiation presence. In chapter four, a more in-depth look at the Dual Sided-Microstructure Semiconductor Neutron Detector (DS-MSND) utilized in the Area Radiation Gamma and Neutron Origin Telemetry System (ARGANOT) modules is described. The development from their basic foundation in the late 1990's to their modern-day existence including their design, fabrication, packaging, and characterization, along with all problems and solutions implemented along the way will be examined. In chapter five, the strontium iodine spectroscopic gamma-ray detector, its operation, signal processing, packaging, performance, and characterization will be reviewed. Once the background of the DS-MSND and strontium iodine gamma-ray detector has been sufficiently reviewed, their implementation into a mobile detection system is described in chapter six. Chapter six will introduce mobile radiation detection, ARGANOT predecessors and then feed into a discussion on the ARGANOTs physical operation. Within chapter seven the capabilities, performance, and post analysis of the ARGANOT system is presented to include the alerting for both neutron and gamma-ray detectors, and the directionality algorithm. Additionally, The GADRAS software's integration to the ARGANOT system is discussed. The GADRAS extension enabled gamma-ray spectroscopic isotopic identification and activity and personnel dose estimation. The ARGANOT system will then be compared to other commercially available

mobile field dual neutron and gamma-ray detection systems from suppliers such as particularly offering dual gamma-ray and neutron detectors. Chapter eight will conclude the work by exploring ARGANOT's future versatile applications. Such applications will include use in active and passive investigative scenarios on a variety of platforms such as drones and other concealment methods.

1.2 Great Power Competition

Political leaders often weigh the threat of war with strategic deterrence opportunities that may in fact deter war itself. No other deterrent has been more effective in human history than nuclear weapons. Careful analysis of the conflict in Vietnam, for example, illustrates that the fear of escalation to nuclear war led American, Chinese, and Russian leaders to show restraint and avoid World War III [3]. With the advent of the nuclear weapon, our world has endured the longest lasting period without a major war between global powers—a result that some say is just coincidental or a nuclear taboo, but is nonetheless fact [4].

One of the key factors that has maintained this relative peace is the US nuclear umbrella, especially in East Asia where North Korea poses a constant threat with its nuclear weapons. The US repeatedly reaffirms its commitment to defend South Korea with its nuclear arsenal in case of an attack from the North, most recently during President Yoon Suk Yeol's visit to Washington in April 2023. The US-South Korea joint declaration signed by Yoon and President Biden included several measures to enhance the credibility and effectiveness of the US extended deterrence strategy, such as sending US nuclear-armed submarines for port visits in South Korea and establishing a consultation group to respond to North Korean nuclear attacks [5]. These initiatives aim to reassure South Korea of the US security guarantee and deter North Korea from any aggression.

Despite the peace brought about with the proliferation of nuclear arsenals, the nuclear era has brought unique challenges to the forefront of global security and stability [6]. Over past decades, drastic changes in capabilities have caused global powers to feel vulnerable and question their own ability to defend against a nuclear attack and destroy an adversary's nuclear arsenal. In response to the inequities in capability, nuclear powers strive to increase the reliability and performance of their arsenals. Within the cycle of perceived or actual imbalance, followed by improvement of capability, and back to imbalance, military planners employ three basic approaches to protect their country's nuclear forces from attack: hardening, concealment, and redundancy [7]. The implementation of these approaches complicates nuclear security and the ability for a national power to accurately determine an adversary's true capability. This could increase reliance on sensors that may be able to defeat approaches to disguising nuclear capabilities.

On the other hand, the nuclear threat from non-state actors or violent extremists (terrorist groups) such as ISIS and Al Qaeda, is a significant national security challenge that requires urgent attention and action. Violent extremists may seek to acquire or use nuclear weapons or materials to cause mass casualties, destruction, and disruption to their enemies or society at large, to provoke a nuclear war or escalation between rival states or alliances, to demonstrate power, capabilities, and legitimacy, or to fulfill ideological, religious, or apocalyptic beliefs [8]. One of the possible ways to deter and prevent the nuclear threat from violent extremists is to enhance the role of sensors in detecting and monitoring nuclear activities and materials. Sensors can be used for various purposes related to nuclear security, such as detecting and locating nuclear weapons or materials at borders, ports, airports, checkpoints, and facilities and tracking and tracing the movement and origin of nuclear weapons or materials in case of theft, loss, diversion, or smuggling. Additionally, sensors may be used to safeguard facilities that may be an attractive target for terror groups through monitoring and verifying the compliance of states with their nuclear non-proliferation and disarmament obligations [9].

Several applications of sensors will be reviewed and allow for the various modes of detector operation to be considered when outlining design and operation goals for upcoming radiation detection systems.

1.2.1 Investigative Sensor

The signatures that sensors detect span many fields including acoustic, seismic, radio frequency, chemical, and biological. One of the most important fields of collection for sensors and the focus of this work is ionizing radiation sensors. Sensors are very powerful and diverse tools that have and continue to confirm the presence or movement of illicit materials, equipment, and or weapon systems as well perform personal monitoring for soldiers.

1.2.1.1 Active Investigative Sensors

To determine an adversary activity at a specific area of interest, sensors may be deployed by physically placing them at or near an area of interest. The use of sensors in this way is referred to as active investigation and typically requires a human or machine to deploy a sensor, which comes with both advantages and disadvantages. One advantage is that a human can more reliably place a sensor or maneuver an area of interest based on native human input for example. A human may be able to adjust as needed and troubleshoot systems at a moment's notice.

Additionally, a human may be able to maneuver within an environment drawing little attention. However, there are also clear disadvantages to human delivery methods. One aspect to consider is that humans in today's technologically advanced environment are extremely trackable and observable. Therefore, it may be hard for a human to deposit a sensor in locations discreetly and without adversary knowledge. Additionally, even once the sensor is placed, the human is still perhaps in an undesirable location and extraction may be necessary. Extraction may become very difficult considering the location where sensors are commonly deployed are typically very difficult for humans to physically access. Machine delivery of a sensor, such as a drone or unmanned underwater vehicle may more easily maneuver such obstacles. Additionally, an unmanned vehicle may be able to stay in place for several days before returning home and either leave the sensor in place or return home with the sensor attached. However, unmanned delivery systems often emit discoverable electronic signals, and therefore the environment for which the unmanned delivery system would enter must be considered along with the consequences of the system's discovery.

1.2.1.2 Passive Investigative Sensors

In addition to actively investigating suspected weapon sites and providing policy makers with details regarding the materials found at the site, passive investigation also serves an important role. In a passive investigation mode, a sensor is placed in a location and passively waits for nuclear material to pass by, on a roadside for example. If such an event occurs in a pattern unnatural to the environment, an alert may be sent to the end user notifying them that illicit activity likely just passed by the sensor location. An example of covert passive monitoring is shown below in Figure 1-1 and Figure 1-2. The sensor panels alongside the railroad tracks are designed to resemble a power station. Such sensors would be especially beneficial to deploy against enemies where the railways or highways are used to ship materials from enrichment to production plants. While it may prove difficult to deploy sensors in a rival country, doing so would be helpful when attempting to maintain an up-to-date estimation on the travel routes of trains and perhaps even convey information regarding the materials on board.



Figure 2-1 An example passive radiation monitor is shown alongside of a railroad [10].

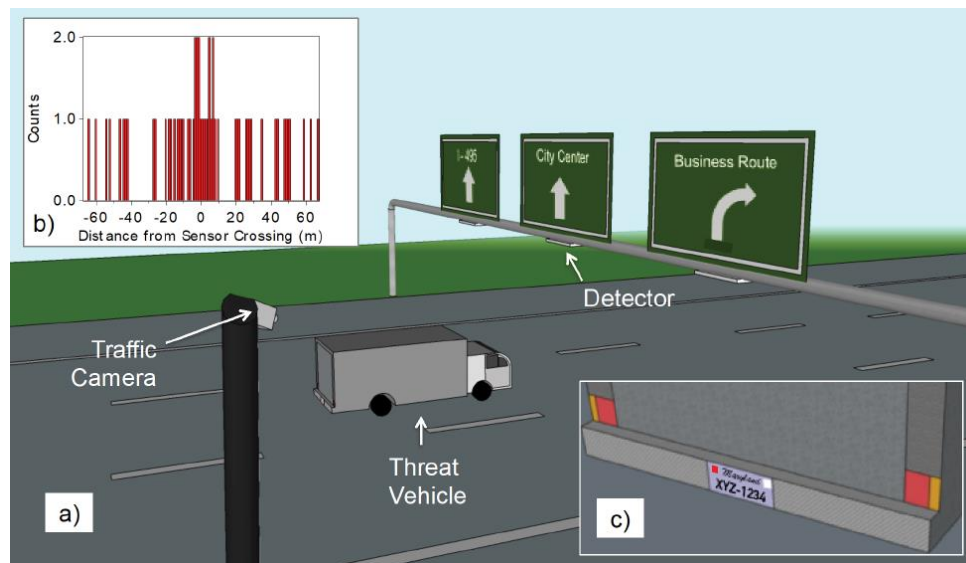


Figure 2-2 An example of concealed radiation sensors above a highway is shown to monitor the passing traffic [11].

1.2.1.3 Defeating Concealment

In this very serious game of hiding and seeking illicit nuclear materials, the job of the seekers therefore becomes to defeat concealment methods. Recent trends in technology are actively eroding illicit material concealment efforts. As sensors become more readily available, manufacturing practices become more established, and new alternative sensors drive competition, prices are likely to decline with time. With the commonality and affordability of sensor systems, widgets will likely become more commonly deployed in investigative scenarios with little monetary loss if such sensors are discovered. If a third party were to discover and effectively decommission a sensor system, the investigative party may just simply redeploy a

new system at a new location for a monetary price worth the sacrifice. As long as the intelligence provided by the system is worth the frequent sacrifice of the system, the active investigation and deployment of the sensors system may be deemed a fair price to pay. Additionally, perhaps the third party's continuous responsibility of hunting for sensor systems will not only be cumbersome and expend resources but may in fact deter illicit behavior because of the fear that the behavior will likely be discovered.

Additionally, with increased sensor deploy-ability, the development of sensor arrays operating in a distributed like mode will likely increase. With a widening number of sensors working in unison deployed in the world, more opportunities for collections are created and therefore a fuller picture of a country's military planning and programs may be pieced together. If sensors such as acoustic and ionizing radiation detectors could be frequently placed along the various railroad lines, the path of movement for illicit nuclear material could be discovered, along with the type of material, how much material, the identification of start and endpoints, and perhaps even potential vulnerabilities of the network could all be uncovered. With a wider lens of collection, more high-quality intelligence will allow for greater transparency of an adversary's military planning and drive future action. Furthermore, sensors are growing increasingly smaller and more compact, allowing the widget to take several different shapes and forms. Sensors today can fit in a pocket and operate in a covert mode if needed. Because of the small form factor, sensors are also taking advantage of concealment methods and can be concealed within rocks, tree branches, and other parts of the environments to covertly collect information from passing targets.

With improved performance, deployment opportunities and expansion of delivery platforms, use of sensors is now a more viable option. Coupled with development in the artificial intelligence sector, sensors and their delivery platforms will deliver higher-quality data for analysis than ever before, allowing an adversary's nuclear strategy to become even more transparent to American leadership. Sensors are overcoming complicated shielding tactics, and are able to gather source info, granting unprecedented transparency to policymakers.

Many nuclear programs also utilize redundancy in everything from command and control to an alternate location, where a system is housed allows for insurance that an adversary would not be able to eliminate the strike option. However, within redundancy, concealment methods

may also be deployed to obscure where the live weapon systems are housed. When decoys are used such as the silos shown in Figure 1-3, analysts must once again rely on a sensor's ability to distinguish empty and populated housings. Additionally, because of how difficult it may be in reality for an individual to gain physical access to such a hard-to-reach environment, a widget that is durable, small, discrete, and self-sufficient may be required to collect signals intelligence on such difficult targets.

In order to overcome the sophistication in offensive abilities, defensive sensor technology must maintain pace with their counterparts. In response to hardening, the concealment, and redundancy of weapon arsenals, sensor development must respond in kind to defeat adversary tactics and provide policymakers and military planners alike with the most reliable and accurate data possible in the strictest of environments.

In addition to use in restricted environments, sensors must also be available for passive and active monitoring both abroad and in the homeland. The need is for compact sensors that are compact, low weight, utilize concealment tactics, and quickly respond to the environment.

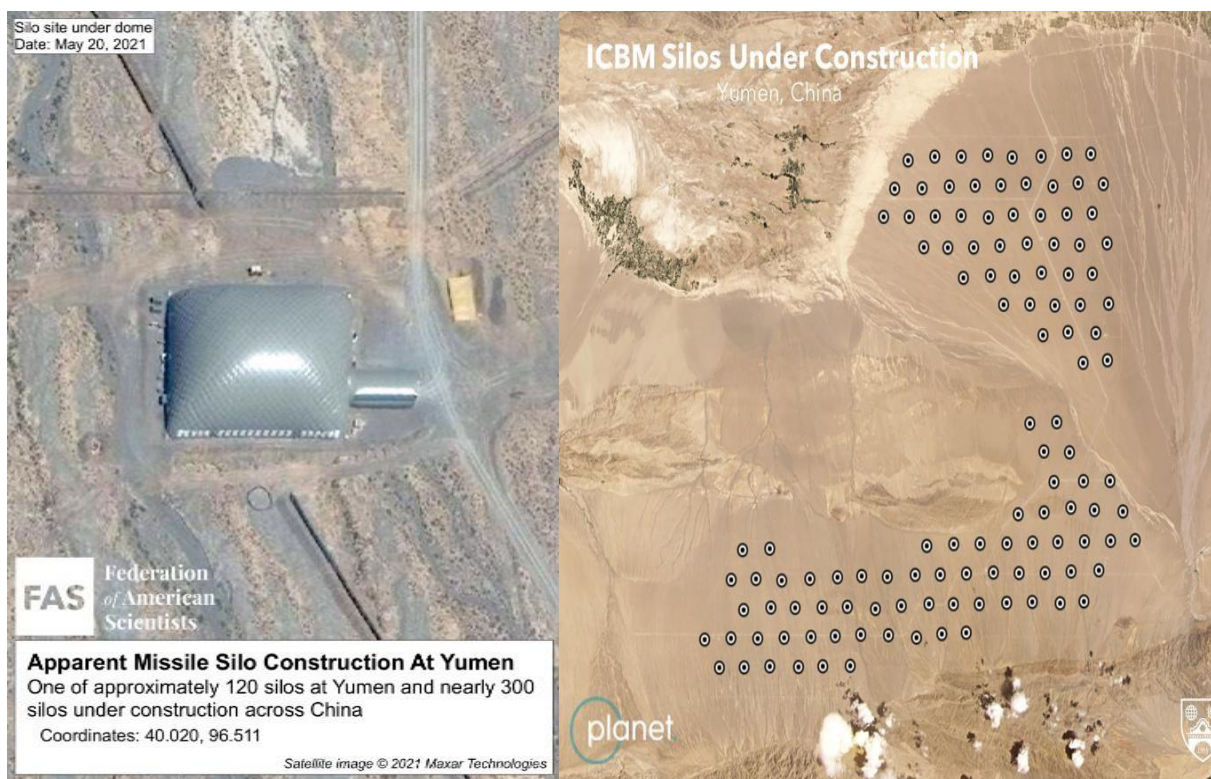


Figure 2-3 Multiple silo fields with up to 300 silos total have been constructed in China as of 2021. [12]

1.2.2 Personal Monitoring

While many debates on whether the next global security threat will be of a chemical, biological, or nuclear nature, one cannot ignore the recent unnerving comments from the Russian president Vladimir Putin in Fall 2022 reminding the west of Russia's more modernized tactical nuclear weapons and Putin issuing a threat that that Russia "will without question use all the means at our disposal in Ukraine. This is not a bluff" [13]. Contrary to the perception many Americans hold about nuclear weapons, countries such as the United States, Russia, and China can tailor effects to achieve widespread destruction or limited destruction. The ability to achieve discreet effects is what makes Russia's large and varied arsenal of battlefield nuclear weapons so dangerous and usable. Even more alarming is China's unprecedented and unbridled nuclear expansion that appears to be looking to the Russian stockpile as a blueprint. Threats such as the one issued by Putin in September are proof that he is well aware of the ability to manipulate nuclear weapons effects to accomplish his agenda. The threshold for low yield weapons has perhaps been lowered and has rightfully served as motivation for recent work focusing on detecting and monitoring minimum safe distances (MSD) in nuclear detonations.

In light of the stockpile status of our two greatest rivals, it therefore is quite reasonable to respond by outfitting military and emergency response personnel with radiation sensors in particular that can help regulate and communicate a minimum safe working distances for each individual. Focus in particular should be devoted to outfitting soldiers and first responders with mobile radiation sensors that can provide real time data regarding dose and count rates (activities). These sensors must be light weight, offer knowledge of radiation type, and perhaps even spectroscopic information [14].

As shown through various simulations, the destructive effects of nuclear weapons depend on the yield, height of burst, and location of the detonation and can be tailored to achieved desired effects and produce a challenging yet approachable environment for personnel to maneuver within [15]. Immediate access to sensor data for all non-civilian personnel working in or near the affected area is crucial especially when one considers the fact that a lower yield weapon at varying heights of burst will produce a smaller effect zone with high ("hot spots") and low activity spots that will drastically change with respect to the dose over time. This therefore means that the radius of safe entry over time will be drastically changing over a matter of hours. Therefore, we must be able to have safe and reliable methods, such as sensors, to answer the

most pressing questions our policymakers may have so that they can make the most informed decisions.

1.3 Motivation for Mobile Detector System

As described above, sensors are tools to overcome the challenges presented when attempting to monitor and inform on the nuclear activities of adversaries. Through encouraged competition and mere time, sensors have and will continue to experience improvements to provide the best intelligence to the end user. Based on the discussion above of how sensors can be deployed in response to modern day threats of adversarial nuclear programs, design goals and objectives can be outlined for a future wearable radiation detection system (RWD). The radiation detection system of the future must be dynamic, versatile, and include adaptive technology to thrive in the advancing nuclear domain.

1.3.1 Goals and Objectives:

Knowing that a successful sensor is one that can bring greater transparency of adversary movements, plans, and programs in multiple application scenarios as mentioned above, the following criteria has been set for the design of a future all-encompassing radiation detection system as directed by the sensor development sponsor, the Defense Threat Reduction Agency (DTRA) as well as by the needs of the industry learned through experience and testimony of analysts and in the field end users.

1. Develop a high-performance radiological detection system suitable for use in a variety of low-profile missions, including covert search, leave-and-retrieve, passive and active interrogation, distributed array, and personnel monitoring.
2. Incorporate communication and control controls into each individual sensor so that each module can function independently or as a network of detectors.
3. Leverage recent innovations in neutron detector design and development and associate electronics to produce neutron sensitive detector modules with higher detection efficiency, lower electronic noise, and lower power usage.
4. Leverage recent innovations in gamma-ray detector design and development to integrate a spectroscopic gamma-ray detector with each sensor module.
5. Produce an Android application for live time communication between each detector module and the end user that provides a simple interface, robust data management, and is ATAK & ITAR compliant.

6. Can be used in either a wireless configuration for ease of operation or in a low radio frequency (RF) emissions mode to evade a signal discovery.
7. Produce an initial set of test articles, refine the design, produce an initial prototype, refine the design, and produce a final set of prototypes that are near commercial level.

1.4 Introduction to ARGANOT

Upon considering the needs of the end users, the questions that leadership wishes to have answered, and in accordance with the goals and objectives previously listed, a mobile radiation detection system was designed under the direction of DTRA. The predecessor systems, which will be discussed in greater detail later, served as an excellent starting point to develop a radiation detection system to satisfy all present and future needs of the end users.

Functional design began by collecting and analyzing requirements from three sources: ANSI Standard N42.53 Performance Requirements for Backpack-based Radiation-Detection Systems, the DTRA SOW, and the performance of the TSG's Eagle Ray detection system as an exemplar for comparison and developed derived requirement from existing technology [16]. Once we established a comprehensive list of requirements (most of which are encompassed in those mentioned above) and began to cross-correlate each requirement to specific system functions, such as neutron detection and electrical design. In this way, the team ensured that every requirement was met, and every system component served a necessary function. The functional design task also included producing a data management plan, a communications architecture, and a system power budget.

In designing the detection system with consideration of the goals and objectives defined above, the naming of the system eventually came to pass. The team longed for a name that encompassed the intelligent, adaptable, arrayed nature of the detection system. Many ideas came and went with little traction, until one inspiration stood above the rest: the Argonaut octopus. The eight-armed Argonaut species shown below in Figure 1-4 is incredibly rare and frankly a bizarre creature. The octopus swims in the open ocean and is consequently very difficult to locate and study due to their translucent seashell-shaped casing and odd methods of mobility [17].



Figure 2-4 The Argonaut octopus is shown above. Notice the translucent shell concealing the octopus.

After considering the unique nature of the Argonaut creature, the Area Radiation Gamma and Neutron Origin Telemetry system or ARGANOT, seemed a natural fit for the technology under development. The wish of the ARGANOT designers and engineers was that like its namesake, the highly advanced eight-module-legged system, some of which are shown below in Figure 1-5, would provide the end-users with a compact, mobile, and highly unique system experience. With those qualities in mind—the ARGANOT system was brought to fruition.



Figure 2-5 Final version of ARGANOT modules produced on the RWD contract.

The conclusion of this work produced a commercial-level modular radiological detection system that is reconfigurable in the field to suit multiple mission requirements. Within each ARGANOT module, there are several Dual-sided Microstructured Semiconductor Neutron Detectors (DS-MSND) to provide high-efficiency neutron detection accompanied by Strontium

Iodide gamma-ray detectors. Together the gamma-ray and neutron detectors are packaged in a modular and ruggedized form suitable for deployment by military forces. Development of the detector system included algorithm development to implement source directionality, isotope identification, and activity estimation through GADRAS software implementation, and user notifications in the form of alarms and warnings. The source directionality function uses the relative differences in count rates produced by multiple neutron and gamma-ray sensors, plus their configuration information (i.e., their relative positions) to determine a general direction to the source (i.e., forward, back, left, right). The final product also included the development of software for a handheld device to provide a graphical user interface (GUI), detector control and health monitoring, data management, analysis, and storage, and be compatible with situational awareness tools (MFK and TAK).

2 Introduction to Radiation Detection



Antoine Coypel, Democritus, 1692 [18]

“Nothing exists except atoms and space, everything else is opinion.” [19]

~Democritus, the “Laughing” Greek philosopher, 460 BCE

Ancient Greek philosopher Democritus was one of the founders of ancient atomist theory and chose the Greek word “*atomos*” to describe the indivisible building blocks of the universe. Democritus theorized that atoms actively repel one another when they collide [20]. Although our understanding of atomic theory is greatly improved, many still regard Democritus as the philosopher who propelled many into a deeper understanding of atomic physics such as Henri Becquerel in 1896. The French physicist through experimenting with photographic film and uranium salts correctly concluded that penetrating radiation came from uranium [21]. In concluding that Uranium emitted energy without any need for excitation by an external energy source, Becquerel was credited as the discoverer of radiation.

As is known today, radiation is fundamentally a transfer of energy through space time and is produced by radioactive materials. Radiation generally is categorized into two sectors, electromagnetic and particulate radiation. Within those two sectors, the radiation type whether it be electromagnetic, or particulate can be further classified into ionizing radiation and non-ionizing. Ionizing simply means that the incident particle is capable of separating an electron from the parent atom while non-ionizing radiation is typically too low in energy to separate an electron from the parent atom. Although non-ionizing radiation can contribute to negative health effects such as Havana Syndrome through heating of tissue, ionizing radiation is largely the

concern for the national security community because of its high penetrating ability and the pronounced damaging effects in human DNA for both the near and long term [22]. Ionizing radiation may interact either directly or indirectly. Directly ionizing radiation such as alpha and beta charged particles and gamma-rays actively transverse a material—such as a detector—and deposit energy through numerous coulombic interactions per unit path length traveled. Indirectly ionizing radiation, such as neutron particles instead interact with individual atoms. As a result of ionization, a measurable current creates a pulse height spectrum. While the formation of spectra will be discussed in greater detail later in this chapter, a spectrum relates back to the initial amount of energy deposited per interaction. A spectroscopist may then determine the radiation source present through analyzing the features that appear in the collected spectrum. The resulting peaks in the pulse height spectrum are formed based on various modes of interaction for the impinging radiation. Each method of interaction and the general detector response to neutrons and gamma-rays is discussed below.

2.1 Neutron Detection

Neutrons are subatomic particles partially comprising the nucleus of atoms. Free neutrons can be produced in a variety of ways such as cosmic-ray interactions, fission (spontaneous or nuclear), fusion, or the bombardment of specific target material with radiation. Cosmic-ray interactions are the primary source of natural neutron background radiation. The neutron background energy spectrum peaks around 1 MeV on Earth's surface, and neutron flux levels vary from approximately 0.01 to $0.03 \text{ n cm}^{-2} \text{ s}^{-1}$ [23]. Because neutrons are not all that prevalent in the background and are fairly unique to special nuclear material (SNM), neutrons are often the preferred mode of alerting to the presence of SNM with gamma-ray emission spectroscopy serving to identify the SNM isotopically.

2.1.1 Interactions

Neutrons are neutral, and therefore, neutrons do not interact coulombically with the surrounding medium as do electrons, protons, alpha particles, and heavy charged particles. Instead, neutrons interact through scattering or absorption with a target nucleus. The neutron cross sections vary greatly with both energy and elements as shown in Figure 2-1. Equation 2-1 describes the uncollided intensity over thickness t , with the initial number of neutrons being I_0 , and Σ being the total macroscopic interaction cross section which can be defined as the probability per unit path length that a neutron interaction occurs.

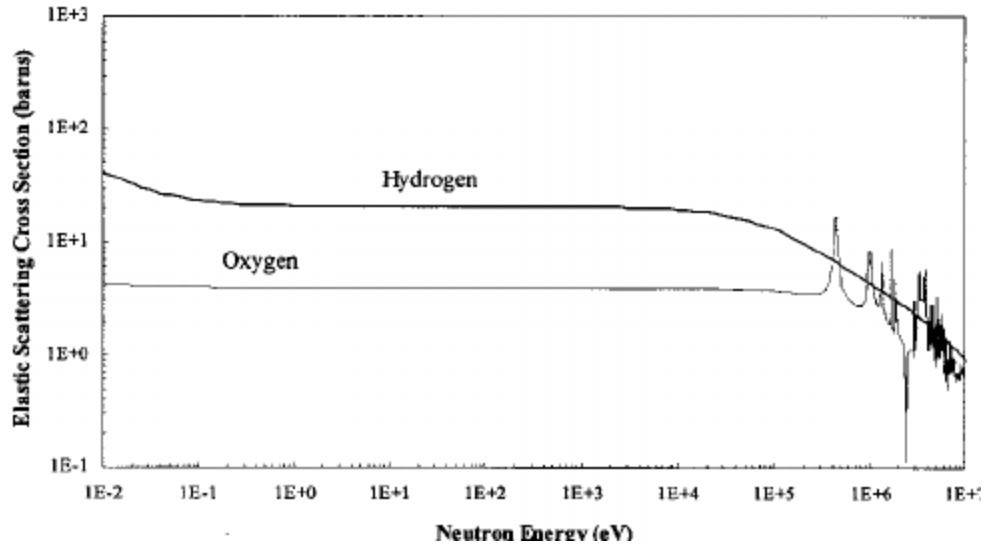


Figure 2-1 The macroscopic interaction cross section for various materials across a range of energies is shown [24].

$$I(t) = Ie^{-\Sigma t} \quad (2--1)$$

Neutrons with energy of approximately 0.025 eV are referred to as slow neutrons and neutrons with energy beyond 0.025 eV up to ~0.1 MeV are referred to as epithermal neutrons. Beyond 1 MeV, neutrons are considered to be fast neutrons. Fissile isotopes are those which can undergo fission upon the absorption of a thermal neutron and sustain a chain reaction. Three well known fissile isotopes are U-233, U-235, and Pu-239. Fissionable isotopes, however, are those which can be made to fission upon the absorption of a neutron with sufficiently high kinetic energy [25].

There are several outcomes for a fission reaction that include the release of gamma-rays, neutrons, and charged particles, such as protons or alpha particles. In neutron detection specifically a large focus is placed on fission events that produce charged particles. This is because the neutron is neutrally charged and therefore difficult to detect. With a fission event releasing charged particles that are easily detectable, the current produced in a device from ionizing charged particles may serve as a mode to indicate the presence of neutrons. Common isotopes used in radiation detectors for those exact reasons are lithium-6 (Li-6), boron-10 (B-10), and Helium-3 (He-3) [26].

The Li-6 nucleus has a thermal neutron cross section of 938.8 barns with a Q value of 4.78 MeV from an interaction, shown below in Figure 2-2 and is represented with Equation 2-2. For slow

neutrons, the alpha particle and triton reaction products are emitted isotopically with 2.05 and 2.73 MeV, respectively, at 180° difference. Lithium is a highly reactive alkali metal and is typically used either as LiF or as the component of a scintillator.

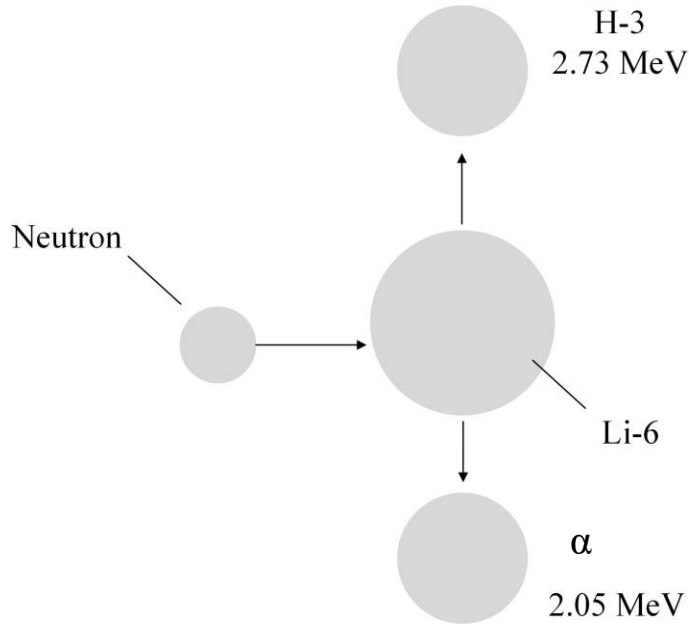
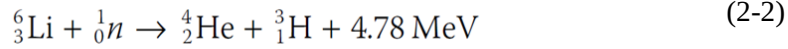


Figure 2-2 The fission of a Li-6 atom upon absorbing a neutron is depicted.

He-3 has a fairly large thermal neutron cross section of 5320 barns for 2200 ms⁻¹, but is relatively rare in abundance. He-3 is in fact a decay product of tritium, a common component in nuclear weapons. Therefore, with the the success of arms treaties in bringing the number of permissible nuclear warheads allowed for both the US and Russia down, along with the rising popularity of He-3 for research and industry, the supply of He-3 has susquently decreased over the past decade or so [27].

For an absorption of a thermal neutron a triton particle with 0.191 MeV of energy and H-1 with an energy of 0.573 MeV are spontaneously emitted as represented in Equation 2-3. The two reaction product as shown in Figure 2-3 are emitted in opposite directions and may ionize the sourounding medium.

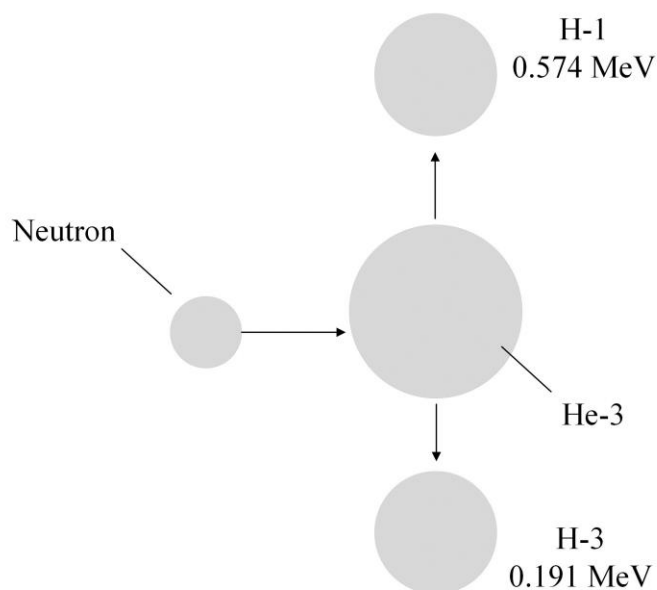
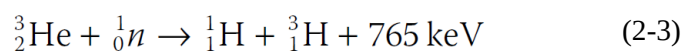
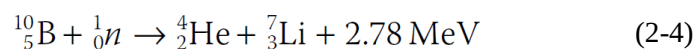


Figure 2-3 The fission reaction for a He-3 atom is shown.

Boron-10 has a thermal neutron cross section of 3847 barns and interacts according to Equation 4, producing a total of 2.78 MeV per fission reaction. However, B-10 is unique in that the fission reaction product outcomes are dictated by a branch ratio. As shown in Figure 2-4, 94% of reactions release an alpha particle with 1.47 MeV of energy and a lithium-7 atom in an excited state with 0.84 MeV of energy. However, only 6% of reactions release an alpha particle with 1.78 MeV and a lithium-7 atom with 1.02 MeV.



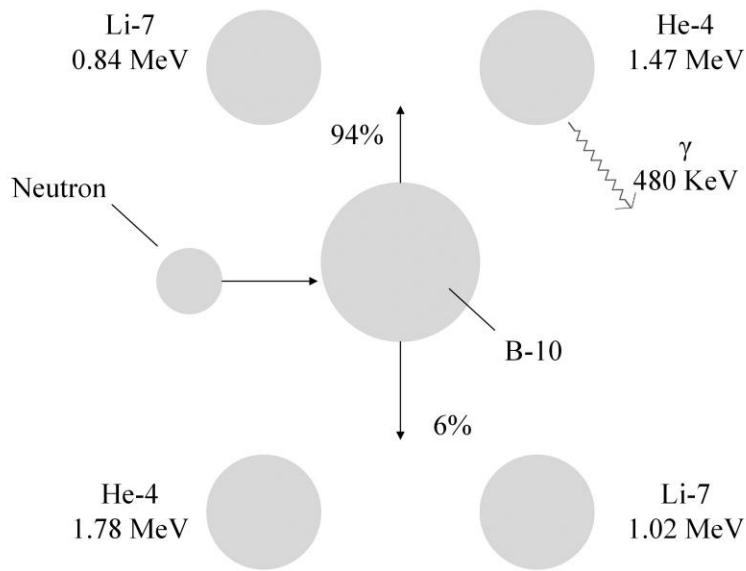


Figure 2-4 A boron fission reaction is shown in which a branch ratio determines whether the final energy of the reaction product. Note that 6% of the time.

2.1.2 Neutron Detector Evaluation

When selecting a neutron detector for a specific purpose in mind, it is important to consider several parameters of evaluation to ensure the optimal detector has been chosen for the measurement of interest. The following parameters will be discussed: the efficiency of the system and the ability to calibrate, the gamma-ray rejection ratio, and the size, weight, Power consumption, scalability, and cost (conveniently summarized as SWaP-SC).

2.1.2.1 Efficiency

The efficiency of a detector may be described in varying ways, resulting in many efficiency definitions. The absolute efficiency (ϵ_A) is defined as the ratio of the measured number of counts (R_{det}) recorded by the detector to the total number of emissions from the radiation source (A) with branching ratio (B) over time. Simply put, absolute efficiency represents the ratio of the count rate to the source emission rate as described Equation 2-5. As depicted below, the flux seen by the detector as position 1 verse position 2 is altered as a function of the solid angle. Therefore, absolute efficiency has little value if the experimental arrangement is not described with enough specificity to allow the experiment to be reproduced [28].

$$\epsilon_I = \frac{R_{det}}{AB} \quad (2-5)$$

The intrinsic efficiency(ϵ_i) corrects for the change in flux due to a change in geometry through the use of the fractional solid angle, Ω which presents the proportional probability that a random radiation event from the source will be emitted and intersect with the detector face for the proposed geometry as represented in Figure 2-5. The intrinsic efficiency may be represented by Equation 2-6.

$$\epsilon_I = \frac{R_{det}}{AB\Omega} \quad (2-6)$$

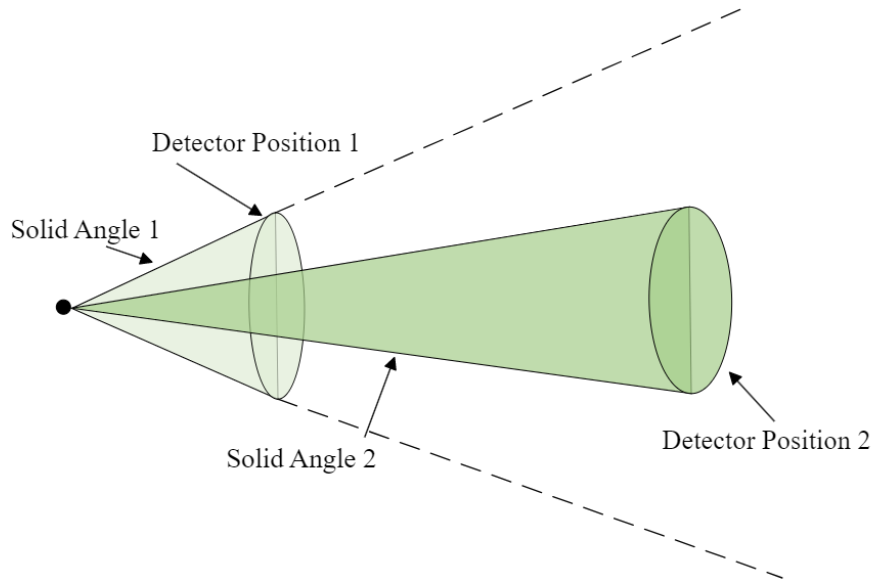


Figure 2-5 The solid angle is demonstrated to change the number of events reaching the detector face as the distance between the source and detector change. Notice that the solid angle at position 1 is much greater than that of position 2.

Not only is the count rate of the detector dependent on the geometry of the detector in relation to the source, but the neutron reaction rate is also dependent on the neutron flux. The neutron interaction rate is dependent on the neutron absorption cross sections in barns. For materials such as Li-6, He-3, and B-10, the cross sections appear linear within the thermal region (<1 MeV) on a log log scale of energy versus cross section. Such materials are commonly referred to as 1/v neutron absorbers. As derived by McGregor and Shultis 2011, 1/v absorbers offer great advantages in that if the target nuclei exhibit 1/v_r absorption behavior, then any arbitrary cross

section $\sigma(v_{ro})$ for relative speed v_{ro} is related to the cross section at v_r as represented by Equation 2-7 [29].

$$\sigma(v_r) = \frac{\sigma(v_{ro})v_{ro}}{v_r} \quad (2-7)$$

With substitution, Equation 2-8 results where $\Sigma_a(E_0)$ is an arbitrarily chosen macroscopic reaction cross section at corresponding speed v_0 , n is the total neutron density, and N is the total target nuclei density.

$$Reaction\ Rate = \Sigma_a(E_0)nv_0 \quad (2-8)$$

The result of Equation 2-8 is very important in that it states that the reaction rate of neutrons with a $1/v$ material is independent of the neutron or target nuclei velocity distributions [30]. Hence, a measurement using a reaction rate to determine detection efficiency is independent of the neutron energy distribution for $1/v$ absorbers and decomplicates the calibration process for thermal neutrons where neutrons must be in $1/v$ region [31].

2.1.2.2 Gamma-Ray Rejection Ratio

Gamma-ray absolute rejection ratio in the presence of neutrons (GARRn) is the absolute neutron detection efficiency in the presence of neutrons and gamma-rays, $|\varepsilon_{\gamma n}|$, divided by the absolute neutron detection efficiency without gamma-rays present. The GARRn is described by Equation 2-9. In determining GARRn a moderated ^{252}Cf neutron source (in 5mm lead and 25mm of high-density polyethylene) should be located at 2 meters from the horizontal and vertical center of the front of the detector and a Co-60 gamma-ray source should be located so the average field strength is 10mR/hr over the front face of the detector [32].

$$GARRn = \frac{|\varepsilon_{\gamma n}|}{|\varepsilon_n|} \quad 2-9$$

An ideal GARRn is 1.0 which indicates that the neutron detectors have no gamma-ray response. A strong gamma-ray rejection ratio is important to avoid invalid detector results that could incorrectly conclude that a neutron source was present or overestimate the neutron source activity. A weak GRR could be especially problematic in arms control scenarios where detectors are regularly employed to determine the amount of nuclear material as part of non-proliferation efforts in which neutron enrichment measurements are regularly conducted. Overestimation may

trigger aggressive unwarranted policy action, further amplifying the importance of a strong GARRn neutron detector.

2.1.2.3 SWaP-SC

In addition to the metrics used to characterize the detector performance, SWaP-SC are arguably the most important parameters of evaluation for the detector system as a whole. The mission space for which the detector may be deployed can completely dominate detector selection. Often times end users will face challenges to obtain a detector that is light weight with a small form factor, which is also highly efficient at an affordable cost. Scalability of detector technology not only affects the cost but could also limit the end user's ability to equip teams with enough detection units, replace malfunctioning units when needed, as well as fulfill order requests within a reasonable time window. All SWaP-SC factors are used to evaluate detector systems, in addition to the detector specific parameters discussed, and thus should also be heavily weighted in the detector design process.

2.1.3 Common Thermal Neutron Detectors

Known as the gold standard for thermal neutron detection, because of its high intrinsic thermal neutron detection efficiency, He-3 gas filled detectors are the most popular neutron detectors today. Typically, He-3 is used for gas filled detectors, while B-10 and Li-6 offer large versatility for the end user in configurations such as gas, semiconductor, and scintillator type detectors.

2.1.3.1 Gas Filled Detectors

Gas filled detectors for neutron detection are typically classified into two sections: Neutron reactive gas types and those lined with neutron reactive material that then rely on a proportional gas to relay the charge generated to the counting electronics. Gas filled detectors, such as He-3, see Figure 2-6, are normally cylindrical in shape, are pressurized, and are capable of operating in high temperature and high flux environments and are therefore hermetically sealed by the manufacturer. The detectors operate by allowing the neutron to fission with the He-3 gas, which generates the charge particles that go to ionize the He-3 gas, creating an electron and ion pair. The charge generated may then be collected by applying a bias. In a number of gas detector steps, as the charge carriers drift through the gas with great momentum. They collide with electrons along the path, transferring a portion of energy to the new charge carrier which also proceeds to collide with other neighboring electrons, triggering a Townsends Avalanche

[33]. Once the charge carriers are collected within the detector electronics, a histogram spectrum such as the one shown in Figure 2-7 will begin to form with tallied counts on the y axis with the x axis representing neutron energy broken into energy bins.



Figure 2-6 Two cylindrical He-3 neutron detectors are shown above.

Within the spectrum below, one will notice that the various features appear in the individual energy of each reaction product, along with the sum of both. It is ideal for all energy to contribute towards the sum peak at 764 keV, however based on where in the detector the reaction products are created, the spectrum will develop stair steps. Because the reaction products are emitted isotopically, there are instances where one of the reactions products is absorbed in the wall of the detector while the remaining product goes on to contribute towards the detector signal. In those instances, the other reaction product energy is lost, and therefore a distribution is created indicating the reaction product that did contribute to signal collection. Additionally, as denoted on the spectrum below, the detector ends also contribute to the formation of the spectrum. Either end of the detector will possess a lower electric field than the rest of the device volume, meaning that a decrease in charge collection for events more in the ends of the detector will be seen. Hence, the decreased charge collection for events will contribute to lower energies on the spectrum.

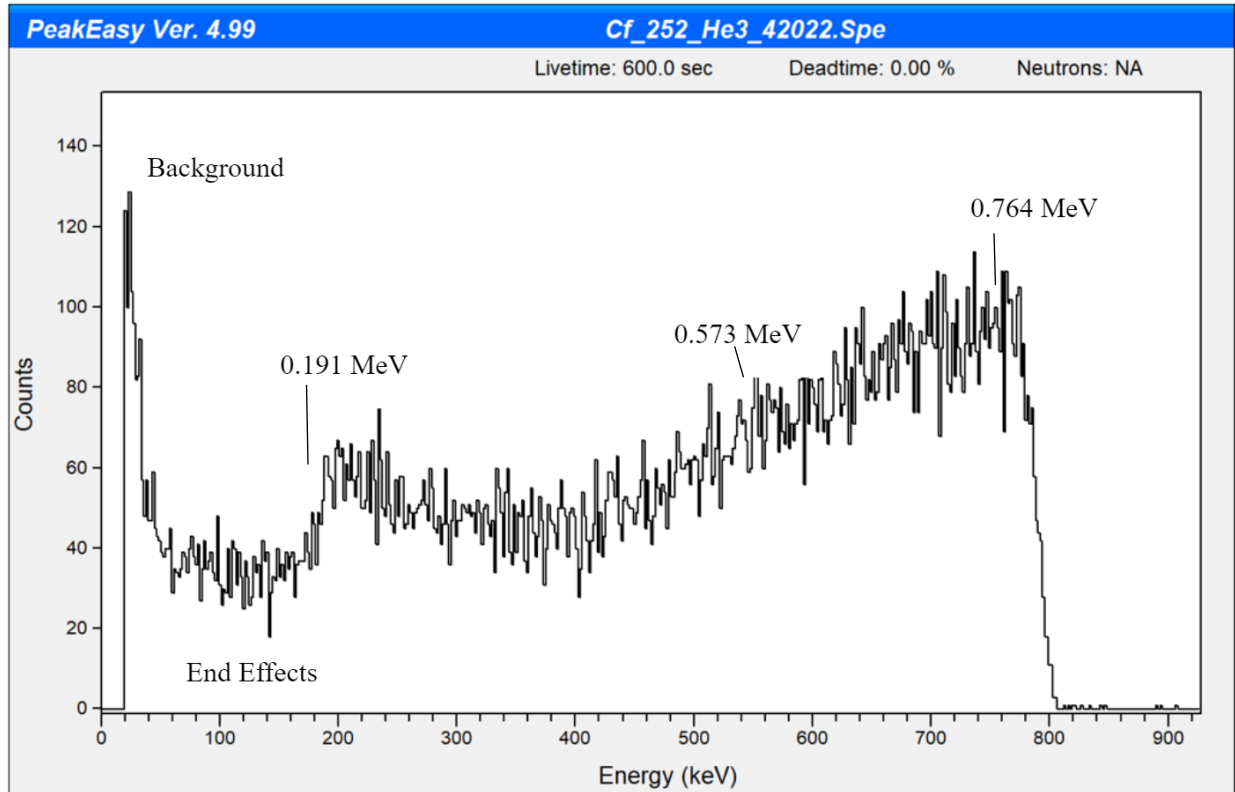


Figure 2-7 A He-3 neutron reaction product spectrum is displayed where several of the wall effects are exhibited.

Another common gas neutron detector with very similar operation is filled with a $^{10}\text{BF}_3$ gas mixture. Similar to He-3, the $^{10}\text{BF}_3$ spectrum is not reflective of the initial neutron energy, but rather the various reactions products that will contribute to various edges and peaks in the spectrum due to wall and end effects. $^{10}\text{BF}_3$ will however have up to four contribute to the wall effect in the spectrum due to the branching ratios of the fission reaction as shown in Figure 2-8. Some $^{10}\text{BF}_3$ tubes utilize thin window entrances at one end, improving the absorption efficiency, however many $^{10}\text{BF}_3$ tubes instead use large end caps with insulating material that severely contribute to the end effect taking place. Therefore, there does exist some directionality dependence for some $^{10}\text{BF}_3$ tubes. Like He-3 tubes, $^{10}\text{BF}_3$ tubes optimize pressure to reach the maximum detection efficiency, though $^{10}\text{BF}_3$ tubes typically have intrinsic efficiencies lower than He-3 [34]. $^{10}\text{BF}_3$ typically operates around 500 to 3000 V, depending on diameter. Argon is sometimes added to $^{10}\text{BF}_3$ mixtures to operate in a proportional mode in which the resolution of the spectral features is improved, but the detector is limited to a count rate of less than $2\text{E}+04$ cps because of pulse pile up. Researchers have also reported $^{10}\text{BF}_3$ exhaustion and degraded

performance over time [35]. $^{10}\text{BF}_3$ is also a level three health hazard according to the National Fire Protection Association, therefore pressure and temperature are a concern and therefore somewhat limits the applicability.

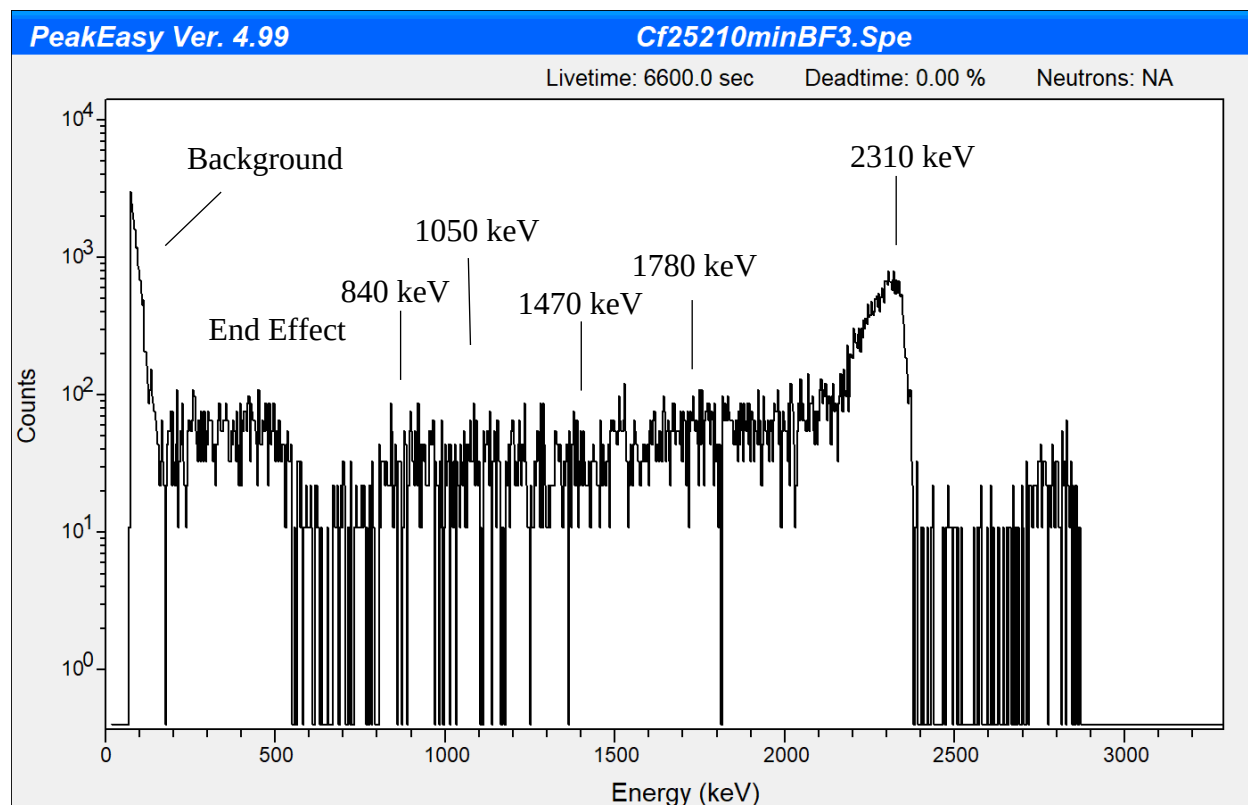


Figure 2-8 A $^{10}\text{BF}_3$ fission reaction product spectrum is shown where all energies and features of the detector are identified.

Perhaps the first type of gas-filled slow neutron detector was coated with B-10 [36]. Boron-lined detectors operate as ion chambers and can use practically any counting gas. Despite the intrinsic thermal neutron detection efficiency of the B-10 lined gas detectors being limited to approximately 9% due to self-attenuation, the B-10 lined tubes are desirable gas detectors for higher temperature operation [37]. Additionally, when operated in current mode, the B-10 lined detector may be used in high radiation environments, such as near a nuclear reactor [38].

2.1.3.2 Scintillator Neutron Detectors

Scintillator detectors, which will be covered in greater detail in a later chapter, are materials that emit light when a radiation event interacts within the material. Upon the interaction, an electron is excited to a higher energy band. Eventually the electron de-excites,

emitting a photon corresponding to the electron's excited state. The secondary photon may be processed through a photomultiplier tube (PMT) which amplifies the signal to then be passed along with the accompanying electronics to produce an energy spectrum. Scintillators may come in an organic and inorganic version and be in a crystal, powder, liquid, plastics, or even ceramic form. Of those that are used for neutron detection can be broken down into two groups: those composed of a single compound that interacts with neutrons and fluoresces and those that are comprised of two compounds with one fluoresces and the other absorbing neutrons for detection. Elpasolite neutron reactive scintillators such as $\text{Cs}_2\text{LiYCl}_6$ (CLYC) and Cs_2LiYBr (CLYB) offer a unique advantage in some cases to be sensitive to both gamma-rays and neutrons. Pulse height discrimination techniques allow for the signal contributions to be sorted and correctly counted. However, in high radiation fields, the ability to discriminate between differing radiation event types becomes challenging because pulse pile up contaminates the neutron induced spectrum [39]. While the elpasolites offer the convenience of dual detection, neither performance is more efficient than a standard compact neutron detectorⁱ, nor does it produce better energy resolution than other standard scintillators such as LaBr_3 . Additionally, because of the addition of Cs-133 and Cl-35, which are both heavily naturally abundant, beta contamination resulting from the transmutation of the two isotopes into beta emitting isotopes upon the absorption of a neutron occurs. Not only does the contribution of the Beta particles degrade the energy resolution and can add significant noise to the gamma-ray spectrum, but the neutron events that interact with the parasitic absorbers are missed and the detection efficiency suffers.

As mentioned, some scintillators are loaded with neutron reactive material in which the neutron reactive material is mixed into the scintillating fluor. In order to maintain the scintillator's transparency whether in a plastic or liquid, the weight percentage of neutron reactants such as Li or B are typically limited to less than 5% [40].

2.1.3.3 Semiconductor Neutron Detectors

Semiconductors thermal neutrons have certainly grown in popularity in recent years with the microstructured semiconductor neutron detectors (MSND) being launched in a variety of platforms such as the Domino in 2013. The MSND, a derivative of the coated planar semiconductor diode neutron detector, utilizes a neutron conversion such as ^6LiF or B-10 on the

ⁱ CLYC:Ce is limited in intrinsic thermal neutron detection efficiency to only 22%.

surface of the semiconductor silicon substrate like the planar device. However, in addition to placing a thin film, the MSND goes a step further by utilizing deeply etched trenches into the silicon substrate that are also then filled with the neutron conversion material such as ^6LiF as shown in Figure 2-9. Not only is the likelihood of a neutron absorption increased from 11% to 74% with the implementation of trenches, but the overall intrinsic neutron detection efficiency also increases to nearly 52% due to the ability to capture both the alpha particle and triton reaction products within the trench fins. Once the reaction product reaches the silicon, electron-hole pairs are generated and a voltage applied to the contacts on the device drifts the charge carriers to the detector electronics, creating a measurable pulse corresponding to the charge particle energy.

Figure 2-9 below displays the ^6LiF reaction product spectrum for a device exposed to a 12,132 ng/s moderated ^{252}Cf neutron source. Several MSNDs comprise the commercially available Domino[®] detector that can tie several MSNDs together in a tiled format, allowing all devices to contribute to the digital output per module. The MSNDs are compact (less than 1cm^2), quite rugged, inexpensive, and require less than 2 V to operate at peak performance. The MSND devices may be arrays to match the form factor of a standard He-3 gas filled tube. The array has the ability to match the performance of a typical 6 atm He-3 tube of the same diameter as the MSND array at a fraction of the cost. MSNDs also have demonstrated the ability to detect fast neutrons with the use of a HDPE block [41]. Several other microstructure designs have been considered such as a pillars design instead of trenches as well as other neutron conversion materials such as B-10 [42]. The design variations have yielded shorter reaction product ranges in some cases as well as reduced charge collection, which have led to reduced thermal neutron detection efficiencies of only 32%.

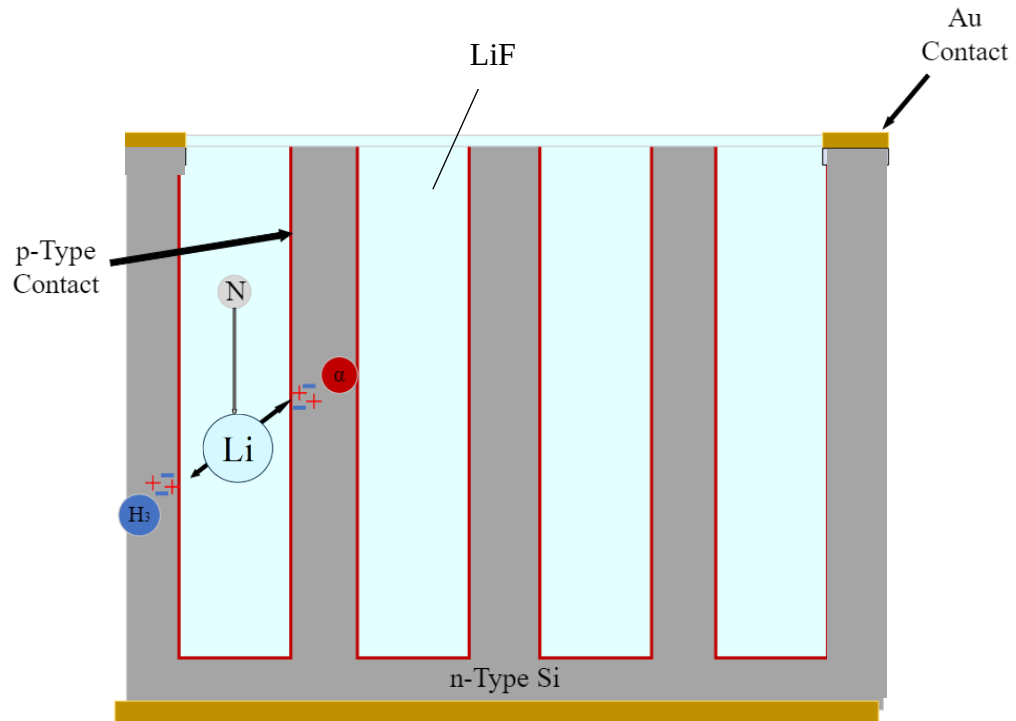


Figure 2-10 A profile view of a thermal neutron fission reaction taking place within a LiF backfilled trench is shown, The alpha and tritium particle each reach the silicon in the above scenario and proceed to ionize the material.

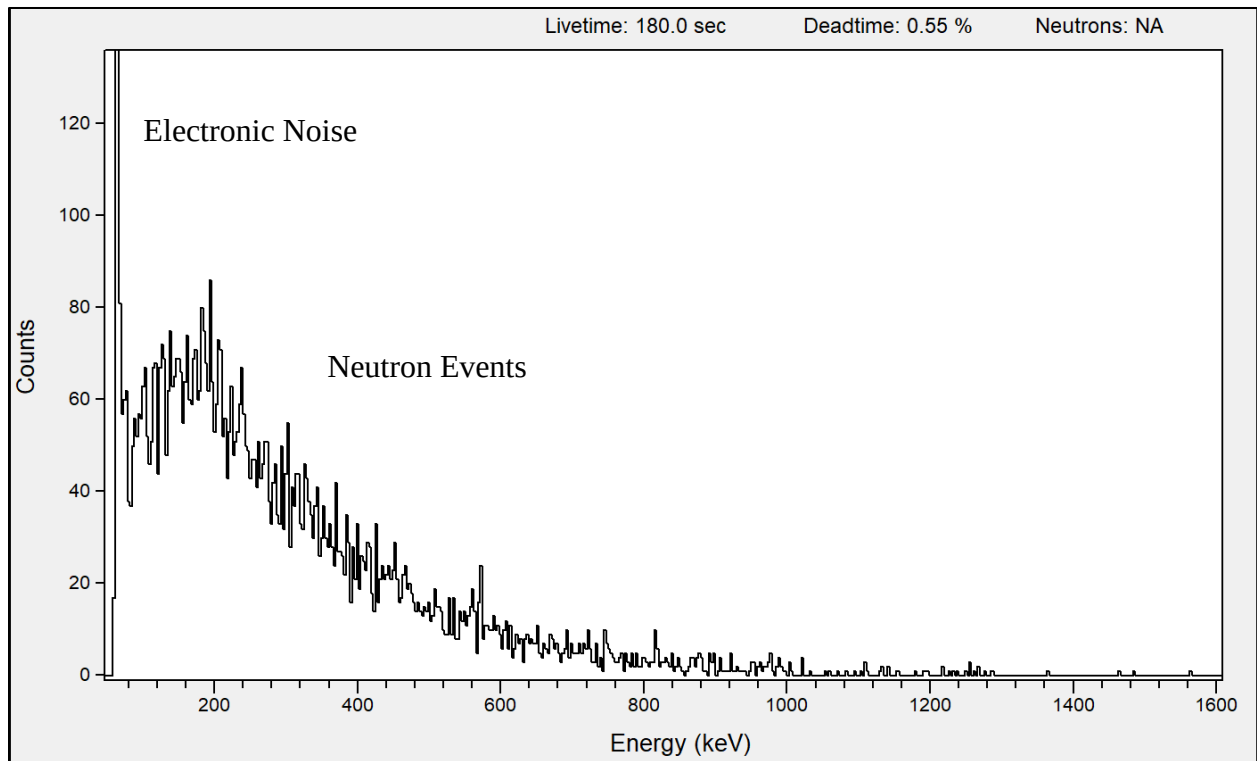


Figure 2-11 An example MSND thermal neutron fission reaction product spectrum is shown. The contribution of the reaction products is noticeably separated from the noise of the spectrum.

In the past few years, modifications have been made to increase the intrinsic thermal-detection efficiency up to almost 70% by mirroring and offsetting a second set of trenches on the back of the device to eliminate any free streaming neutrons as shown in Figure 2-11. The device known as the dual sided microstructure semiconductor neutron detector (DS-MSND), is the device utilized in the ARGANOT device and will be discussed in greater detail in the following chapter.

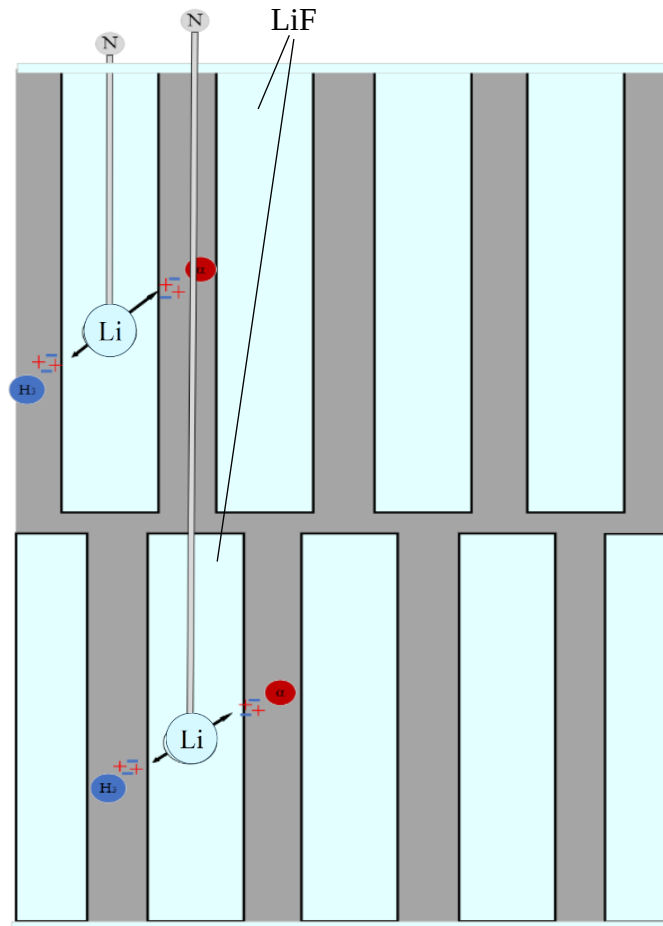
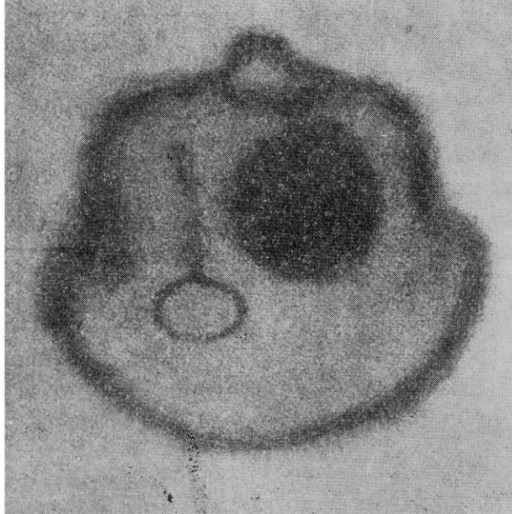


Figure 2-12 A DS-MSND device is shown in which a free streaming neutrons is successfully captured in the backside set of trenches.

3 Introduction Gamma-Ray Detection



Gamma radiograph of a porte-monnaie (purse) with coin and key by Marie Curie [43].

“I think that this effect is due to the presence of non-déviante rays, which are less absorbent than the ones [alpha rays] that have been described by Mr. Curie. . . . It follows from the facts presented above that the non-déviante rays emitted by radium contain some very penetrating radiations, capable of traversing metal foils and affecting a photographic plate.” [44]

~ Paul Villard, 1900

Gamma-ray detection follows a similar method to neutron detection in that gamma-ray detection is mostly based on indirect ionization [45]. The following section will discuss gamma-ray interactions, gamma-ray spectroscopy, evaluation techniques for gamma-ray detection, and an introduction to common gamma-ray detectors.

3.1 Interactions

Gamma-ray can interact in matter through many methods. However, it is usually photoelectric effect, Compton scattering, and pair production that are important to radiation detection. Each interaction method has an associated probability of taking place within a material. The probability is dependent on a number of factors including the energy of incident gamma-ray, the matter composition, and density of the interacting medium as described by Equation 3-1 where number of interactions over thickness x , with the initial number of gamma-rays being I , and μ_t being the total linear interaction coefficient. Shown in Figure 3-1 are the probabilities of gamma-rays of various energies of interacting through these three methods in

various materials. As shown in the NIST figures, gamma-ray photons, at lower energies typically interact through the photoelectric effect. The photoelectric effect takes place when a photon interacts with bound electrons, usually in the k shell, resulting in the emission of a photoelectron. The photoelectron carries the residual kinetic energy T_{pe} , described by Equation 3-2 is a result of the initial gamma-ray energy, E_γ , minus the electron binding energy, E_b .

$$T_{pe} = E_\gamma - E_b \quad (3-2)$$

$$I(x) = I_0 e^{-x\mu_t} \quad (3-1)$$

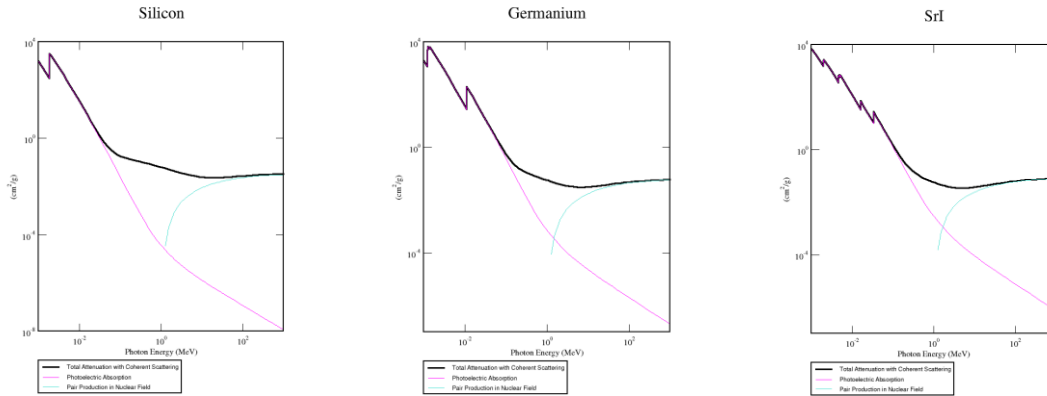


Figure 3-1 The interaction probability for various radiation detectors over a range of energy is shown. (NIST XCOMM)

The photoelectron may then be processed by the detector electronics in a similar way to the directly ionizing radiation that may be measured, contributing to what is referred to as the photopeak at the energy proportional to the impinging gamma-ray. In the example HPGe Co-60 gamma-ray spectrum show in Figure 3-2, two prevalent full energy peaks labelled as ‘Co-60’ are shown, thereby indicating that both the 1173 KeV and 1332 KeV photons interacted through a photoelectric absorption and the full energy of the photons were collected and measured by the detector. Although it is not visible on the spectrum below; it is also possible for smaller x-ray peaks to appear just below the full energy peak due to the emission of a K shell electron. The small energy of the shell electron is then removed from the full energy peak, sometime allowing

for a much smaller peak to appear at the energy of the impinging gamma-ray minus the shell electron.

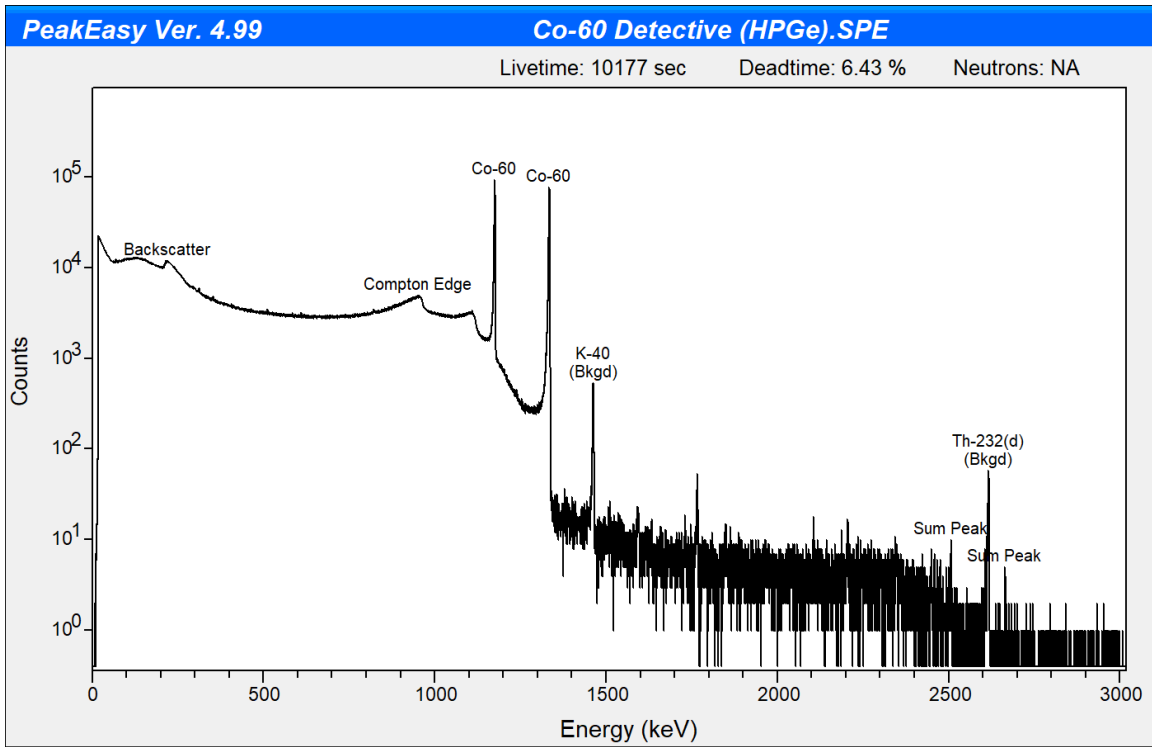


Figure 3-2 An example Co-60 gamma-ray spectrum collected with a HPGe detector is displayed. The various features of the spectrum are indicated. The peak labeled as the Th-232 is a combination of Thorium daughter products, namely Tl-208 with minor contributions from Po-212.

To the left of the photopeak, two Compton edges created by Compton scattering appear to the left of each full energy peak and continue into the lower energy of the spectrum. In the Compton scattering process, a photon is scattered by bound electron. The interaction results in a scattered photon with a lower energy that if lost, may then go on to interact within a target medium. The energy of scattered photon, E' is highly dependent on the angle θ_s for which the initial photon with energy E interacts, as represented by Equation 3-3 where m_e is the mass of an electron and c is the speed of light.

$$E' = \frac{E}{1 + \left(\frac{E}{m_e c^2}\right)(1 - \cos \theta_s)} \quad (3-3)$$

Therefore, as indicated by Equation 3-3, the scattered photons will possess a wide range of energies throughout the down scattering process, which contributes to the prevalence of Compton edge to the left of each full energy peak. Adding to the scattering contribution is backscatter. When a photon interacts with the surround shielding of a measurement, often lead for example, it is possible that the resulting scattered photon is scattered back into the detector, thereby contributing to the backscatter peak. All contributions of Compton scattering within the detector may be collected and when summed. If all energy from the initial gamma ray is re-absorbed, then an equivalent current generated with the photoelectric interaction will be collected. In such case, the photopeak may now incorporate counts from interaction types aside from photoelectric effect and should thus be referred to as a full energy peak when considering all counts tallied in the peak.

The last method of gamma-ray interaction, pair production is a bit more complex in that an incident photon is completely absorbed by the inner most potential of the atom. Energy is converted to mass as the atom emits a positron-electron pair. The positron and electron ionize the detector medium with the positron eventually colliding with an electron and annihilating, creating two 511 keV annihilation photons that are emitting in opposite directions. The two 511 keV annihilation photons may go on to interact by either Compton scattering or photoelectric effect within the detector, and will contribute to the full energy peak. The threshold of original photon energy needed for pair production to take place is 1.02 MeV or greater, making pair production a generally higher energy process, increasing with gamma ray energy. In the HPGe Co-60 gamma-ray spectrum shown in Figure 3-2, the ‘single escape’ and ‘double escape’ peaks represent the detectors’ inability to successfully capture and measure either one (single escape) or both (double escape) of the 511 keV annihilation photons. Therefore, the ‘double escape’ peak should appear exactly 1022 keV less than the full energy peak whereas the ‘single escape’ peak should appear exactly 511 keV less than the full energy peak. Consequently, because of the nature of how the escape peaks are formed, a Compton edge does not appear for escape peaks in the spectrum, although it should be noted that source energies below 1.5 MeV are unlikely to contribute to large escape peak presence [46]. However, as the probability of a pair production event taking place increases, the appearance of escape peaks will also grow, which may in turn assist in identifying the original gamma-ray energy.

In addition to the various peaks directly produced as a result of detector interactions, artificial peaks may also appear in the gamma-ray spectrum. Artificial peaks (sum peaks) are the results of two or more photons interacting within the resolving time of the detector. The current produced from both photon events is then read simultaneously, resulting in the current being summed together. The larger current therefore produced a true sum peak within the spectrum that may be mislabeled as an unknown energy peak. In some instances, the processing time of the detector is long enough to where two gamma-rays are processed within the same short amount of time, resulting in an accidental coincidence sum peak. Identifying these artificial peaks may be tricky considering they often times will be accompanied by Compton edges and look to be a genuine source feature. However, by simply determining if any combination of the full energy peaks add up to the energy of unknown peak, the sum peak can be identified. Therefore, artificial peaks are especially common with poly-energetic sources, such as Co-60. One may further distinguish a true sum peak from the accidental sum peak by simplifying decreasing the solid angle of the source to the detector. In increasing the distance between the source and the detector, fewer accidental coincidence events should take place due to the decrease count rate, thereby leaving the spectrum with a majority of true sum peaks.

3.2 Gamma-Ray Spectroscopy Evaluation

There are several metrics to characterize a gamma-ray spectrometer's performance such as the detection efficiency, energy resolution, noise resolution, and peak to Compton ratios. When selecting a detector for gamma-ray detection, all of the performance parameters should be considered in order to answer the key question(s) of interest. The following parameters of gamma-ray detectors evaluation are presented to aid in the detector characterization process.

3.2.1 Detection Efficiency

Detection efficiency may be measured or defined in several different ways. The useful efficiencies for gamma-ray spectroscopy are the total intrinsic detection efficiency, the intrinsic peak intrinsic, absolute, and escape peak efficiency. As discussed in the section on neutron detector evaluation parameters, intrinsic efficiency ϵ_I is the ratio of the detector response either

as a net count rate or net total countsⁱⁱ to the particle emission or emission rate. The intrinsic efficiency once again presented for the reader in Equation 3-4 below considers the effect of geometry on the detector's response by dividing the ratio by the solid angle and thereby reducing the reliance of geometry in the detector's reported performance. The solid angle, Ω , represents the probability that emitted radiation intersects the detector's entrance aperture, while A_{sp} represents the total counts in the entire spectrum, A is the source activity, t is the measurement livetime, and B_i is the branching ratio.

$$\epsilon_I = \sum_i^N \frac{A_{sp}}{A\Omega_f t B_i} \quad (3-4)$$

Intrinsic peak efficiency is similar to the total intrinsic efficiency described above with the difference being that instead of considering the entire spectrum's counts, A_{sp} represents the counts under a single peak of interest and becomes A_p in Equation 3-5.

$$\epsilon_I = \frac{A_p}{A\Omega_f t B} \quad (3-5)$$

Because detectors often have encapsulation or equipment blocking the line of sight to the detector itself, the peak efficiency method is perhaps more reliable to describe the detector's efficiency across a variety of energies. Therefore, it is common to use a mixed gamma-ray calibration source spanning a variety of energies to collect the peak intrinsic efficiency of the detector. The many data points collected may then be plotted to create an efficiency verses gamma-ray energy curve, which may be then fitted with a curve and then with an equation to represent the curve allowing the efficiency to be determined at any given energy within the calibrated energy range.

The escape peak efficiency is used to measure the detector's ability to effectively recapture the 511 keV escape peak from the pair production interactions. If a large amount of 511 keV photons escape the detector, a large escape peak will be present in the spectrum and

ⁱⁱ The net count or count rate refers to the background measurement or the measurement of the nature background when no source is present subtracted from the foreground, or the measurement with the source present. Stripping the background from the foreground should therefore remove any indigenous peaks from the measurement used in the final analysis.

therefore indicate that a larger loss of impinging events of interest may be occurring. The intrinsic escape peak efficiency can be found, as described in McGregor & Shultis and others such as Cline 1968 and Nafee 2011, using Equation 3-6 where A_{ep} represents the number of counts in escape peak of interest after stripping the background from the foreground measurement of interest, A represents the source activity, Ω_f represents the solid angle of the measurement, t represents the time of the measurement, and B represents the source branching ratio [47].

$$\epsilon_I = \frac{A_{ep}}{A\Omega_f t B} \quad (3-6)$$

3.2.2 Energy Resolution

Perhaps the main utility of gamma-ray spectroscopy is the identification of isotopes based on the collected energy of the gamma-rays impinging on the detector. Therefore, if the purpose of using a gamma-ray detector is to identify the radiation present, broad peaks within the gamma-ray spectrum make determining the impinging energy difficult due to the large number of energy possibilities to match the unknown source. Therefore, semiconductors such as HPGe are often the detectors of choice due to their superior energy resolution.

In gamma-ray spectroscopy, the pulse generated from the collected current in the device resulting from energy (E) deposited in the detector will vary. The function relationship between the initial energy deposited and the pulse height collected is represented by f . Due to the variation in the current collection, the resulting pulse in the gamma-ray spectrum will appear within a distribution. The peak will be distributed about a mean value $h_0 = f(E_d)$ by some kernel $g(h|E_d)$ where $g(h|E_d)dh$ is the probability an event that deposits energy E_d in the detector with a variation dh about h . The resulting gamma-ray spectroscopy peak can be appropriately described as gaussian, however due to the variation in charge collection of the device, the peak will usually be skewed slightly to lower energies.

Energy resolution therefore is used to describe the degree of variation in the pulse height and therefore the gaussian spread of a peak in the gamma-ray spectrum. The energy resolution is described by the Full Width Half Maximum (FWHM) in keV and may be communicated in either a percent or energy units and may be found using Equation 3-7 where σ represents the standard deviation [48].

$$FWHM = 2\sqrt{2\ln 2}\sigma \cong 2.355\sigma \quad (3-7)$$

The number of channels also affects the ability to accurately perform spectroscopic identification of sources. The height of each pulse collected from the device correlates with a specific energy bin. The range of the energy bin is directly related to the number of channels which is a binary system that may be subdivided with 2^N over a 10 V range. In high resolution gamma-ray spectroscopy, systems typically operate with either 2^{13} or 2^{14} (8,192 or 16,384) channels. Therefore, having an increased number of channels allows for small energy bins across the spectrum and therefore more precise energy assignment.

3.2.3 Peak Ratios

Another common metric for radiation spectrometers is to determine the peak-to-Compton ratio (PCR) which can be a sort of signal to noise ratio as the PCR is conveying the detector's ability to discern lower energy and lower count rates in the vicinity of larger peaks and the accompany spectral effects. The PCR is determined by examining a Co-60 gamma-ray spectrum and is defined as Equation 3-8 where C_{max} is the number of counts in the peak channel of the 1332.5 keV peak. $\bar{N}$ is the average number of counts per channel between the channels representing 1040-1096 keV, which will fall in the 1173.2 keV Compton gap, leaving most counts attributed to the Compton region from the 1173.2 keV gamma-ray [49]. With improved energy resolution and size of the detector, the counts under the 1332.5 keV peak should increase resulting in a large PCR [50].

$$PCR = \frac{C_{max}}{\bar{N}} \quad (3-8)$$

A metric used most commonly for compound semiconductor detectors is the peak-to-valley ratio (PVR) and is found with a ^{137}Cs spectrum and is defined as the number of counts in the peak 662 keV channel divided by the number of counts in the middle of the Compton gap at 569 keV. The last common ratio method is the peak-to-total (PTR) which is the ratio of the total counts in the full energy peak to the number of counts in the total spectrum. The PTR is proportional to the intrinsic full energy peak efficiency, ϵ_I as describe as Equation 3-9.

$$\epsilon_{peak} = \epsilon_I \frac{\text{peak counts}}{\text{total counts}} = \epsilon_I PTR \quad (3-9)$$

3.2.4 Background in Gamma-Ray Spectra

Following the discussion of how various peaks appeared in the HPGe Co-60 spectrum shown above, only two peaks in the spectrum remain unexplained: The peaks labels as K-40 and Th-232. As mentioned, the spectrum shown was for a Co-60 source but the accompanying unwelcomed contamination of K-40 and Th-232 and its most likely daughter, Ti-208 were hard to evade as they are primordial isotopes, see Figure 3-3, often present in building materials such as concrete and limestone and are difficult to shield against during measurements [51].

Pb (lead) is an ideal material to shield high levels of gamma-ray and charged-particle radiation to eliminate background radiation from entering highly sensitive gamma-ray radiation detectors for low-level measurements. Highly sensitive detectors require operation in extremely low background environments to produce measurement results down to the picocurie of accuracy and require significant shielding to produce pure radiological spectrums. Previous research has indicated that experimentalists must be careful when constructing low background environments after revealing that Pb material commonly used in radiological measurement is tainted with undesired radiological isotopes. Undesired radiological isotope presence, referred to simply as radiological contamination, arises from four sources: uranium decay to the Pb isotope, cosmic radiation activating the Pb, purposeful addition of radiological impurities, and even radiological fallout.

In a 1991 paper, Alessandrello et al. reported that spectra collected from modern Pb contained a considerable number of radioactive isotopes such as Pb-210 and Ra-226. Additionally, when examining Pb recovered from shipwrecks that had resided at the bottom of the ocean for centuries, gamma-ray spectra of that material revealed that nearly all contaminants were limited to only 0.19 ± 0.03 counts per hour compared to 8.8 ± 0.4 counts per hour observed with modern Pb material [52].

Similarly, Smith et al. in 2008 examined contamination of Pb spanning different time periods (1700 to 1960) and continents (Europe and Australia). Findings revealed that while certain regions of Pb mining and production may be less contaminated than others, all Pb examined contained contamination from uranium, thorium, and their daughter products [53].

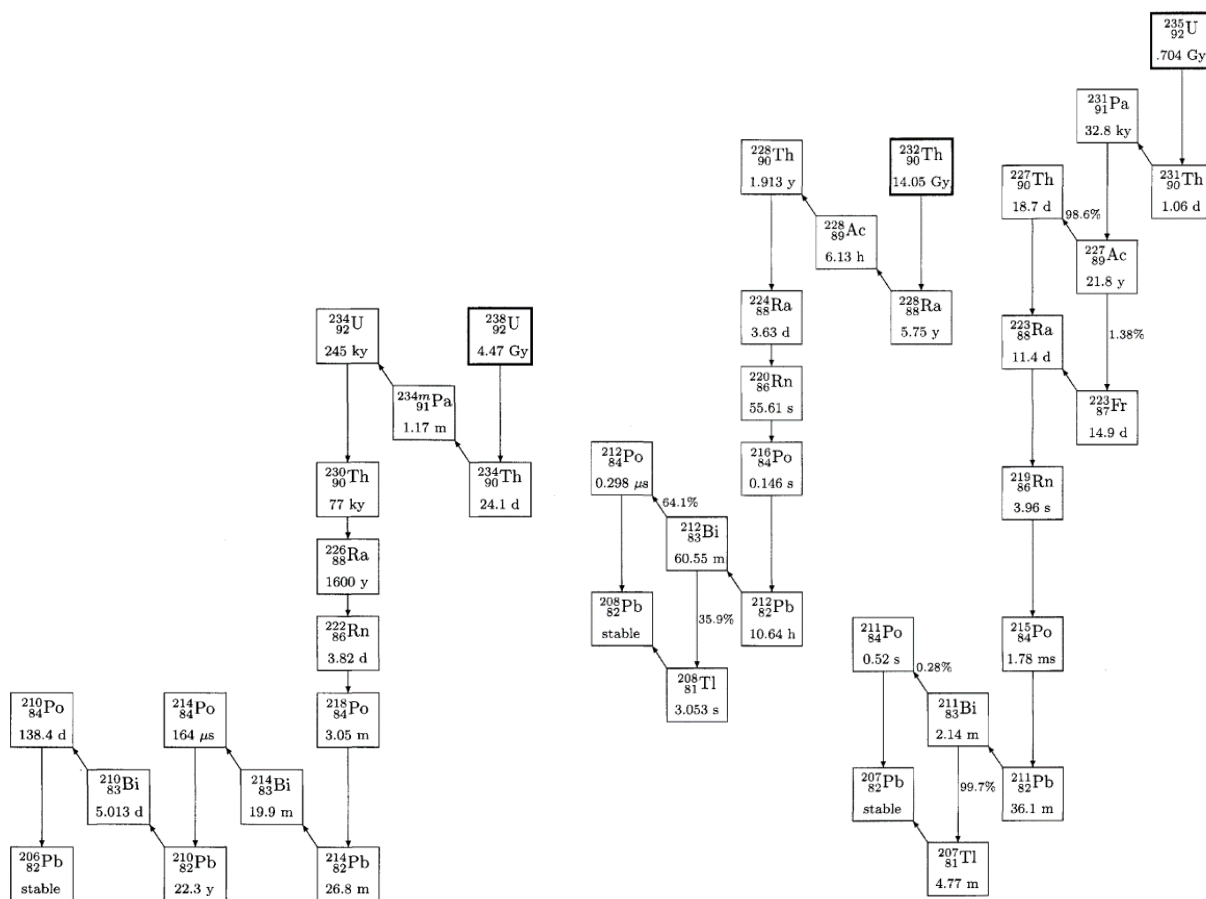


Figure 3-3 The decay chain of two common primordial isotopes is shown. Po-210 is a possible source of contamination in spectrum, however Ti-208 is the more probable source.

Alessandrello accredited the lack of contamination in shipped wreck Pb to the ocean's shielding of any cosmic radiation that could have activated the Pb, a uranium daughter product. Additionally, Smith results supported Alessandrello in exhibiting that the regional uranium content from which the Pb originated from and regional cosmic weather both impact the overall activity of Pb [54].

Others however speculate that impurities purposefully included in production and manufacturing hold some responsibility for the radiological contamination. One such example is the addition of impurities such as potash for chemical separation in the smelting process as well as the lesser explored culprit, radiological fallout from the nuclear era [55]. Radiological fallout is the product of a nuclear denotation in which activation products, the environment, and even the original radiological material is dispersed across an area and distributed back to earth and likely contaminated Pb material initially through surface contamination of lead mines around the world [56]. Voelz et al. published work examining fallout distributions patterns concluding that

“fallout would trace the pattern of rainfall, such that the California-Arizona region had the lowest level of fallout, whereas the Northeast and Northwest had the highest.⁵⁷ Often such scenarios result in vast fallout dispersal ranges as fallout will gradually enter the lower atmosphere and the particles settle to the earth with precipitation or are wafted with the winds leading to contamination due to fallout of lead material likely varied with mine and lead production smelting facility location.

Additionally, these detonations consequently caused terrestrial and aerial contamination that ultimately coated materials such as Pb during fabrication in the post-war world. To allow researchers to quantify the innate activity whether from nuclear fallout or contamination from other means in shielding materials such as Pb, researchers may employ simulations to determine the activity of an empirical unknown source measurement, see Figure 3-3 and Figure 3-4. In one instance Pb brick appeared to be contaminated with both Co-60 and ¹³⁷ Cs and an activity analysis was performed to determine the suitability of the shielding material for measurements. Comparing simulated and empirically collected data, as reflected in Figure 3-6, resulting simulated spectrum was then scaled to match the activity of the Pb contaminated brick as shown in where the empirical spectrum is represented by black and the simulated is represented by the blue. The analysis yielded an estimated activity for the Co-60 of approximately 1.0458 ± 0.0095 pCi and 2.923 ± 0.053 pCi of ¹³⁷ Cs contaminated the Pb brick. The activity analysis process described here may be implemented in a variety of scenarios to determine the material contamination. Through following the steps used above, one can ensure that the shielding materials selected for lower background measurements are not contaminated due to isotopes product of the nuclear weapons era, cosmic radiation activation, or other production means.



Figure 3-4 The suspected contaminated Pb brick was place over top of the HPGe detector head and a long measurement was conducted.

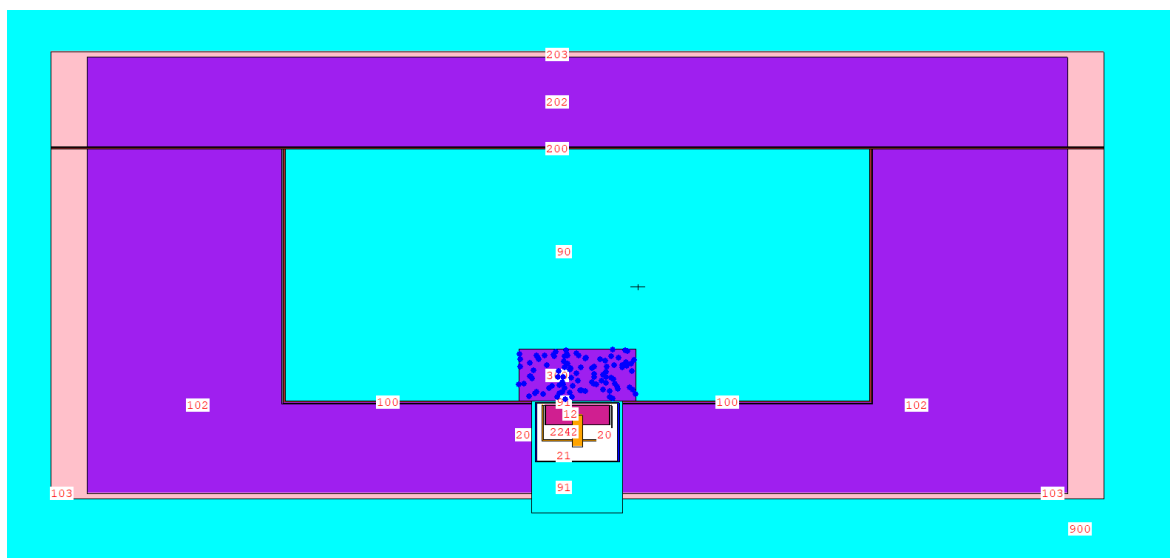


Figure 3-5 A MCNP simulated HPGe detector was designed to emulate the HPGe detector shown in figure 18. The Pb brick on intestest was then placed above the detector head.

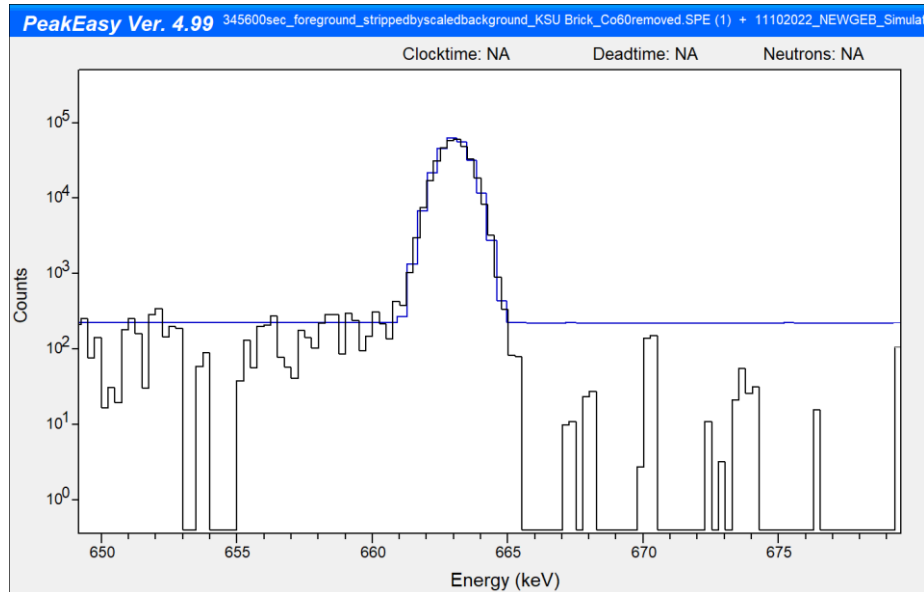


Figure 3-6 The simulated HPGe detector spectrum response to a contaminated Pb brick (blue) was scaled to match the empirical data collected with a real life HPGe detector.

3.3 Gamma-Ray Detector Types

3.3.1 Scintillations Gamma-Ray Detectors

Scintillator detectors, which were introduced earlier in the neutron discussion are also very common gamma-ray detectors. While the operation is quite similar to the neutron sensitive scintillators described earlier, the desired material to detect gamma rays is instead a high Z number material with high density. One of the most common inorganic scintillator materials is sodium iodide (NaI) that often contains small amounts of thallium (Tl) to improve the population of excited electrons and excitons [58]. NaI:Tl produces approximately 43,000 photons per MeV from absorbed gamma-rays with an average wavelength of 415nm, which perfectly aligns with most common bialkali photomultiplier tubes (PMT) [59]. NaI:Tl has a decay time of 230ns, which is relatively long compared to other scintillators [60]. NaI:Tl, for which Moszyński reported a FWHM of $5.2 \pm 0.35\%$ at 662 keV, does not require cooling, making them an ideal candidate for field operation when properly hermetically sealed [61].

In addition to Na:Tl, other common scintillators for gamma-ray detection are cesium iodine (CsI:Tl) and lanthanum bromide with small amounts of cerium (LaBr₃:Ce). CsI:Tl has a reported light yield of 54,000 to 61,000 photon/MeV and is much more durable than NaI [62]. CsI:Tl may even be machined in some cases with less fracturing. In addition, CsI:Tl has a

gamma-ray attenuation coefficient that is higher than that of NaI, allowing CsI:Tl's intrinsic detection efficiency per unit volume to be greater than that of NaI with an energy resolution of approximately 8.3% at 511 keV [63]. LaBr₃:Ce while being quite hygroscopic, has an excellent reported energy resolution below 2.9% at 662 keV for various crystal sizes [64]. Along with possessing great gamma-ray absorption efficiency, LaBr₃:Ce also has a decay constant of 16 ns, allowing LaBr₃:Ce to be a great candidate for high flux measurements.

While scintillators are excellent detectors for mobile radiation detection, high resolution detectors are often needed for precision measurement. To achieve high resolution, semiconductors are often employed.

3.3.2 Semiconductor Gamma-ray detectors

By far the most common semiconductor gamma-ray spectrometer is high purity germanium (HPGe) which developed from the 1960 germanium-lithium drifted (GeLi) detector [65]. HPGe has a narrow bandgap of only 0.66eV at room temperature, which allows a great deal of charge to be created per radiation event. However, the small bandgap also allows for excessive thermal excitation which contributes to a significant leakage current. To ameliorate the thermal noise, HPGe detection of electron-hole pairs require liquid nitrogen cooling to achieve the energy resolution as low as 1.8 keV for an n-type coaxial detector at 1332 keV [66].

Upon the interaction of a photon within a semiconductor, ionization of an electron takes place, and the electron will have an increased amount of energy [67]. If the imparted energy is great enough, a secondary electron can be excited to a higher energy band within the atom. Upon the electrons elevating from the valence band to the conduction band, an empty state is left behind. The absence of the negative electron charge is therefore referred to as a 'hole' within the valence band. The hole is modeled to have a positive charge, opposite of the missing electron. A voltage can then be applied to 'drift' the charges to either a positive or negative electrode for collection. Within semiconductor radiation detection, the extraordinary process through which electron hole pairs are created within semiconductor material and then collected can then be used to back out the incident radiation energy. This can be done through first considering Equation 3-10 in which a number of photons (I) with energy (E) can interact within the length of the material (x) with a predefined likelihood, G . Equation 3-11 can be used to determine the total amount of electron hole pairs created for the amount of energy (E_γ) deposited into the detector.

Equation 3-12 may then be used to approximate the amount of charge collected from the device where q is the charge of an electron.

$$E_\gamma = I(x)G \quad (3-10)$$

$$\#E/Hpairs = \frac{E_\gamma}{E_{ionization}} \quad (3-11)$$

$$Q_0 = q \frac{E_\gamma}{E_{ionization}} \quad (3-12)$$

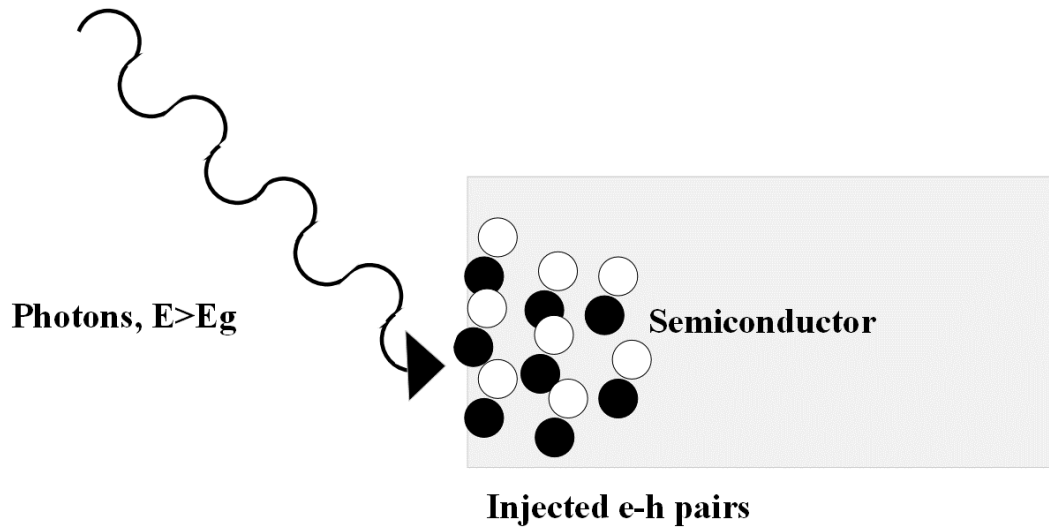


Figure 3-7 A gamma-ray is shown interacting in a semiconductor, producing electron hole pairs. As the charge carriers move towards their respective electrode, they induce a mirror charge. The induced charged, see Equation 3-13, is characterized by the charge collection efficiency (CCE) electric field at position of interaction, x within the device. The charge collection efficiency is a property of the devices with indicates the energy resolution and photopeak efficiency [68].

$$Q(x) = Q_0\varepsilon(x) \quad (3-13)$$

A charge sensitive amplifier then converts the induced signal to a proportional voltage to its charge sensitivity (s) or rather the inverse of the feedback capacitance. The pre-amplifier signal is then fed to an amplifier when a gain, G, is applied along with shaping. The signal is then passed to a multi-channel analyzer (MCA). The MCA input voltage, V_A is found through Equation 3-14.

$$V_A = QsG \quad (3-14)$$

The peak voltage signal is then binned into channels based on the corresponding peak height. The voltage channel relation can typically be represented by a linear equation, as shown in Equation 3-15. Channel(x) represents the channel number in which the count from an interaction at location x is binned, and m and b are the slope and intercept of the regressed line. A substitution of the previous equations may be made, resulting in Equation 3-16 and the CCE may then be found. The CCE can be predicted from a variety of factors including the electric field, the weighting potential, and the charge carrier transport properties of the device [69].

$$\varepsilon(x) = \frac{(Chn(x) - b)w}{msGqE_\gamma} \quad (3-15)$$

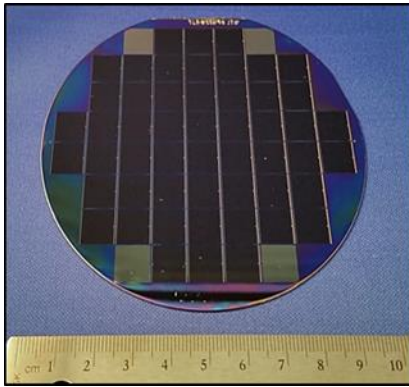
$$Channel(x) = mV_A(x) + b \quad (3-16)$$

3.3.3 Gas Filled detectors.

One of the easiest to operate and cost-effective detector options is a Geiger Mueller (GM) counter. The GM tube operates in a very high voltage region—typically around 500-2000V. The normally cylindrical in shape coaxial detector chamber is often filled with a Noble gas that is then contained with an insulating cap and an anode wire traveling the inner and a cathode contact traveling along the wall of the detector volume. While the GM tube may be used to detect all types of radiation and produce the same response regardless of energy, GM tubes are especially common for gamma-ray detection and are fairly straightforward counters.

As impinging radiation enters the detector volume, it will likely interact, creating an electron-ion pair. The electrons will rapidly drift towards the anode wire due to the high electrical field created from the high operating voltage. As the electrons reach closer to the anode wire, the electric field is so great that the electrons, through collisions, begin to liberate additional electrons from neighboring atoms, resulting in what is described as an “avalanche” of electrons. As result, a large dense number of electrons gather near the anode wire and will start to lose energy through recombining or deexciting emitting a photon. The photon will continue to contribute towards the avalanching, creating more electrons along with a large collection of positively charged ions. The ions gather near the anode and begin to reduce the electric field, eventually leading to a flip in polarity. The avalanching effect is no longer sustainable, and the ions begin to drift towards the cathode. The detector electronics are then able to use the ions to produce a signal pulse. As the ions are removed, the electric field about the anode wire begins to recover and the interactions begin once more. Due to the mode of operation, the particle type and energy cannot be determined with GM tube. Despite GM counters’ inability to distinguish between radiation types, beta particle and alpha particle may be selectively detected through the use of window filter on the outside of the detector face. Additionally, while the GM tube may not indicate the energy of the initial impinging radiation, they serve well for simply alerting to and roughly determining the source activity.

4 The Dual-Sided Microstructure Semiconductor Neutron Detector



DS-MSND Wafer picture captured at the Semiconductor Material and Radiological Technology Laboratory at Kansas State University

“It was at Bell Labs that I first made direct contact with real semiconductor experts and thus began to fully understand what amazing materials they were and what they could do.”

~ Robert B. Laughlin, Noble Prize in Physics, 1998

The Microstructured Semiconductor Neutron Detector (MSND) of which the Dual-Sided MSND (DS-MSND) is derived from, first evolved from a simple thin film device with minimal efficiency, but unlimited potential. The thin film device was a simple semiconductor substrate coated with a thin film of a neutron conversion material, such Boron-10 or LiF—leading to the name, “thin film device” [70]. When neutrons came into contact with the neutron conversion material coated surface of the device, neutrons would interact with a fission reaction, creating fission products. The fission products would then go onto ionize the semiconductor substrate if they were able to reach it. The thin film devices were then equipped with a read-out contact allowing any current produced from fission products to be converted into a readable output pulse, indicating the presence of neutrons [71].

While the original thin film device was quite cleverly designed, they suffered from obvious weaknesses. For one, the device was only designed to support a thin film layer of the neutron conversion material, making the likelihood of neutron interaction quite low. Unfortunately, the original designers were unable to increase the thickness of the thin film to promote great absorption due to the fact that any fission products generated in a thick film would be unable to reach the semiconductor substrate to ionize and electron hole pairs, resulting in missing the neutron count all together. Additionally, because the fission products would be

produced and ejected 180° from one another, the best-case scenario was that one fission product was ejected in the direction of the semiconductor substrate. The reaction product self-absorption in the neutron sensitive coating restricted the thermal neutron detection efficiencies to typically less than 4.5% intrinsic efficiency [72].

To remedy the multiple weakness of the thin film device in the mid-2000s, the MSND was developed [73]. The MSND was constructed with 10 k Ω cm n-type Si with thousands of perforations etched vertically into the silicon that were approximately 20 μm wide and 100 μm deep. The perforations, which came to be known as “trenches”, were conformally diffused with a p-type source to form a shallow junction. Each trench was then backfilled with nanoparticulate ^6LiF . Li-6 was chosen in particular because the fission products produced energetic reaction products with a superior range, allowing for an increase volumes of neutron conversion material, which allowed for an increased absorption efficiency. With the Silicon trenches surrounding the LiF material, the likelihood of absorbing the fission product also increased, thereby increasing the thermal neutron detection efficiency of 11% [74]. The intrinsic thermal-neutron detection efficiency depends strongly upon the geometry, size, and depth of the perforated microstructures and through manipulating many variables such as the trench depth to approximately 500 μm deep, the MSNDs have consistently achieved an intrinsic thermal neutron detection efficiency greater than 30%. Throughout the development of MSNDs, the major loss in efficiency was found to be due to the loss of free streaming neutrons through the vertical silicon trenches setting the theoretical limit to be approximately 54%.⁷⁵ The MSNDs which were originally developed as Helium-3 (He-3) replacement technology, are comparable in efficiency while MSNDs yield a price tag of only \$125 compared to thousands of dollars for typical He-3 tubes [76].

From the MSND technology, the DS-MSNDs were developed to reach higher intrinsic thermal neutron detection efficiency and therefore become more competitive with the traditional He-3 . DS-MSND are fabricated from 1.5-mm thick, n-type 5 k Ωcm resistivity silicon in a class 100 cleanroom using photolithography processes to pattern the trenches on both sides of the wafer as well as the contact pads. The trenches are then etched using potassium hydroxide (KOH) and then titanium (Ti) and gold (Au) metals are then evaporated onto the contacts to improve the connectivity of the device. Deep trenches etched into the substrate on either side of the device are slightly offset to eliminate free steaming neutron paths through the device. The

DS-MSND devices are about 3mm thick and have an active area of 0.95 cm^2 and are individually diced from a 4" silicon wafer and are wire bonder and connected to a ceramic readout board. The final devices are then able to be incorporated into a number of different detection platforms [77].

Just as with the MSND devices as shown in Figure 4-1, the absorption of a neutron by a ^6Li nucleus induces fission, resulting in the emission of a 2.73-MeV triton and 2.05-MeV alpha particle that travel in opposite directions. When the charged particles interact with the adjacent silicon fins, electron-hole pairs are created in the silicon and the motion of the electrons and holes induces current on the DS-MSND electrodes that is simultaneously measured and ultimately forms a voltage pulse. The present generation of DS-MSNDs have a vertical partial-conformal pnp n-type ring diode encompassing the topside trenches. The n-ring design allows for good charge collection regardless of where the fission reaction products are produced within the substrate of the device. The decreased dependency of the reaction position origin therefore allows for more uniform pulse heights. Today's robust devices biased at 2.5V can achieve 65-70% intrinsic thermal-neutron detection efficiency [78]. Due to the low power consumption, small form factor, and cost-effectiveness, DS-MSNDs are optimal for implementation into a mobile detection system such as the ARGANOT system. A more detailed discussion of semiconductor physics, the fabrication of DS-MSNDs, and their performance is discussed below.

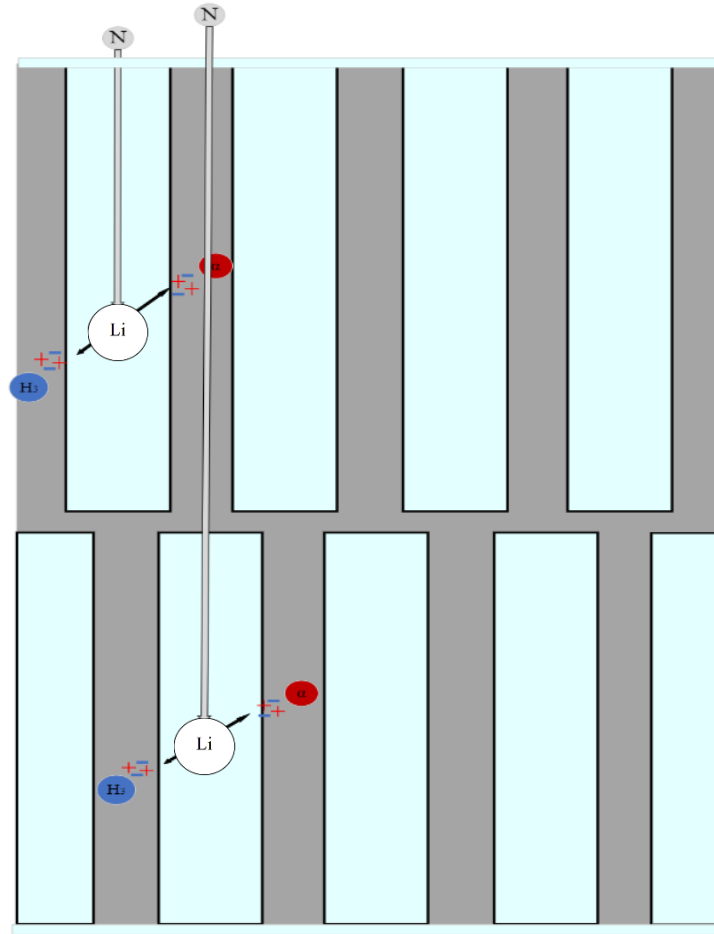


Figure 4-1 A neutron is absorbed and fissions to produce an alpha and triton particle that proceed to ionize the silicon fins of the device.

4.1 Review of Semiconductor Radiation Detector Basics

Semiconductor detectors have been used to detect radiation as early as 1947 and have since developed into a number of common high resolution detector types such as cadmium zinc telluride, high purity germanium, and many others due to their low energy threshold for the creation of charge carriers.

4.1.1 Physical Properties

Semiconductors are solid state material that lie somewhere in the middle of being an insulator or conductor. The band gaps or rather the energy space between the valence energy band and the conduction energy band of the atoms drastically vary between insulators, semiconductors, and conductors. As shown below in Figure 4-2, the insulator band gap (left) is larger than the band gap of both the semiconductor and conductor shown in the middle and right

side of the Figure. Typically, the insulator's valence band has no free electrons while the semiconductor uniquely contains a few thermally excited electrons. Conductors on the other hand have a partially electron populated conduction band, allowing the electrons to easily transfer energy states [79].

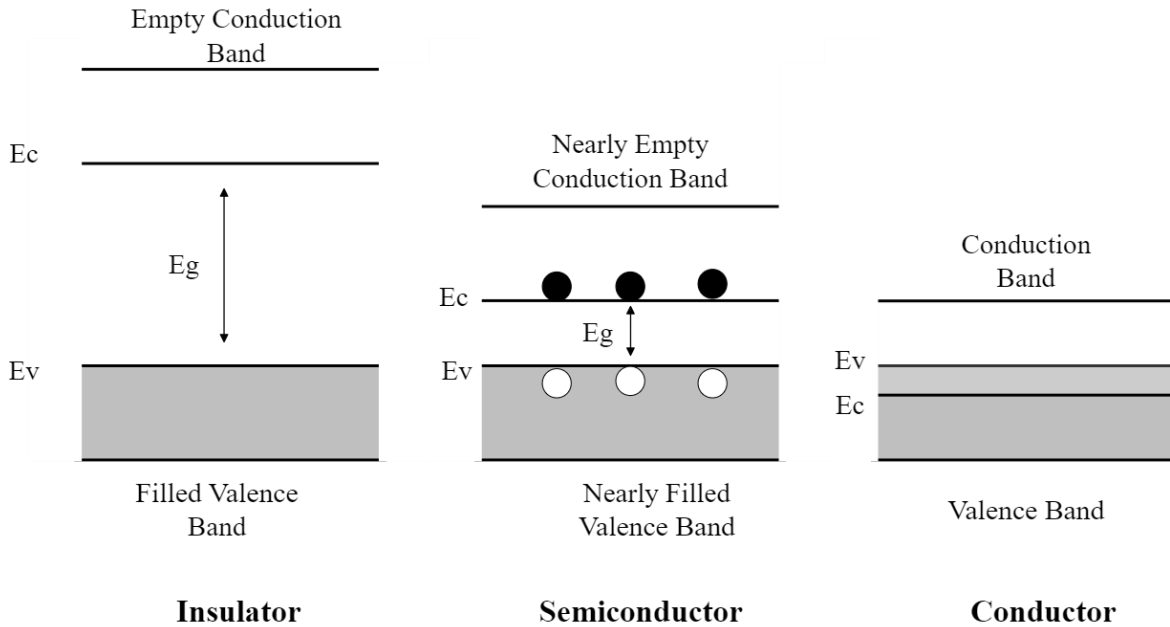


Figure 4-2 A comparison between an insulator, a conductor and a semiconductor are shown.

A single silicon atom has 14 protons and 14 neutrons in its nucleus surrounded by 14 electrons. Ten of those electrons are in tightly bound bands while the remaining four are located in the outer 3s and 3p energy bands. As shown in Figure 4-3, these electrons must each exist at discrete energy states as stated in the Pauli Exclusion Principle. When two atoms are within close vicinity of one another, the allowed energy levels overlap, causing the energy state to split, creating a quasi-continuum of allowable energy states and the two atoms can share the outer electrons. With the addition of more atoms, more quasi states are created where the lower of the split states will become the valence band (E_v) while the high energy band will become the conduction band (E_c) separated by an empty region band gap (E_g) for which electrons may not exist. In silicon for example, the band gap energy is approximately 1.12eV at room temperature. Through correctly applying Equation 4-1 and Equation 4-2 to either carrier density population of the conduction or valence band, the density for both states may be determined by Equation 4-3 where N represents the effective density of states for the band of interest. As shown in Figure

4-4, the fermi level (E_F) must maintain charge neutrality and is therefore very close to the middle of the bandgap as the population of carriers in each band is about equivalent as shown.

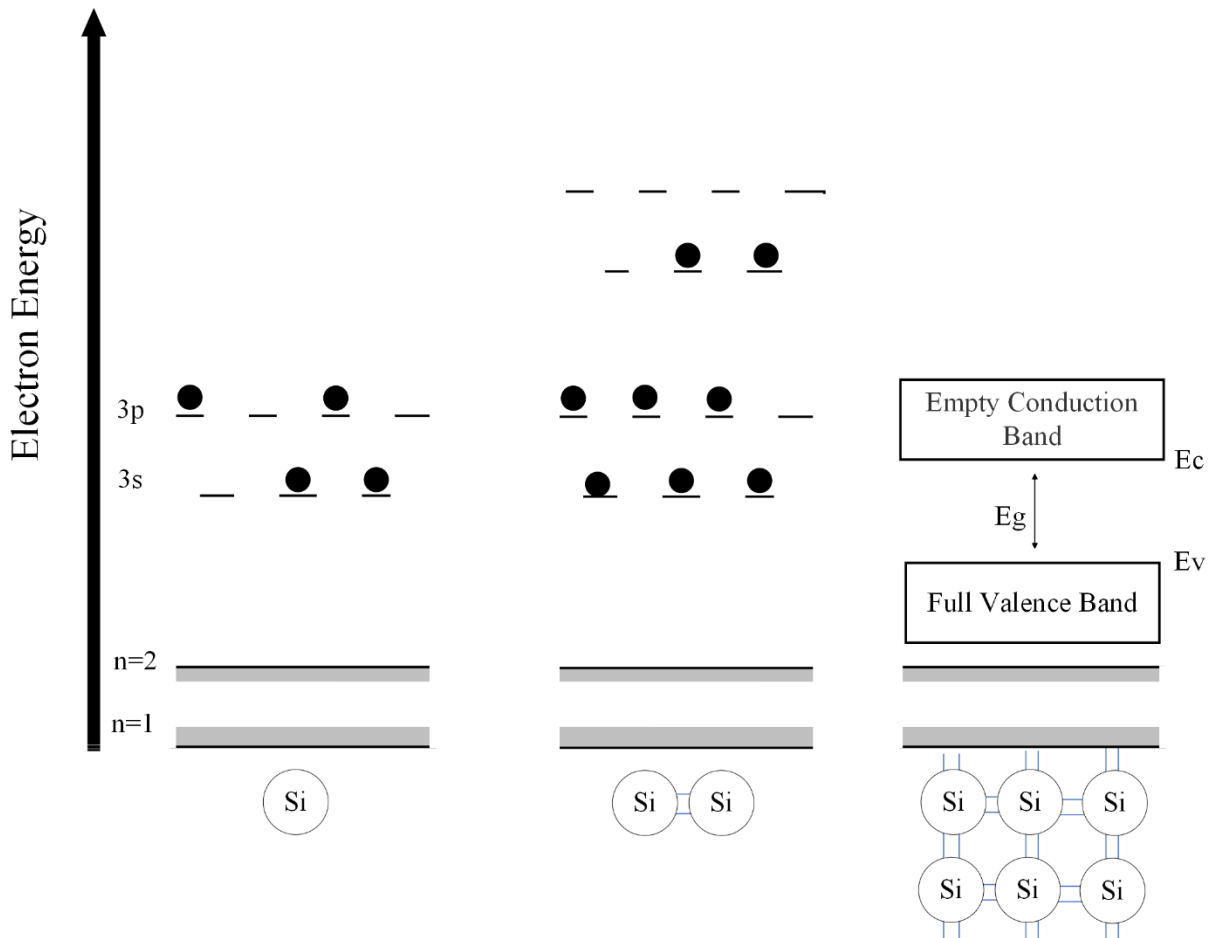
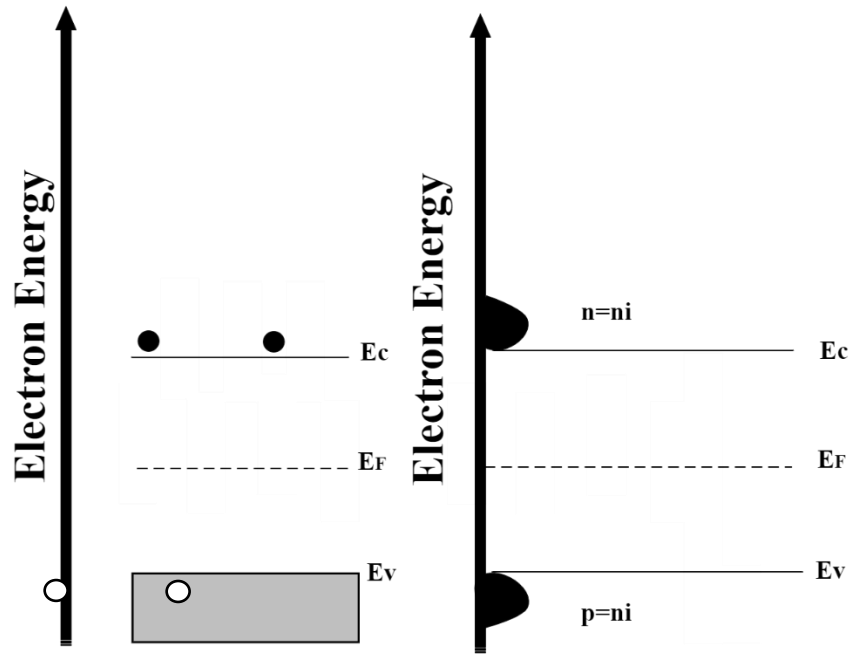


Figure 4-3 The electron shells for which electrons can exist at along with the band gap energies are shown.

Within these semiconductor models, electron charge carriers are located in the conduction band while holes or rather electrons with the opposite charge are located in the valence band. With intrinsically altered Si for example, the carrier density may be described as the integral over the energy band of the density of states of that band multiplied by the Fermi-Dirac distribution function $F(E)$, as shown in Equation 4-1. The Fermi-Dirac distribution function, which is described mathematically below, conveys the probability an electron occupies an energy state

while the density of states is simply the number of allowed electron energy states in an energy range per unit volume.



Intrinsic Semiconductor

Figure 4-4 The fermi level of the intrinsic semiconductor is demonstrated. The fermi level is adjusted to maintain neutrality.

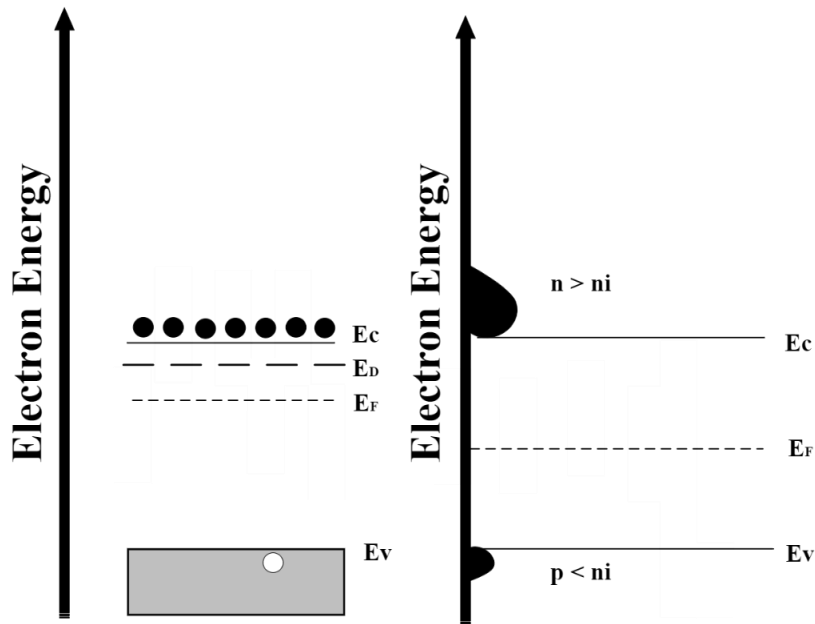
$$n = \int_{\text{Lower Band Energy}}^{\text{Top Band Energy}} N(E)F(E)dE \quad (4-1)$$

With Equation 4-2, k represents the Boltzmann constant, and T represents the absolute temperature.

$$F(E) = \frac{1}{1 + e^{\frac{E-E_F}{kT}}} \quad (4-2)$$

$$n = N_c e^{-\frac{E_C - E_F}{kT}} \quad \text{or} \quad p = N_v e^{-\frac{E_F - E_v}{kT}} \quad (4-3)$$

However, it is common practice in many Si based electronics, including MSND based technology, to intentionally manipulate the electrical characteristics of the semiconductor through the introduction of impurities or dopants such as phosphorous (P) or boron (B) to precisely control the carrier population of either energy band. The dopant concentrations, N_D or N_A , are normally an order of magnitude larger than the intrinsic carrier concentration ($\sim 10^{-10} \text{ cm}^{-3}$). In the case of phosphorous doped material, a P atom with five valence electrons replaces the Si atom, and a negatively charged electron is “donated” to the conduction band. The fermi level as shown in Figure 4-5. An example of an n type donation is shown. Notice the movement of the Fermi level with the addition of charge carriers in the conduction band. Figure 4-5, is then altered to ensure charge balance of the atom. The Si thereon refers to as “n-type” Si because of the addition of the negative charge carriers and the P atom becomes a “donor” atom. Additionally, the donor level location is very close to the conduction band and therefore can be considered ionized at room temperatures.



Extrinsic (Artificially Doped) Semiconductor

Figure 4-5 An example of an n type donation is shown. Notice the movement of the Fermi level with the addition of charge carriers in the conduction band.

When B is introduced to the Si material, the Si becomes p-type because the B with three valence electrons substitutes for a Si atom. The additional electron is “accepted” and forms four covalent bonds around the B atom. The positively charged hole is then created in the valence band. The acceptor level is located close to the valence band. Figure 4-5 demonstrates the fermi level locations for both n and p type materials.

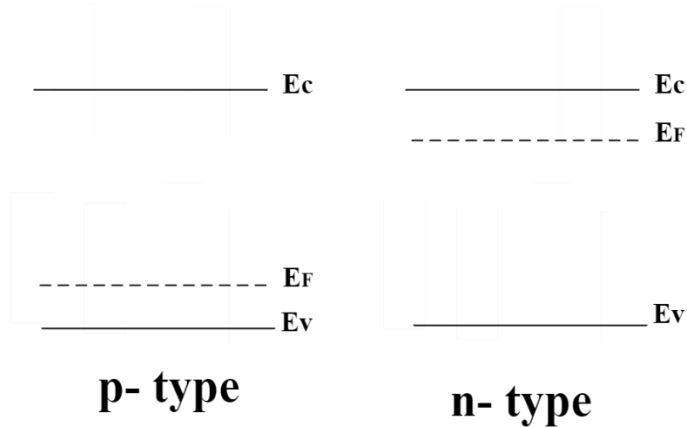


Figure 4-6 The Fermi level for a both a p and n type doped semiconductor is shown.

4.1.2 Depletion region formation

For DS-MSND devices in particular, the phosphorous doped Si wafer has an initial $5\text{k}\Omega\text{-cm}$ resistivity with a donor concentration of approximately 9×10^{11} atoms per cm^3 . DS-MSND do not operate as simple resistance devices and therefore used a pn junction to limit leakage current across the device. The pn junction represented in Figure 4-7 is simply the junction between the p and n type regions of silicon of the device. The high resistivity pn junctions of the DS-MSNDs ensure that the leakage current noise signal from the device is lower than the neutron signal generated. In using the high resistance pn junctions, a significant electric field is established which is critical for the charge carrier collection across the device. Within the pn junction formation, the fermi level continues to remain charge neutral and therefore adjusts or bends the bands to connect the conduction bands as well as the valence band between the n and p type regions. However excess electrons from the n type region and excess holes from the p type region diffuse into the opposing region. The migration of these carriers will leave behind vacancies that will effectively take on the opposite charge and contribute towards the electric field in that region that will deny any further diffusion across the junction but will promote the drifting of the minority carrier across the junction opposite the direction of the diffusion, see

Figure 4-7. With no additional thermal or bias influence on the device, the current contributions from both the drift and diffusion should reach equilibrium.

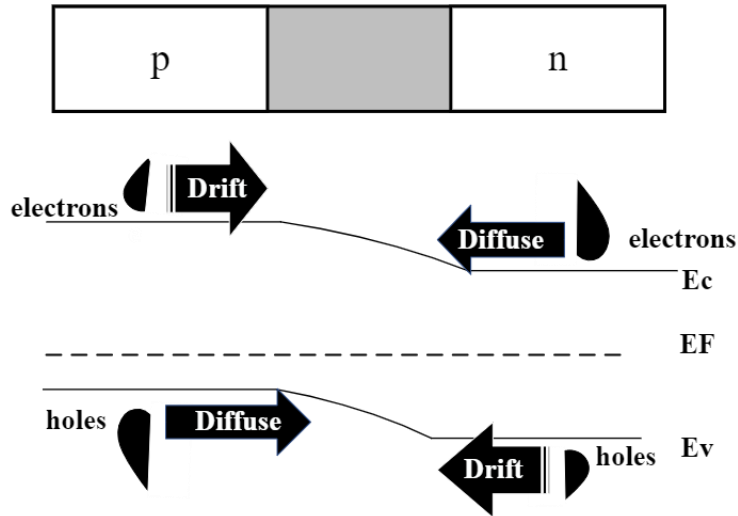


Figure 4-7 The drift and diffusion of charge carriers is shown for a simple pn junction semiconductor. The gray region on the device may present a slightly p-typed doped region, a slightly n-type doped region, or in some cases, an intrinsic region.

Naturally, there is a built-in self-bias as result of the difference between the charge densities and one side of the junction with a much higher dopant concentration, the depletion region will extend deep into the lighter doped side. In DS-MSND devices for example, the internal built-in bias is significant enough to operate the detector without an externally applied bias. The advantages of this design feature will be further discussed later.

The electric field $E(x)$, voltage potential $V(x)$, and space charge using Equation 4-4 and Equation 4-5 for an example one-dimension abrupt pn junction device is shown below in Figure 4-8. Within Poisson's equation, ρ represents the net charge density while ϵ_s presents the dielectric permittivity. As shown in Figure 4-8, the maximum electric field occurs at the junction interface between the two material types, and it was assumed that $V(x_p)$ was equal to zero. $V(x_n)$ however represents the built-in bias of the device, which is solved mathematically using Equation 4-5 where N_A and N_D are the dopant concentrations, n_i represented the intrinsic carrier concentration, and k represents the Boltzmann constant and T represents the absolute temperature.

$$\frac{\partial^2 V(x)}{\partial^2 x} = \frac{\partial E(x)}{\partial x} = \frac{-\rho}{\epsilon_s} \quad (4-4)$$

$$V_{Built\ in} = \frac{kT}{q} \ln \frac{N_A N_D}{n_i^2} \quad (4-5)$$

Using equations Equation 4-6 and Equation 4-7 which are dependent on the region of interest to solve for the electric field and voltages potentials, the plots in Figure 4-8 were created.

$$E(x_{ptype}) = \frac{-qN_A}{\epsilon_s}(x_p + x) \quad AND \quad E(x_{ntype}) = \frac{-qN_D}{\epsilon_s}(x_n - x) \quad (4-6)$$

$$V(x_{ptype}) = \frac{-qN_A}{2\epsilon_s}(x_p + x)^2 \quad AND \quad V(x_{ntype}) = \frac{-qN_D}{2\epsilon_s}(x_n - x)^2 \quad (4-7)$$

In cases such as the DS-MSND where the dopant concentration was drastically different, Equation 4-8 may be used to determine the length of the depletion region by assuming a single junction. To maximize the depletion region and in turn the charge collected, the pvp DS-MSND type devices initially developed by Fronk et al. can have a negative voltage applied to the p type region [80]. Alternatively, a positive voltage may be applied to the n type region. The reverse polarity of these applied voltages is therefore referred to as reverse biasing the device. The charge collection from these devices is shown in Figure 4-9, where the internal electric field guides the charge carriers to their respective electrodes. For low electric fields, the average drift velocity of the electrons and holes are simply the product of the E is the electric field at the location of interest and the carrier mobility μ .

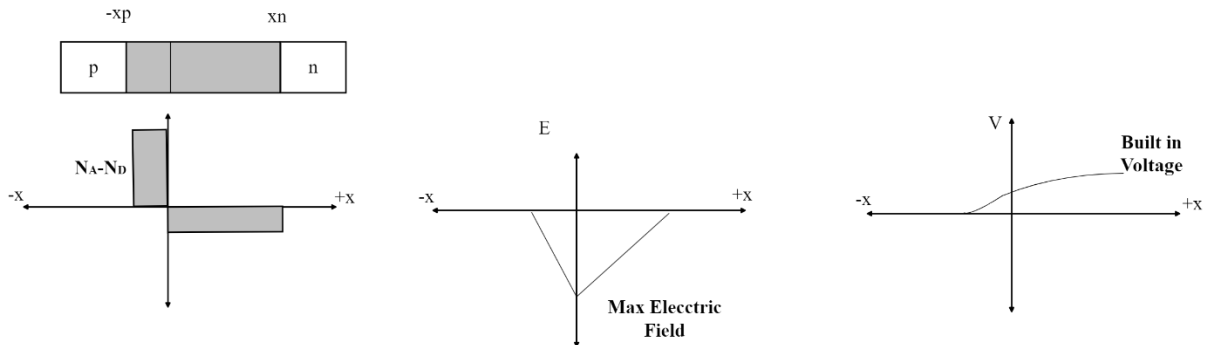


Figure 4-8 The electric field $E(x)$, voltage potential $V(x)$, and space charge for a pn.

$$W \cong \sqrt{\frac{2\epsilon_s(V_{Bi} - V)}{qN_D}} \quad (4-8)$$

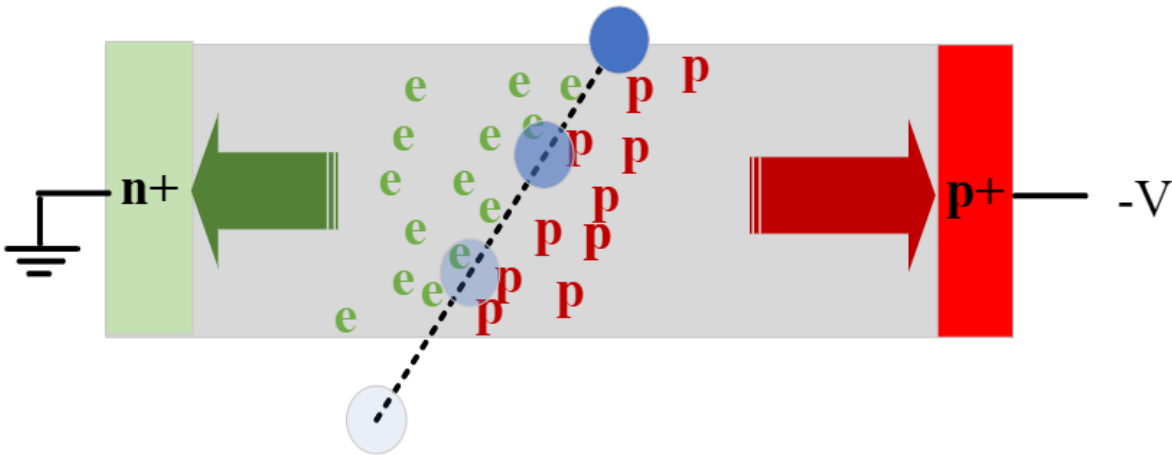


Figure 4-9 A neutron is shown entering into the semiconductor with both positive and negative electrode. The charge carriers produced from the reaction product's ionization are then drifted toward their respective electrodes.

The quality of the pn type junction may be quantified through the use of measuring the current and capacitance of the device for various voltages. The data produced from the measurements are then used to construct IV and CV curves. The deviation point of the empirical data from the ideal curve is referred to as the breakdown point. The breakdown point serves as a threshold level for which the noise of the devices is too substantial.

In summary, as the bias voltage is increased to increase the width of the depletion region which lowers the capacitance, the leakage current tends to increase as indicated in the ideal vs empirical IV curve. Therefore, it is of high priority to find the optimal operating voltage that avoids unnecessary leakage current, but also provides a high enough internal electric field for good charge collection to maximize device performance.

4.2 DS-MSND Diode Design and Fabrication

4.2.1 Photolithography

While theorized, simulated, and initially as a promising alternative in Ochs 2020 [81], the n-ring type device was first fabricated and characterized in this work herein. As mentioned, the trenches of the DS-MSND device are patterned using photolithography process. Performed in the classroom 100 cleanroom at Kansas State University, DS-MSNDs are fabricated on 5 k Ω -cm resistivity, n-type, phosphorus-doped, silicon which corresponds to a donor concentration, N_D , of about 9×10^{11} atoms cm^{-3} . The fabrication of the dual sided n-ring series begins by first patterning the n-dot outline across the wafer using AZ 1512 photoresist [82]. Upon application of the photoresist the wafer is exposed to mercury UV light wavelength between 200 to 600 nm and the regions of the photoresist protected by the mask pattern from the UV light are maintained while those exposed to the UV light for approximately 10 seconds experience broken molecular bonds and are therefore removed in the photolithography developer chemical mixture, as reflected in Figure 4-10 [83]. The n-dot pattern shown in Figure 4-11 is then created and hardened to protect the silicon dioxide underneath. The wafer is then placed into a Buffer Oxide Etchant (BOE) chemical that etches the unprotected oxide at approximately 75 μm per hour, creating the n-dot pattern within the SiO_2 layer, see Figure 4-12 [84]. The wafer is then cleaned and placed into a diffusion furnace for approximately 2-4 hours where phosphorous, and n-type dopant is introduced to the silicon of the etched away n-dot pattern. The wafer is then packaged and returned to an oxide growth manufacturer where a silicon dioxide layer is regrown over the entire wafer, covering the n-dot pattern with a replenished oxide layer.

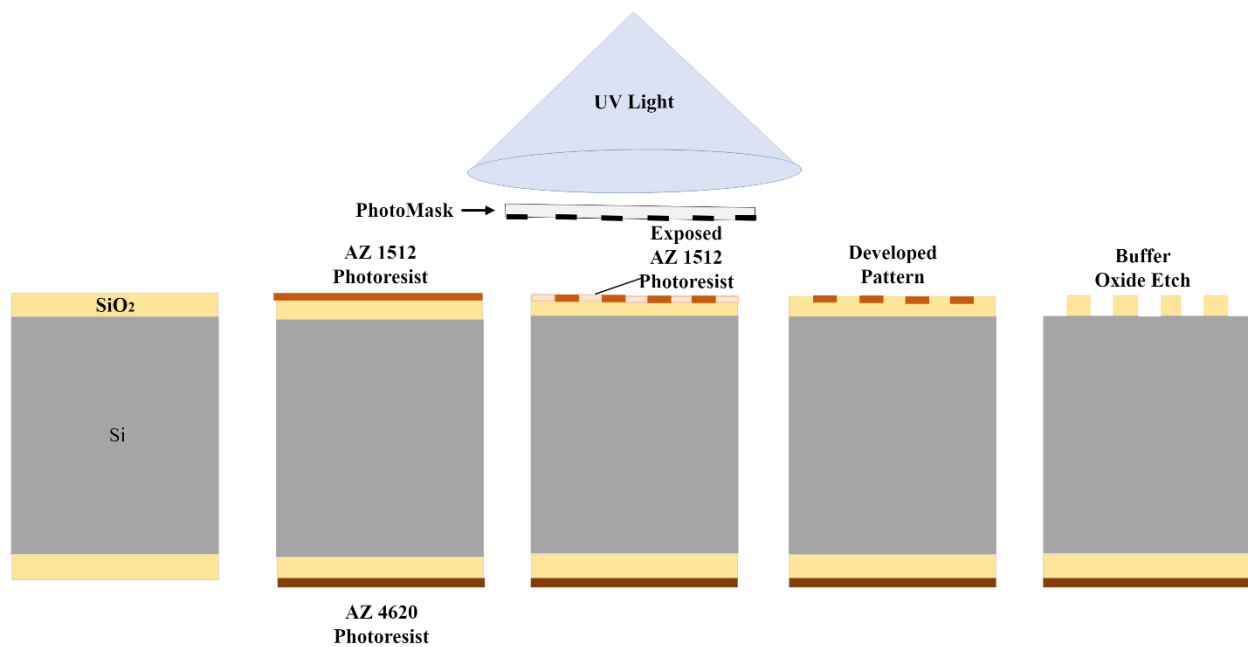


Figure 4-10 The photolithography process up to the oxide etch step for a DS-MSND device is shown.

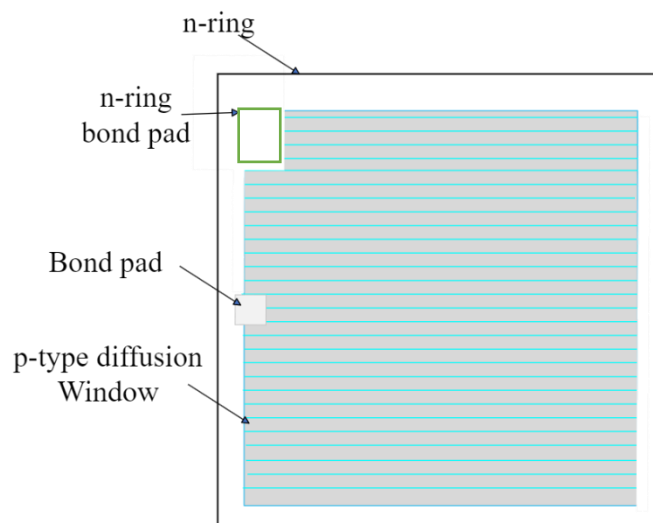


Figure 4-11 A sketch of a resulting DS-MSND n-ring patterned device is shown.

The wafer is then returned and once again patterned using AZ 1512 to pattern both of the wafer in a trench formation. Like the n-dot pattern, the trench pattern is then placed into a BOE mixture to etch away a certain amount of the unprotected SiO₂. The wafer is then cleaned, and the same photolithography process is followed to create the diffusion window fully encompassing the trenches of each of the 52 devices across the wafer. Upon the completion of the

photolithography patterning. The wafer is then prepared for Potassium Hydroxide (KOH) etch of the silicon. Within the KOH wet etching process, all regions protected by SiO_2 are etched at a much slower rate than the bare Si. Therefore, the regions of the devices, namely the trenches that are not protected with SiO_2 are etched anisotropic while the fins of the trenches that maintain a healthy layer of SiO_2 are largely unaffected, see Figure 4-13.

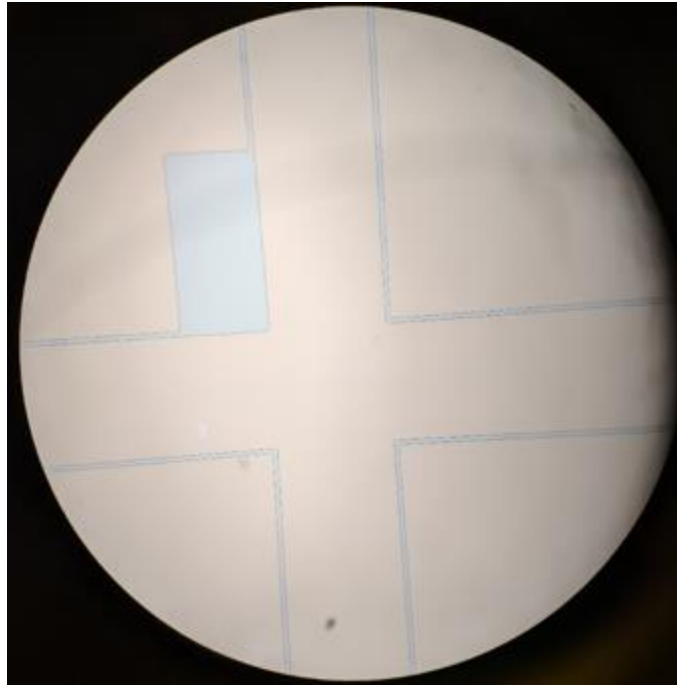


Figure 4-12 The junction of four diodes across a wafer is shown following a BOE etch.

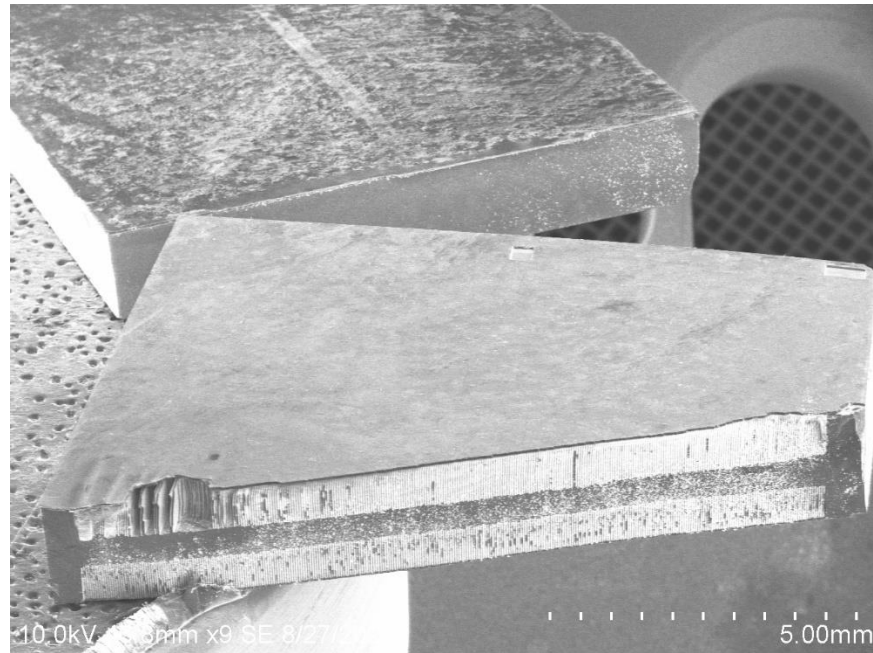


Figure 4-13. The fins of the devices that were protected from the KOH during the etching process are visible in the cleaved wafer. Notice the n-dot bond pad in the upper right-hand corner of the device along with the center bond pad are clearly visible on the individual diode. During the KOH etching process, the bond pad is protected by SiO_2 .

4.2.2 Etching

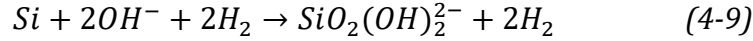
While DS-MSND etching had previously achieved depth as deep as $600\ \mu\text{m}$, the depth, uniformity and structural integrity of the trench was not achieved consistently [85]. To allow for significant trench volume on two sides of the wafer, the thickness of the wafer was increased to 1.5 mm from the 1.0 mm thickness of the dual sided MSND presented in previous work [86]. Multiple challenges were encountered in the transition to the new 1.5-mm thick DS-MSND design. Specific attention was allotted to address the root causes of non-uniformities in the trench depths. While the non-uniformities are small in comparison to the total trench depth, it was believed that uniformity of the DS-MSND devices served to improve the overall reliability and performance of the detectors in the long run as well as adhere more closely to the performance suggested by simulations. It was concluded that one of the primary causes of trench non-uniformity was the presence of hydrogen bubbles that can cling to the surface of the trenches as they are etched. The presence of the hydrogen bubbles prevented the KOH solution from fully contacting the Si and therefore stalling the etch of the area, creating un-uniform results.

Results published by Park et al. confirmed that megasonic agitation is an effective way to improve the etching characteristics, such as the etch rate, etch uniformity and surface roughness, and is suitable for the fabrication of devices with complex structures [87]. As suggested by Park et al., to overcome the non-uniform etching issue, a “megasonic” tank was used to agitate the KOH solution and improve the etch quality [88]. The megasonic device produces vibrations tuned specifically to induce hydrogen bubbles created during the etching reaction to un-adhere from the trench walls during etching.

A second challenge to etching dual-sided MSNDs was the occasional occurrences of “undercutting” in which the silicon material is over-etched near the surface, creating undesirably thin fins. The very thin fins were then unable to fully capture the fission reaction products, resulting in a decrease in the charge collected throughout the device. Undercutting produces irregularities in the device and occasionally produces undesired electrical pathways across the surface of the MSND and reduces production yield of useful devices. The root cause was determined to be poor adhesion of photolithography as well as a poor exposure to UV when patterning the devices. The poor development of mask pattern likely led to poor oxide etch, which then allowed the undercutting to take place with the KOH Si etch.

The new DS-MSNDs have dual 600-mm deep trenches, as opposed to the single 400-mm deep trenches in the MSNDs. In theory, the megasonic vibrations produced in the KOH solution will increase the interaction between the KOH solution and the silicon material producing H₂, thereby increasing the speed at which material is removed. In addition, the vibrations induce detachment of air bubbles that have, in the past, inhibited consistent etch depths by clinging to silicon fins and blocking movement of KOH. Testing of the megasonic bath presented challenges in terms of maintaining a consistent operating temperature and in turn the etch rate.

As represented in Equation 4-9, the KOH Si reaction as described by Seidel in which the KOH molecules form K^+ and OH^- ions as they disassociate in water [89]. When the two hydroxide atoms are in proximity to open Si bonds, two electrons are held near the surface of the Si by a potential well. The Si-Si bonds that hold the Si to the neighboring Si atom lattice are broken by bonded electrons with excess thermal energy. The Si-Hydroxide molecule may then react with the remaining OH^- to form $Si(OH)_4$ and the etch process continues.



As result, the number of atomic bonds in Si crystalline plane of the etch directly impacts the rate of etching. The Si etch rate is also impacted by factors such as the solution temperature and KOH concentration. As shown in Figure 4-14, the trenches were vertically etched in the [110] direction, aligning the trench walls with the (111) plane. With higher temperature and a more concentrated solution, the etch rate drastically increases. While a quicker etch rate allows for a quicker processing time, there was concern that a high etch rate could introduced rough, uneven feature, and may be difficult to control in terms of etch depth. On the other hand, a slower etch rate introduces concerns as well. For one, as the etch time drastically increases, the window for potential errors due to power outages, temperature variations, unoccupied lab space overnight, and more, also increases. Furthermore, the integrity of the protective SiO₂ over a significant amount of time was also of concern. Therefore, there appeared to be an optimal etch rate to achieve the unprecedented average dual sided 600 μm depth trenches.

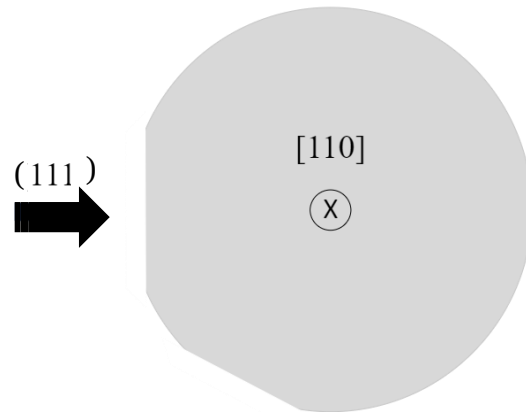


Figure 4-14 The KOH etching into the 110 directions against a 111 plane is shown.

The previous KOH recipes used 20 wt % KOH salt and water at 75 °C which would etch Si approximately 100 μm per hour, so to reach a target depth of 600 μm, the etch time would be 6 hours. With the new recipe, the etching temperature window was chosen to be between 60-70 °C [90]. At 60 °C and 20 wt % KOH the Si etch rate was empirically determined to be 40 μm per hour, up to 70 μm hour, as shown in Figure 4-15. Etch rates published by MicroTech Systems, Inc. demonstrate that KOH silicon etch rates over a temperature range of 20 °C to 100 °C increase exponentially [91]. To reach the desired 600 μm depth, 14.5 hours was determined to be

the approximate etch time window. Though the longer etch time was a bit slower than desired, consistent periods of observation and evaluation were maintained throughout the 14 hour etch period.

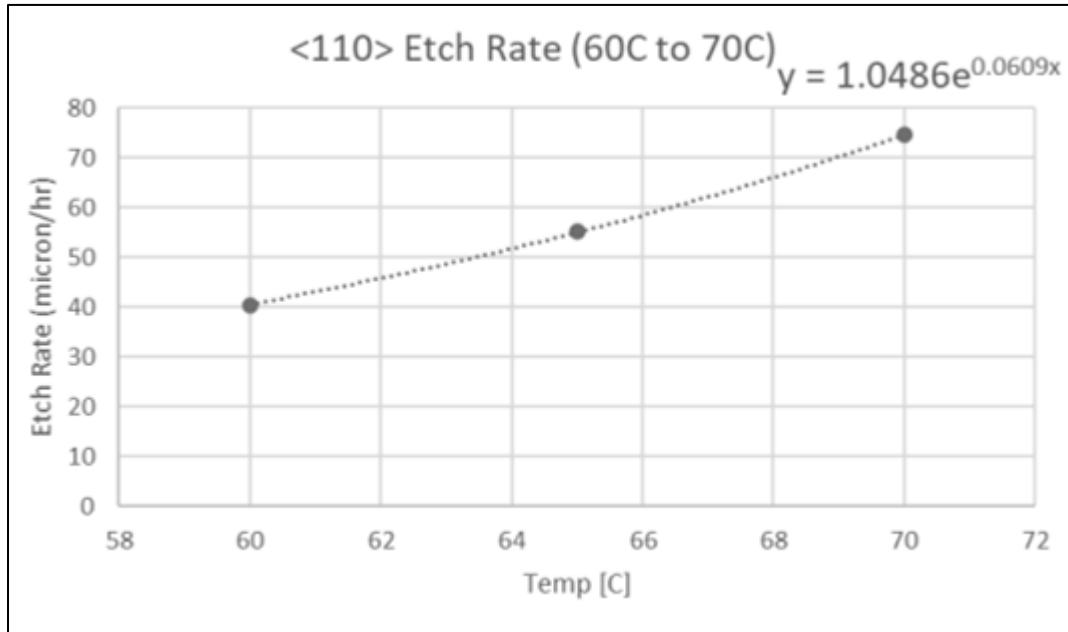


Figure 4-15 The curve of etching rate for KOH over increasing temperature is shown.

With the temperature impact on etch rates in mind, a timer/switch was to set an on and off cycle of the vibration to best control temperature fluctuations with agitation. The vibration on-off parameters were determined empirically after monitoring the temperature increase when the vibration was in the on state. When the temperature rose above 70 °C and period of off times were implemented to determine an optimal ratio of on to off times. Two minutes was selected as the maximum amount of time the solution could be agitated before the temperature began to rise rapidly above 70 °C. Ten minutes was determined to be the optimal amount of off time to allow for the solution temperature to stabilize. The cycle was then repeated throughout the duration of the 14 hours etch period, see Table 4-1. Figure 4-16 depicts the KOH megasonic tank in the “on” state with a wafer in the solution. Once the wafer was fully etched, it was removed from the cleanroom environment for evaluation. Scanning Electron Microscope (SEM) images of the internal side view of the wafer are shown in Figure 4-17. As shown, the average target trenches depth of 600 μm was successfully achieved. Additionally, the average trench fin thickness of approximately 10 μm were close to ideal for an average trench width of approximately 20 μm

[92]. Furthermore, the integrity of the oxide coating the top of the trench fins appeared to be maintained throughout the etch as evident by the uniform quality of the trench widths.



Figure 4-16 The mega sonic tank in operation with the wafer placed into the wafer boat is shown.

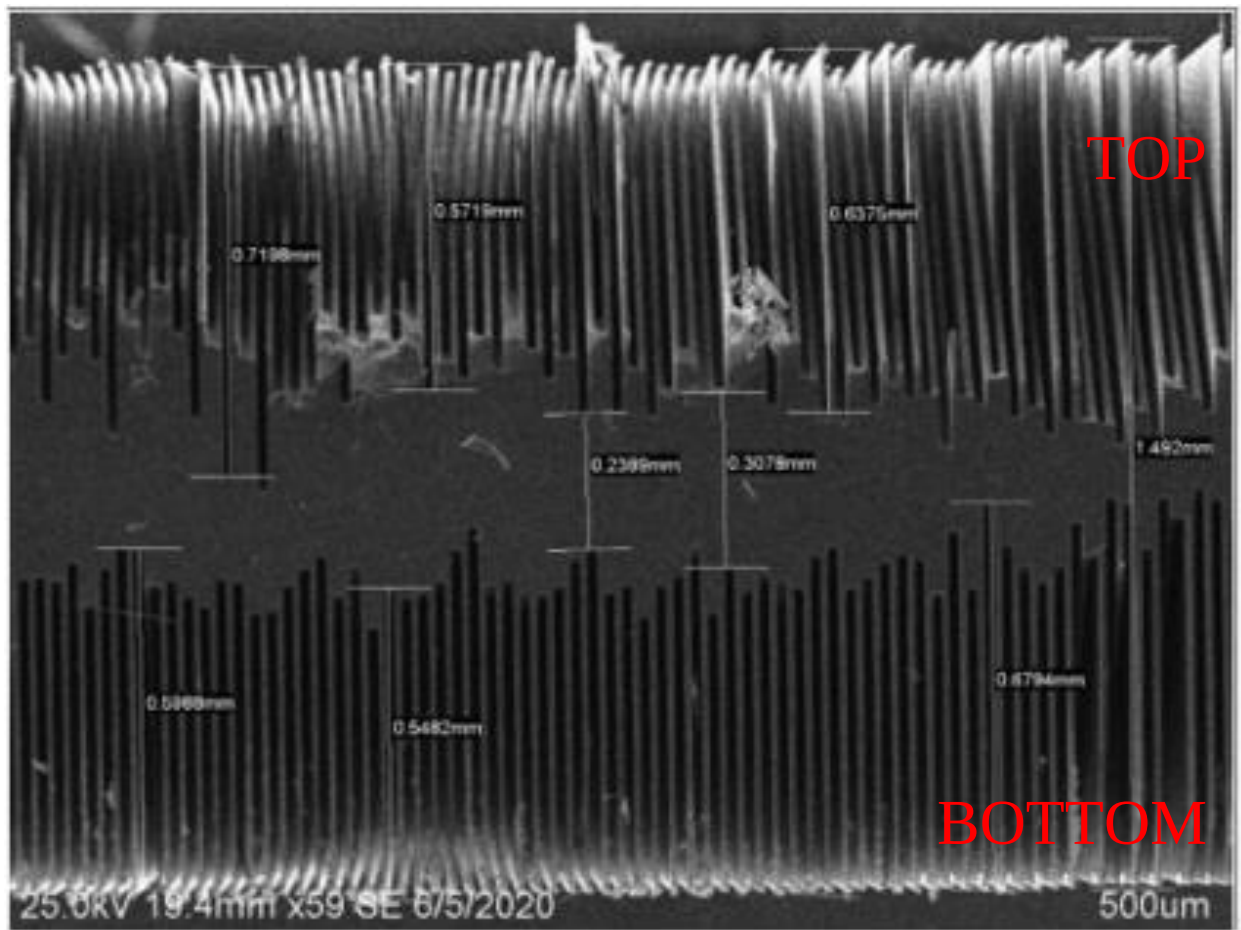


Figure 4-17 An SEM image of an etch silicon wafer from once cleaved is shown. The top fins appear to be bent, but this feature is commonly attributed to stresses when the wafer is severed for analysis purposes.

While the etch achieved record depths, several weaknesses became apparent following the SEM analysis. For one, the trench depth was quite non-uniform and much deeper on the “top” side of the device. This may have been caused by residual oxide material remaining on top of the trenches, which would retard the KOH silicon etch until the remaining SiO_2 layer was sufficiently removed.

To study the uniformity in etch depth between the top and bottom set of trenches, trials were run in which the wafer was positioned both perpendicular and parallel to the ultrasonic waves. Several diodes from across each wafer were examined. Two representative SEM images from both trials are shown below. The perpendicular set of SEM images, see Figure 4-18, indicated a large discrepancy in depth, with the top again etched much deeper ($\sim 700 \mu\text{m}$) than the bottom ($\sim 400 \mu\text{m}$), see Table 4-2. In the case in which the wafer sat perpendicular to the oncoming

ultrasonic waves, the trenches were positioned parallel to the wave. The trench width to fin ratio was then examined for the perpendicular etched wafer. The fin was approximately 10 μm wide compared to an approximately 20 μm wide trench, see Figure 4-19. The parallel wafer was then examined in which the ultrasonic waves ran perpendicular to the trenches, see Figure 4-20. Results from the parallel case indicated much greater uniformity of trench depth of both the top and bottom sets of trenches with both estimated to be about 525 μm and 600 μm respectively. However, the perpendicular trenches deviation between top and bottom was close to over 300 μm . The fins became quite thin with the parallel etch approach, see Figure 4-21. It is possible that the oxide was thinner than desired with this device, leading to the rather thin fin. Ensuring enough oxide on the top of the fins to limit the lateral etch was made a priority following the positional study.

It was possible the perpendicular etch, which allowed the entire trench to be in direct line with the sonification waves was greatly affected by wave formations. Therefore, if a wave was formed in close proximity to the front wafer face, the rate at which the agitation occur was likely greater than the back of the wafer which may have been further from a standing wave formation within the bath, contributing towards a reduced agitation and slower etch rate. Therefore, it appears that the parallel wafer in which the trenches were not directly open face to the waves were more mildly affected in a more uniform fashion. The milder treatment resulted in a more uniform etch rate on either side of the wafer and ultimately more uniform trench depths. Based on these results, all etches following were conducted in parallel to the ultrasonic waves.

Table 4-1 The KOH megasonic testing parameters below were experimentally determined to produce the best deep wafer etch.

KOH Megasonic Testing Parameters
20 wt% KOH Solution

Megasonic Mode	20 minutes off 15 seconds on
Set Temperature	59 C
Minimum Temperature	55 C
Over-Temp	64 C
Average Temperature	60.5 C
Theoretical etch rate <110>	41 mm/hr
Theoretical etch rate, SiO ₂	0.055 mm/hr

Table 4-2 The results of etching the wafers in parallel and perpendicular to the wave form is shown.

KOH Megasonic Testing Results

Wafer 1 - Parallel	BoE	BS Depth	FS Depth	Etch Time	AVG Etch Rate	Fin Width Top	Fin Width Bottom
	1 min	663 μ m	580 μ m	14 hrs	44 μ m/hr	4 μ m	10 μ m
Wafer 2 - Perpendicular	BoE	BS Depth	FS Depth	Etch Time	AVG Etch Rate	Fin Width Top	Fin Width Bottom
	3 min	760 μ m	490 μ m	14 hrs	44.6 μ m/hr	4 μ m	10 μ m
Wafer 3 - Parallel	BoE	BS Depth	FS Depth	Etch Time	AVG Etch Rate	Fin Width Top	Fin Width Bottom
	2 min	700 μ m	575 μ m	13.75 hrs	46.75 μ m/hr	SiO ₂ ruined	Etched out
Wafer 4 - Parallel	BoE	BS Depth	FS Depth	Etch Time	AVG Etch Rate	Fin Width BS	Fin Width FS
	1 min	541 μ m	514 μ m	12.53 hrs	42.6 μ m/hr	8 μ m	8.5 μ m

* FS - Front Side

BS - Back Side

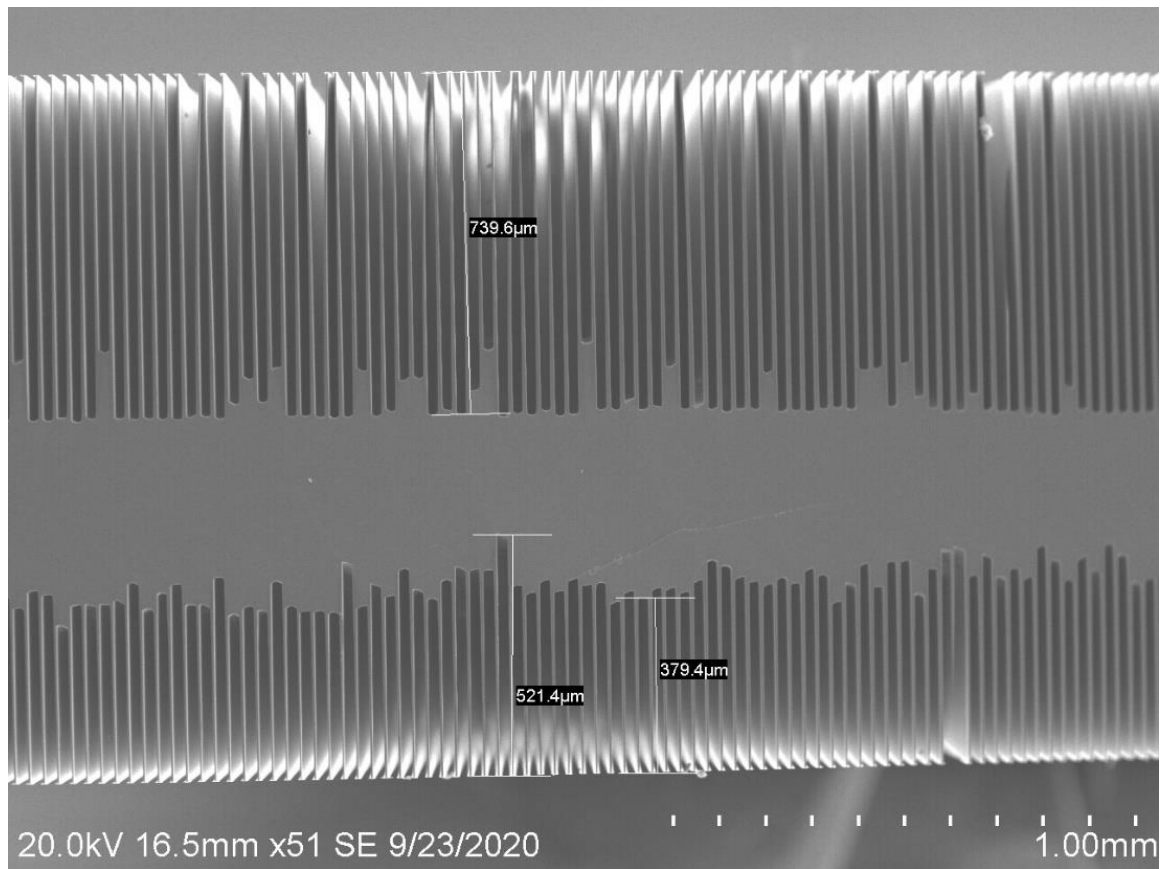


Figure 4-18 The SEM image of a cleaved single diode etched perpendicular to the wave form is shown.

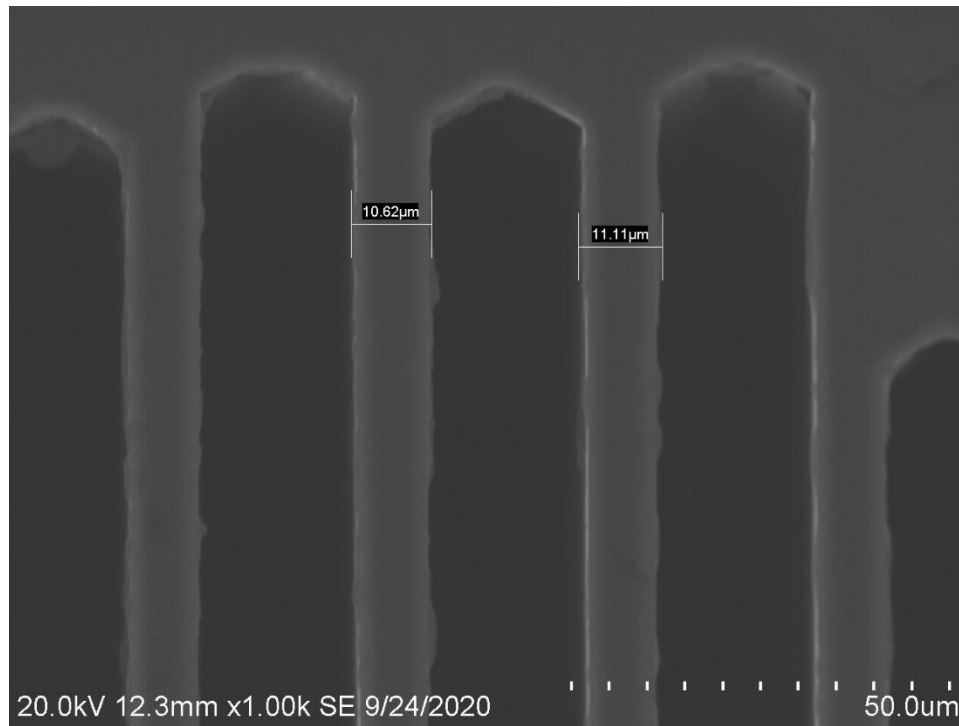


Figure 4-19 The view of single trenches demonstrating the trench to fin ratio is shown. The perpendicular etch achieved high aspect ratio features.

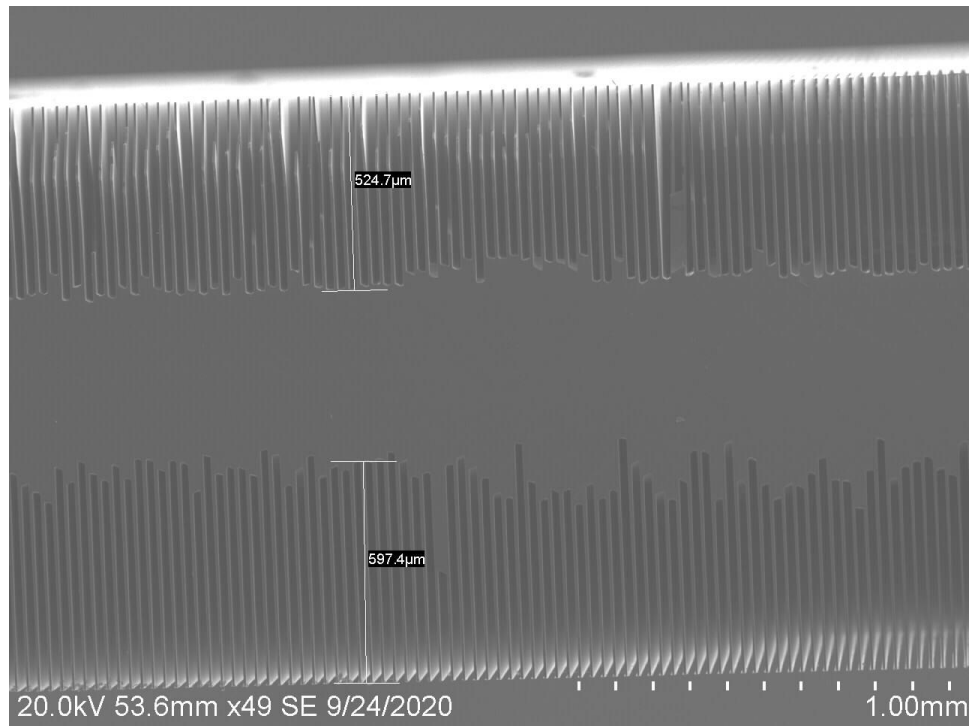


Figure 4-20 A single diode was cleaved after undergoing a KOH etch in which the wafer was placed parallel to the oncoming wave.

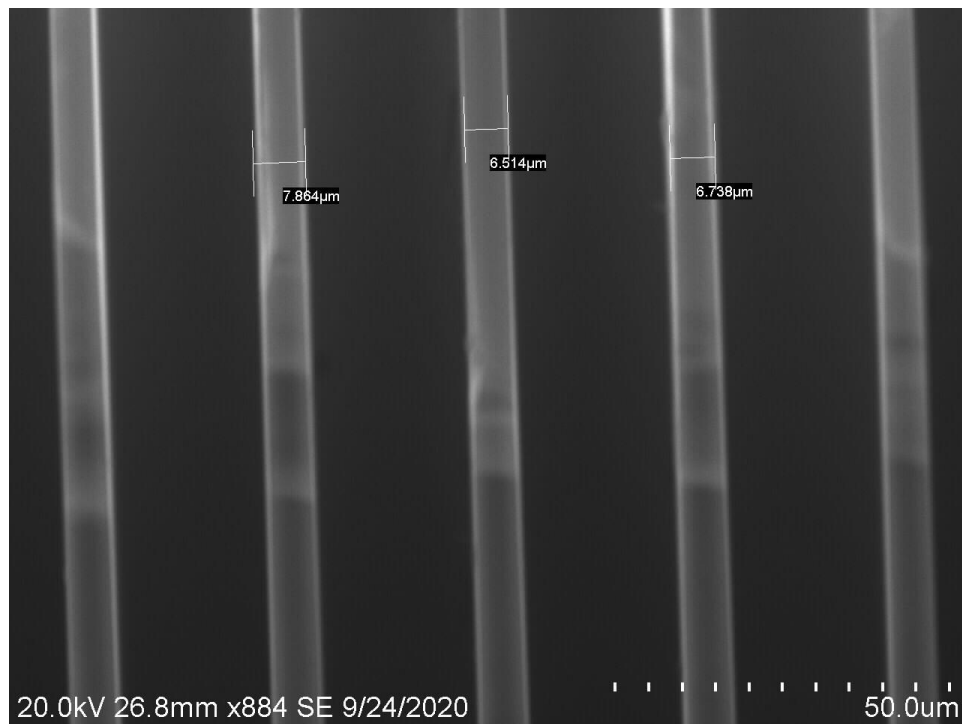


Figure 4-21 The view of single trenches demonstrating the trench to fin ratio is shown. The parallel etch achieved high aspect ratio features.

Following the etch, the wafers were cleaned in hydrochloric bath to remove alkali ion contaminant such as K^+ and the residual oxide layer on the tops of the fins was removed using a BOE etch, see Figure 4-22. Any organic residuals on the surface of the wafer were then removed with a hot “Piranha” solution ($H_2SO_4:H_2O_2:H_2O$). The wafers were once again BOE etched and then placed into a $NH_4OH:H_2O_2:H_2O$ bath. The wafers were placed into a final BOE solution, rinsed, dried and then placed into a diffusion contamination-controlled space to prepare for the pn-junctions creation on the DS-MSNDs.

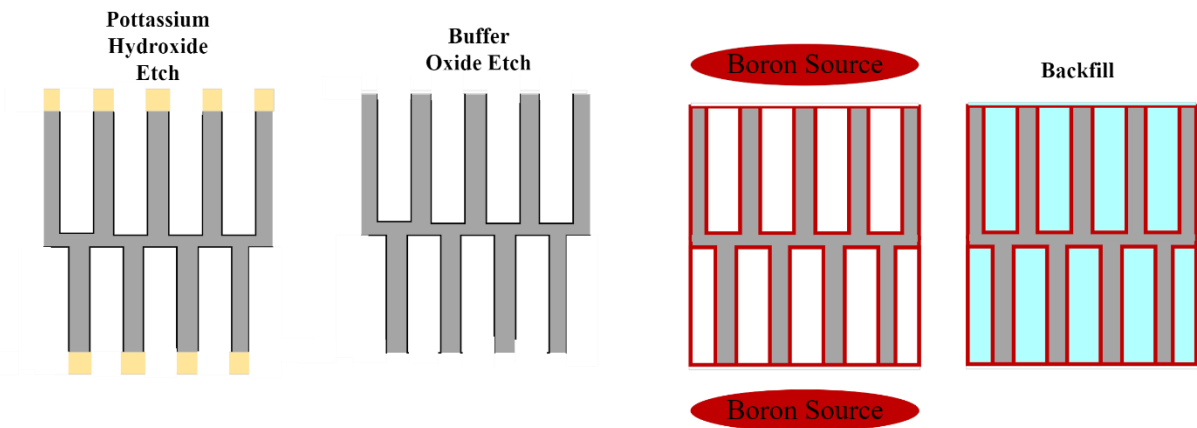


Figure 4-22 Following the KOH etch, the wafers undergo a rigorous cleaning process to ensure that all possible contaminants have been removed prior to introducing desired dopants. Following the doping process, the wafer is removed from the class 100 cleanroom and the trenches are filled with a neutron conversion material.

4.2.3 Doping

As discussed at the beginning of the chapter, the doping profile of the DS-MSND devices heavily influences the performance of the device in terms of charge collection and noise levels. McNeil, Bellinger, Fronk, had previously investigated the design and doping profile of the MSND and DS-MSND devices [93]. Most recently, Ochs extensively examined the performance of DS-MSND devices with various doping schemes [94]. Ochs work examined the comparison of conformal and partially conforming (see Figure 4-23) pvp and partial-conformal pvn doping along the trenches of DS-MSND devices.

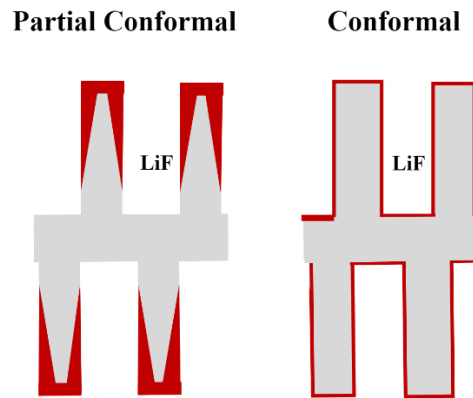


Figure 4-23 An image of a conformal and partial conformal trench doping scheme is compared.

4.2.3.1 Review of Partial-Conformal p_nn DS-MSND

The partial-conformal p_nn-DS-MSND was initially studied by Fronk et. al., but was abandoned because the performance did not meet expectations [95]. The partial-conformal p_nn devices which had a p-type dopant extending partway down the top-side fins and n-type dopant extending partway down the back-side fins, experienced good charge induction for interactions in the p-type doped fins. However, interactions in the n-type doped fins induced current in the detector electrodes much slower. Because of the electronic circuitry time parameters, a large percentage of the backside current was not collected in time to contribute towards the device signal, which led to poor detection efficiencies. Additionally, the devices had operated with an applied bias to increase the charge collection, however significant electronics would have been needed. Additionally, while the charge collection efficiency was initially determined to be the culprit leading to the degraded performance, later work performed by Ochs determined the leading cause of the degraded performance was also likely due to insufficient backfill of the LiF neutron conversion material [96].

4.2.3.2 Review of Conformal p_pp DS-MSND

The conformal p_pp-DS-MSND which had boron p-type contacts uniformly covering the fins and trenches on both sides of the DS-MSND were developed to eliminate charge collection problems with early DS-MSNDs fabricated with a vertical p_nn-type structure [97]. The anode and cathode became the top-side fins and back-side fins and the device operated with just the built in potential, which conveniently fully depleted the 10 μm wide trenches. Because of this design, the

device output produced voltage pulse of varying polarity, depending on which set of fins the reaction occurred. While the output was scientifically informative, the device required additional complex electrical equipment that would prove inconvenient in applications such as the ARGANOT in which detector module size is of the utmost importance. Unfortunately, simulations conducted by Ochs determined that the amount of charge induced in the detection circuit was largely dependent on the depth of the charged-particle interaction and ultimately the charge generated in the conformal pvp devices was less than that produced with the pvn devices and called for longer pulse processing times [98].

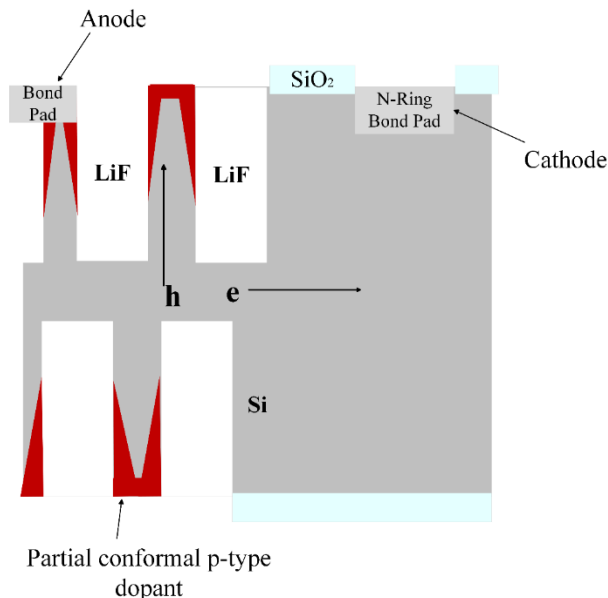
4.2.3.3 Review of Pvp Partial-Conformal

The partial conformal pvp-DS-MSND had p-type dopant on both sides of the diode extending partway down the fin and tapering off before reaching the bottom of the trench. The amount of charge induced was a function of interaction location and was more uniform than the fully conformal pvp-DS-MSND design. However, the average amount of charge induced was less. Ochs concluded that the motion of the electrons appears to be diffusion dominated causing a significant portion of the current to cancel out, because current with opposite signs was induced in both electrodes simultaneously. Ultimately while the amount of charge induced as a function of interaction location was more uniform than the fully conformal pvp-DS-MSND design, the reduced charge induction downshifted the neutron spectrum towards the noise floor which negatively impacted the gamma-ray rejection properties of the detector [99].

4.2.3.4 Pvp With n-Ring

To remedy the problems presented with the previous DS-MSND generations, the partial-conformal pvp-DS-MSND with an n-type ring was designed by Ochs in 2019 but first fabricated here [100]. Because the partial conformal in every doping scheme outperformed the conformal doping, it was selected for the n-ring design. The front and back-side fins were both doped with p-type (boron) dopant while an n-type ring was deposited to a single side surface. The ring fully surrounded the p-type doped features that were then separated by an oxide layer to prevent the contacts from shortening on the surface. The n-ring design was thus capable of operating with an applied bias which would allow for an increase in the electric field if needed. The two p-type

sides of the diode were connected to form the anode once packaged while the n-type ring became the cathode. The outermost trenches parallel to the bulk silicon were intentionally left intrinsic to prevent unneeded depletion into the non-trenched areas of the diode, see Figure 4-24 .



Following various simulations conducted by Ochs and preliminary physical testing initially performed by this author with varying voltages and charge carrier doping profiles, the pop with n-ring design emerged as the optimal design for compact high efficiency neutron detection within the ARGANOT system.

4.2.4 Backfill

4.2.4.1 Review of Backfill

As previously mentioned in an earlier chapter, several neutron conversion materials can be used in radiation detection. In past work conducted by McNeil and Bellinger, LiF, a stable alkyl halide, was determined to produce the best signal leading to higher efficiencies, while providing the most ease to handle [102]. Boron for example was ruled out due to the short range of the fission products. Additionally, the signal produced from the boron fission products was too close in energy to the noise background of the device, making it difficult to set a sufficient threshold without subtracting neutron events of interest.

In the initial DS-MSND devices, the detection efficiency was limited to about 55% [103]. As mentioned earlier in the dopant discussion, it was assumed by Fronk that the pun doping scheme was responsible for the limitation because it was thought that the ^6LiF was sufficiently packed in all trenches through a centrifuge method. The SEM image shown in Figure 4-25 below demonstrates how the device appeared 100% full through visual inspection. Through extensive modeling performed and presented in Ochs, 2020, the expected device performance as a function of trench dimensions and the density of the neutron conversion material was determined. Initially the density quantification or rather the backfill packing fraction, which is simply the density of the LiF power divided by the density of the device, was estimated visually. However, once a packing quantification process was implemented, it was revealed that the packing fraction of ^6LiF material was only 30%. When the 30% packing fraction was simulated using MCNP (Monte Carlo N-Particle, a nuclear transport code by Los Alamos), the 55% efficiency was the theoretical limit for 30% packing fraction of the devices manufactured at that time [104].

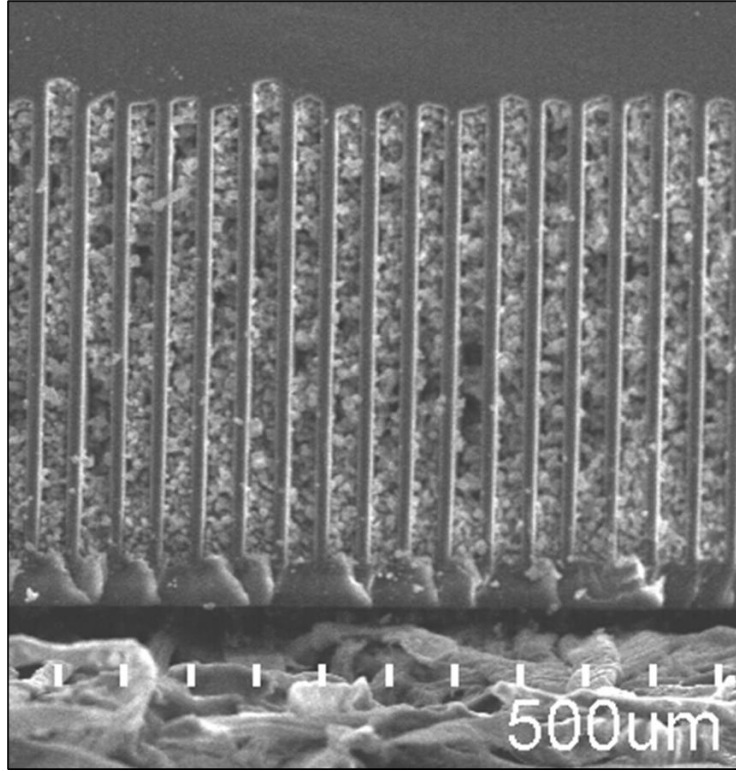


Figure 4-25 The backfill of a single diode once an individual diode was cleaved is shown.

As result of the simulations published by Ochs, it was determined that the optimal packing fraction of LiF material in the MSND trenches was between 50% and 60% [105]. Ochs concluded that increasing the packing fraction, see Equation 4-10, above 60% reduced the detectors' neutron response through self-absorption and therefore limited the range of the fission reaction products. Additionally, Ochs's work identified the importance of optimizing the LiF particle size, particle size ratio, humidity levels, lubricants, and trench sizes. Results yielded that uniform rounded particle would be optimal, however the cubic lattice structure of LiF somewhat limited the shape manipulating capabilities. The cubic restructure was instead modeled to optimize the particle size and dimensions.

$$\text{Packing Fraction} = \frac{\rho_{\text{LiF in Trenches}}}{\rho_{\text{LiF}}} \quad (4-10)$$

After identifying many of the parameters at play in maximizing the LiF packing fraction, investigation of the best method to place the LiF material began. As described by Ochs, an isostatic press was applied as means to backfill without the use of a wetting agent [106]. With

pressure ranging from 1000 psi to 1500 psi and using both micro- and nano-sized powder in various combinations, the highest resulting packing fraction was 35.6% (24.3mg gained) for micro-sized powder with a trench of depth 365 μm . A vibration table method was investigated with limited results as well with possible superior lubricants such as Ethylene glycol. Through the various backfilled methods, it was determined through experiments that the optimal quantity of nano-sized particles in the micro-sized ${}^6\text{LiF}$ mixture to fill in void spaces between adjacent micro-sized particles ranged between 15%-25% by weight.

4.2.4.2 DS-MSND Backfill

A method of inputting a LiF methanol solution into trenches through a micropipette was also briefly explored, see Figure 4-26. However, a good deal of difficulties arose with the microtip from becoming congested. The images within Figure 4-27 demonstrate the approximately 10 μm inner diameter of the tip filled with LiF material. The consistent tip congestion would likely make any LiF injection on a mass production nearly impossible.

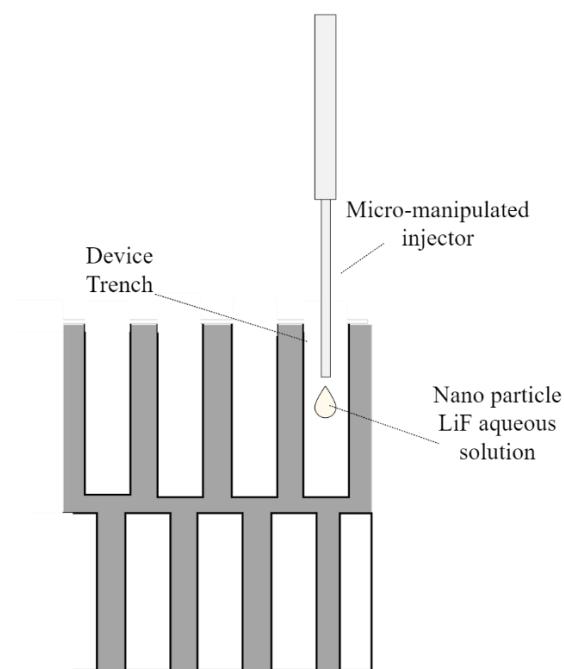


Figure 4-26 The micro-manipulator would inject a slurry of a LiF based solution into the trench. Injection however became challenging because of a build up of LiF particles.

Injection of molten metallic lithium was also briefly considered based on backfill performance work conducted by Watson, 2023 [107]. Watson investigated the potential for many lithium-based compounds such as metallic lithium, LiF, LiBr, LiCl, LiOH, Li₃N, LiI, and Li₂O₂ and

through simulations, indicated that metallic lithium for a 20 μm wide, 400 μm deep trenched device achieved the highest maximum efficiency. Unfortunately, because the metallic lithium is unstable, an n-type dopant, and had been reported by Watson to be quite destructive to the silicon, the method was never pursued [108].

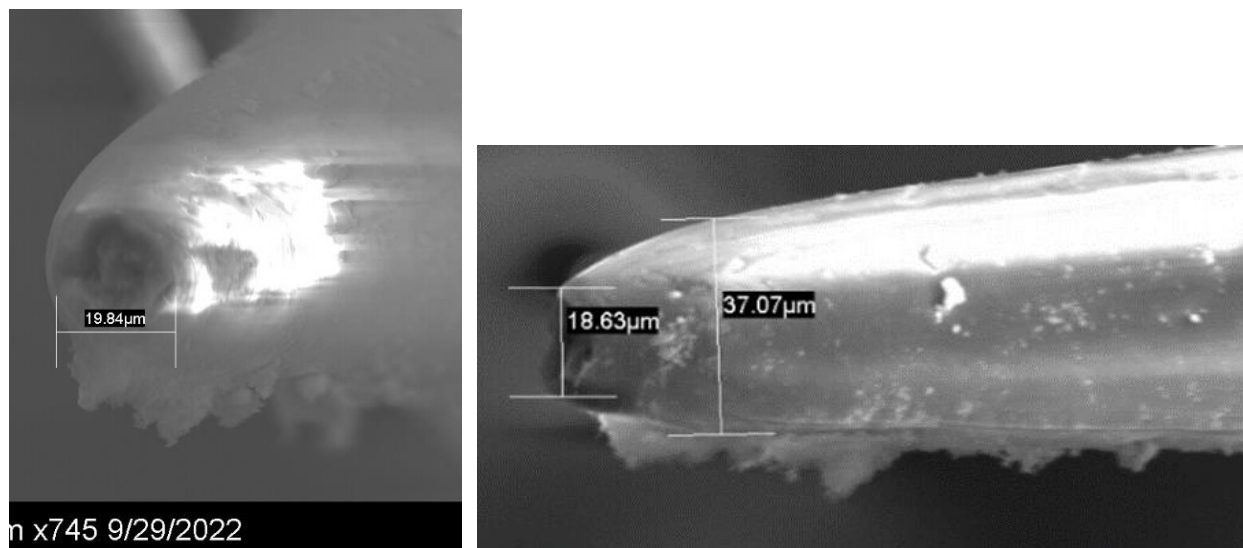


Figure 4-27 The micromanipulator tip is shown. The nano size Lif particulates are shown accumulated in the tip.

Ultimately, the wafers used in the ARGANOT modules were backfilled with an ultra-sonic method. The wafers were sonicated for one hour with 333 mL methanol and 10g of treated ^6LiF . The sonicated backfills produced negligible damage to the semiconductor diodes, while maintaining a moderately consistent backfill packing fraction. Following sonication, the wafers were dried, weighed, and then packed by hand with methanol-wetted treated ^6LiF , using silicone brayers. Devices that were sonicated and roller backfilled produced higher packing fractions compared to those that just underwent a sonification backfill. The two devices shown in Figure 4-28 demonstrated the difference with the dual backfill processed wafer (left) having a calculated packing fraction of approximately 43% whereas the sonification diode (right), was found to have a packing fraction of only 18%. Therefore, a combination of sonification and the ink press were implemented for the backfill due to higher packing fraction outcomes that were more consistent and inflicted limited damage to the devices. The average resulting packing fraction for the dual

sided diodes used in the ARGANOT modules with an estimated 550-600 μm trench depth, an approximately 10 μm width fin, and approximately 20 μm wide trenches was determined to be anywhere between 30-40%, depending on the wafer and the diode location of origin from the wafer [109]. MCNP based simulations indicated that the expected intrinsic thermal neutron detection efficiency would be approximately 50% or greater which would meet the initial design goals for the DS-MSND development [110].

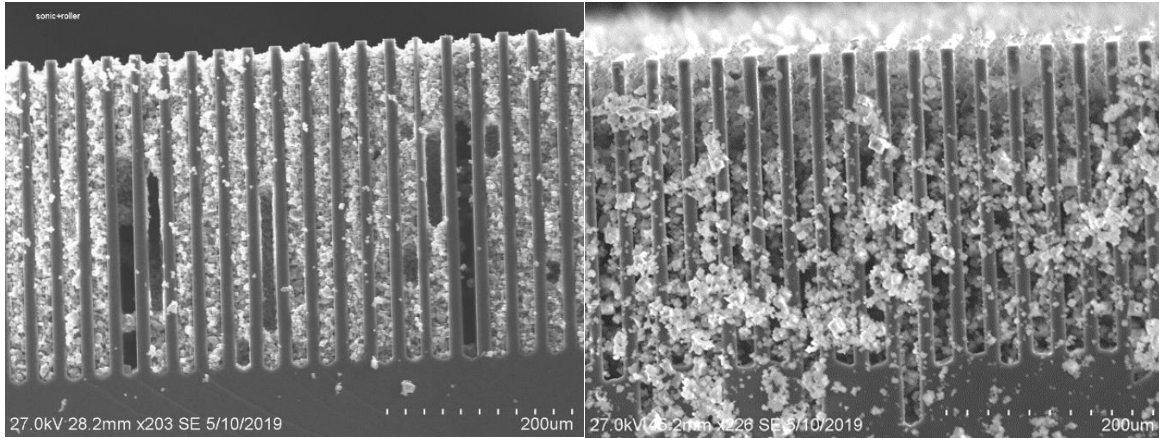


Figure 4-28 The backfill of single diode after undergoing a sonification followed by an ink press method (left) and the backfill of a single diode after only undergoing sonification (right) is compared.

4.2.5 Packaging

Following the backfill process, the diodes were mounting into a case to protect the silicon trenches from damage, make electrical connection, and shield any external radio frequency (RF) signals from interfering with the device signal. In the beginning of the ARGANOT project, it was assumed that the procedure for packaging the single sided MSND devices could easily be adopted for DS-MSND because of how perfected the packing process for MSND had become. With the MSND devices, the detector was placed with the backside bare silicon substrate directly adhered to the custom designed ceramic detector boards (CDBs) with a silver epoxy. A gold wire bond was then used to connect the singular detector bond pad to the CDB case. An alumin RF shield was then placed and sealed with an insulating epoxy to the CDB, concluding the packing stage.

Immediate challenges were presented in packaging the DS-MSND devices as result of the 2020 and 2021 supply chain crisis when CDB were not available for the DS-MSND devices. In place

of the custom designed boards, printed circuit board (PCB) were instead used. There were several problems with the PCB platform depicted in Figure 4-29 such as the lack of vertical wall support. The lack of horizontal constraint made placing the device in a uniform fashion difficult. Not only did the lack of placement uniformity affect the ease of placing the two wire bonds from the DS-MSND to the much lower PCB platform, but the device edges and the wire bonds were left exposed. Wire bonds and devices were regularly discounted from the PCB platform and were unreliable to deploy in the field. Additionally, the PCBs were found to be extremely flexible, which also led to broken wire bonds and cracked DS-MSNDs. To further protect the wire bonds from contacting the RF aluminum shield, a hydrogenous based clear epoxy was coated over the bonds and around the edges of the devices while mounted in PCB. It was however determined that the epoxy produced unacceptable neutron scattering and clearly decreased the performance of the devices.

To remedy the problem, the team began to use Humi-Seal spray-on sealant to insulate the DS-MSND/PCB assembly prior to mounting the outer RF shield as well as purchased a thicker PCB while the project waited for a cost-effective CBD design with vertical wall support to securely hold the device in place to arrive.

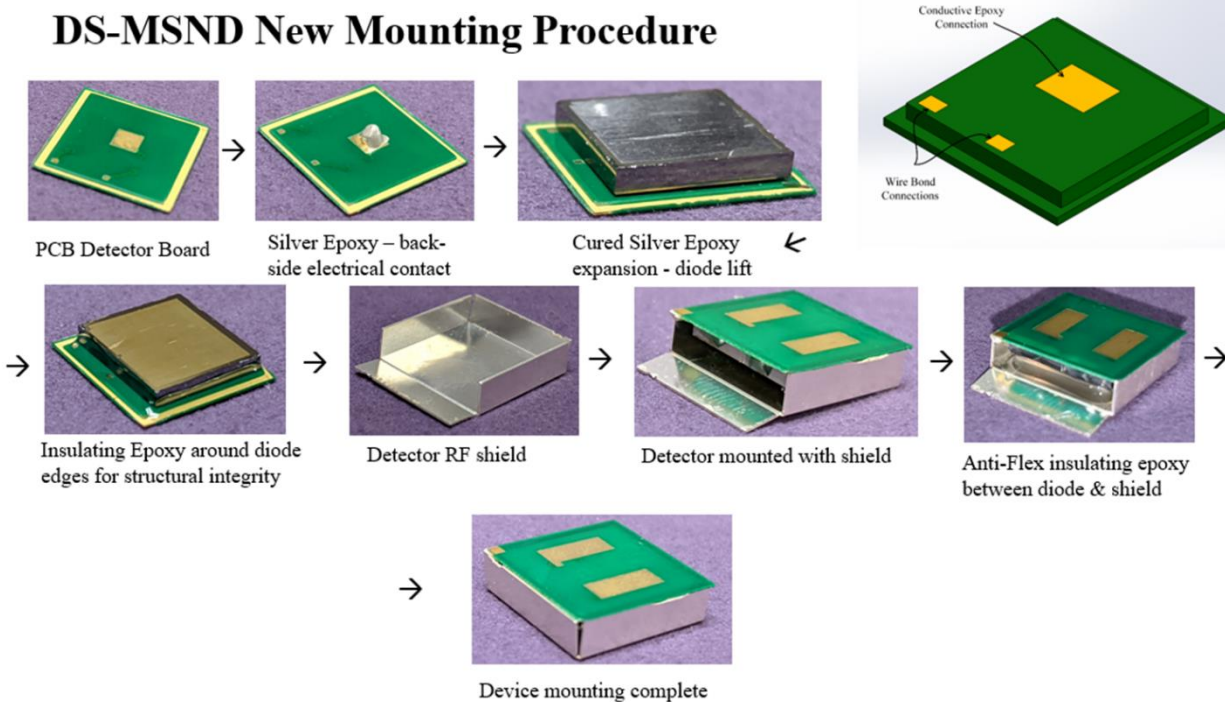


Figure 4-29 The improvised DS-MSND mounting procedure is described.

Furthermore, an extra layer of complexity the dual sided trenches introduced to packing the assembly was realized almost immediately when the same silver epoxy procedure was adapted to the DS-MSND devices. It was discovered through IV testing throughout the mounting process that the silver epoxy was coating the bottom set of trenches [111]. Once cured, the epoxy expanded within the bottom trenches, causing fins to break or crack, resulting in an increased leakage current. Figure 4-30 below demonstrates how the silver epoxy was penetrating the trenches and curing. To remedy the problem, an insulating dry film coat was applied to coat the trenches from silver epoxy, leaving a small in the center of the device along with the two bond pads on the front side of the device.



Figure 4-30 The silver epoxy used to adhere the device to the PCB was found to leak into the trenches. Upon heating, the epoxy would cure and expand within the trenches of the devices, increasing the leakage current.

The application and removal of the dry film in these areas proved difficult. As shown in Figure 4-31, residual dry film remained on the bond pads even after attempted removal. The XRF (X-ray fluorescence) images and plot of counts per the detected elements for both the bond pads and the device center demonstrate the large amount of dry film, as indicated by the large carbon presence (red), and silicon dioxide as indicated by the presence of oxygen (green), that remained on the surface, see Figure 4-32 and Figure 4-33. The unwelcome presence of the dry film and oxide acted as an insulator and made electrical contact nearly impossible.

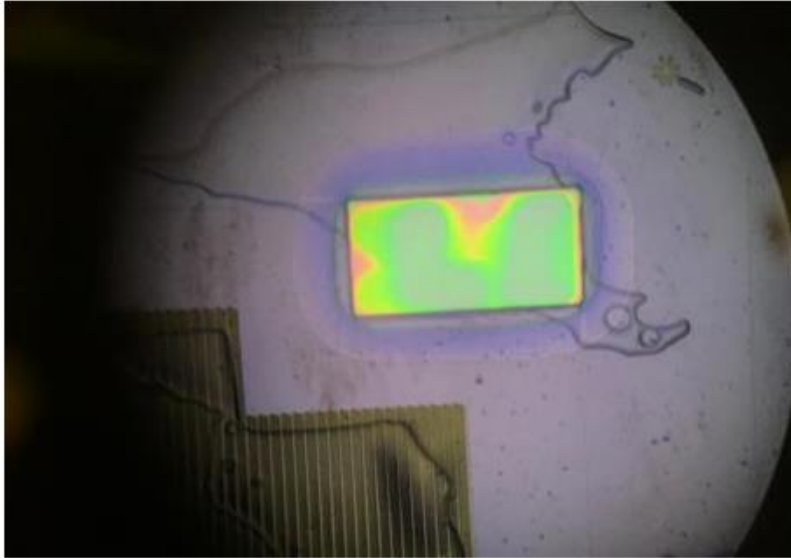


Figure 4-31 The residual dry film and oxide is visible on the n-dot bond pad.

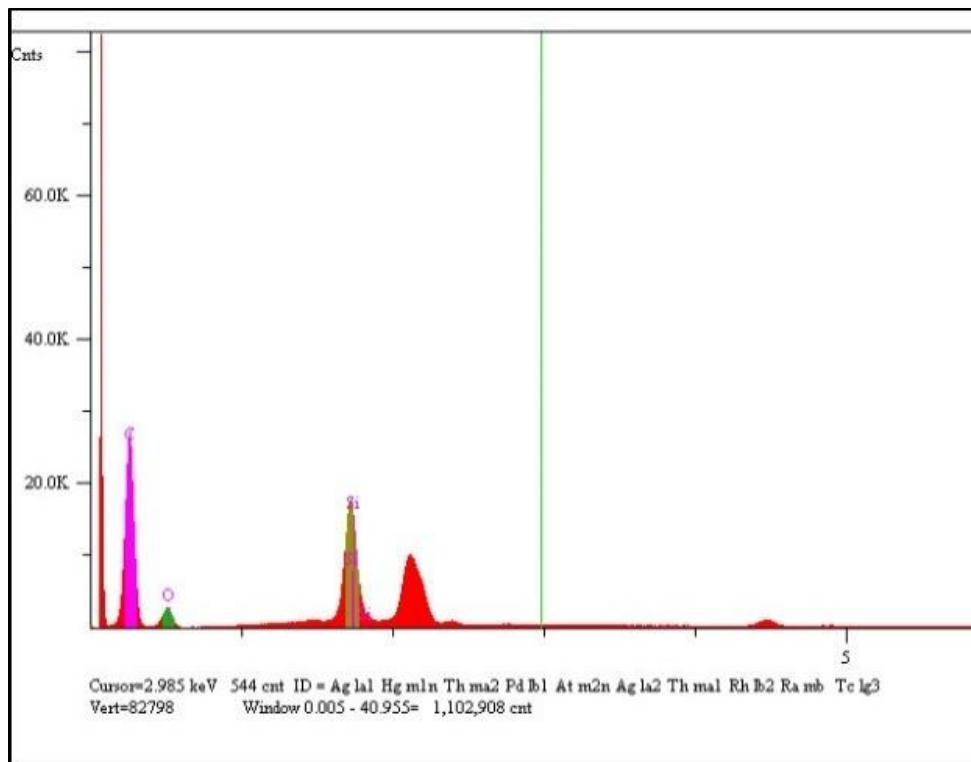


Figure 4-32 The XRF analysis of the bond pad area confirms the presence of dry film and a silicon oxide layer.

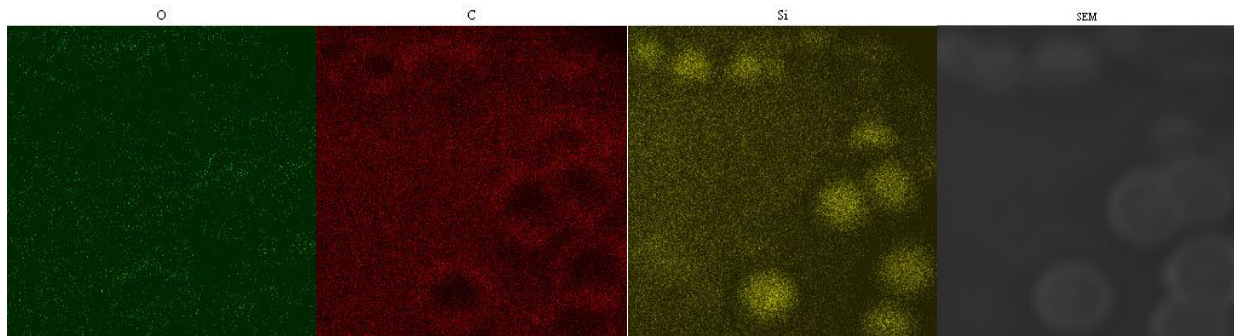


Figure 4-33 A fast map of the bond pad using the XRF techniques reveals the location of the bare silicon, the dry film, and the silicon oxide is shown.

In the left SEM images of a bond pad in Figure 4-34 below, residual dry film was clearly present. In the middle image, the former location where two attempted wire bond connection were made is visual due to the clear missing residual dry film patches present beside each present wire bond. Such images acted as evidence that the wire bond would not be able to adhere to the dry film coated surface, yet alone convey a current. The far-right image demonstrates the bond pad with two wire bond connections, surrounded by dry film over top trenches.

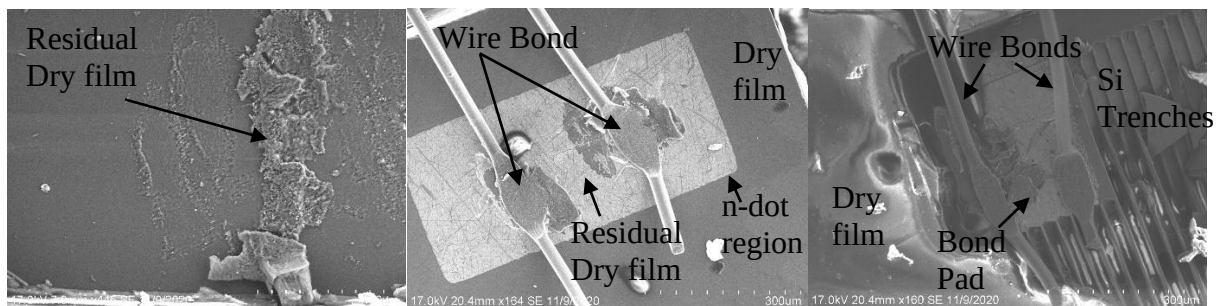


Figure 4-34 The residual dry film was present on the bond pad of the device, making it difficult to securely bond the device to the casing.

When comparing the ease of removal of dry film from the bond pad to the encapsulating Si trench region, the center trench region proved much easier to remove due to the poorer adhesion of the dry film to the trenches as shown below in Figure 4-35. However, there still appeared to be residual silicon oxide as reflected in Figure 4-36 and Figure 4-37. The dry film encapsulation, however, proved to be quite helpful in preventing the leaching of the silver epoxy on and into the trenches of the device. The gold evaporated contacts combined with the dry film encapsulation drastically aided in reducing leakage currents.

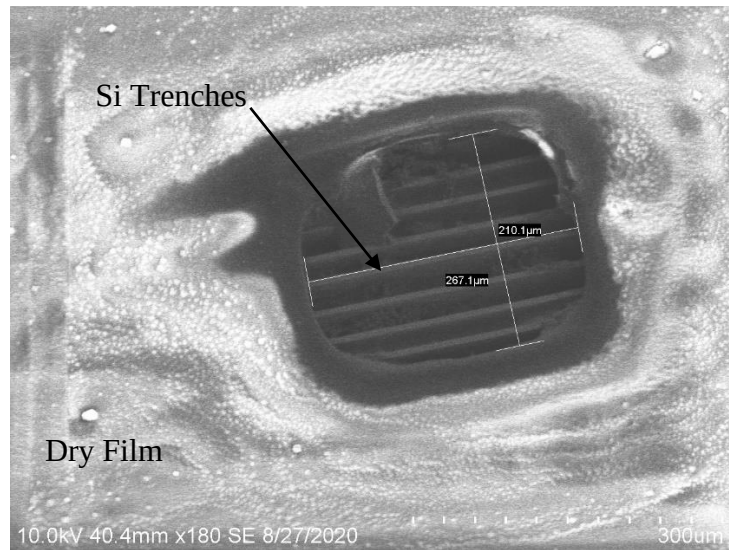


Figure 4-35 The bottom of the DS-MSND device in which a small opening was created to bond the bottom of the device is shown.

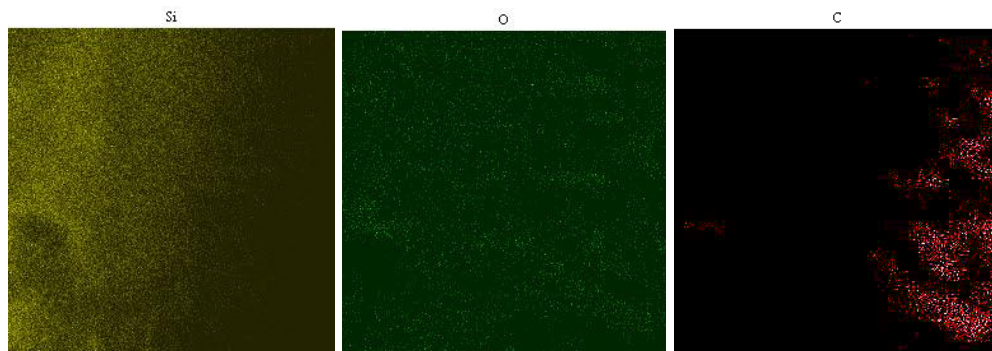


Figure 4-36 The XRF fast map of the bottom of the device where the dry film and the silicon oxide should have been removed is shown.

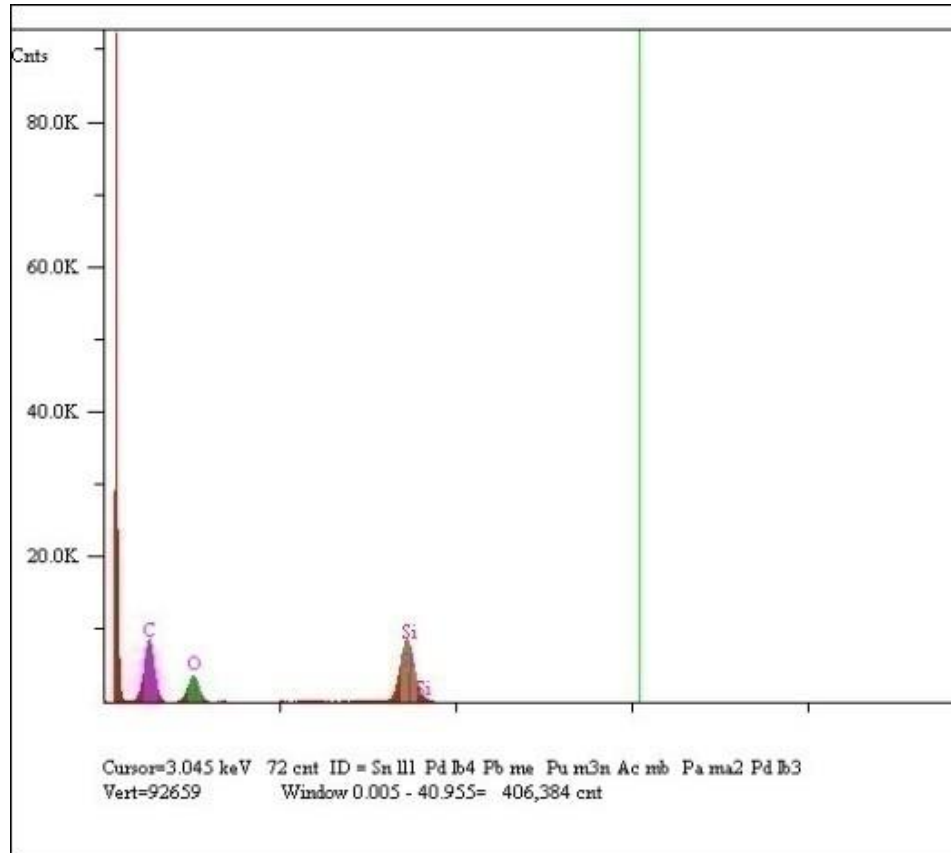


Figure 4-37 The XRF analysis of the bottom bond pad is shown.

Following the identification of the foreign material on the bond regions, additional precautions were added to the production process to ensure removal such as increased dry film exposure, developer, and oxide etching times. Additionally, the wire bond mechanism was further strengthened by bonding to conducting gold metal surface. To do so, the E-beam evaporator was used to evaporate a layer of titanium onto the bare silicon and then an outer layer of gold on top of the titanium. Once the evaporation was complete, the wafer underwent a “metal lift off” in which the dry film photoresist with the layer of Ti+Au on its surface was stripped away from the wafer, leaving behind only the metallized contacts in which the dry film over top was previously removed.

4.3 Characterization

Throughout year two of the ARGANOT project, 10 silicon wafers produced 520 DSMSND diodes in June and July of 2021. Of those 520 devices, 290 were successfully processed and packaged into functional DS-MSNDs. Of the 290 completed DS-MSND devices, 82 failed

connectivity testing, due to either dry film contamination or broken wire bonds. Multiple factors contributed to this 55% yield, including potassium hydroxide etching errors, failures caused during dicing, and mounting issues. The mounting issues include wire bond breakage and separation of the diode from the mounting PCB. Of the 202 completed DSMSND devices that passed initial connectivity testing, 127 exhibited high electronic noise and/or high leakage current that render the devices unsuitable for use in the ARGANOT modules. Approximately 75 DS-MSNDs were then characterized to be used in the ARGANOT modules.

4.3.1 Electrical Properties

The metallized contacts not only improve wire bond adhesion, but also contributed to a lower leakage current and better signal than bare silicon contacts as observed in Figure 4-38. Figure 4-39 shows the reaction product spectra for a moderated ^{252}Cf source for DS-MSNDs with only the backside trenches connected. The device was tested using NIM electronics. The shaping time used in both measurements was 12 μs . The “bm” detector had the native silicon-oxide stripped on the bottom-side bond area and metalized to prevent re-oxidation before mounting into the CDB. The strip and metallization steps improved the signal to noise ratio of the DS-MSND. Figure 4-40 shows the reaction product spectra for DS-MSNDs with front-side only, backside only, and both front and backside connection. These DS-MSNDs had the backside metalized. The number of counts with both sides of the DS-MSND connected is twice the number of counts of only one side connected (within a standard deviation), which was the expected result. By removing the native oxide and metalizing the backside connect, both sides of the DS-MSND could be readout simultaneously with single polarity pulses.

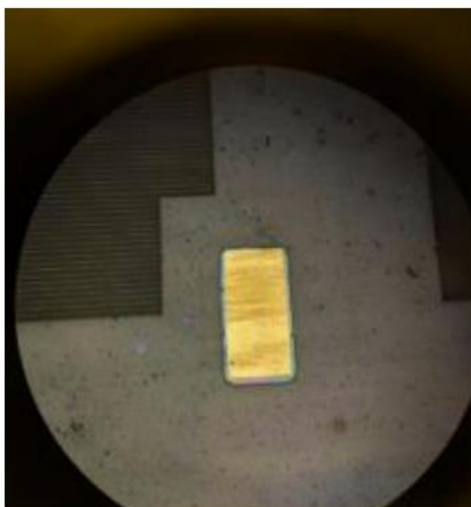


Figure 4-38 The gold lift off on the n-dot bond pad is shown.

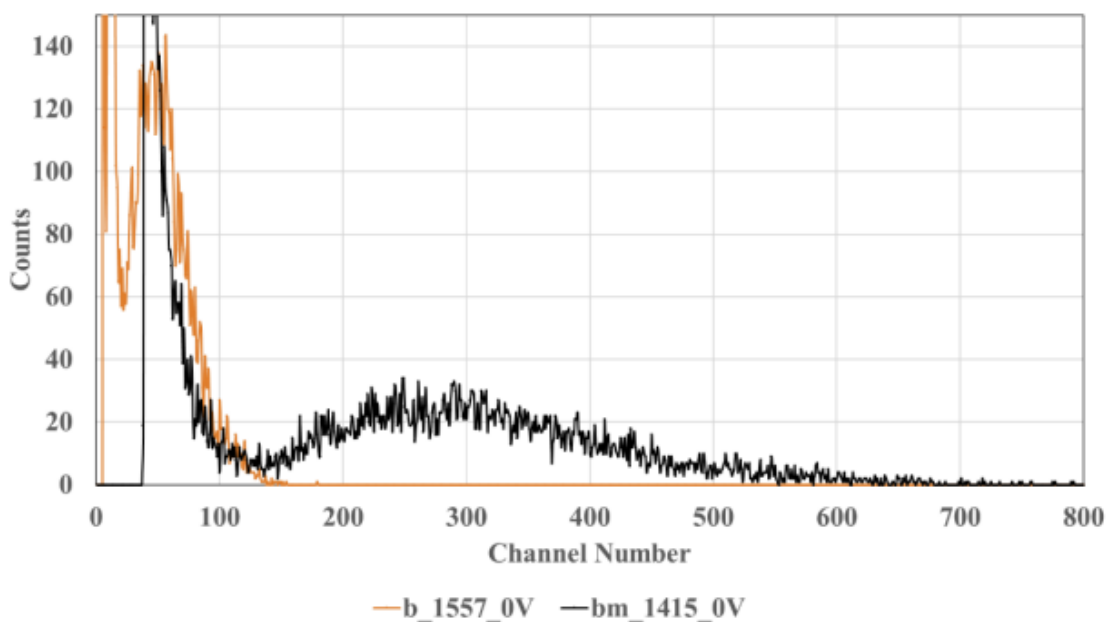


Figure 4-39 Reactor product spectrum for DS-MSND with only the backside trenches connected. The orange line shows a DS-MSND that was silver-epoxied to the CDB without prior metallization. The black line DS-MSNDs had the native oxide etched and then the backside DS-MSND contact was metallized to prevent oxidation. Removing the oxide appears improves the signal by removing the resistive oxide layer from the circuit

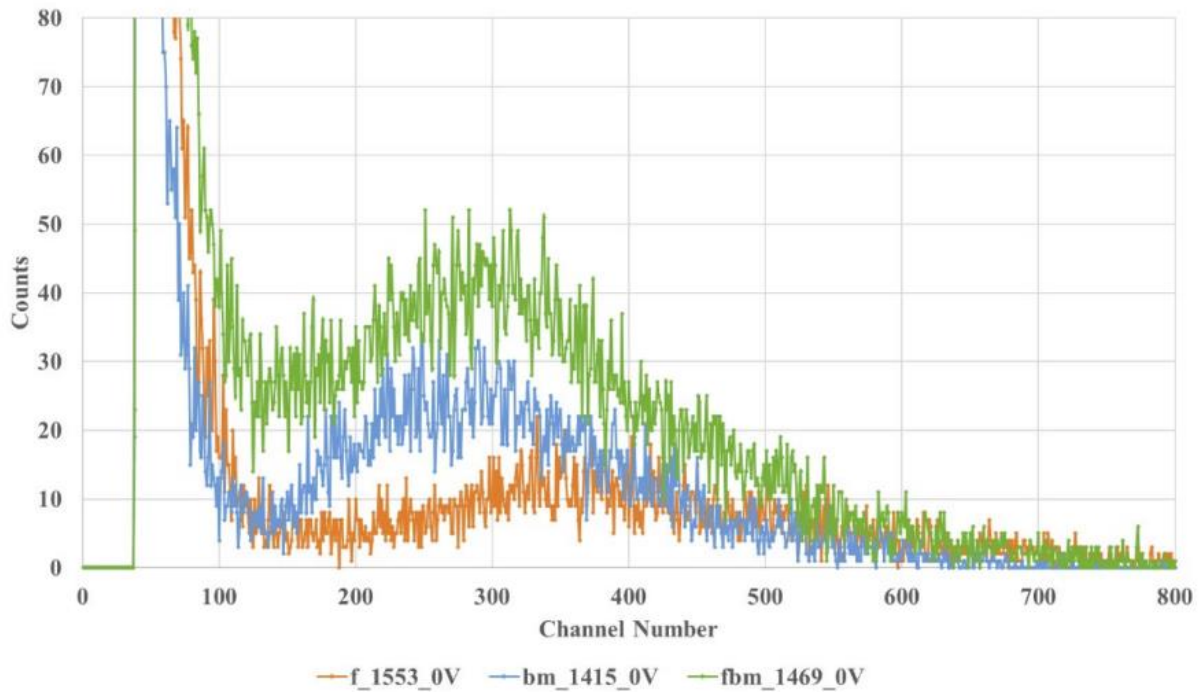


Figure 4-40 Reaction product spectra for front-side only, backside only (metallized), and full DS-MSND electrically connected to a CDB. These results show that we can now readout both sides of the DS-MSND simultaneously.

Throughout the packaging process, it was determined that following each step, a quality control IV measurement should be performed. Not only did this additional step immediately identify devices that were no longer viable detectors, but problems throughout the process were immediately identified and addressed. Table 4-3 below demonstrated the leakage current measurements performed on five devices made at the end of the DS-MSND processing research and development stage. Notice that the leakage current drastically increases with the wire bond attachment but were silenced to tolerable levels with the RF shield.

While the processing quality control study was able to identify problems in the packaging that would contribute to a reduced yield, the wafers were still entering the packaging step with a higher than desired leakage current. Therefore, the investigation and mitigation of the leakage current resulting from the dopant schemes was a top priority. To do so, various dopant schemes were studied to understand if the leakage current could be reduced simply through altering the dopant recipe. As part of the study, two dopant schemes were tested and compared for the DS-

MSND n-ring patterned wafers. The four wafers used for the study were processed together up to that point, so all variables (such as the oxide thickness, photolithography quality, etch depths and qualities, etc.) were assumed to be equal.

Table 4-3 IV is reported in nA at -3V and -10V following various steps in the packaging process.

Device ID	IV After n-ring Window & Silver Epoxy		IV After Top Side Trench Wire bond		IV After Epoxy and Shield Application		IV After Neutron Testing	
	-3 V	-10 V	-3 V	-10 V	-3 V	-10 V	-3 V	-10 V
1	15	24	24	42	22	39	26	46
2	20	34	34	58	35	59	49	80
3	4	6	130	600	10	150	5	8
4	24	50	350	170	25	61	23	43
5	5	6	17	28	30	50	18	29

The recipes may be viewed together in Table 4-4 and Table 4-5 and varied only in the time and temperature of exposure to the dopant source. The longer exposure times at high temperatures were thought to have created a more conformal dopant scheme within the trenches. Though simulation performed by Ochs 2020 had indicated that partial conformal patterns yielded the best performance in terms of charge collection, the devices produced struggled with higher-than-expected leakage currents. To verify the simulation results, a comparison was performed between what at the time was thought to be a conformal doping scheme developed and discussed in Fronk 2017 (recipe 14, [112]) and the partial conformal doping scheme that was consistently used through the modern DS-MSND production (recipe 22). The idea at the time was that the longer “Drive in” time and gas flow rates would mean more mass transfer from the Boron dopant source wafer and would therefore result in a greater Boron deposition. A longer and quicker mass transfer was expected to create a more conformal doping pattern. Recipe 14 specifically called for a twice as long “Drive in” time, however at a lower gas flow rate of 6 liter per minute (lpm). The gas flow rate was considered important as well for controlling the deposition rate. Once in the furnace, the N₂ and O₂ gas, which would facilitate the Boron mass transfer would sweep over the individual wafers in what was perceived to be a turbulent flow.

Table 4-4 The recipe for the shorter drive in time is shown.

Step time (min)	Zone Temperature (C) Flow	Gas Flow (LPM)	Rec #22 Run 2 wafers for production-pvp ndot-DSMSND (1.5mm) front and back. Goliath1.5 Series.
15	700	8	Load
10	800	8	Ramp up
5	800	8	Stabilize N2 O2
15	800	8	Forming Gas
20	975	8	Ramp up to drive in
60	975	8	Drive in
30	650	8	Ramp down to idle
0	400	8	Idle

Table 4-5 The recipe for the longer drive in time is shown.

Step time (min)	Zone Temperature (C) Flow	Gas Flow (LPM)	REC #14 (Anneal Fronk) 2- pvpndot-DSMSND (1.5mm) front and back. Goliath1.5 Series. Close Boat
15	800	6	Loaded wafers N2
3	800	6	Stabilize N2 O2
30	800	6	Forming Gas
20	1000	6	Ramp up in N2
120	1000	6	Drive in with N2
30	700	6	Ramp down in N2
0	400	6	Idle

Two wafers were treated with each recipe of interest and then compared. Four randomly chosen devices across each wafer were then used to sample an IV curve. The IV at -3V was also recorded for 16 randomly measured devices across the various wafers, see Table 4-6.

Initially, results indicated that recipe 14 produced lower leakage current in viable devices than recipe 22. However, when yields per recipe were examined, only 50% of the devices measured from recipe 14 were useable, compared to a 63% yield by recipe 22. For that reason, recipe 22 was maintained as the preferred doping recipe for the DS-MSND n-ring devices.

To determine acceptable operating biases, the leakage current as a function of operating voltages were plotted and may be viewed in Figure 4-41 and Figure 4-42. The gray IV curves on each plot represent viable devices while the devices curves highlighted by the red and blue represent nonviable devices. These devices with excessive leakage current at the time were considered damaged and little attention was paid to understand the causation behind their failure. Recent work conducted by Radiation Detection Technology (RDT) has indicated that the devices may have been over doped into a degenerative state. Further investigation of the initially collected IV curves from each recipe revealed that the DS-MSND of both recipes were almost certainly over doped. In the IV curve of recipe 14 shown in Figure 4-41, the four devices in which the leakage current drastically increases with a reverse bias highlighted in blue, indicated that the fermi energy level has likely exceeded the energy of valence band. The same drastic increase of leakage current, was also present in the IV curves of recipe 22 in Figure 4-42 and are highlighted in blue as well. The red IV curve of recipe 22, which was initially overlooked provides additional evidence that the wafer was drastically over-doped. As shown in Figure 4-43, an excess amount of the boron dopant was introduced to the wafers and created a surplus of electron charge carriers. The surplus forced the fermi level to exceed the energy of the valence band as previously mentioned. As result, the band gap between the conduction and valence band became quite small, which would easily allow for electron tunneling through the junction. The tunneling would undoubtedly contribute significant leakage current to the devices, which drastically worsens with increasing bias. However, an interesting artifact within the red IV curve was that as the reverse bias increased, the leakage current suddenly rebounds, and then once again plummets. This feature may be explained by once again referencing Figure 4-43 in which once moving to voltage 2 from voltage 1, the size of the band gap grows. With the growth, the charge carriers are no longer able to tunnel through the junction, thereby limiting the leakage current. As the voltage is increased, the leakage current worsens again. This explanation aligns with the data collected from recipe 22.

Table 4-6 The leakage current at -3V for many diodes across two different devices, treated with dopant recipe 14 (left) and 22 (right) are shown.

Leakage Current (amps) at -3V	
Recipe 14	Recipe 22
-7.48E-07	-1.49E-06
-8.03E-07	-1.91E-08
-4.61E-08	-2.26E-08
-7.62E-07	-6.20E-07
-1.48E-08	-3.18E-08
-2.37E-08	-9.05E-08
-1.24E-08	-4.79E-08
-1.67E-08	-2.18E-06

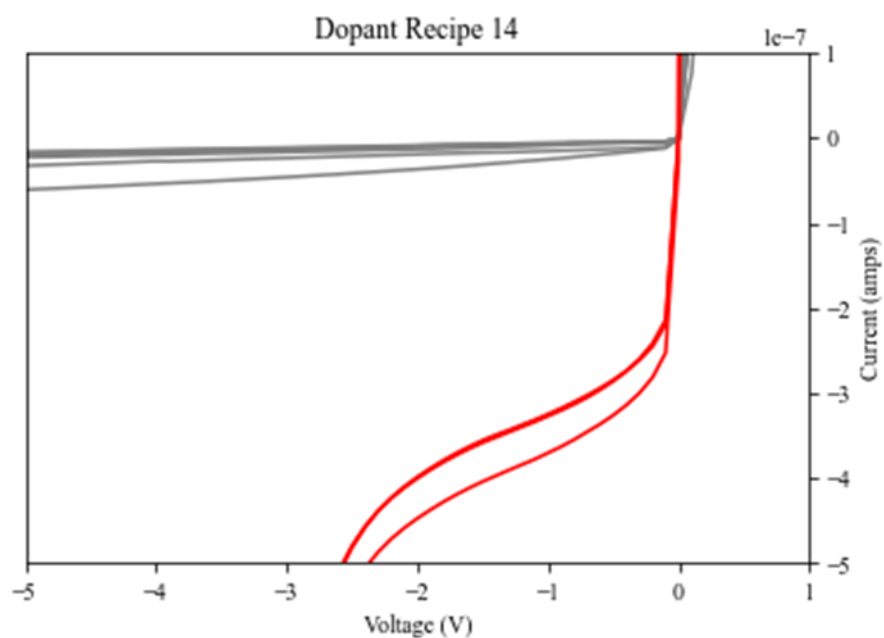


Figure 4-41 The IV curve of several diodes from a wafer treated with dopant recipe 14 is shown. The red heightened curves indicate the devices were unusable at any voltage.

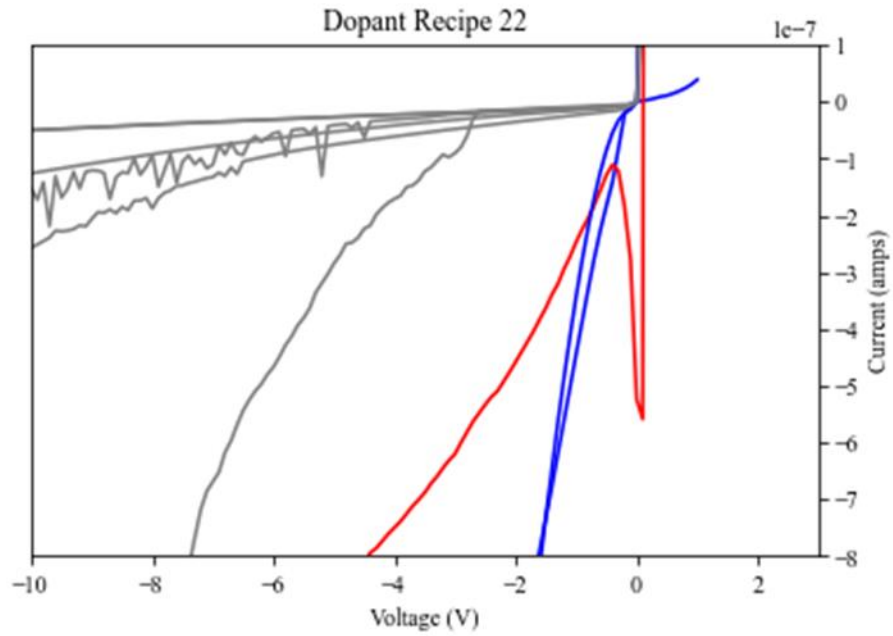


Figure 4-42 The IV curve of several diodes from a wafer treated with dopant recipe 22 is shown. The blue highlighted curves indicate those diodes were unusable while the diode highlighted in red indicates the diode was operating in a degenerate state.

To eliminate the excessive dopant source disposition that was leading to degenerate semiconductors, RDT began to conduct work to understand the relationship between flow rates and dopant conformity. It is theorized today that the flow rate of recipe 14 at 6 lpm may have in fact created a more laminar flow boron deposition and therefore eliminated any turbulence against the wafer face that may have been present with recipe 22. It is hypothesized that the laminar flow is less disruptive to the Boron mass transfer and may have allowed the dopant to drive deeper into the wafer trenches which would lead to a more conformal dopant scheme and possibly introduced an excessive amount of boron to the wafer. The difference between the gas flow rate of recipe 14 and 22 in addition to exposure times, likely contributed to the over-doping of the silicon wafers.

Reverse Bias Where $V_1 > V_2$

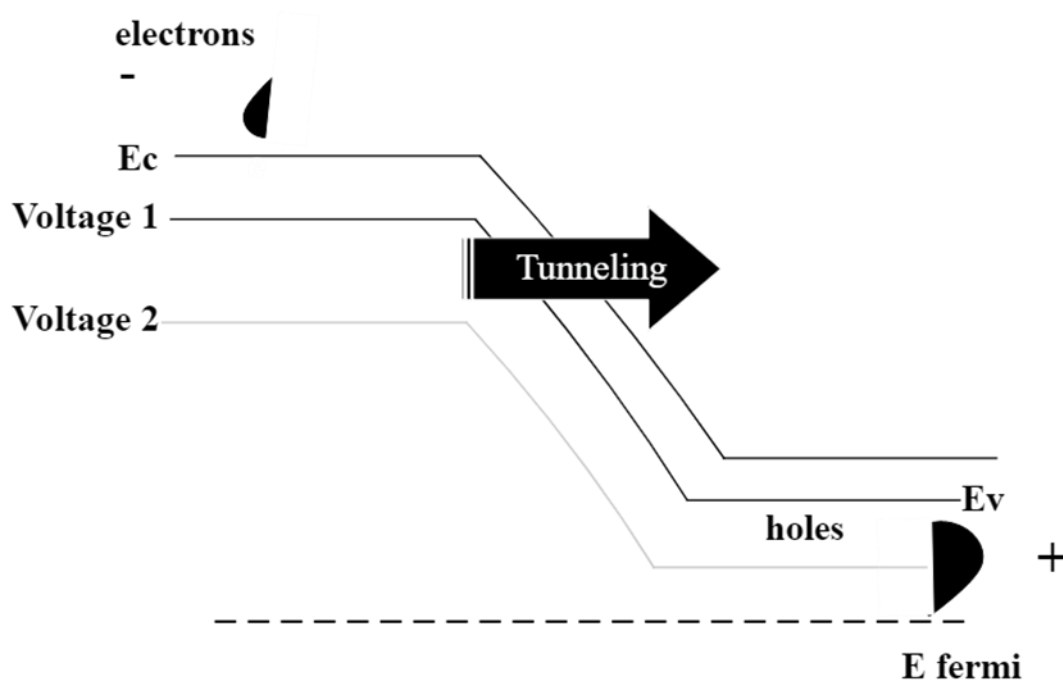


Figure 4-43 Within a degenerate diode, the fermi level exists outside of the bandgap and therefore charge carrier may easily tunnel through the gap, resulting in high leakage currents that fluctuate with voltage as the band gap shifts with voltage.

Presently, how deep the generative region is, along with how far down the trench the dopant goes is unknown. It is possible that the heavily doped region may have been segregated to the top of

the fins as reflected in Figure 4-44, leaving a large proportion of the fin and bulk silicon undoped. If the doped regions were segregated, the device would likely have a very limited active region, which would contribute to a poor charge collection and as result poor efficiency.

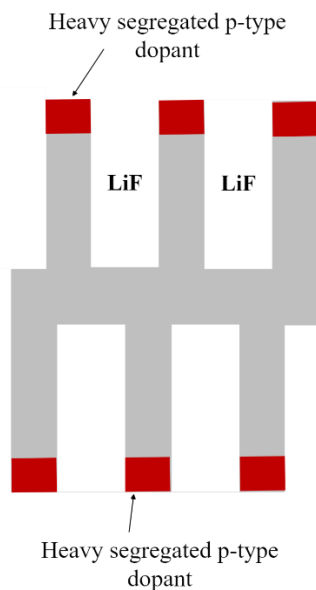


Figure 4-44 The heavy doping recipe likely resulted in a segreamated p-type region close to the wafer surface.

4.3.2 Detector Response

The expected intrinsic neutron detection efficiency from simulation based on the DS-MSND fin trench geometry and the ^6LiF backfill packing fraction was ~50-60%. However, the measured detection efficiency was similar to a single-sided MSND at approximately 25-40%. Additional measurements were performed to determine the cause of the decreased efficiency after it was suspected that only the backside trenches were contributing to the signal, and interactions from the front-side (which also contains the n-ring) were not being measured. Figure 4-45 shows the results from measurements where the DS-MSND was readout through a 142A preamplifier and 2022 amplifier with a shaping time of 12 μs . The “b” and “fb” labels in the legend denote whether if only the “b”ackside (black) or “f”ront and “b”ackside (gray) were electrically connected, respectively. With only the backside connected, a clear noise floor and neutron peak was observed. When both sides are connected, the noise floor increases, and the neutron peak widens. The areas under the neutron peak were approximately equal which may indicate that

interactions in the front-side trenches were not adding to the spectrum and therefore not being measured.

The measurements for the “b” and “fb” DS-MSNDs were then repeated with the detectors connected to the ARGANOT analog electronics. The shaping amplifier output was connected to a multi-channel analyzer. The results of the measurements are shown in Figure 4-46. In these measurements, the only additional counts in the pulse height spectrum for “fb” (gray) vs “b” (black) detectors appear in the low energy channels (5-20). The discriminator for these detectors is usually set between channels 25 and 40. The area in the upper peaks are similar, but their shapes changed depending on if one or both sets of trenches are electrically connected. Due to time constraints of the ARGANOT delivery date, the n-ring devices which appeared to operate similarly to a single sided device were still used within the ARGANOT modules.

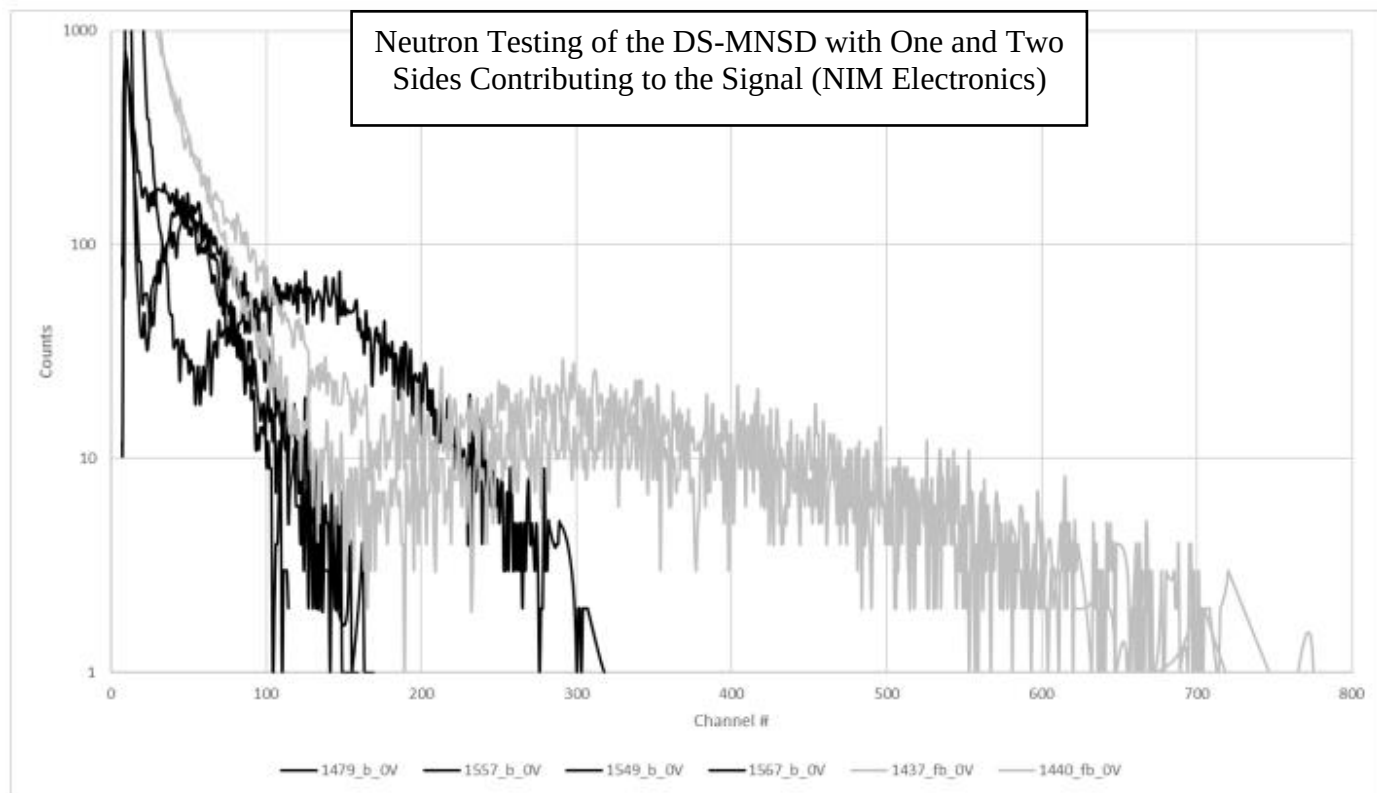


Figure 4-45 Neutron source measurements performed with 142A preamplifier - 2022 amplifier.

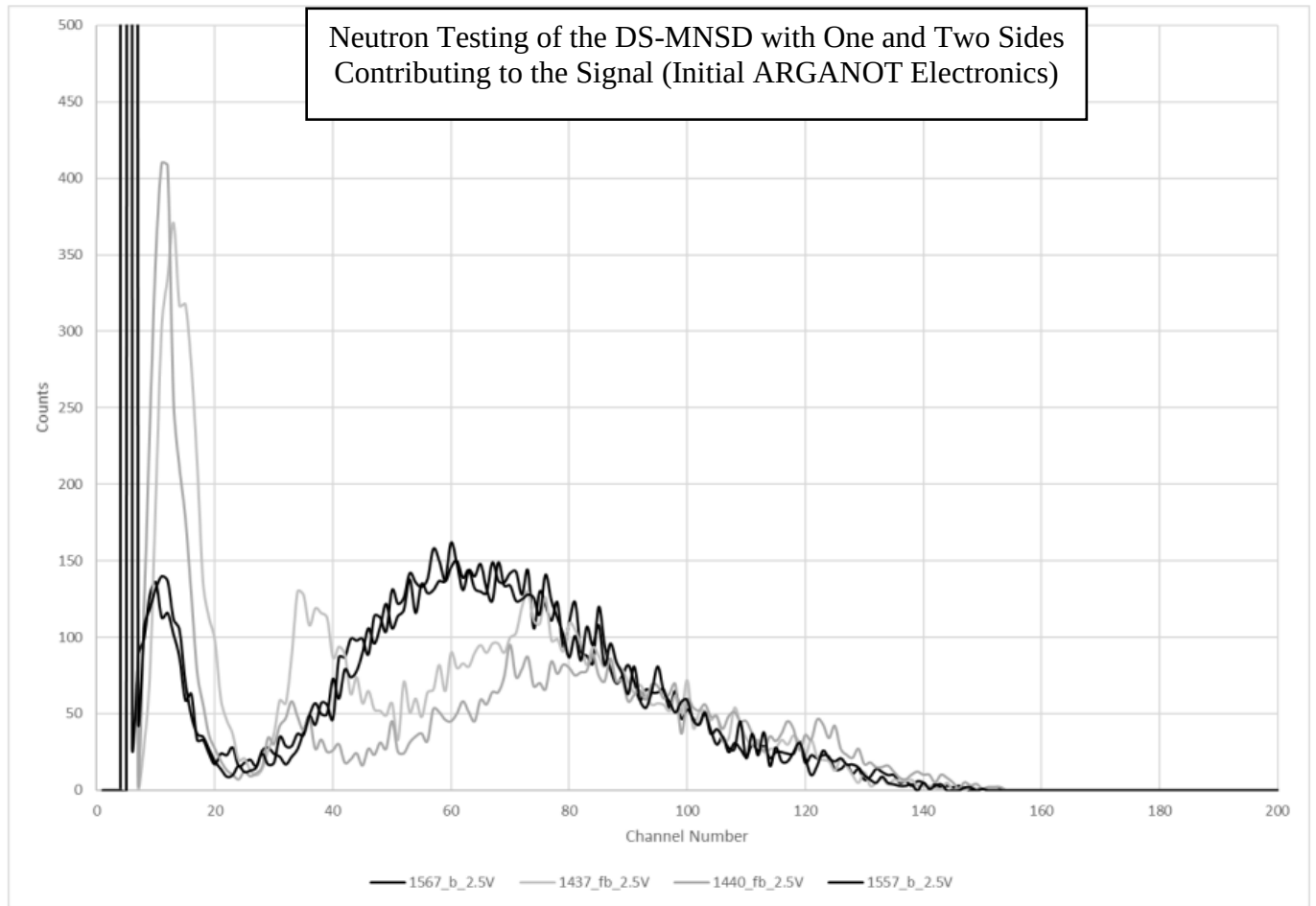


Figure 4-46 Neutron source measurements performed with Y2 ARGANOT electronics. “b” detectors were connected only on the backside p-type contact and the n-ring. “fb” detectors were connected to backside and frontside p-type contacts and the n-ring. The shaping time used within the ARGANOT electronics was 21 μ s.

Three factors as to why the devices behaved similar to single sided devices have developed up to this point. The first theory is related to the degenerate doping scheme previously discussed. As mentioned, had the dopant been segregated to a small portion of the trench fins, less signal would be collected from the device. It was suspected that the depletion region only extends 10’s of microns beneath the p-type dopant in our DS-MSNDs meaning the diode is not fully depleted between the p-type trenches and the n-ring. The second factor that may have impacted the DS-MSND performance was that a large amount of undoped material existed in between the front and backside of trenches, are reflected in Figure 4-46. In this situation, the un-depleted Si region between the p-type and n-type contacts would act as a resistor with a value proportional to the resistivity of the Si, resulting in a deteriorated device performance. To reduce the amount of

current lost in the Si bulk resistor, lower resistivity Si can be used. However, as resistivity decreases the depletion width for a given bias also decreases.

The third factor that was only realized years after the manufacturing of the DS-MSND devices was completed was the effects of the SiO₂. The SiO₂ was used to insulate the surface of specific regions of the DS-MSND device such as between the trenches and the bond pads [113]. As discussed in Sharma 2022, a positive space charge was formed inside the oxide film due to fixed oxide charge during the thermal oxidation process [114]. These charges induce negative image charges in the Si substrate and resulted in the accumulation of n-type carriers near the Si-SiO₂ interface. As result, Sharma reveals that the fixed oxide charge in an n-type substrate decreased the resistivity of the substrate around the Si-SiO₂ interface and increased the required bias to deplete the diode [115]. The required bias to fully deplete the device when factoring in the effects of the fixed oxide charge was likely not able to be reached due to the leakage current limitations of the device.

4.3.3 Efficiency Measurements

To determine the intrinsic thermal neutron detection efficiency, the northwest thermal diffracted beam port at the Kansas State University's Triga Mk II nuclear reactor was utilized to perform the initial characterization measurements. At 50 kW, the thermal neutron beam was moderated to approximately 0.5 mm in diameter through the use of a cadmium (Cd) filter. A He-3 tube (8 cm by 5.25 cm diameter) with an intrinsic thermal neutron detection efficiency of $80.7 \pm 0.5\%$ was placed approximately 12 inches from the exterior of the beam port. A measurement with the Cd filter fully closed was collected for 120 seconds. A secondary 120 second measurement was then collected with the Cd shutter open. The He-3 measurements were then used to determine the efficiencies of the five DS-MSND devices. The DS-MSND devices were then placed in the same location as the He-3 tube with an electronics testing case holding each device in place. The same Cd shutter open and Cd shutter closed measurements were then conducted for each DS-MSND device. The efficiency was then determined on average to be $34.2 \pm 0.61\%$ for those five devices and may be viewed in Table 4-7.

Table 4-7 The results of calibrating four DS-MSND devices at the NW beamport against a He-3 with a known efficiencies are shown.

Device	shutter closed	shutter open	Time (s)	Efficiency
329	123 ±11.09	7875 ±88.7	180	41.53 ±0.73%
316	129 ±11.36	6568 ±81.0	180	34.49 ±0.64%
323	111 ±10.54	4604 ±67.9	180	24.07 ±0.49%
406	107 ±10.34	5853 ±76.5	180	30.78 ± 0.58%
Average Efficiency 34.2%± 0.61%				
He-3	8753 ± 91.56	23705± 153.96	180	80.10%
	Flux He	103.70		

The devices for which the efficiencies were known were then able to be used to calibrate the other DS-MSND devices also according to the McGregor and Shultis method using the configuration shown below in Figure 4-47. Within Figure 4-47, the device was placed upon the platform of the test box and a high-density polyethylene moderate cast was placed over top of the device [116]. The 12.3-ng (28,500 n/s) ²⁵²Cf neutron radiation source was then placed into the moderator cast pocket. In order to ensure accuracy of results, the position of the device with respect to the source was consistently maintained. A foreground and background measurement were taken using the calibrated DS-MSND device of interest and was then efficiency recorded. Table 4-8 below displays the top ten highest efficiency detectors from the batch along with the recorded leakage current. Only devices with acceptable leakage currents were tested as potential devices to populate the ARGANOT modules. As shown in Table 4-9, only 11 devices with an average leakage current of 2880 nA were above the 40% thermal neutron detection efficiency limit. At 30% efficiency, however, 13 devices were accepted with an average leakage current of 276 nA.

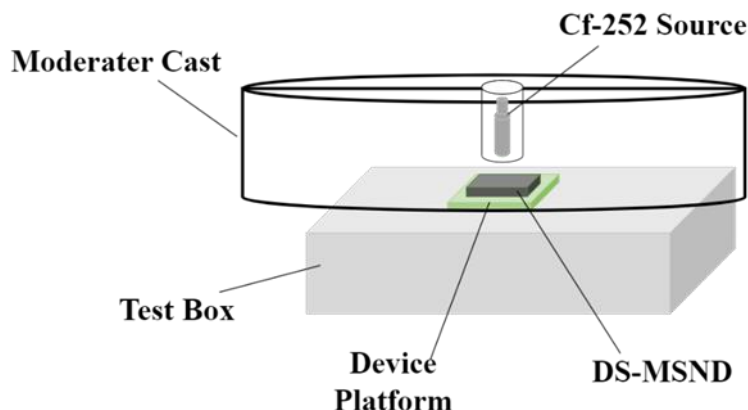


Figure 4-47 The testing configuration for DS-MSND devices using NIM electronics is shown.

Table 4-8 The calibration data for several devices and the calibrated standard device is conveyed.

Calibration Standard Detector		Background	Foreground	Time (s)	Flux
DS-MSND 406		2 ± 0	1239 ± 35.20	120	$31.31 \pm 1.12\%$

Device	Leakage Current (Amps)	Background	Foreground	Time (s)	Calibrated with Device 406
358	6.00E-07	8 ± 2.83	2572 ± 50.71	120	$64 \pm 2.50\%$
359	1.50E-08	132 ± 11.49	2574 ± 50.73	120	$61 \pm 2.43\%$
0414	7.00E-08	1 ± 1.00	1980 ± 44.50	120	$50 \pm 2.00\%$
0523	5.00E-08	0 ± 0.00	1750 ± 41.83	120	$44 \pm 1.81\%$
0467	2.20E-05	0 ± 0.00	1742 ± 41.74	120	$44 \pm 1.80\%$
0530	4.30E-08	0 ± 0.00	1737 ± 41.68	120	$43 \pm 1.80\%$
0575	1.50E-07	10 ± 3.16	1744 ± 41.76	120	$43 \pm 1.80\%$
0495	1.10E-07	17 ± 4.12	1736 ± 41.67	120	$43 \pm 1.79\%$
0451	7.10E-08	1 ± 1	1589 ± 39.86	120	$40 \pm 1.67\%$
333	9.00E-09	151 ± 12.29	7533 ± 86.79	180	$40 \pm 4.39\%$

Table 4-9 The averager leakcurrent for upper efficiency devices are shown.

Number of Devices	Efficiency	Average Leakage (Amps)
11	>40%	2.88E-06
13	>30%	2.76E-07

A comparison of the most to minimally efficient DS-MSND detector reaction product spectrum is shown in Figure 4-48 when exposed to the moderated ^{252}Cf source. The 64% efficient detector

noise floor clearly shifts rightwards and there does not appear to be good separation between the reaction product spectrum and the background noise level as expected with MSND devices.

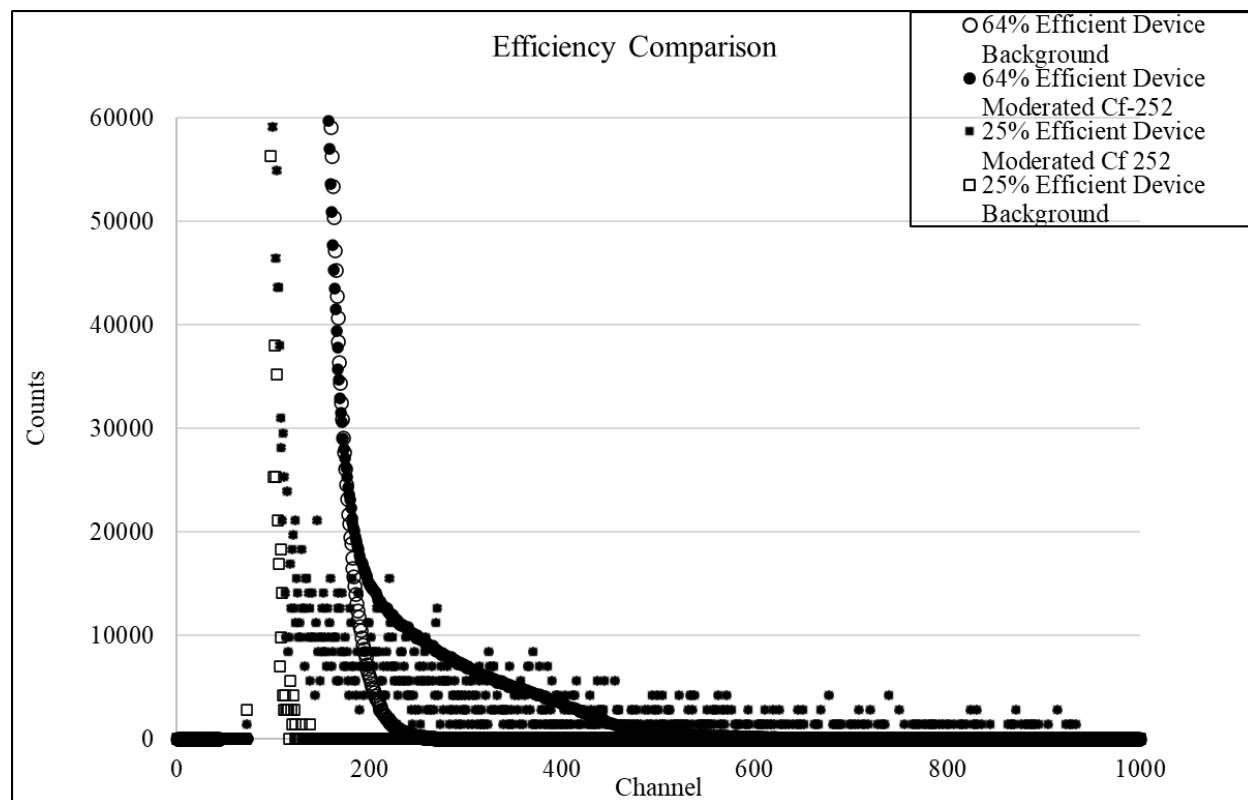


Figure 4-48 A comparison of a 64% efficient and a 25% efficient device is shown.

Spurred by speculations that the DS-MSND was not reaching full depletion as well as concerns with leakage current, a study again was performed to determine the optimal bias voltage at which to operate the detectors through examining the efficiencies. Previous versions of the MSND were operated at 5.0V bias. However, it was concluded that operating the DS-MSND at 2.5V bias produced better results, as shown by the ^{252}Cf spectra in Figure 4-49. The grey/dark grey data represent the background and moderated ^{252}Cf , respectively, collected with a DS-MSND biased at 2.5V. The yellow/brown data represent background and moderated ^{252}Cf , respectively, collected with a DS-MSND biased at 5.0V. The 2.5V results indicate a lower background count rate and a stronger response to moderated neutron signals.

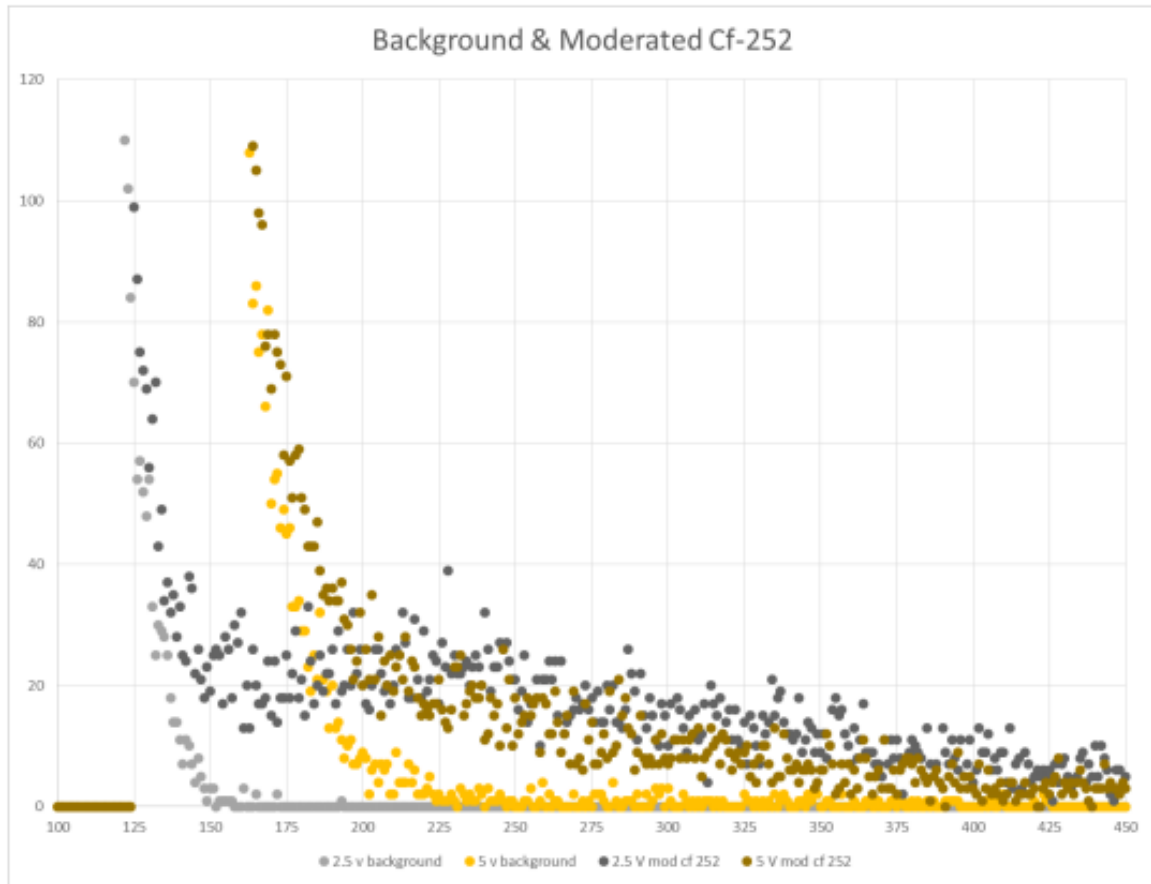


Figure 4-49 The DS-MSND were tested with a ^{252}Cf source with varying operating voltages.

Lastly, a comparison between the performance of a DS-MSND and single sided MSND were compared, see Table 4-10 . When exposed in the same configuration of Figure 4-47, DS-MSND clearly displayed the superior net count rate when exposed to the moderated ^{252}Cf , demonstrating the superiority of the DS-MSND despite all the design complications that were encountered.

Table 4-10 A comparison of the MSND and DS-MSND device in the same testing configuration is shown.

Detector	Source	Neutron Counts	Net Count Rate
MSND	^{252}Cf (moderated)	12270	2.03 ± 0.02
DS-MSND	^{252}Cf (moderated)	11445	3.09 ± 0.03
MSND	background	1080	0.02
DS-MSND	background	5403	0.14

5 Strontium Iodine Gamma-Ray Detector



Image from Capesym High resolution SrI2(Eu) Scintillation detectors Datasheet [117]

“On darkening the room you see the diamond shine with as much light; as a candle, phosphorescing of a bright green.” [118]

~ William Crookes, 1879

The second key piece to a successful ARGANOT system was a detector array also sensitive to gamma-rays. Incorporating the detection of both gamma-ray and neutrons into one detector was considered but ultimately decided against the method for three reasons.

1. Form factor

Detector systems that employ both gamma-ray and neutron detection in one volume struggle to maintain a small form factor in both the detector appearance as well as the accompanying electronics.

2. Gamma- Neutron rejection

Detectors of this type struggle with discrimination of events in the detector, making neutron and gamma-ray detections undesirably intertwined.

3. Cost

As the complexity of dual detection systems increases, as does the overall price of the system.

The dual detector approach alleviated the struggle of event discrimination, allowing a small and slender form factor, all while preserving the cost and quality of the detector for the end user.

5.1 Detector Selection

Once the decision was made to pursue an individual gamma-ray detector capable of gamma-ray spectroscopy, many scintillators and a couple semiconductor options were explored, see Table 5-1. As previously discussed within the ARGANOT design goals, the gamma-ray detector unit had to be capable of offering good energy resolution, be extremely durable, compact, cost efficient, and require minimal power inputs.

Table 5-1 Various gamma-ray detectors that were considered the detection system are shown below.

Scintillator Options									
Crystal Type		Dimensions		Cs-137 Energy Resolution	Peak λ [nm]	Density [g/cm ³]	Luminosity [γ 's/MeV]		
LYSO		13 mm x 13 mm x 50 mm		9.00%	420	7.1	34,000		
NaI		18 mm x 18 mm x 18 mm		6.80%	420	3.67	45		
GAGG		3 mm x 3 mm x 3mm		8.20%	520	6.63			
LaBr3		d=2" h=2"		3.20%	380	5.1	70		
OGO		2.4 mm x 24 mm x 15 mm		9.70%	480	7.13	20		
SrI ₂		8 mm x 8 mm x 20 mm		3.80%	435	4.6	100		
CsI		18 mm x 18 mm x 18 mm		6.40%	425	4.51	50,000		
BeBr ₃		d= 3" h=3"		4.00%	380	5.2	70		
Semiconductor Options									
Crystal Type	Manufacturer	ID	Dimensions [mm]	Intrinsic Resistivity [Ω cm]	Density [g/cm ³]	Voltage [V]	Shaping Time [us]	Cs-137 Energy Resolution	Price
CZT	Redlen	CPG	10 mm x 10 mm x 10 mm	1.00E+11	5.8	1500	1	2.50%	\$1,100
		Al	6.1 mm x 6.1 mm						
CdTe	Acroad	Schottky	x 1.0 mm	1.00E+09	5.9	500	0.5	1.80%	\$180

Semiconductor detectors, while they offered superior energy resolution compared to the scintillators, were determined to be less favorable due to the high voltages required for such small volumes. Once scintillator became the obvious choice for the type of gamma-ray unit in the ARGANOT, the focus then turned to selecting the desired readout electronics. It was

determined that a Silicon Photomultiplier (SiPM) would meet all of size, power, and durability requirements better and easier than a traditional photomultiplier tube (PMT) and therefore became the obvious readout choice to pair with the scintillator so that the entire module could be powered with LiPO batteries. SiPMs are able to operate with a very lower power requirement due to the innate operation of the device that depends on the avalanche breakdown condition across thousands of tiny silicon photon avalanche diodes. The breakdown voltage for many types of SiPMs is reported to range from 24-65V [119]. The maximum operating voltage for the $\text{SrI}_2(\text{Eu})$ scintillator often ranges from 10-15% above the breakdown bias [120].

Ultimately, europium doped strontium iodide coupled with a SiPM array was selected for the ARGANOT module. $\text{SrI}_2(\text{Eu})$ was selected because of its consistent energy resolution of $\leq 4\%$ at 662 keV as compared to other scintillators, see Table 5-2. Additionally, $\text{SrI}_2(\text{Eu})$ possessed good elasticity and hardness values that fell between that of $\text{CsI}(\text{Tl})$ & $\text{NaI}(\text{Tl})$. However, $\text{SrI}_2(\text{Eu})$ is much less fragile than $\text{NaI}(\text{Tl})$ and has no preferential cleavage plane, making the crystal a bit more impervious to cracks while in the field. Furthermore, $\text{SrI}_2(\text{Eu})$ doesn't have an internal radioactivity, which was one of the leading factors that lead to LaBr:Ce being eliminated as a candidate to populate the ARGANOT modules even though it boasted superior energy resolution. $\text{SrI}_2(\text{Eu})$ ($Z_{\text{eff}} = 49$) can operate with a low power silicon photomultiplier (SiPM) and has a high light output of 100,000 photons per MeV [121]. Finally, SrI_2 gamma-ray detectors can operate in a range of common thermal environments, with operation limited below 60°C , mainly because of the temperature compensation available through the bias applied to the SiPM. The only concern with the $\text{SrI}(\text{Eu})$ detector was its extremely hygroscopic nature, meaning that the casing would have to be impermeable. Ultimately the radiation detector manufacturing company CapeSym was selected to provide a $\text{SrI}_2(\text{Eu})$ detector package with a SiPM array.

Table 5-2 The $\text{SrI}_2(\text{Eu})$ energy resolution at various energies is shown [122].

Energy (keV)	Energy Resolution
1333	2.20%
1173	2.40%
662	3.30%
609	3.40%
122	6.00%
32	14.50%

To confirm expectations regarding the performance of various $\text{SrI}_2(\text{Eu})$ at varying volumes, MCNP simulation was performed. The shape of the gamma-ray $\text{SrI}_2(\text{Eu})$ detector shifted drastically as a result of MCNP simulations to evaluate tradeoffs in spectroscopic gamma-ray detector choices. For the simulation, a moving ^{137}Cs source with an activity of 500,000 Bq and an array of 16 SrI_2 detectors on a PMMA phantom, as specified in ANSI N42.53 was used. The simulation was conducted by modeling CapeSym's 2.34 cc and 0.47 cc size crystals with the associated energy resolution characteristics provided by the manufacturer. The net count rate (cps) recorded for the moving source test in the X and Y direction for the 2.34 cc and 0.47 cc detectors clearly revealed that the 2.34 cc design was the better performing crystal and was chosen.

5.2 Detector Design

The detector initially selected and received from Capesym was a newer model of the SC-13c-SiPM, with dimensions 16.5 mm x 16.5 mm x 21mm. Schematic images of the detector are shown below in Figure 5-1. Within the SiPM the T-sensor, also known as the temperature sensor, is built in to acts as a high-accuracy, low-power remote temperature sensor monitor [123]. The manufacturer specifications document states that 2.9% energy resolution was achieved with this specific device using a 10 μ Ci ¹³⁷Cs (662 keV) source 3 cm in front of the detector. The spectrum collected by Capesym is shown in Figure 5-2.

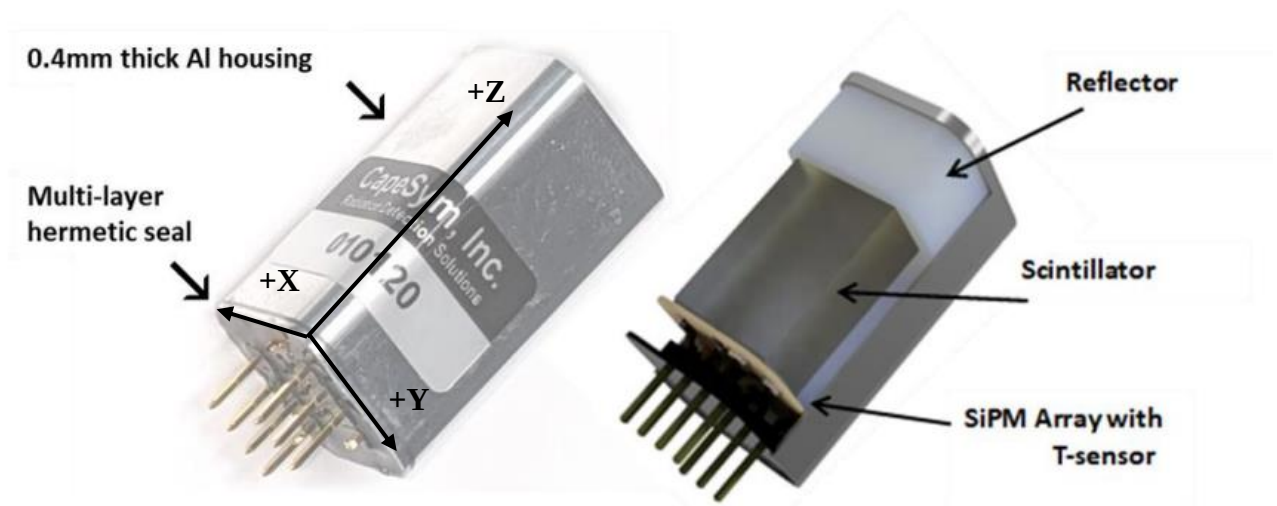


Figure 5-1 Capesym was a newer model of the SC-13c-SiPM, with dimensions 16.5x16.5x21mm [124].

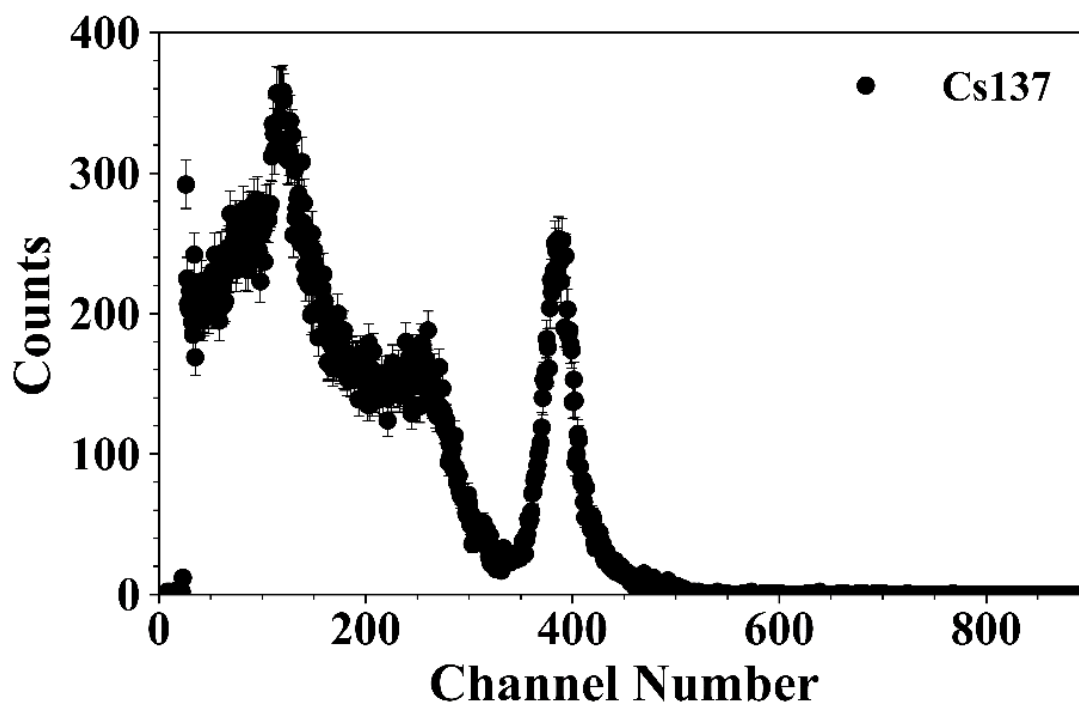


Figure 5-2 The resulting $\text{SrI}_2(\text{Eu})$ response to a $1\mu\text{Ci } ^{137}\text{Cs}$ source.

The electronics design partner, Electronics Design Laboratory (EDL), fabricated an electronic readout board that tested the detector and characterized its response with different sources and may be viewed in Figure 5-3. The initial connection of the detector to the newly designed board first raised questions regarding the best way to connect the gamma-ray spectrometer to the module circuitry. The SensL SiPM's, which CapeSym implemented on the detector had 3 pins: cathode, anode and fast output. It was quickly determined that the ARANOT had no application for the fast output, and the decision was made to remove it in the following integration of the detector designed by CapeSym. A different SiPM, the Broadcom AFBR-S4N66C013 with only two pins (cathode & anode) was then selected to best fit the geometry restrictions. Additional questions were raised regarding whether pins should be connected internally with two shared pins, or to connect each SiPM individually. At the time, it was thought that an individually connected design would allow for the most redundancies in the system, meaning that if one connection was faulty, the other connections would be maintained and still allow for data to be communicated to the user.

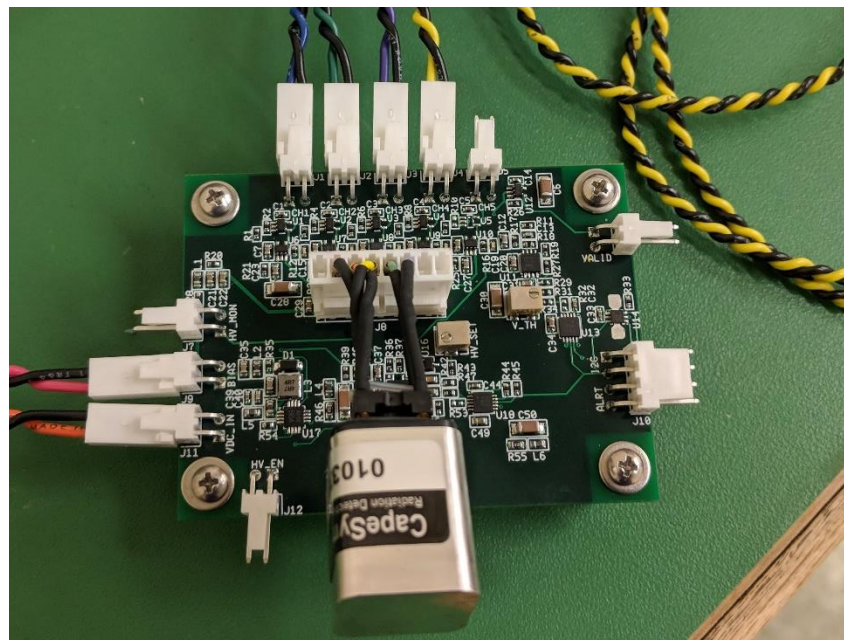


Figure 5-3 The initial testing assembly for the SrI(Eu) detector is shown.

For many months work was conducted to outfit the detector board with the appropriate electronics to maximize the gamma-ray detector performance. During the various assemblies, testing, and re-assemblies, it was realized that the hermetic seal of the $\text{SrI}_2(\text{Eu})$ crystal had been breached and the hygroscopic detector material had deteriorated. The contents of the detector began to swell and leaked a yellow substance as shown in Figure 5-4.

At the time, CapeSym had reported a similar failure once before caused by a hermetic seal failure due to direct soldering on the pins of the module. The pins were soldered to the board by a skilled technician, making it possible the breach was simply due to excessive heat, though there remained concern for the integrity of future detectors because heat application was inevitable to connect the detectors to the board. Because the initial $\text{SrI}_2(\text{Eu})$ detector package was larger than what was envisioned to fit in the new RWD module and there were concerns regarding the detector integrity, a following generation of $\text{SrI}_2(\text{Eu})$ detectors was designed. In the following generation, to lessen the effects of heat to the hermetic seal, an epoxy seal was used in place of the plastic top that used ribbon cables instead of pins for the current detector packages which minimized any heating of the package.

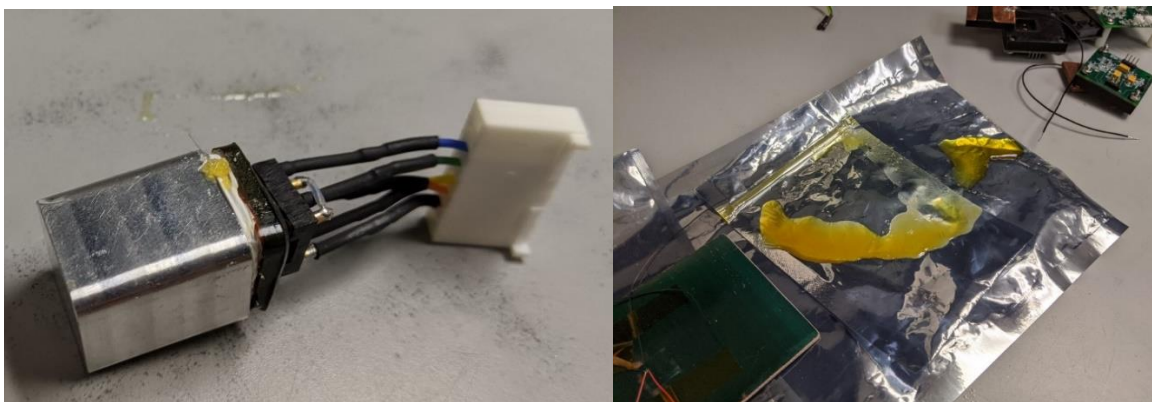


Figure 5-4 The hygroscopic crystal was exposed to moisture and was reduced to a yellow substance.

The free space that was available in the new board design for the footprint of the gamma-ray detector was 9x18x31mm, which would account for the $\text{SrI}_2(\text{Eu})$ crystal, aluminum casing, and the SiPM. CapeSym had offered to design and build a custom geometry detector which would reach the 6x13x30mm requirement that would include the $\text{SrI}_2(\text{Eu})$ crystal coupled to 5 SiPMs on the 6x30mm side and packed into a 0.5mm aluminum casing [125].

To decide whether the interaction efficiency of the gamma-ray detector would still yield an acceptable interaction efficiency if the shape of the detector was drastically changed, a plot for interaction probability with respect to energy for various device thickness was generated and may be viewed in Figure 5-5. The NIST XCOM attenuation coefficient data shown in Figure 5-6 below and the known $\text{SrI}_2(\text{Eu})$ density of 4.55g/cc, were used to determine the values displayed in Figure 5-5. As shown in Figure 5-1, the initial design of the rectangular 16.5cm wide $\text{SrI}_2(\text{Eu})$ crystal had an approximately 58% interaction probability, which would increase to approximately 67% should the photon interact on the z axis of the crystal. Though, with the reduction of the thickness, the expected interaction probability decreased to approximately 38% and as high as 80% should the interaction take place along the x axis of the crystal. It was therefore agreed that the decrease in overall interaction probability was a worthy yet moderate sacrifice to achieve the sleek and discrete module size.

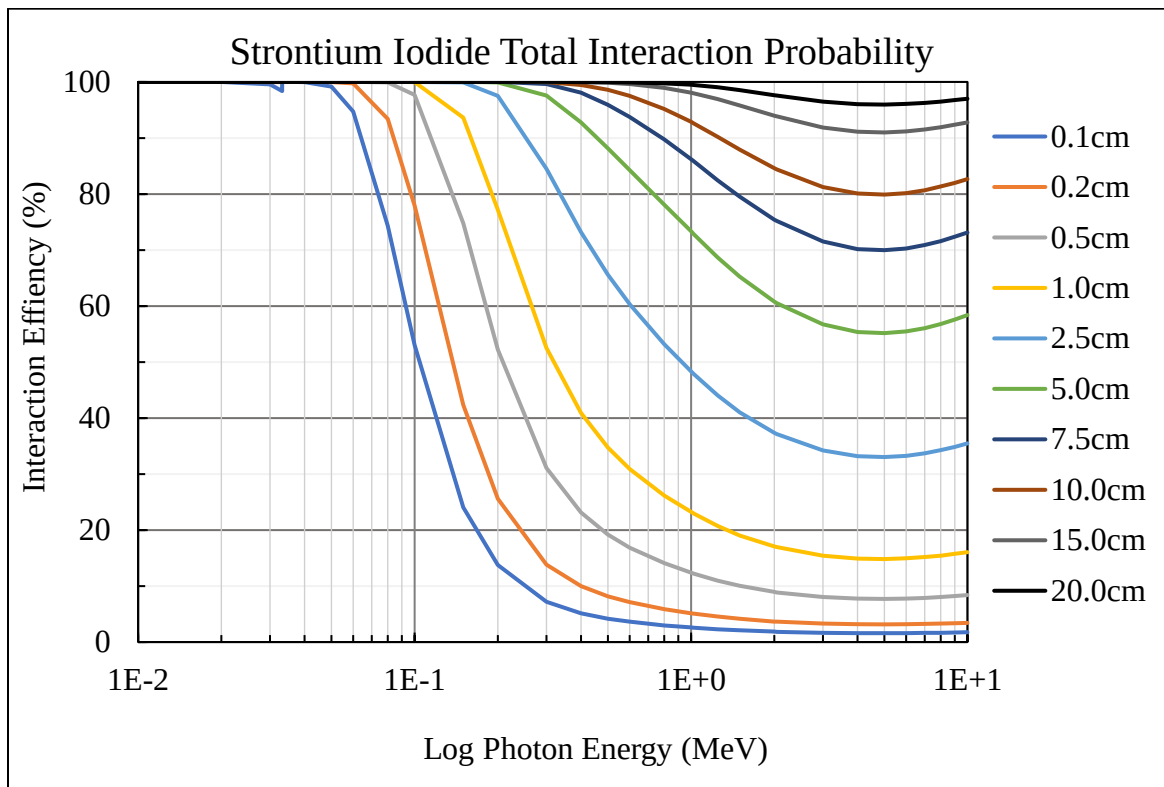


Figure 5-5 The interacting efficiency for various thickness of $\text{SrI}_2(\text{Eu})$ material over a range of energies.

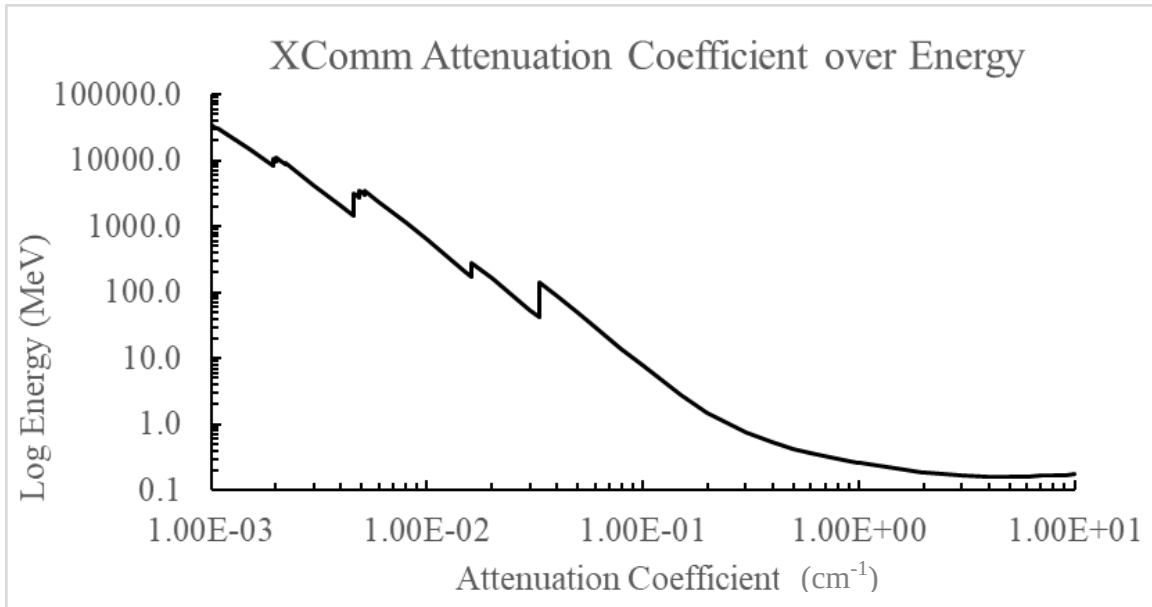


Figure 5-6 The attenuation coefficient over increacsing energy is shown [126].

The next generation of $\text{Sr}_2\text{I}(\text{Eu})$ detectors utilizing the 2.34 cc design were much slimmer and enabled the final board shape and casing to be reduced by several centimeters. At the time, there was also interest in multiple options for how the detectors, electronics, and batteries were housed in the ARGANOT modules. In order to explore all ways to optimize the design constraints, several factors were explored including reducing the thickness, overall volume, and rounded corners while maximizing the detection capability and battery life of each module. The new casing design was flat, with a LiPo battery located adjacent to the electronics and DS-MSNDs were no longer stacked as with the previous designs. These design changes allowed for a thinner module (11.75 mm vs 12.7 mm) to be accomplished, which would distinguish the ARGANOT from its competitors.

Figure 5-7 shows the second-generation detector with 5 of the SensL SiPM's coupled to the 6 mm x30mm side of the SrI_2 crystal in which the two shared pin method was implemented. The idea was that in using the two shared pin method, space on board within the module would be conserved. Figure 5-8 shows the connected gamma-ray detector with the redesigned board. The second generation of boards, which was maintained in appearance for the duration of the ARGANOT project, looks drastically different from the previous version in order to achieve the module size expectations. An additional mounting concern was whether to use pinned or wired

connections. While pinned connections would be rigid and sturdy, this could potentially introduce a torque between the detector and the board it is mounted to. A wired connection seemed more favorable due to the flexibility, adjustable length and having the option to insulate or shield the wires from RF interference.



Figure 5-7 Five of the year 2 gamma-ray detectors are shown.

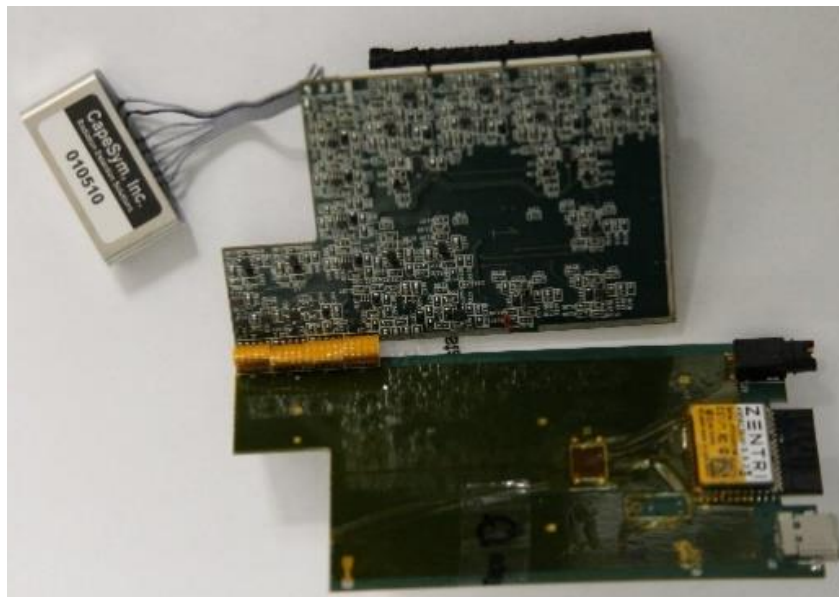


Figure 5-8 The second generation $\text{SrI}_2(\text{Eu})$ detector is shown attached to the new digital electronics board.

Throughout the time with the generation two detectors, work began to characterize the detector and SiPM response when operating in various temperature environments. Throughout preliminary temperature testing, the drastic response with respect to temperature, mainly

attributed to the SiPM, was realized. As shown in Figure 5-9 below, the energy resolution rapidly degraded as the peak began to shift leftward with the 4°C degree change over a one-minute period. As a result, a temperature correction algorithm was developed.

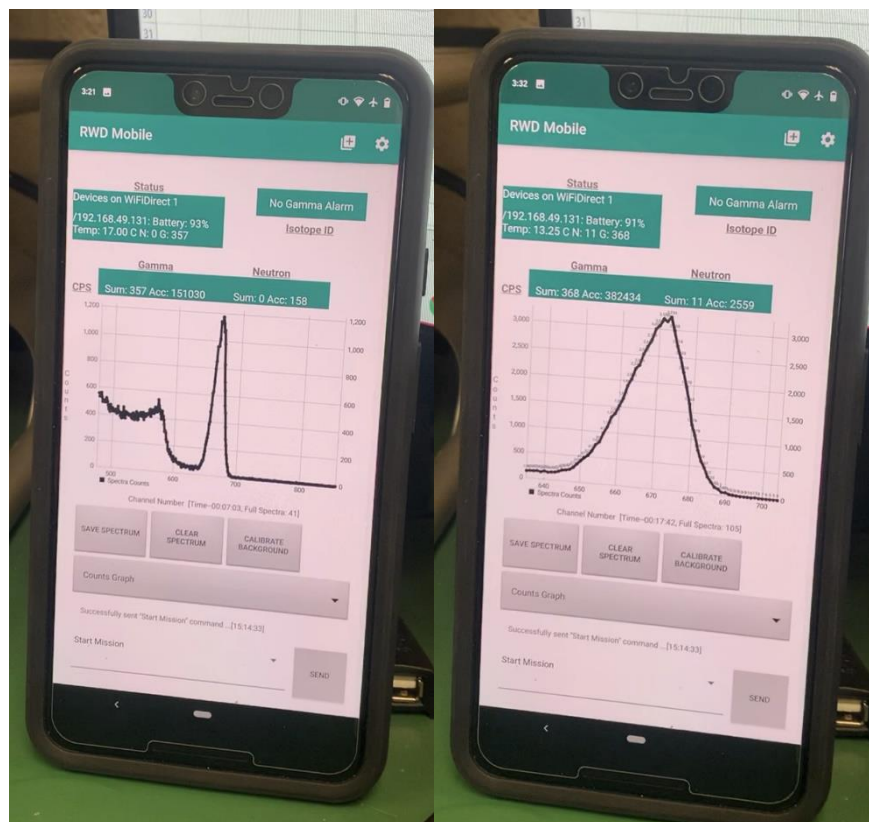


Figure 5-9 The shift of the ^{137}Cs peak from the $\text{SrI}_2(\text{Eu})$ detector as the temperature shifted from 17°C to 13°C is demonstrated.

Within a few months of use and testing of the detector, several defects were realized that instituted the third generation of $\text{SrI}_2(\text{Eu})$ detectors. A widespread problem was that the SiPM- $\text{SrI}_2(\text{Eu})$ modules appeared to be greatly affected by light. When the module was placed into a dark environment, the noise level decreased to the expected level, but almost immediately increased when removed from the dark environment. Since the SiPM is light sensitive and housed near the pins, it was determined that the epoxy coating was not proficient in shielding the sensitive electronics from the light of the environment. The problem was communicated to the manufacturer who concluded that the detector dimensions requested were tight, and therefore it was challenging to have a thick enough epoxy layer to block the light coming from the outside

completely. To ensure the SiPM was properly sealed, a plastic lid shown in Figure 5-10, was designed to fit the future generation of detectors.

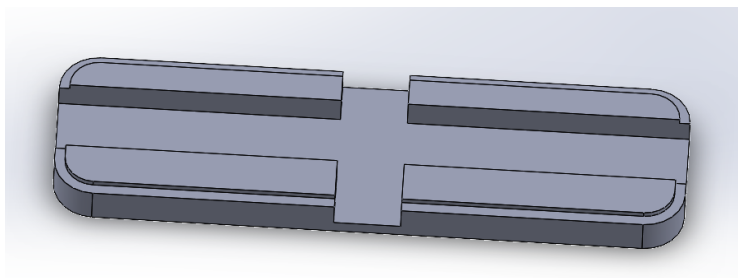


Figure 5-10 A CAD image of the specially designed cap to better seal the hygroscopic light sensitive detector is shown.

In addition to the light sensitivity, a few of the detectors began to present a dual or split peak when exposed to a $1\mu\text{Ci } ^{137}\text{Cs}$ source six inches away. An example of the dual peak as shown in Figure 5-11 below was collected over a one-minute period. Within Figure 5-12, the ten-minute measurement resulted in two peaks that grew at nearly the same rate. The dual peak was first noticed during temperature calibration when the temperature was altered within the measurement. Due to the innate operation of the detector, the temperature change instigated a spectral shift as indicated by the large number of counts in channels 1,100 to 1,500. Once the temperature stabilized, the counts between channels 1,100 and 1,500 ceased to grow. To ensure the dual peak was a product of the detector, not the environmental conditions, several measurements were completed in a stable temperature environment to ensure that there was no spectral shift due to temperature. After collecting several measurements in which the dual peak was present, it was determined that the dual peak was likely caused by a crack within the $\text{SrI}_2(\text{Eu})$ crystal. The idea that the damage was to the SiPM was also briefly considered but was ruled out because of the peak and detector behavior. If the SiPM had been damaged, the count rate would have likely significantly increased due to noise and the peak resolution would have rapidly deteriorated. Because the dual peaks still maintain their integrity, a crack in the $\text{SrI}_2(\text{Eu})$ seemed the most likely culprit. X ray images of several detectors, including the suspected cracked detector were collected at various detector orientation. As shown in Figure 5-13, there were no visible grain boundaries within any other sampled crystals. The lack of a clear grain boundary does not necessarily rule out the presence of a crack, but rather indicated that the crack

may have occurred in an orientation not perpendicular to the x-ray making it nearly impossible to clearly see the boundary.

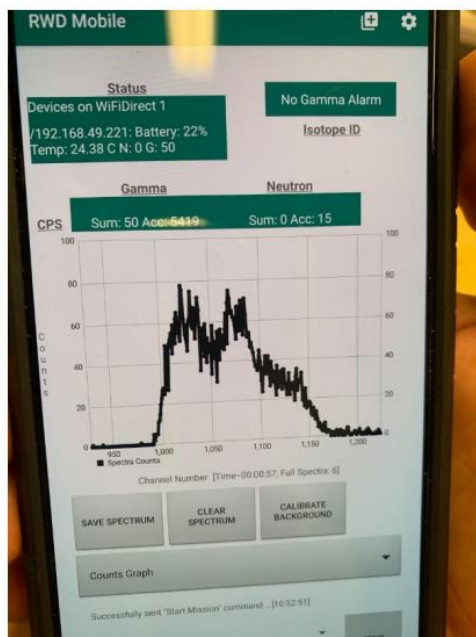


Figure 5-11 The ^{137}Cs dual peak initially viewed on the ARGANOT application during temperature calibration is shown. It was determined that the detector crystal likely suffered from a crack in the $\text{SrI}_2(\text{Eu})$ crystal.

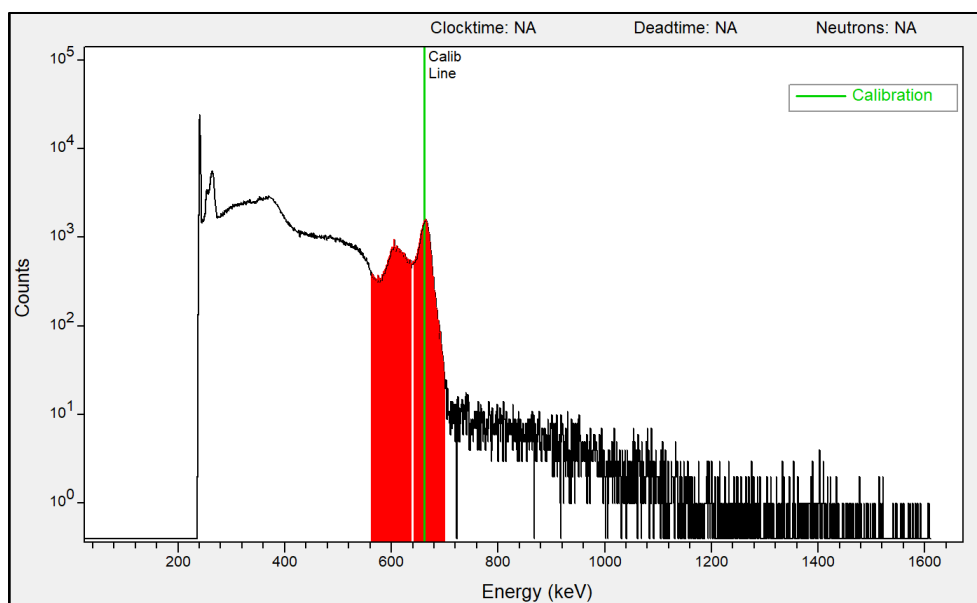


Figure 5-12 The full spectrum from the cracked $\text{SrI}_2(\text{Eu})$ crystal is shown.

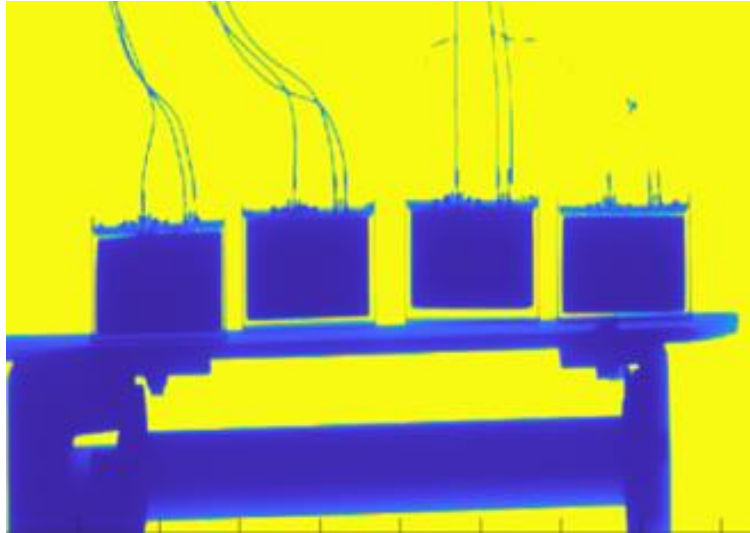


Figure 5-13 Several X ray images of the $\text{SrI}_2(\text{Eu})$ detectors were made to identify devices suffer from weak hermetic seals. Several features of the detectors are distinguishable while several detectors appeared with shades of blue in areas that should be clear.

Simulations in MCNP6.1 were performed to model the expected performance of a cracked $\text{SrI}_2(\text{Eu})$ scintillator crystal using an F8 tally. Figure 5-14 below details the simulation results for a $\text{SrI}_2(\text{Eu})$ detector below in which the crack created two equal crystal sizes, according to Figure 5-15. The spectral contributions for both halves (blue and orange) and the joint contribution (gray) are shown. The contributions from regions 10 and 12 appear to be very similar as reflected in Table 5-3 which displays the total energy (eV) per source particle collected in both regions of the crystal.

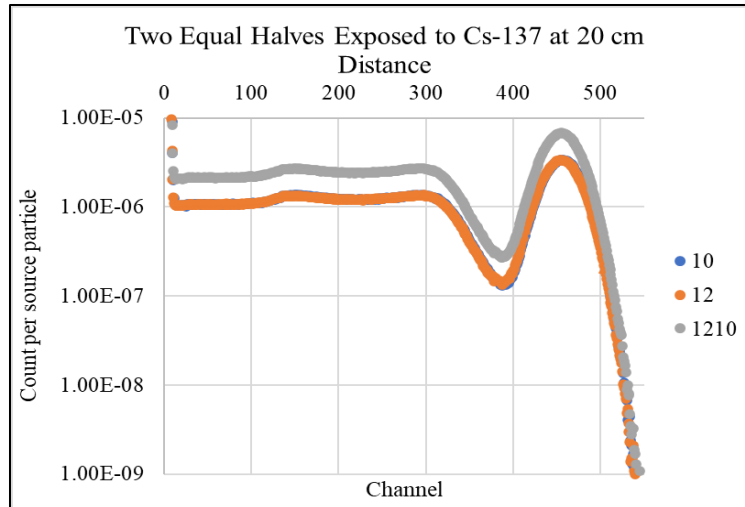


Figure 5-14 The spectral output of both regions of the cracked crystal and the conjoined output from MCNP6.1 simulations are shown.

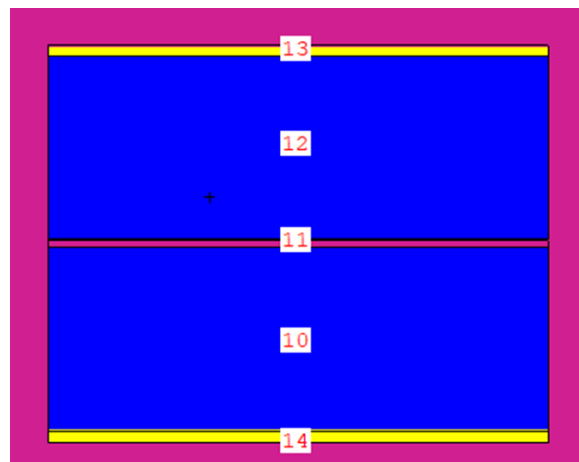


Figure 5-15 The MCNP Vised geometry of the simulated cracked $\text{SrI}_2(\text{Eu})$ is shown. The purple region represents a vacuum while regions 13 and 14 represent the silicon of the SiPM readout.

Table 5-3 The resulting energy per source particle in both regions of interest is tallied.

Total Energy (eV) per source particle (Region 10)	Total Energy (eV) per source particle (Region 12)
5.8517E+01	5.7948E+01

Following the simulation of two equal halves, a crack in which region 12 was large and region 10 was small, as reflected in Figure 5-16 was simulated. As shown in Figure 5-17, the amount of energy per source particle collected in region 12 was much than that of region 10.

When examining Table 5-4, region 12 contributed over three times more energy to the amount of energy absorbed per source particle. When the two spectra were summed and plotted together (gray), the energy resolution broadened. When the absorbed energy contribution was then translated to a lightout by the scintillating material, the number of secondary photons emitted was more significant for region 12. However, a signal contribution from region 10 still appeared, but at a reduced amount. Therefore, the dual peak would be especially prevalent from a detector with the geometry described in Figure 5-16 [127].

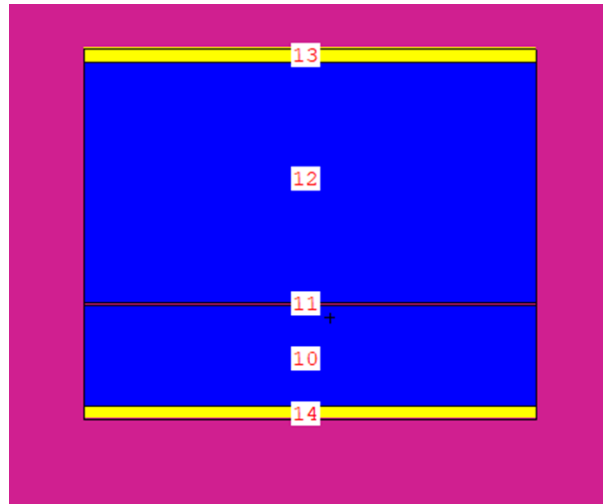


Figure 5-16 The MCNP Vised geometry of the simulated cracked SrI(Eu) is shown. The purple region represents a vacuum while regions 13 and 14 represent the silicon of the SiPM readout.

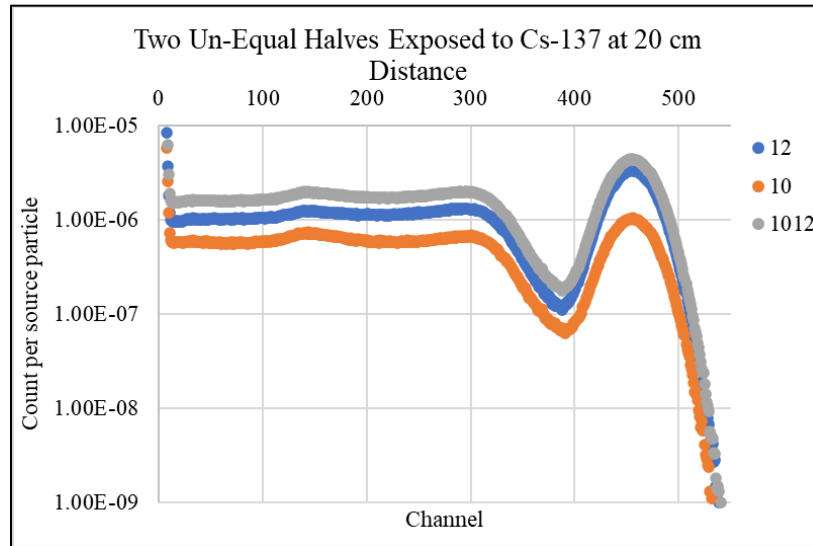


Figure 5-17 The spectral output of both regions of the cracked crystal and the conjoined output from MCNP6.1 simulations are shown.

Table 5-4 The resulting energy per source particle in both regins of interest is tallied.

Total Energy (eV) per source particle (Region 10)	Total Energy (eV) per source particle (Region 12)
1.7883E+01	5.7725E+01

While looking to identify the cracks within the crystal, it was noticed that several of the crystals appeared to be amorphous in shape with a light blue tint appearing at both the base of the crystal as well as near the SiPM, see Figure 5-18. In many of the crystals, the SiPM area and base appear transparent, which caused concern for of the outlier detectors.

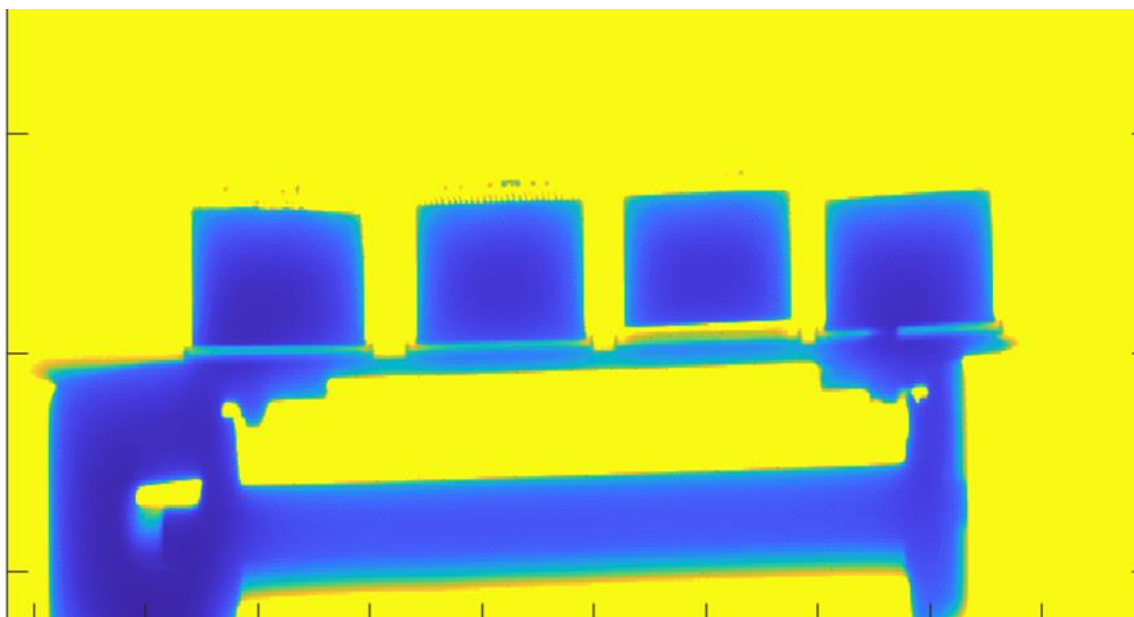


Figure 5-18 The x ray revealed material on the bottom on the detector casing. It was suspected that the light blue regions were possibly the result of a faulty hermetic seal and the start of a degrading $\text{SrI}_2(\text{Eu})$ crystal.

The detectors of concern were still, however, operating as expected and were pushed forward into calibration phases. Within a few months' time, several detectors began exhibiting excessive noise and failed to output data. The detectors were stored separately and within weeks began to swell and leaked similar yellow tinted material that was leaked from the previous $\text{SrI}_2(\text{Eu})$ generation as shown in Figure 5-19. Detector 010462 was included within the failed detector group. When the x ray image collected in the previous months were reexamined, it was clear that there was an abnormality with the package contents when the detector was placed

upright perpendicular to the x ray beam and laid parallel to the x ray beam as shown in Figure 5-20.



Figure 5-19 Several detectors swelled and often leaked yellow fluid. When detectors were identified to have a faulty hermetic seal, the detector was isolated and properly disposed of.

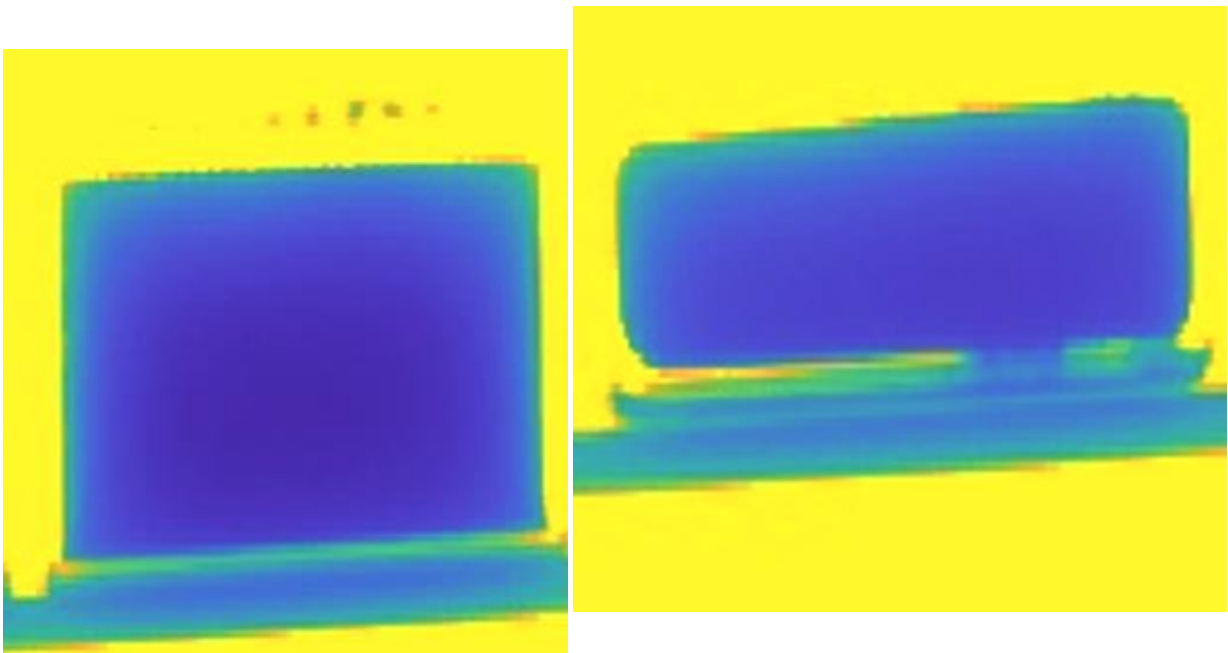


Figure 5-20 Detector 010462, which was later identified as having a faulty hermetic seal, was x-rayed. The images exhibited color disorientation along with evidence of a dense substance leaking into void regions.

In addition to the cracks and hydroscopic vulnerabilities, various connection wires became loose and disconnected from the detector. The decision was made to internally group the five SiPM readouts and have only three output wires (Cathode, Ground, and Anode) that would then tie to the ARGANOT module electronics board to migrate the risk of damaging the readout wires. The schematics provided by CapeSym of the third generation $\text{SrI}_2(\text{Eu})$ detectors are shown below in Figure 5-21. The third generation, as shown in Figure 5-22 below, were primarily used to develop the temperature correction algorithm, which will be discussed further in the following section. Throughout the temperature testing, sporadic noise indicated that the generation three $\text{SrI}_2(\text{Eu})$ detectors were still light sensitive. A fourth and final generation of $\text{SrI}_2(\text{Eu})$ detectors, see Figure 5-23, maintained a large amount of the internal design from generation three with the main addition being a Delrin caps shown in Figure 5-10, along with the existing epoxy.

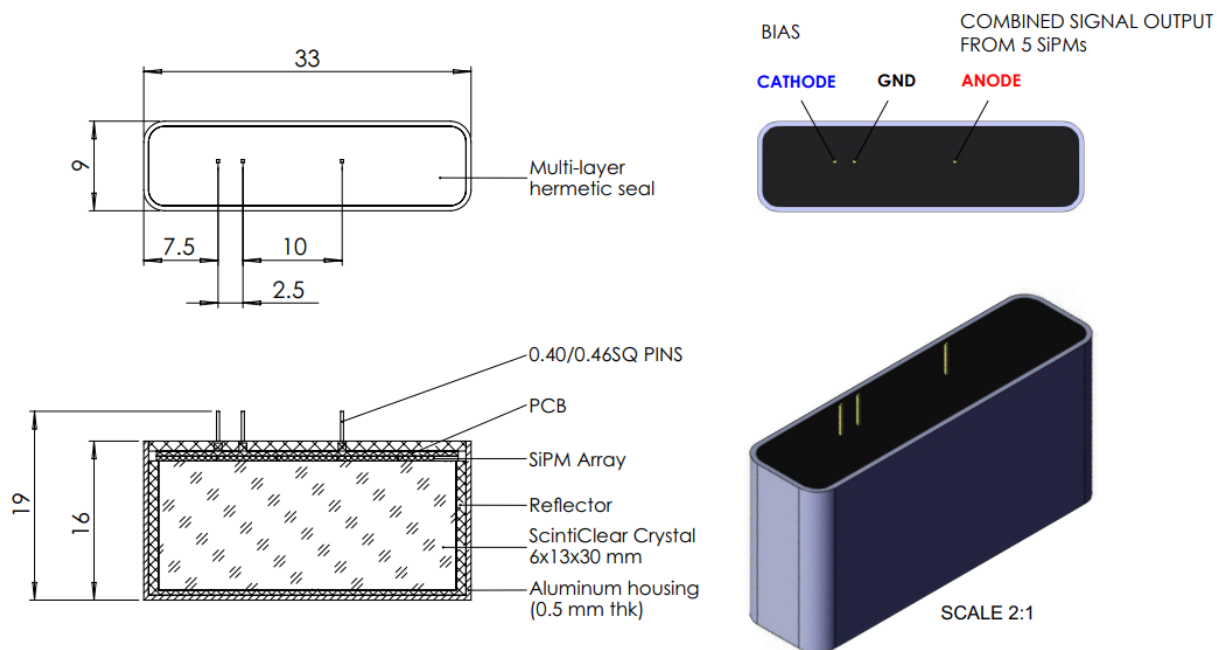


Figure 5-21 The CAD drawing of the final design of the $\text{SrI}_2(\text{Eu})$ detector is shown. The final designed largely reduced the wires emerging from the epoxy to limit the hermetic seal's vulnerability.



Figure 5-22 The generation three, CapeSym SrI₂ detectors (2.34cc) are shown.

Additionally, the wires were routed out the side of the lid instead of the back to improve the sensor's fit within the case and limit the wires and hermetic seals hydroscopic vulnerabilities. Figure 5-24 below demonstrates the final energy resolution of a single SrI₂(Eu) detector to be 4.04% for 662 keV ¹³⁷Cs source approximately 15 cm away.



Figure 5-23 Version four of the Capesym SrI₂ gamma-ray detector with light-tight lid and relocated wire output.

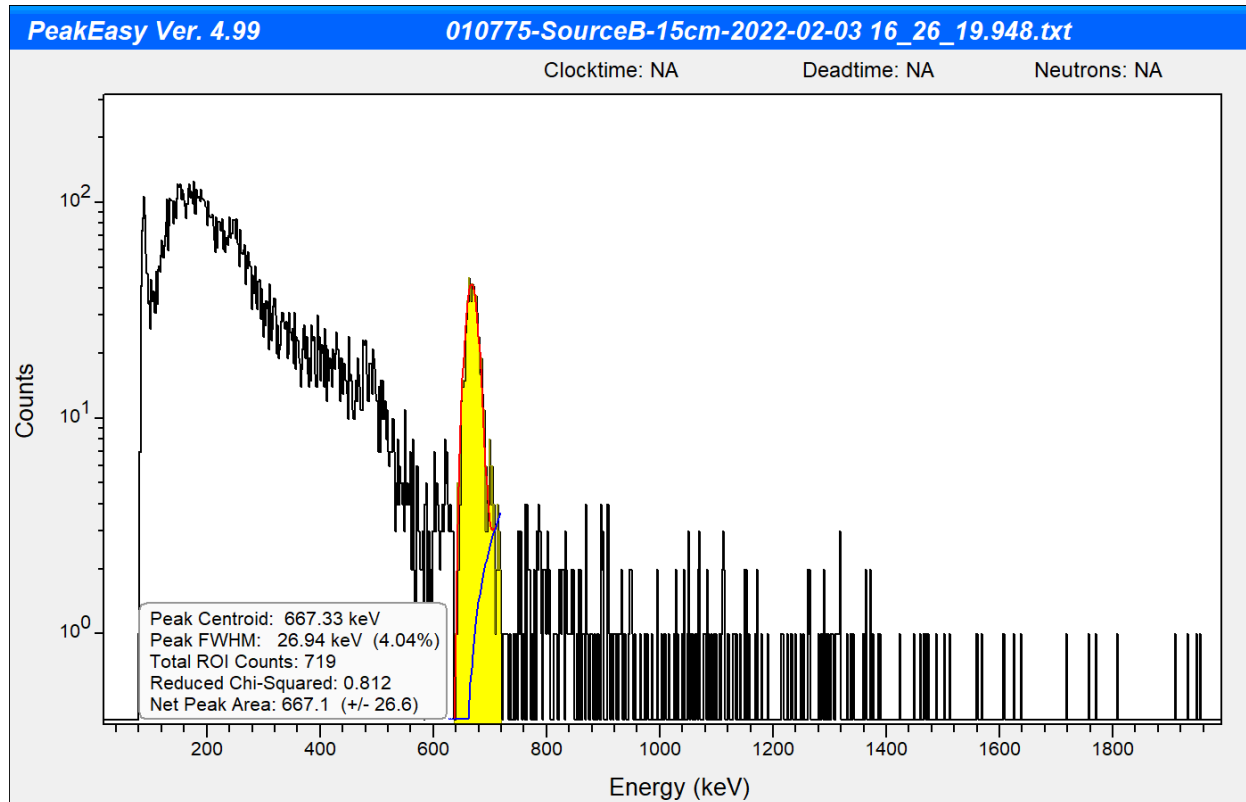


Figure 5-24 A $\text{SrI}_2(\text{Eu})$ spectrum employing the ARGANOT electronics when exposed to a $0.85\mu\text{Ci } ^{137}\text{Cs}$ source at 15 cm produced a peak with 4% energy resolution.

In addition to the $\text{SrI}_2(\text{Eu})$ detector, a $\text{CsI}(\text{Ti})$ crystal was also purchased from Berkeley Neutronics Corp (BNC) to offer the end user the option of a higher efficiency, but slightly inferior energy resolution option. BNC was also able to design the $\text{CsI}(\text{Ti})$ crystal to approximately the same size standards as CapeSym's $\text{SrI}_2(\text{Eu})$ detectors as shown in Figure 5-25, allowing the module size to remain constant regardless of the gamma-ray crystal option. The energy resolution of the $\text{CsI}(\text{Ti})$ detectors as reported by BNC ranged per crystal from 7.6% to 8.5%. The $\text{CsI}(\text{Ti})$ was selected as an alternative option due to being less hygroscopic than $\text{SrI}_2(\text{Eu})$ and ability to withstand a good deal of shock without cracking while maintaining pliability, making it a strong candidate for field operation. Because the $\text{CsI}(\text{Ti})$ crystal emission spectrum was altered with respect to the temperature differently than $\text{SrI}_2(\text{Eu})$, additional rigorous temperature testing and calibrations were performed for $\text{CsI}(\text{Ti})$ as well.

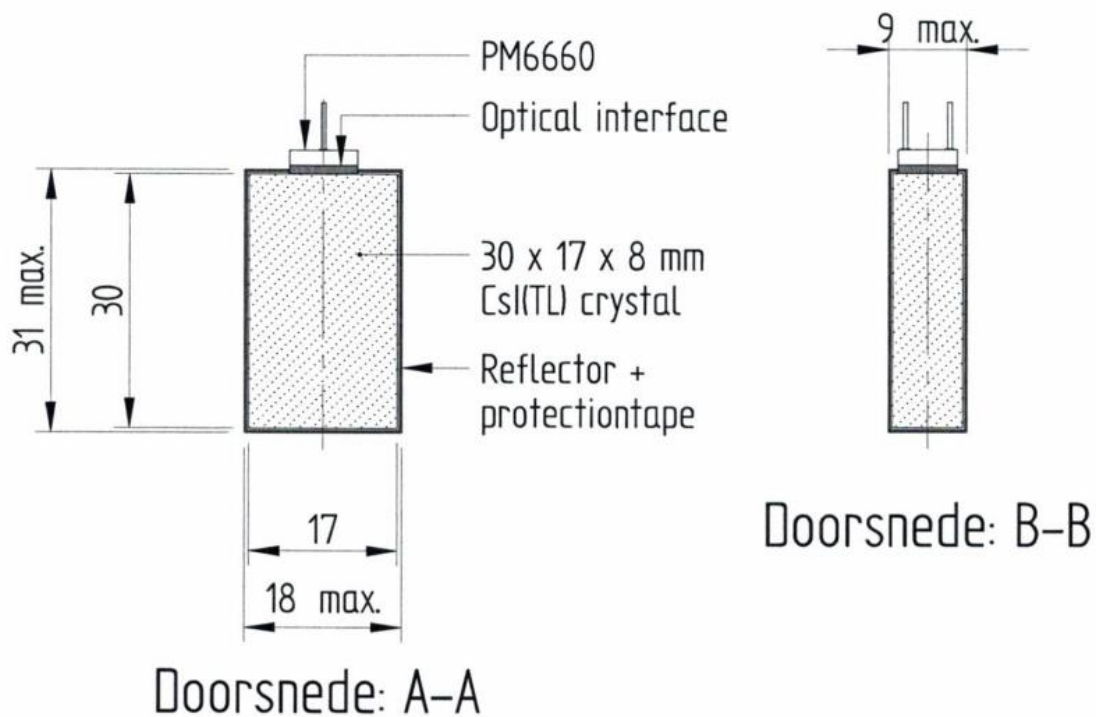
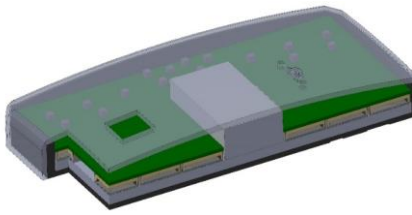


Figure 5-25 The schematics of the BNC provided CsI(Tl) gamma-ray detector is shown. The option was offered to allow for greater detection efficiency.

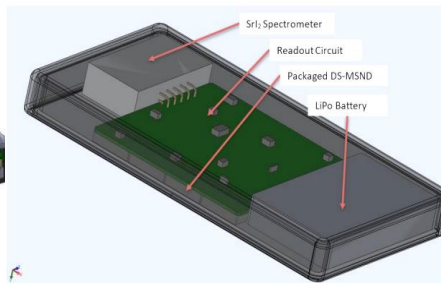
5.2.1.1 Packaging

As mentioned, the compact size of the ARGANOT was the single most important characteristic of the modules that set the system apart from competitors. Due to the extremely slim profile of the DS-MSND devices, the thickness of the module became driven by the gamma-ray detector. Through the four iterations of SrI₂(Eu) design, the size of the detector, the accompanying electronics located on the electronics board, and the module itself were able to be improved and consistently shrank, as shown in the evolution of board designs of Figure 5-26. The improvements allowed ARGANOT to achieve the paramount goal of delivering the customer an extremely compact appearance with the maximum operating capability by maintaining a light weight and compact appearance.

Generation I



Generation II



Generation III/IV

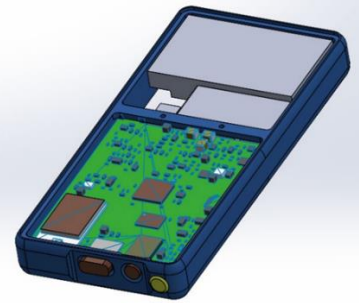


Figure 5-26 The evolution of the ARGANOT module is shown. With each generation, the module size was reduced.

5.3 SiPM

To offer more flexibility to scintillator operation, Semiconductor Photodetectors (SiPM) were chosen to operate with the scintillator detector over traditional Photomultiplier tubes (PMT). Not only do SiPM have the rigidity and small form factor that traditional PMT lack, but they also offered superior probability of absorption of scintillator photons in the active volume which offers a higher photon detection efficiency than traditional PMTs can offer.

5.3.1 SiPM Operation

SiPMs signal production is created by a compact array of thousands of tiny (each diode is approximately $35\text{ }\mu\text{m} \times 35\text{ }\mu\text{m}$ in size) single avalanche photon diodes (APD) tied together. A type of APD is a Single Photon Avalanche Diode (SPAD). Each SPAD is a silicon pn diode with a junction breakdown that allows current to flow across the semiconductor when charge carriers are excited across the bandgap. Upon a single charge carry impact with other charger carriers, energy is transferred at a rapid pace, creating what is known as an avalanche of charge carriers. The amount of charge carriers excited due to impact can grow to be as great as $10\text{E}+9$ free charger carriers per a single photon event [128].

Due to the of the tremendous efficiency of self-sustaining charge production, the current created must be quenched in order to prepare the APD for the sequential impinging photons. In order to reset or quench the avalanche, the current can be directed to flow through a quenching resistor. Using Equation 5-1, as the current across the resistor builds, the resistor will draw voltage from the device, creating a voltage drop across the device well below the voltage required for a breakdown avalanche to occur. Charge carriers within the device will mostly cease impacting as well as exciting across the bandgap. Once a subsequent photon strikes the APD, the

avalanche will occur once more and continue until the shunt resistor once again cuts off the avalanche.

$$V_{\text{output}} = V_{\text{in}} - iR_{\text{shunt}} \quad (5-1)$$

Based on the operational physics of SiPM devices, each SPAD merely acts as a counter. Regardless of how many photons strike each SPAD in a given moment, the amount of voltage produced will remain relatively constant. A trick however to enable the SiPM to act as a spectrometer is to quantify the number of SPAD that counts an event in a given moment. Because each pixel produces an avalanche of a similar magnitude for each event, the total photon count can just be determined by dividing the total measured charge by the avalanche gain. The number of events counted can then be correlated with a suspected amount of photons produced and therefore the energy of the original impinging photon. The backing out of the energy of each individual photon then allows for an energy spectrum to be formed.

While there are many advantages to the SiPM such as compactness, just with any technology, there are also weaknesses. The major problem with the SiPM is related to how accurately the number of photons detected reflects the real amount actually produced from a radiation event of interest. As described above, the SPADs are merely counters and therefore become insensitive to impinging photons while processing a previous photon. Therefore, as both source strength and decay times of scintillator material increase, the SiPM will further underestimate the amount of radiation events impinging upon the detector.

One phenomenon that decreases the efficiency of SiPMs is afterpulsing. Afterpulsing describes an event in which a charge carrier that was once trapped is excited out of the trap and consequently triggers a false avalanche within the semiconductor [129]. The false avalanche triggers the SPAD to report a radiation event took place and becomes insensitive to photons of interest for a specified amount of time. Therefore, in the case of SiPM, two or more photons can strike the same cell and the response is not proportional to the energy of detected gammas. Hence, the detector response should be expressed in N_{fired} . The linear range of SiPM response depends on the number of APD cells, the effective recovery time, and length of input light pulses. For a light pulse in which the light pulse time is less than the effective recovery time, the response of SiPMs can be described by Equation 5-2. N_{fired} is the number of excited APD cells,

N_{total} is the total number of APD cells, N_{photon} is the number of incident photons, PDE is the photon detection efficiency and t_d are the effective deadtime [130].

$$N_{fired} = N_{fired} \left[1 - \exp \left(\frac{-PDE (N_{photon})}{N_{Total}} \right) \right] P_W \leq t_d \quad (5-2)$$

A possible culprit of the charge carrier excitation from traps could be thermionic photons striking the trapped carrier. Because the charge carrier populations vary at various allowable energy levels within the semiconductor, the thermal noise will increase with increases temperatures. Equation 5-3 describes the intrinsic carrier concentration in the situation in which an increase of temperature will eventually cause highly doped extrinsic material to behave as intrinsic material. N represents the density of states for the Conduction and Valence bands respectively, E_g represents the ground level energy, T is the temperature in Kelvin, and K is the Boltzmann constant.

$$n_i = \sqrt{N_c N_v} \exp \left[\frac{-E_g}{2kT} \right] \quad (5-3)$$

The behavioral change is due in part to the breakdown voltage and the depletion region of the device. The increase of thermal noise within the semiconductor at high temperatures makes it difficult to use SiPMs in such high temperature conditions.

5.4 Characterization

5.4.1 Temperature Calibration

In order to correctly implement Gamma-ray Detector Response and Analysis Software (GADRAS) for isotope identification, the gamma-ray spectrums had to be energy calibrated to ensure a uniform and consistent isotope peak formation across all modules. For reasons explained above, in order for the ARGANOT system to operate dynamically in the field and maintain gamma-ray spectrum quality, a temperature correction algorithm was needed. The easiest way to control the energy calibration of the detector was to alter the bias.

As displayed in the flow chart of Figure 5-27, the temperature correction algorithm first began by referencing the digital thermometer built on the ARGANOT electronics board. Every ten seconds the temperature sent by thermometer was compared to the previous temperature. If

the deviation was larger than two degrees Celsius, a signal would be sent to the bias controller which would adjust the SiPM voltage to accommodate temperature changes.

Temperature Bias Loop

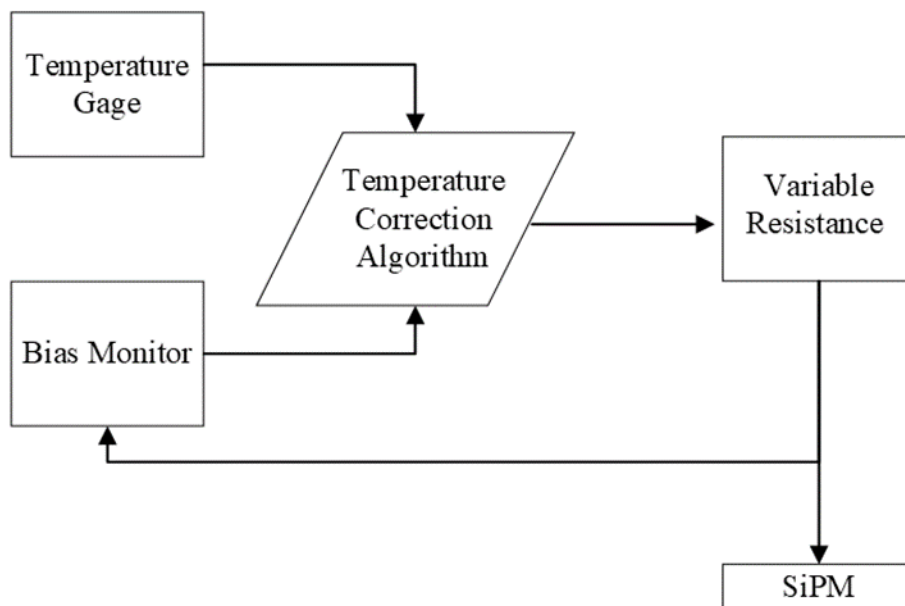


Figure 5-27 The Temperature bias loop used in the ARGANOT temperature correction algorithm is shown.

Using a ^{137}Cs source, the $\text{SrI}_2(\text{Eu})$ gamma-ray detectors were energy calibrated within the environmentally controlled chamber as shown in Figure 5-28 below. During the calibration process, each detector module was placed into the chamber and was allowed to soak in the temperature point of interest. After several minutes, a device was selected and connected to the android app. As shown in Figure 5-29, all individual module health statuses, including the temperature, were communicated, and displayed to the user. After allowing the module temperature to stabilize, the 662 keV gamma-ray ^{137}Cs check source was placed approximately four inches from module of interest. A measurement was then begun on the android app and the monoenergetic gamma-ray peak began to appear. If the peak was not already appearing at channel 967, the bias was adjusted, the spectrum was clear, and a subsequent measurement was started. Once the peak stabilized at channel 967, the device was disconnected, and the following module was connected. Following the same process, each device was then calibrated for the

temperature. Once all devices had been calibrated at the temperature of interest, the temperature was altered to the next point of interest. Throughout the calibration process, temperature increments of ten degrees Celsius were used. The range of testing was determined based on both the limits provided by Capesym as well as the anticipated environmental parameters an end user would operate by in the field. Based on both factors, temperature parameters for testing were set between -10°C and 40°C. While it is probable that the end user will need to operate on the periphery of these bounds, testing was not conducted beyond these ranges to avoid unnecessary damage to the detectors in the research and development stage of the ARGANOT project. In the future, should a user desire to operate out this range, more attention can be paid to fine tuning the operation parameters, rather than extrapolating from the current data.



Figure 5-28 The module devices were placed in a temperature and humidity-controlled chamber to maintain a stable environment for testing and calibration. A ¹³⁷Cs source is shown above in front of five of the ARGANOT modules while conducting a mission (right).

The 967-channel number for energy 662 KeV was decided based on the number of spectrum channels available and the 1.4 MeV energy range. Figure 5-29 (left) is a screen grab taken during the energy calibration process. The peak (highlighted in blue) was originally located at channel 1,025 at 18 °C. In order to shift the peak to channel 967, the gamma-ray bias setting was reduced. To allow the temperature algorithm to be implemented automatically, a temperature bias calibration curve was first developed for each module, Figure 5-29 (right).

Initially, it was hoped that a single calibration curve could be developed, and all modules could adapt the determined linear equation within its specific temperature algorithm. However, as demonstrated below in Figure 5-30, the majority of SrI₂(Eu) gamma-ray detectors differ just

enough in response to temperature to warrant a calibration of each module. A linear equation was derived from testing of each module and then applied within the module's individual temperature correction algorithm. Once several SrI₂(Eu) detectors were continuously transmitting data to the ARGANOT phone application in unison, minimal deviation was observed.

It should be noted that while the spectrum displayed on the phone indicates the axis represents energy, it in fact represents channels. Additionally, during the software development stage, a third party programmer placed keV at channel 300. In future models, this problem will be corrected.

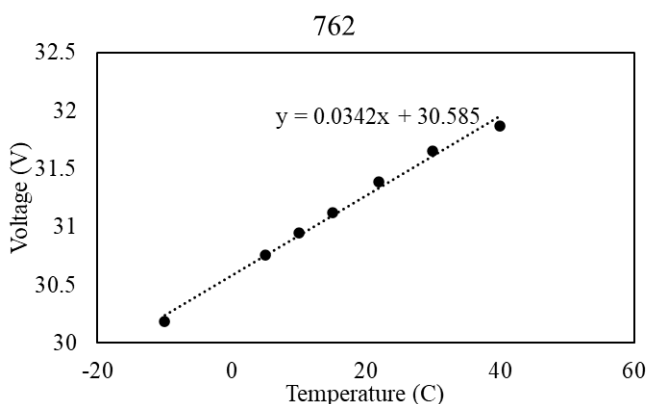
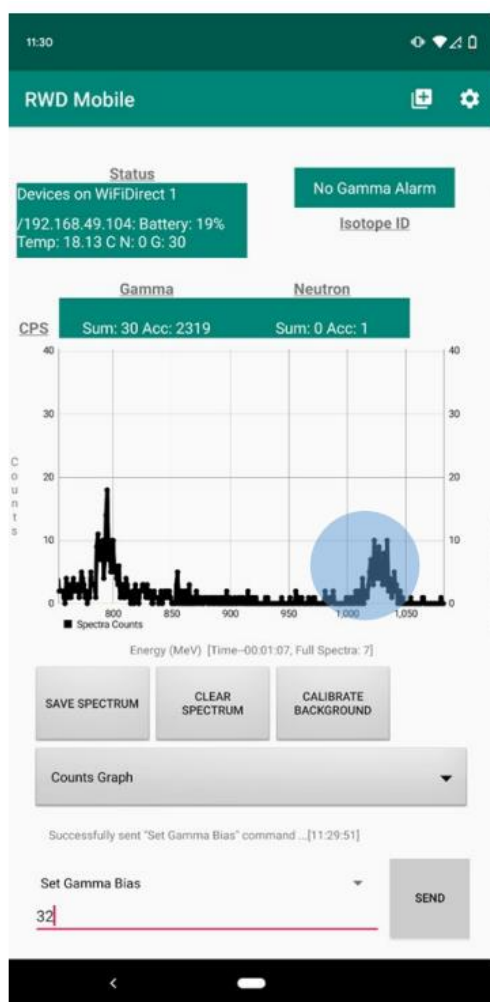


Figure 5-29 The required bias setting for the gamam detector is maintain a constant calibration was determined across a range of temperatures and then plotted. The resulting data was used to construct a line of best fit to determine the correction factor throughout a range of operating conditions.

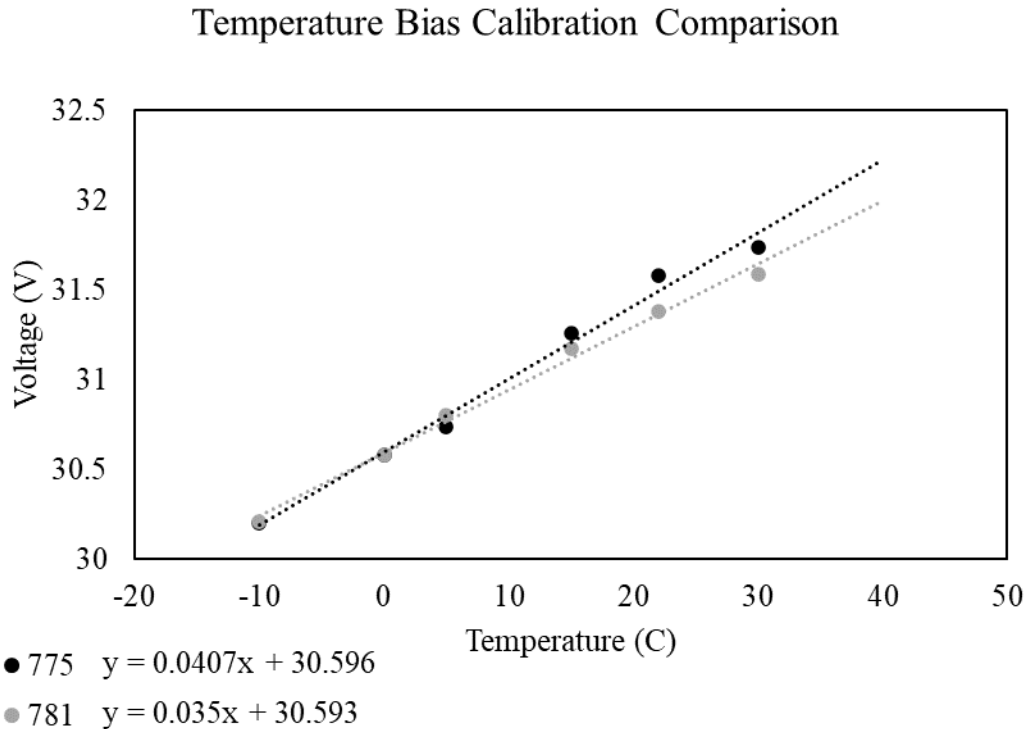


Figure 5-30 A comparison of two gamma-ray detectors' temperature calibration curves are shown. The responses differed enough to require a personalized temperature curve to be collected with each detector.

Once all temperature calibration work was completed and several modules were connected to the android app transmitting data, the spectrum produced very little deviation particularly at stable temperature conditions. With drastic temperature changes (walking indoors to outdoors), the spectrum was unaffected while the modules adjusted to temperature and temperature algorithm was implemented. An example spectrum is shown in Figure 5-31 in which several CsI(Tl) populated modules were contributing to the hour long measurement where a $1\mu\text{Ci } ^{137}\text{Cs}$ source was placed 10 cm away.

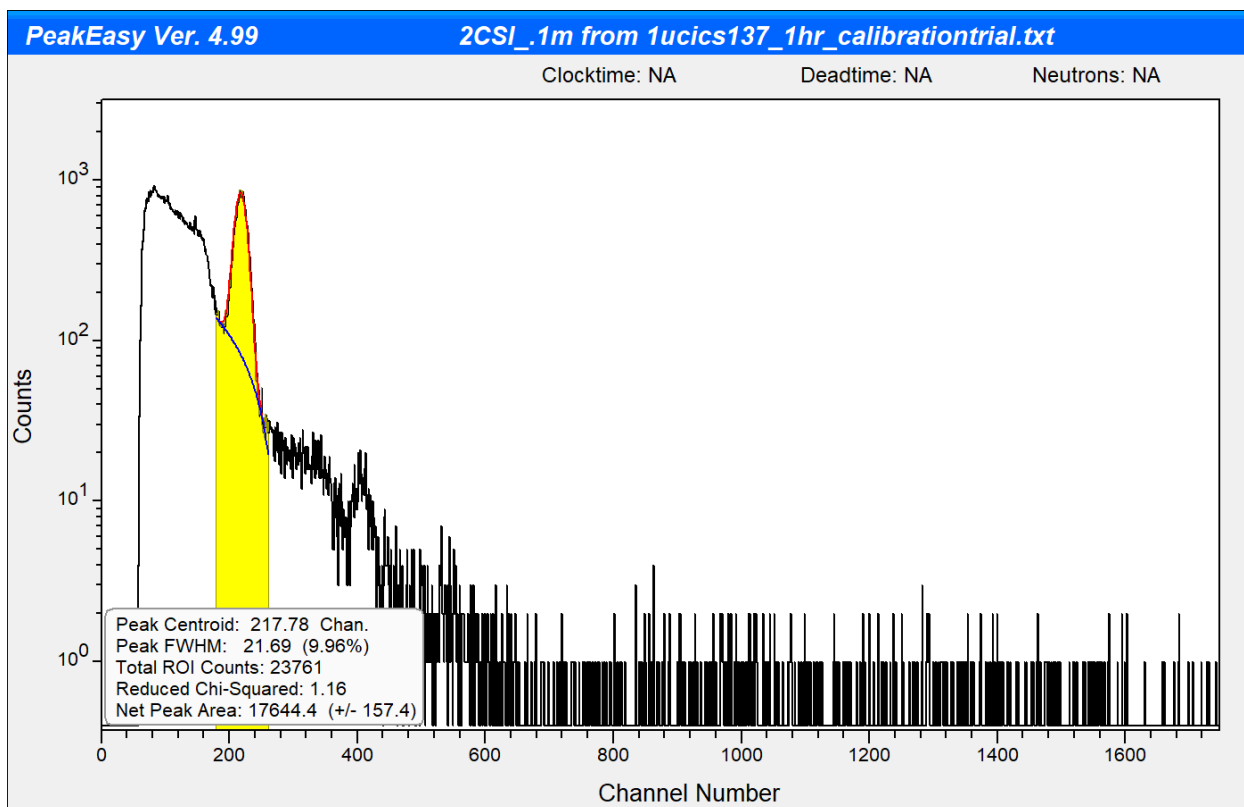


Figure 5-31 The resulting spectrum from two CsI(Tl) gamma-ray detector s when exposed to a 1uCi ^{137}Cs source at a distance of 10 cm.

6 Wearable Radiation Detection



International Atomic Energy Agency (IAEA) officials are pictured conducting a Radiological survey of the land surrounding the Chernobyl Nuclear Power Plant. Image courtesy of the IAEA imagebank [131].

“There are concerns that terrorist or criminal groups could gain access to high activity radioactive sources and use these sources maliciously. The IAEA is working with Member States to increase control, accounting and security of radioactive sources to prevent their malicious use and the associated potential consequences.” [132]

~ IAEA Officials

Special Nuclear Materials (SNM) are categorized as either fissionable (^{233}U , ^{235}U and ^{239}Pu) or fertile (^{232}Th and ^{238}U) materials. According to the IAEA, fissionable materials are a mixture of isotopes capable of nuclear fission. Some fissionable materials are capable of fission only by sufficiently fast neutrons ($> 1 \text{ MeV}$), while isotopes that undergo fission by neutrons of all energies, including slow (thermal) neutrons, are usually referred to as fissile materials. Fertile materials on the other hand are materials which can be converted into a special fissionable material through capture of one neutron and be converted to fissionable or fissile material. Nuclear fuel materials emit neutrons by spontaneous fission in even nucleus (^{238}U , ^{238}Pu , ^{240}Pu , and ^{242}Pu), neutron induced fission in fissile isotopes (^{233}U , ^{235}U , and ^{239}Pu), and alpha particle induced (α, n) reactions in light elements. In fission, multiple neutrons (average 2–3) and gamma-rays are emitted and serve as material signatures are for SNM detection [133].

Detection and accountability of fissile and fertile materials are incredibly important because can

be used to understand the operational status of any nuclear facility that is involved or suspected of being involved with fuel processing, nuclear stockpiles, and safeguards [134]. Nuclear counter-proliferation includes tracking, searching, and locating special nuclear materials (SNMs). In the post 9/11 era, ^3He thermal-neutron detectors are quite expensive. The Microstructured Semiconductor Neutron Detectors (MSND) are a high thermal neutron detection efficiency alternative that can be bundled with an inexpensive, low-profile, power-efficient electronics package. For these reasons and more, MSNDs are an ideal candidate for implementation into a wearable technology known as the Wearable Neutron Detector (WND) [135].

6.1 Review of Predecessors

Several developmental wearable detector systems such as the Wearable Detector Device (WDD V.1) came before the ARGANOT system.

6.1.1 WDD

The WDD was comprised of 16 Modular Neutron Detectors (MNDs), each with an average thermal neutron detection efficiency of 30% and two Modular Gamma-Ray Detectors (MGDs). MND incorporated Microstructured Semiconductor Neutron Detectors (MSNDs) while MGD incorporated LND model 713 Geiger-Müller detectors to count and alert to gamma-rays [136]. The MNDs were connected to controller boards and then encased in a 12.7-cm x 7.6-cm x 1.5-cm HDPE moderator case, see Figure 6-1. Each MND was then connected to a network of quad-connector boards that were daisy-chained to form the communications backbone linking the MNDs to the master control board within an inconspicuous vest. Each module of the vest then communicated to a handheld Android phone or tablet, see Figure 6-2. The WDD system provided live-time data for each of the 16 MND to android application on the mobile device notifying the operator of SNM presence. The intrinsic detection efficiency of the entire vest for the bare ^{252}Cf source ranged from 0.46% to 1.56% from 0.5 m to 2 m. The MNDs showed a low sensitivity to the incident gamma-rays with a GRR of 1.8×10^{-8} [137].

A wireless WDD was also constructed which was comprised of 16 MND and two MGD modules that could communicate with the android device either wired or wirelessly through Bluetooth Low Energy (BLE) to transmit data, see Figure 6-3. The wireless transmission eliminated the need for a master control module and central battery pack. The dimensions of the wireless MND are 11.2 cm x 6.1 cm x 1.7 cm which were slightly thicker than the wired MND due to the

individual batteries of the modules. The MND was tested to have a GRR of 6.2×10^{-9} [138]. Additionally, the intrinsic efficiency of MSND within the MND modules were inferior to common He-3 tubes. The MGD, a GM tube, was slightly larger than the wireless MND measuring 10.9 cm x 6.1 cm x 2.1 cm. One of the limited aspects of the MGD device was the inability of the GM tube to produce a gamma-ray spectrum to enable isotope identification. The next generation of wearable detectors sought to improve upon these shortcomings.



Figure 6-1 The MGDs (Left) & MNDs (Right) communicated with the android application (Top) either through a wired zero-RF low voltage differential signal (LVDS) or wirelessly through a low energy Bluetooth.

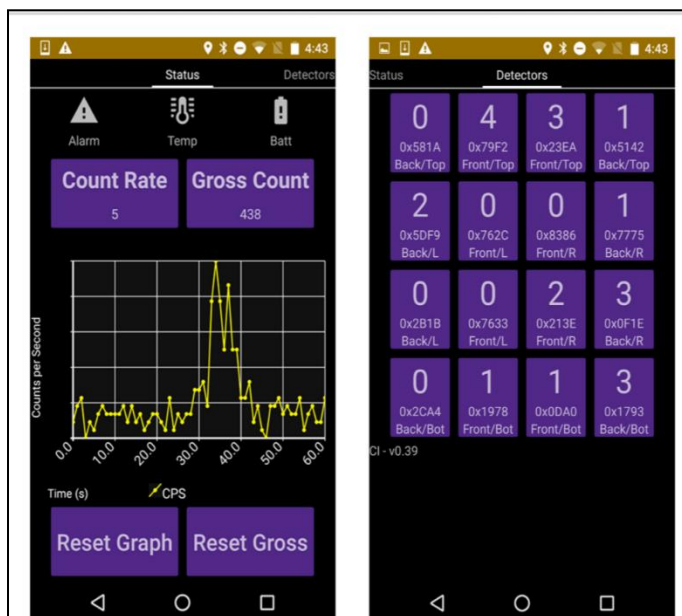


Figure 6-2 The WDD real time response were displayed on the system status page (left) while individual count rates were displayed on the detector page (right).



Figure 6-3 16 MND modules and 2 MGD modules were discretely implanted into the vest worn by the phantom. All data was fed to the users lifetime via the android cell phone.

6.2 Introduction to the ARGANOT system

Building off the WDD processors, the Area Radiation Gamma and Neutron Origin Telemetry (ARGANOT) mobile detection system was developed implementing Dual Sided Microstructured Semiconducting Neutron Detectors (DS-MSNDs) to aid in the alerting to and locating of Special Nuclear Material (SNM). Because gamma-ray background is prevalent and can vary drastically over an environment, detection of neutrons, have been chosen as the primary mode of SNM alerting with the gamma-ray detection feature acting as the confirmation and identification piece to the puzzle. To perform the source identification, each DS-MSND module has been paired with a high-resolution strontium iodide ($\text{SrI}_2(\text{Eu})$) CapeSym detector or a $\text{CsI}(\text{Ti})$ BNC detector as presented in Chapter 5 to aid in the confirmation and identification of SNM and other naturally occurring radioactive material. The final ARGANOT product may be viewed in Figure 6-4.



Figure 6-4 The final ARGANOT module (Left) is compared to the conceptualized CAD module (Right).

6.2.1 Physical

The ARGANOT modules (16 x 69 x 150 mm) used an aluminum oxide frame to allow for field rigidity, see Figure 6-5. To allow for the Wi-Fi signal to communicate with the phone still effectively and consistently, a carbon fiber lid was used on either side of the frame. The two lids were then permanently adhered to the frame. As demonstrated in Figure 6-6, the USB-C ports are slightly recessed to ensure nothing will harm it during operation. Additionally, the final button design was engineered to guarantee that it would not accidentally be activated undesirably. To power on and off the module, the user must firmly press the blue button for three seconds. The indicating light then blinks light green indicating the module is on or red if the module is powering off. Due to the seal of the casing, the ARGANOT modules can achieve IP67 protection rating upon customer request. They therefore are optimal candidates in a variety of field scenarios.



Figure 6-5 The Aluminum oxide casing frame is shown.



Figure 6-6 The final casing housing a single Module is shown.

6.2.2 System Operation

Once the individual modules are powered on and the application is open, modules will automatically begin to connect. The name of the modules should appear on the screen and the user can continue to the placement screen shown in Figure 6-7. Through following the instructions on the screen, each module can be placed as directed. For directionality to work, an equal number of modules must be placed on the front and back. Presently, the system only offers directionality if two, four, or eight modules are correctly placed on the user's body due to the current methods used in the software to calculate the directionality vector. Once all modules have been placed, the users can go on to the mission screen as displayed in Figure 6-8. The mission screen displays all module's mission status, all count rates, accumulative count rates, and the gamma-ray spectrum. Data are continually streamed every second from the module to an Android phone through a Wi-Fi wireless connection. Within the application, both gamma-ray and neutron live time count rates are updated every second. Every 10 seconds, a gamma-ray spectrum displayed on the android device screen receives an update. Based on the collected gamma-ray spectrum and GADRAS software, a user can determine the isotopes emitting radiation around them. As an operator surveys an area, an alarming threshold can be established

to alert a user to an immediate increase in the count rate. The alarming algorithm used in the ARGANOT equipment can be set by the user based on the operator's discretion or based on Equation 6- 1

$$L_C = k_{\alpha} \sigma_0 \quad (6- 1)$$

where the critical level, L_C , or an alerting count rate will be identified over a two-minute background measurement. If the incoming count rate is larger than the set critical rate, an alarm will sound to the operator. The critical level can be found with Equation 6-1 where α is the tolerance for error of the first kind (false positive), k_{α} is the Gaussian probability level corresponding to $1-\alpha$, and σ_0 is error for when net count rate is equal to zero [139].

When more than one ARGANOT module is communicating data to the application, all count rates and energy spectra are compiled live time to display overall radiation count rate data to the operator. Individual detector data can also be accessed. The ARGANOT system possesses a 1 Gb serial flash memory for on-board storage. Each ARGANOT module operates with a 1400 mAh rechargeable battery, which allows for a 10–18-hour mission without facing interruption. For these reasons, the ARGANOT system can operate in several modes. These modes include passive missions such as mobile sensing by a single or many users operating ARGANOT modules. In addition, the ARGANOT system allows for ARGANOT modules to operate in “leave behind” mode to monitor an area. The ARGANOT modules can also be placed within clothing, within a backpack, or function as a hand-held instrument. Due to the form of the modules, several modules may be distributed across several people to operate in a distributed mode. Regardless of the mode of operation, the ARGANOT system is quite sleek and compact, allowing for all operations to be conducted either overt or covertly.

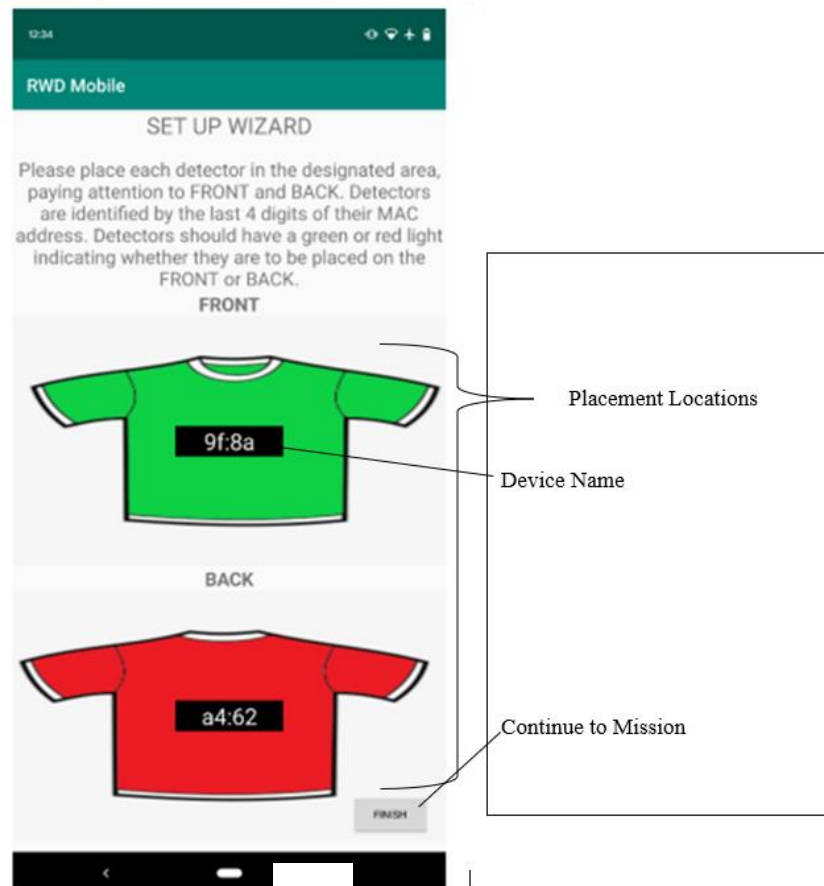


Figure 6-7 The starting page for loading modules onto the body is shown.

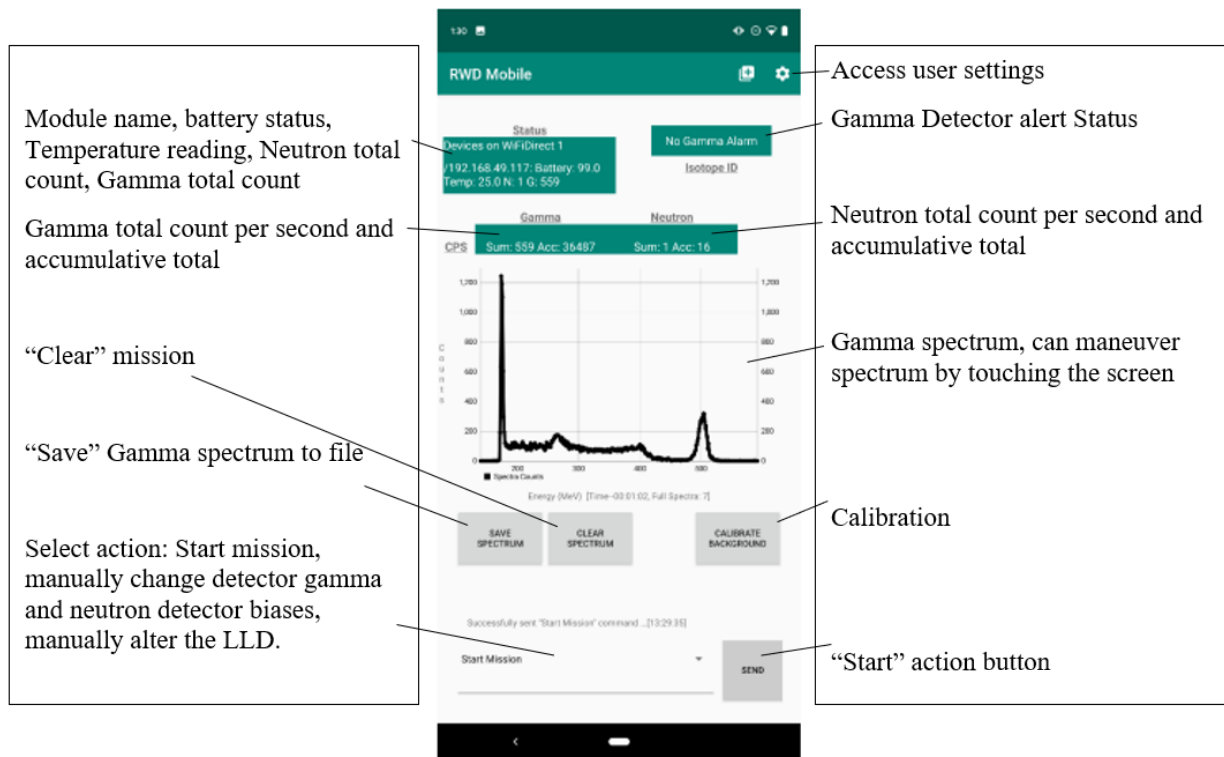


Figure 6-8 The mission screen of the system is shown. All detector data is communicated and displayed here.

While the DS-MSND bias and threshold are stagnant, the gamma-ray detector bias and discriminator thresholds are automatically adjusted based on ambient temperature measured on the electronics board to mitigate the effect of temperature on the performance of the silicon diodes. Through selecting the select action dropdown menu demonstrated in Figure 6-8, the user may select to halt the mission or if access permissions has been previously been granted, access advanced user settings such as the manual detector biases, thresholds, and alerting options.

6.2.3 Neutron Detectors

The ARGANOT modules incorporate nine DS-MSND devices across three channels. To populate the final delivered modules populated with DS-MSND, 15 wafers were processed. The wafers on average achieved a backfill percentage of only 36%. They were therefore limited in efficiency to much less than the desired 60%. Throughout processing implementing the new doping and diffusion processing parameters, the leakage currents were higher than expected. Table 6-1 below displays the top side only leakage current results before completing the packaging of the DS-MSND device. After completing the packaging stage, the leakage current again increased. When the ARGANOT electronics were designed by EDL, three DS-MSND

devices were assigned to a channel readout. Therefore, in avoid amplifier saturation, the total leakage current of all three devices per channel was not to exceed ~100 namps. Therefore, were assigning the location of DS-MSND devices within each module, the leakage current of each device was heavily considered.

			27.8	22.6	22.2			
	15	10.1	9.1	8.5	9.5	12.1	28	
	10.1	6.1	7.7	7.7	230	8.2	12	
7.7	5.2	5.3	7.1	1200	1300	724	13	14
18	9.5	6.4	7.9	10.6	9.4	10.6	24.6	10
	72	9.6	12	12.4	14.4	13	22	
	12.6	13.8	18.1	18.8	19.3	20	1200	
			900	24	17.9			

Table 6-1 The leakage current in nA at -3V across a wafer (top side only) mid-processing for each diode is shown. Once the backside was connected, the leakage values on average doubled. The devices in orange represent a “no compliance”, indicating they were un-usable devices.

All devices with reasonably lower leakage currents underwent neutron testing with a moderated ^{252}Cf source to determine the intrinsic thermal neutron detection efficiency of each device. Using an MSND device with a known detection efficiency of $30.8\% \pm 0.4\%$, the flux of the source was determined according to the McGregor & Shultis calibration method. Each DS-MSND device biased at -3V was then placed into the exact same position and a two-minute foreground, followed by a two-minute background was collected. Resulting intrinsic thermal neutron detection efficiencies range from $25\% \pm 0.2\%$ to $30\% \pm 3.7\%$, once again leading to suspicion that the top side of the device was largely the sole contributor to the spectrum. When the p bond was removed, making the device a single sided device, the efficiency for three of the five devices sampled were largely unaffected, see Table 6-1 for the comparison. Only two devices (in bold) experienced an efficiency decrease. One theory as to why the backside contribution was minimal was that the SiO_2 and or dry film was not completely removed.

Table 6-1 A comparison of the DS-MSND device when operating as a dual sided device and a single sided device once the p type contact was removed is shown. Notice that the efficiency of only two devices were significantly reduced when operating as a single sided device.

NIM testing 8-26 Mod ²⁵² Cf	Background	Foreground	Time (s)	Efficiency	Error ±	
406	0	234	60	31%	0%	
Acting As Dual Sided Device	Leakage Current (Amps)	Background	Foreground	Time (s)	Efficiency	Error ±
DS-MSND 312	6.00E-07	0	232	60	30.7%	3.1%
DS-MSND 366	1.50E-08	0	250	60	33.1%	3.3%
DS-MSND 331	7.00E-08	0	296	60	39.2%	3.6%
DS-MSND 384	5.00E-08	0	232	60	30.7%	3.1%
DS-MSND 320	2.20E-05	0	336	60	44.5%	3.7%
Acting As a Single Sided Device (p Bond Wire Removed)						
DS-MSND 312	6.00E-07	0	229	60	30.3%	3.0%
DS-MSND 366	1.50E-08	0	239	60	31.7%	3.1%
DS-MSND 331	7.00E-08	0	151	60	20.0%	2.0%
DS-MSND 384	5.00E-08	0	286	60	37.9%	3.7%
DS-MSND 320	2.20E-05	2	166	60	21.7%	2.2%

At the conclusion of the device processing stage of the project, 75 DS-MSNDs devices were deemed to have acceptable leakage currents and efficiencies and underwent extensive testing in a board resembling the final electronics, see Figure 6-9. A sample spectrum from these tests is shown in Figure 6-10 in which an Electronic Design Laboratory (EDL) specially designed circuit with a shaping time of 21μs is compared to a spectrum collected with a shaping time of 12μs. The results revealed the fission reaction product peak using the 21μs shaping time shifted to

higher channels than with the NIM electronics and was selected as the final shaping time to be used in all modules. Additionally, the noise floor appeared to shift to lower channel numbers, allowing a decreased Lower-Level Discriminator (LLD) to be used and a clear noise to signal separation to become apparent. The shift in channel enabled a more accurate distinction between noise counts and neutron counts. With the previously shorter shaping times, the neutron peak was meshing with the noise peak, resulting in the loss of an unknown number of neutron counts. Because of the distinguishable difference in noise and neutrons, the efficiencies increased. These results therefore motivated a decision to implement the longer shaping time in the refined ARGANOT digital board design. A final comparison of the EDL versus NIM spectrum is shown in Figure 6-11. The EDL circuit was able to shift the noise floor to a lower channel than the NIM electronics when exposed to a moderated ^{252}Cf , allowing for improved discrimination and efficiencies. Figure 6-12 demonstrates device performance when tested in the final digital board. The reaction product spectrum shown in Figure 6-12 was a signal device's response to a moderated ^{252}Cf source over a two-hour measurement followed by the background.

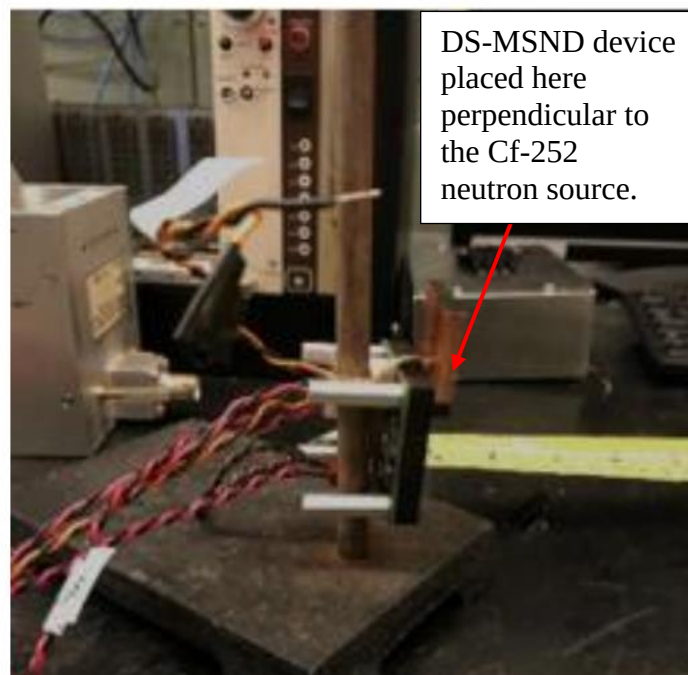


Figure 6-9 The initial EDL circuit designed to resemble the final module circuit is shown. The initial circuit was optimal for quickly altering various parameters such as the shaping time.

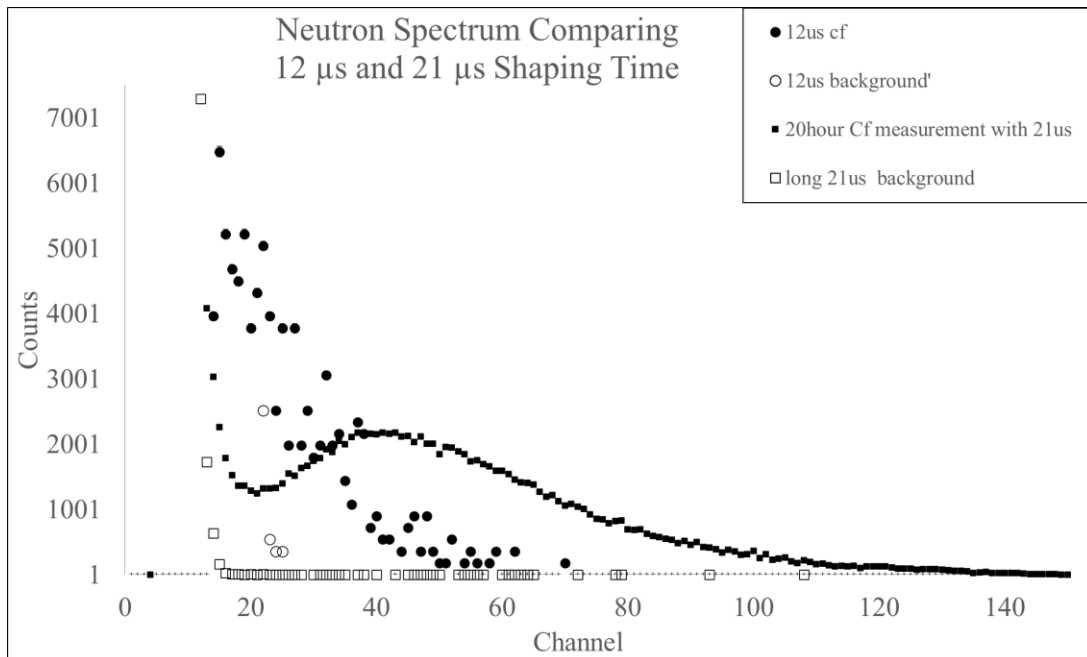


Figure 6-10 The Comparison of a shaping time of 12 μ s and 21 μ s is shown. Notice the clear separation of noise and signal with the increased shaping time.

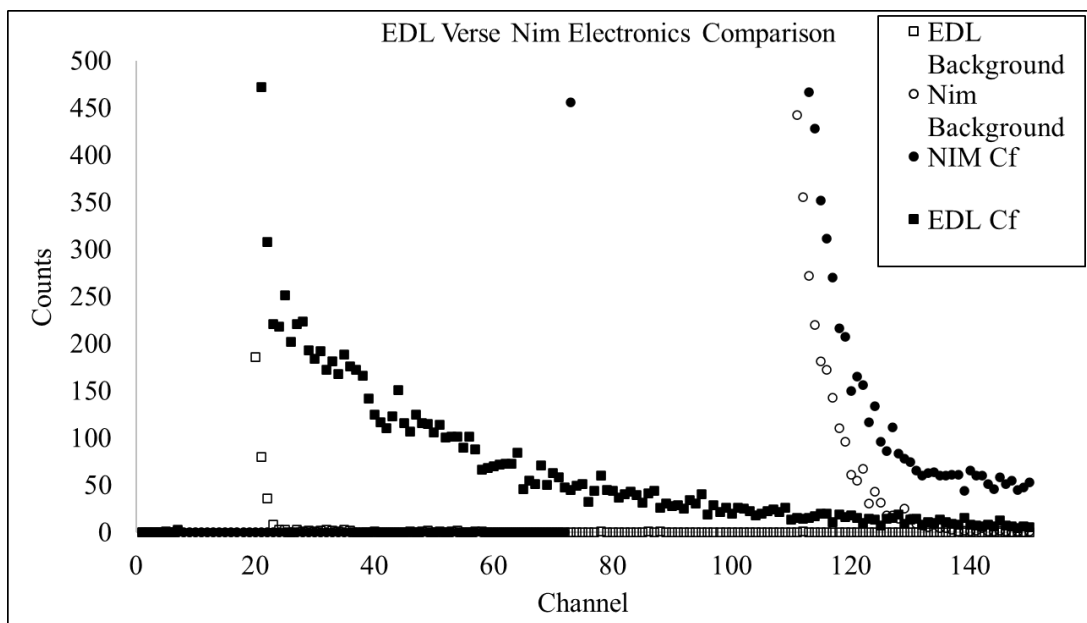


Figure 6-11 A comparison between a DS-MSND NIM electronics collected spectrum and a Electronics Design Laboratory (EDL) designed electronics spectrum is shown. Notice the EDL specialized circuit shifted the noise floor to a lower channel, allowing for a decreased lower level discriminator.

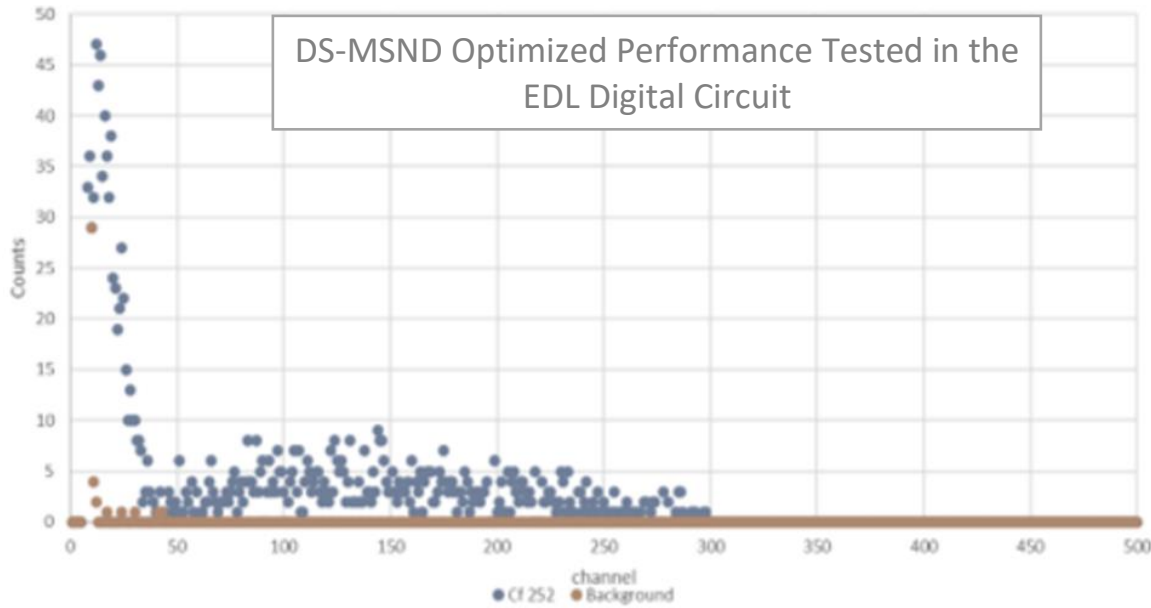


Figure 6-12 The 2-hour moderated ^{252}Cf source and background spectrum for DS-MSND device 466 collected with the final EDL designed circuit is shown.

6.2.4 MSND Based Detector Lifespan

MSND based technology's radiation hardness to neutron exposure was examined to determine the point and degree of device degradation with increasing thermal neutron exposure. MSNDs used were constructed with 10 k Ω cm n-type Si with trench dimensions of approximately 20- μm wide and 450- μm deep with pvn dopants. While the ionization of the silicon by the reaction products is fundamental for device operation, these same energetic alpha particles and tritons traversing the silicon cause displacement damage to the silicon lattice structure [140].

Energetic alpha particles and tritons are capable of creating anti-sites, voids, and interstitials within the Si crystal lattice that create electron and hole traps within the band gap of the semiconductor. At high neutron fluences, this displacement damage will cause performance degradation [141]. Assuming a displacement energy (E_{th}) of 20 eV, SRIM simulations indicate that vacancy production in Si can exceed 70 displacements ion^{-1} for a 2.73 MeV triton and 140 displacements ion^{-1} for a 2.05 MeV alpha particle [142]. Although the alpha particle introduces more damage to the lattice, they are less important in the detection scheme than the tritons by a

ratio exceeding 3:1 [143]. The simulated vacancy density per ion as a function of the ion track length is shown in Figure 6-13 for 2.05 MeV alpha particles and 2.73 MeV tritons.

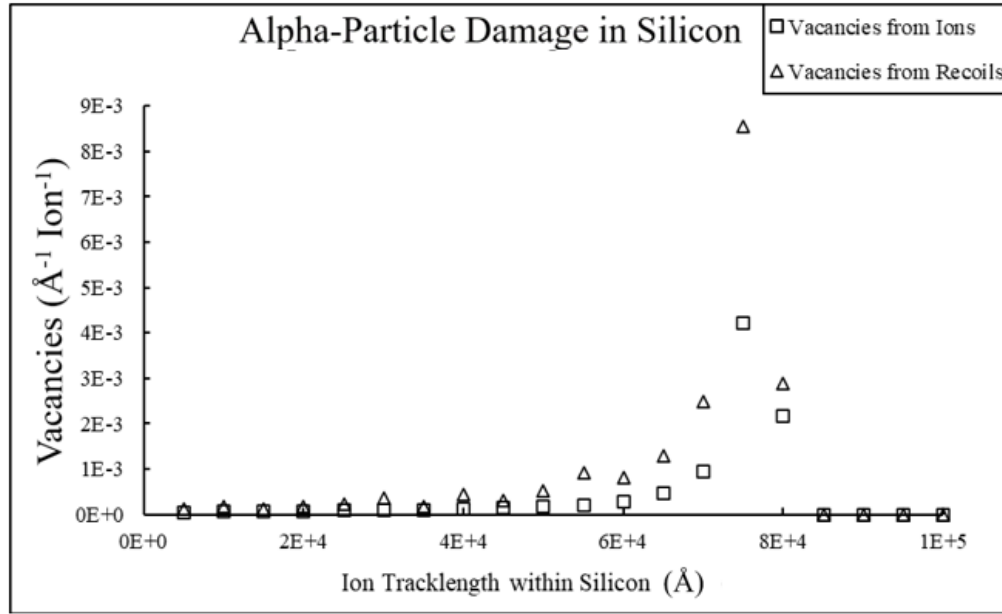


Figure 6-13 The alpha particle damage to silicon was simulated with SRIM software.

Electron-hole losses increase with the trap density, generally described by the Shockley-Read-Hall (SRH) electron capture rate R_{cn} , [144]

$$R_{cn} = \frac{nB_{cn}N_{Tn}}{1 + \exp\left[\frac{E_F - E_T}{kT}\right]}, \quad (6-2)$$

where N_{Tn} is the unfilled electron trap density, E_F is the fermi energy level, E_T is the electron trap energy, n is the electron population, β_{cn} is a proportionality constant for electron capture, and k is the Boltzmann constant with temperature T in Kelvin. In a similar manner, the hole capture rate R_{cp} from the valence band is, [145]

$$R_{cp} = \frac{pB_{cp}N_{Tp}}{1 + \exp\left[\frac{E_T - E_F}{kT}\right]}, \quad (6-3)$$

where N_{Tp} is the unfilled hole trap density, p is the hole population, and β_{cp} is a proportionality constant for hole capture. Note that thermal carrier generation is usually considered a minor effect in the semiconductor depletion region. As the electron and hole trap densities N_T increase, so do the carrier losses to capture, which works to diminish the detector signal current and voltage output signal. Consequently, the resulting reaction product spectrum features shift to

lower energy channels as damage increases. Because direct ionization of silicon with thermal neutrons is improbable and relatively small for fast neutrons, damage contributions from thermal and fast neutrons were assumed to be negligible [146].

To determine the point of degradation, five devices with average characteristics were selected from a pool of MSNDs. These detectors were each 1 cm² area. IV characteristics were documented for each device before conducting neutron measurements, shown in Figure 6-14.

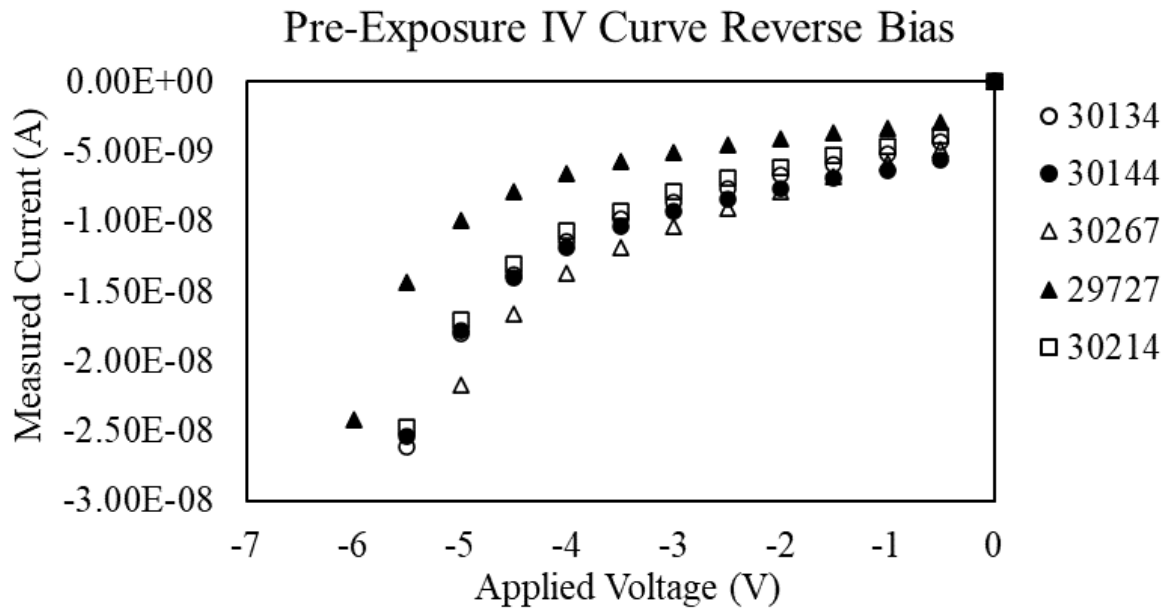


Figure 6-14 The pre-exposure IV curves for each device in reverse bias are shown.

The detectors were tested at the Kansas State University (KSU) TRIGA Mk II nuclear reactor facility. Samples were tested in three locations, identified as follows. A diffracted neutron beam port, located in the northwest beam port (NWB), has a 0.0253 eV thermalized neutron beam from a graphite diffractometer. Due to the graphite crystal structure factor, the even harmonics ($n = 2, 4, 6, \dots$) cancel; hence some beam contamination may be present from the third harmonic, but is well past the average in the Maxwellian distribution [147]. A penetrating beam port is located in the northeast beam port (NEBP) and serves to produce a higher neutron flux as well as a combined fast and thermal neutron flux. The beam port penetrates through the bio-shield, the water moderator, and the graphite reflector up to the reactor core face. The thermal and fast neutron fluxes are comparable, similar to the ratio in the reactor core. Dry tubes adjacent to the graphite reflector around the reactor core, referred to as the inter-reflected irradiation

system (IRIS), were used for the highest neutron fluence exposures Figure 6-15. These six dry tubes are in the same relative positive with respect to the core and provide a method to insert samples for irradiation and activation near the reactor core.



Figure 6-15 The Triga Mk II nuclear reactor core is shown. The control and fuel rods are labels along with the IRIS port numbers 1 through 6.

The diffracted thermal neutron beam (0.0253 eV) at the KSU TRIGA Mk II nuclear reactor was used for initial intrinsic thermal neutron detection efficiency (ϵ_{tn}) characterization. The nuclear reactor was operated at 50 kW, and the diffracted beam was stopped down to a 0.25 cm diameter beam with a Cd collimator sheet for the tests. This reduction in beam diameter ensures that the witness detector (^3He gas-filled detector) and the test MSND completely intersect the neutron beam. The method of McGregor and Shultis was used to measure the thermal neutron flux and detector efficiencies [148]. The He-3 gas-filled detector had a working length of 8 cm and is 5.25 cm diameter with measured ϵ_{tn} of $80.7 \pm 0.5\%$. The He-3 detector was set approximately 12 inches (30 cm) back from the collimator exit. A background measurement with the Cd collimator closed with a shutter was collected for 180 seconds. A second 180 second measurement was then collected with the shutter open. Afterwards, each MSND device was placed in the same location as the He-3 tube within an aluminum test box (with electronics).

Neutron measurements were then conducted with the Cd shutter open and shut as before for each of the five MSND devices. The efficiency of each device was then calculated from the He-3 witness detector results.

Following the initial characterization of the detectors, the He-3 tube was placed into a 14-cm long Cd sleeve with a hole (5-mm diameter) in the center. The shielded He-3 detector was aligned with the closed NEBP and a background measurement was collected for 300 seconds at 0 kW. The reactor was then brought up to 100 kW with the He-3 tube in the Cd sleeve still aligned with the penetrating fast/thermal neutron beam port. A measurement was collected for 33.4 seconds live time, where the He-3 detector registered 67% deadtime. At the conclusion of the measurement, the reactor power level was reduced to 1 kW and a second measurement was collected for 329.7 seconds live time, where the He-3 detector registered 9% deadtime.

As discussed elsewhere, the neutron reaction rate in the He-3 detector is practically constant for He-3 detectors in the $1/v$ region of the neutron spectrum [149]. Further, the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction neutron cross section for energies exceeding 100 keV, at the end of the $1/v$ region, is over 1900 times smaller (< 2 barns) than thermal neutron energies (3844 barns at 2200 m s^{-1}). A similar situation exists for the $^6\text{Li}(n,t)^4\text{He}$ reaction. Therefore, contamination from fast neutron interactions in the measurements was determined to be inconsequential, meaning the thermal neutron response calibration is valid for the NEBP. Using the calculated flux, it was determined that the NEBP would be capable of reasonably attaining a range of thermal neutron fluences from 10^7 to 10^9 cm^{-2} . The irradiation times and power levels were calculated to reach the desired fluences as listed in Table 6-2.

Table 6-2 The determined irradiation times and power levels determined to achieve the desired fluences at the NEBP are shown. Notice that fluences greater than 10^9 were achieved using the IRIS port.

Northeast Beam Port

Desired Fluence	Irradiation times (seconds)	Reactor Power level (kW)
1.00E+07	180	4
1.00E+08	300	40
1.00E+09	540	400

Each MSND was then placed into the NEBP at the predetermined power for time required to reach each fluence decade. At the conclusion of each fluence decade irradiation, each device was taken to the northwest beam port (NWBP), the diffracted thermal neutron beam, and placed in the same location as before for an efficiency characterization measurement. At 50 kW, each device was used to collect a 120 second foreground and background measurement and the reaction product spectra were recorded. Using the initial He-3 calibration measurement, the intrinsic thermal neutron detection efficiency was then determined for each device. Following the characterization, the devices were returned to the NEBP to receive the next increment in fluence irradiation.

After testing the MSNDs that received a fluence of 10^9 cm^{-2} , they were placed in the IRIS dry tubes adjacent to the TRIGA reactor core (shown in Figure 6-15) to reach thermal neutron fluences of 10^{10} to 10^{12} cm^{-2} . The anticipated exposure times for each reactor power level were determined from previous information available regarding the fluence profile of the reactor core and are listed in Table 6-3. Between each neutron fluence exposure, the devices were taken to the diffracted NEBP for characterization measurements. As before, at 50kW, a 120 second foreground and background measurement was collected with each device, spectra recorded, and the ϵ_{tn} values were determined. The devices were then returned to the IRIS dry tubes for the next decade of exposure. To ensure uniform results, the devices were consistently irradiated in the same dry tube. It is noted that the MSND 30214 placed in the dry tube port 3 of the IRIS tubes

(IRIS 3) returned wet, indicating that moderator water breached the port, which could have affected the fluence received by device 30214.

Additionally, to ensure that each MSND received the desired neutron fluence, pure Au foils were placed inside plastic pill bags with the devices of IRIS ports 1-4 for neutron activation analysis calibration. An additional Au foil activation analysis performed with cadmium filters was used to determine the fast neutron component (using the resonance integral neutron activation cross section of Au-186) for which the thermal fluence of the unfiltered devices could be calculated [150].

Table 6-3 The determined irradiation times and power levels determined to achieve the desired fluences within the IRIS porta are shown.

<i>IRIS Port</i>		
Desired Fluence	Irradiation times (seconds)	Reactor Power level (W)
1.00E+10	180	10
1.00E+11	180	100
1.00E+12	180	1000

Following the individual exposures under the conditions listed in Table 6-3, the samples were removed from the IRIS dry tubes and placed into a storage location in the reactor pool, but away from the reactor core. For fluences of 10^{10} cm^{-2} and 10^{11} cm^{-2} , the MSND and Au foil samples were removed from the pool storage location within an hour of irradiation to perform activation analysis and the neutron measurements. Following the 10^{12} cm^{-2} fluence exposure, the MSNDs became moderately activated and it was determined that the devices should remain at the top of the IRIS dry tubes within the reactor pool storage for three days before removal. Following the 10^{12} cm^{-2} exposure and characterization, the MSNDs were placed in storage for two years until the longest-lived activated isotope decayed to comparable background radiation levels. Afterwards, the MSND IV electrical characteristics were evaluated.

Using the spectra collected at 1 kW and 100 kW with the ^3He detector in the Cd sleeve, the thermal neutron fluences for the MSND devices in the NEBP were determined. The calculated

NEBP thermal neutron fluences intersecting the MSNDs are listed in Table 6-4 and were in close agreement with expectations from legacy documented neutron fluxes for the same location and power levels [151]. Previous reactor reports documented the thermal to fast neutron flux ratio as 1.92 for the NEBP [152]. Using this same ratio, the fast neutron flux incident on the devices for the exposures was estimated to range from $9.57 \times 10^7 \pm 2.15 \times 10^3 \text{ n cm}^{-2} \text{ s}^{-1}$ to $5.63 \times 10^9 \pm 1.64 \times 10^4 \text{ n cm}^{-2} \text{ s}^{-1}$ (see Table 6-4).

Table 6-4 The empirically determined thermal fluence that the MSND devices received at the NEBP for all three fluence magnitudes were found. Using existing knowledge of the thermal to fast ratio (0.52) of the particular beam port, the fast neutron fluence was estimated.

Desired Thermal Fluence	Power (Watts)	Time (s)	Thermal Neutron Fluence	Fast Neutron Fluence
1.00E+07	4000	180	$4.98\text{E}+07 \pm 1.1\text{E}+3$	$9.57\text{E}+07 \pm 2.15\text{E}+3$
1.00E+08	40000	300	$5.85\text{E}+08 \pm 3.82\text{E}+3$	$1.13\text{E}+09 \pm 7.36\text{E}+3$
1.00E+09	400000	540	$2.93\text{E}+09 \pm 8.55\text{E}+3$	$5.63\text{E}+09 \pm 1.64\text{E}+4$

Calculated thermal-neutron fluxes for the various IRIS tubes as determined with Au foil activation are listed in Table 6-5. Witness Au foils were placed in IRIS tubes 1 – 4 and the average used for the calculations, including for samples placed in IRIS tubes 5 and 6. IRIS tube 3 developed a water leak, thereby introducing error in the thermal-to-fast neutron flux ratio. The fast-neutron flux component measured three months later after repairing the water leak in IRIS tube 3 yielded a power-normalized fast-neutron flux of $1.5 \times 10^6 \pm 4.27 \times 10^4 \text{ n cm}^{-2} \text{ s}^{-1} \text{ W}^{-1}$, which would have resulted in a higher activation than observed during the experiment described herein. Consequently, the neutron fluence incident upon MSND devices irradiated within IRIS tube 3 during the experiment described herein remain unknown.

Using the values listed in Table 6-5, the thermal and fast neutron fluences were calculated for each device per the desired thermal-neutron fluence as shown in Table 6-6 and 6-7. As listed in Table 6-6, the measured thermal-neutron fluence for the targeted fluence of $10^{10} \text{ n cm}^{-2}$ irradiation were lower than anticipated. All other fluences were relatively close to the

target values. The calculated fluence that device MSND 30214 experienced in IRIS tube 3 is uncertain due to the water breach of the dry tube.

Table 6-5 The Fast and thermal fluxes achieved with each IRIS port was determined through Au foil activation. IRIS port three was discovered at the conclusion of the exposures to be waterlogged.

Iris Port	Thermal Flux (n cm ⁻² s ⁻¹ W ⁻¹)		Fast Flux (n cm ⁻² s ⁻¹ W ⁻¹)	
1	4.81E+06 ±	1.58E+04	9.04E+04 ±	2.04E+03
2	4.53E+06 ±	1.68E+04	8.78E+04 ±	2.17E+03
3	-5.72E+05 ±	1.46E+04	9.37E+04 ±	1.99E+03
4	4.77E+06 ±	1.18E+04	8.24E+04 ±	1.52E+03
5	4.69E+06 ±	1.45E+04		
6	4.69E+06 ±	1.45E+04		

Table 6-6 Thermal neutron power-normalized fluxes, as determined by Cadmium filtered gold foil neutron activations within the corresponding IRIS ports.

	Desired Thermal Neutron Fluence	1.00E+10		1.00E+11		1.00E+12	
Iris Port	Achieved Thermal Neutron Fluence (n/cm ⁻²)						
1		8.66E+09 ±	2.84E+07	1.44E+11 ±	4.74E+08	1.73E+12 ±	5.69E+09
2		8.16E+09 ±	3.03E+07	1.36E+11 ±	5.05E+08	1.63E+12 ±	6.06E+09
3*		-1.03E+09 ±	2.62E+07	-1.72E+10 ±	4.37E+08	-2.06E+11 ±	5.24E+09
4		8.59E+09 ±	2.13E+07	1.43E+11 ±	3.55E+08	1.72E+12 ±	4.26E+09
5		8.44E+09	2.61E+07	1.41E+11	4.36E+08	1.69E+12	5.23E+09
6		8.44E+09	2.61E+07	1.41E+11	4.36E+08	1.69E+12	5.23E+09

Table 6-7 Fast neutron power-normalized fluxes, as determined by cadmium filtered gold foil neutron activations within the corresponding IRIS ports.

Iris Port	Desired Thermal Neutron Fluence	1.00E+10		1.00E+11		1.00E+12	
	Achieved Fast Neutron Fluence (n/cm ⁻²)						
1		1.63E+08 ±	3.68E+06	2.71E+09 ±	6.13E+07	3.26E+10 ±	7.35E+08
2		1.58E+08 ±	3.90E+06	2.63E+09 ±	6.51E+07	3.16E+10 ±	7.81E+08
3*		1.69E+08 ±	3.59E+06	2.81E+09 ±	5.98E+07	3.37E+10 ±	7.18E+08
4		1.48E+08 ±	2.73E+06	2.47E+09 ±	4.55E+07	2.97E+10 ±	5.46E+08
5		-	-	-	-	-	-
6		-	-	-	-	-	-

Using the pre-fluence measurements conducted at the NWBP, the intrinsic thermal neutron detection efficiencies of the five MSND devices were determined as listed in Table 6-8 using the McGregor and Shultis method of calibration [153]. To correct the neutron flux between the difference in aperture settings of 1mm and 2.5 mm, the ratio between the apertures was scaled by 6.25. The resulting spectra following each fluence increment are shown in Figure 6-16 to Figure 6-20. For all devices, as reflected in Table 6-9, there was an initial increase in peak area for fluences 10^7 cm^{-2} and 10^8 cm^{-2} . This increase was attributed to changes in the region of interest LLD as the spectrum changed and increased noise. However, by 10^9 n cm^{-2} , the peak height began to decrease and shift to lower energy channels (Figure 6-21) which displays the peak area centroid shift with increasing fluence. By $10^{10} \text{ n cm}^{-2}$, the spectra drastically decreased in net counts and shifted further to lower energy channels. By $10^{11} \text{ n cm}^{-2}$ and $10^{12} \text{ n cm}^{-2}$, the spectra were drastically reduced in both net counts and energy, caused by charge carrier trapping and indicating that substantial damage of the silicon crystal lattice had occurred. The energy peak area per device as a function of fluence, including the associated error, are shown in Figure 6-22.

Table 6-8 Using an 80.7% efficient He-3 detector (top) in accordance with the McGregor and Shultis calibration method, the intrinsic thermal neutron detection efficiency for each device when measured in a collimated diffracted NWBP was determined for the before (middle) and after (bottom) irradiation exposures.

<i>Pre-Irradiation</i>					
Device	Foreground (counts)	Time (s)	Intrinsic Thermal-Neutron Detection Efficiency		
30267	7361	180	21.46%	±	0.19 %
30134	14216	180	31.85%	±	0.26 %
29727	12103	180	35.29%	±	0.24 %
30144	9635	180	22.07%	±	0.22 %
30214	11617	180	33.87%	±	0.24 %
<i>Post-Irradiation</i>					
30267	1560	180	3.49%	±	0.36 %
30134	1782	180	3.99%	±	0.41 %
29727	2968	180	6.65%	±	0.67 %
30144	3553	180	7.96%	±	0.80 %
30214	1974	180	4.42%	±	0.45 %

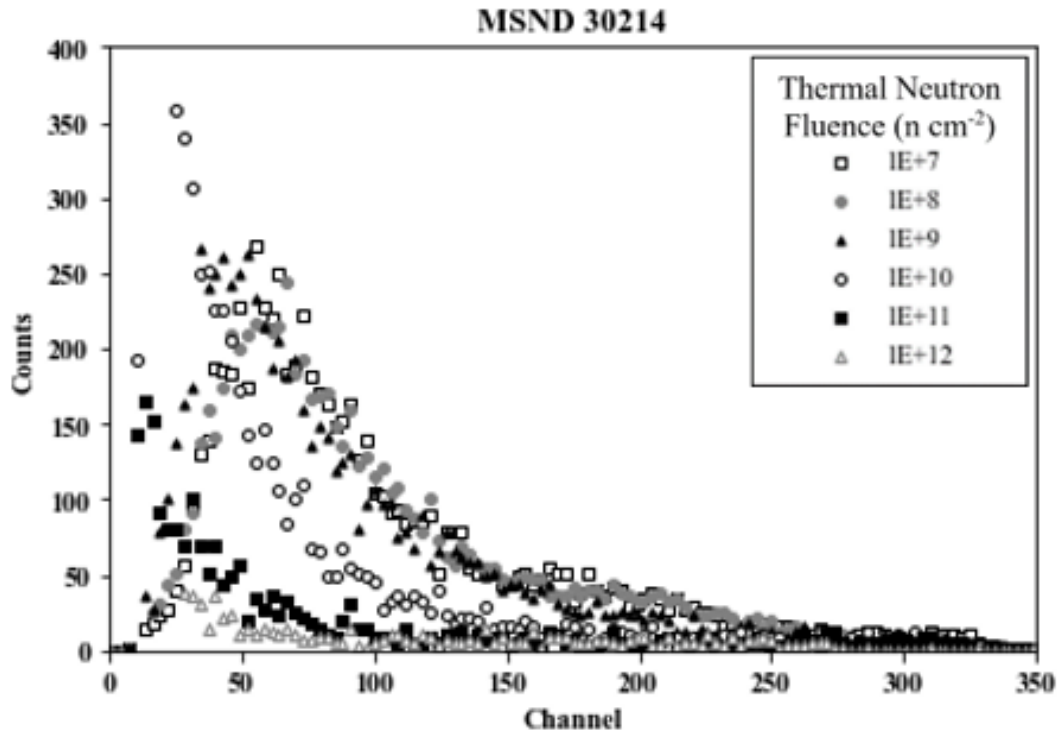


Figure 6-16 The reaction product spectrum from the Triga Mk II NW thermal neutron beam port following each decade of thermal neutron exposure is shown for device 30214.

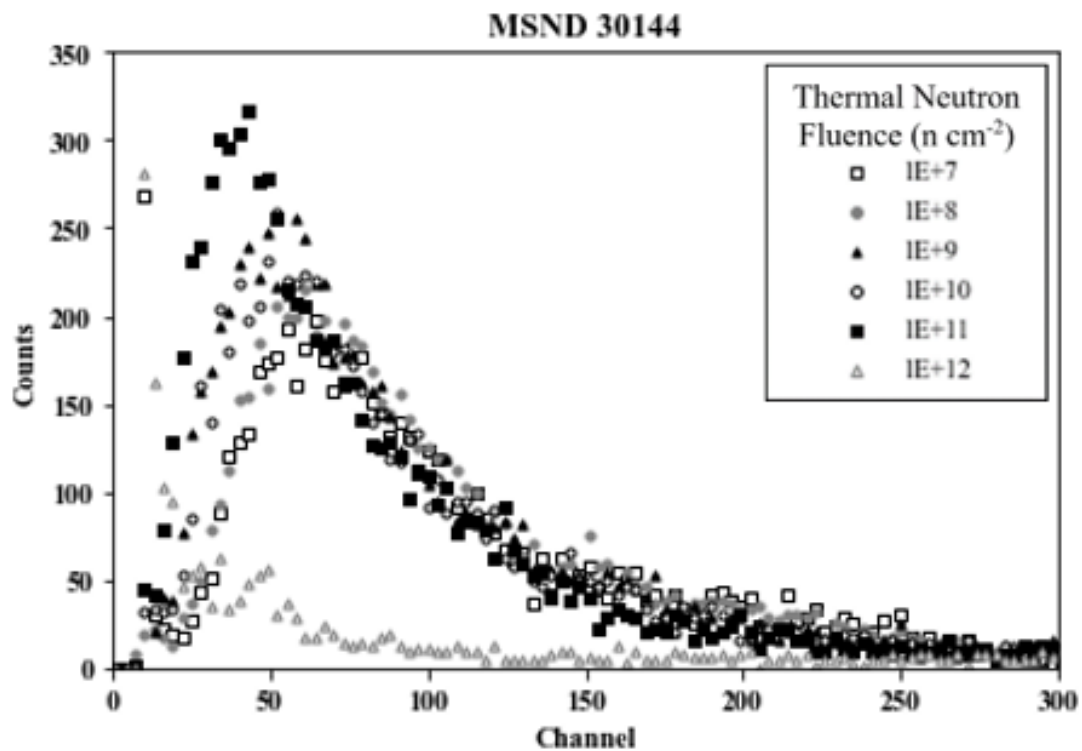


Figure 6-17 The reaction product spectrum from the Triga Mk II NW thermal neutron beam port following each decade of thermal neutron exposure is shown for device 30144.

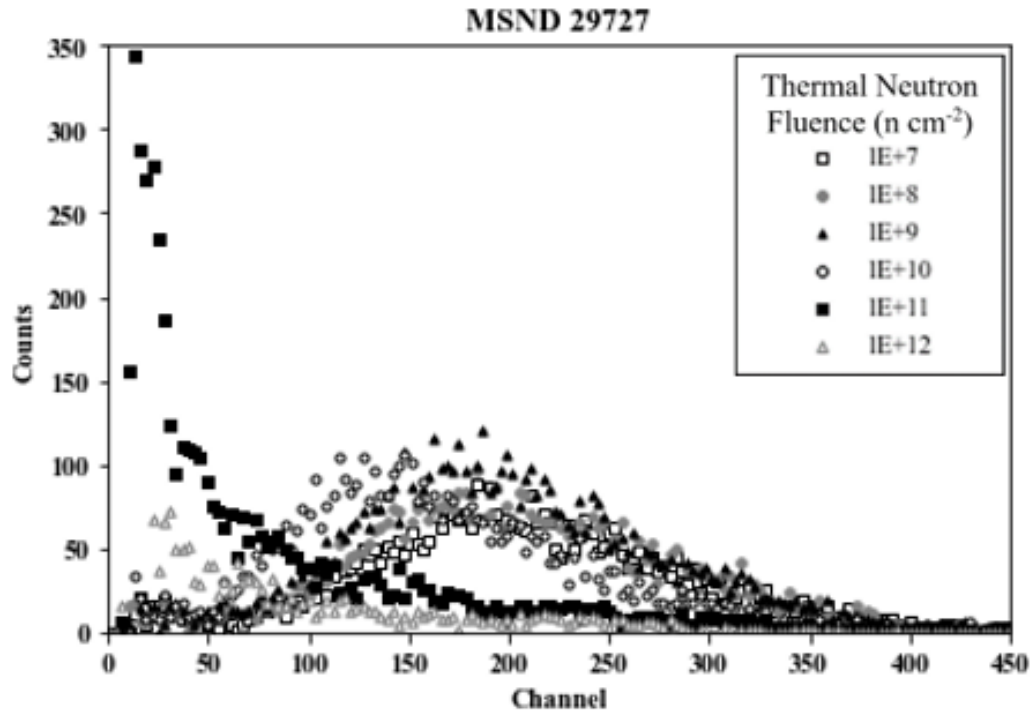


Figure 6-18 The reaction product spectrum from the Triga Mk II NW thermal neutron beam port following each decade of thermal neutron exposure is shown for device 29727.

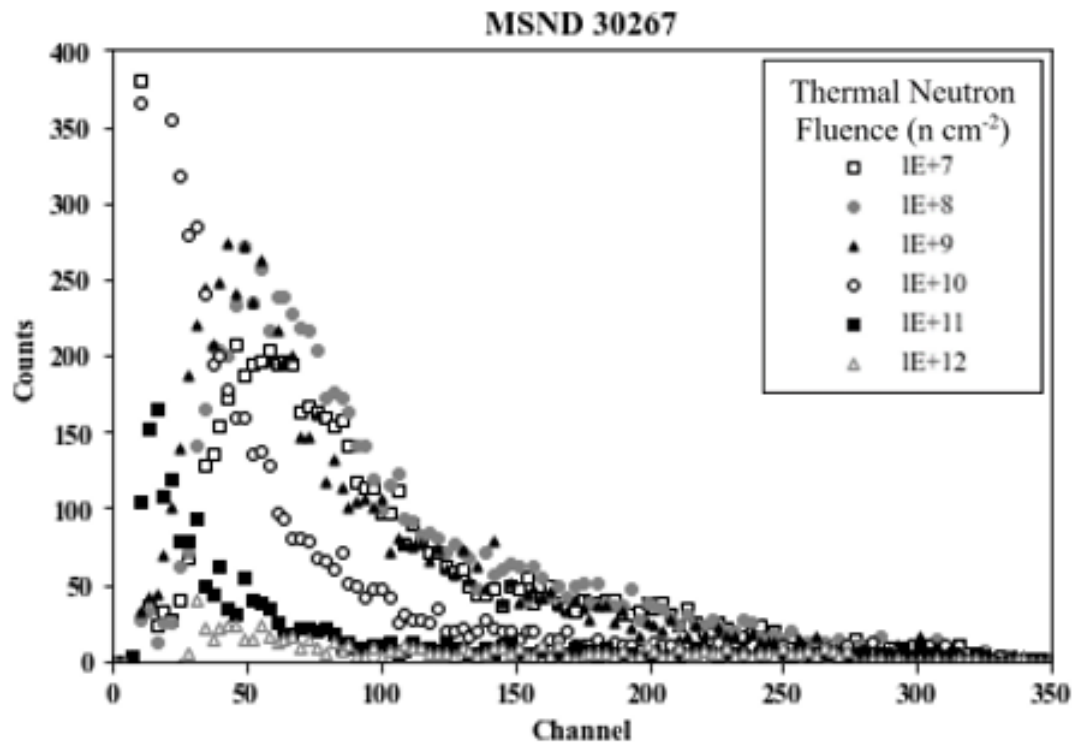


Figure 6-19 The reaction product spectrum from the Triga Mk II NW thermal neutron beam port following each decade of thermal neutron exposure is shown for device 30267.

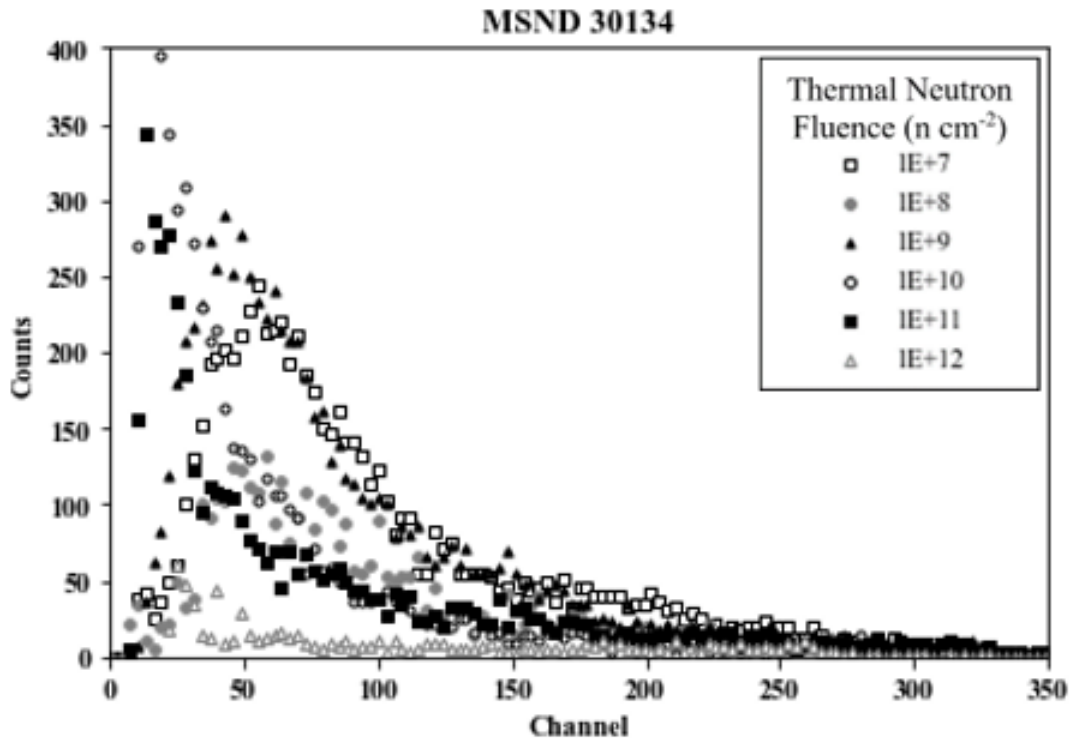


Figure 6-20 The reaction product spectrum from the Triga Mk II NW thermal neutron beam port following each decade of thermal neutron exposure is shown for device 30134.

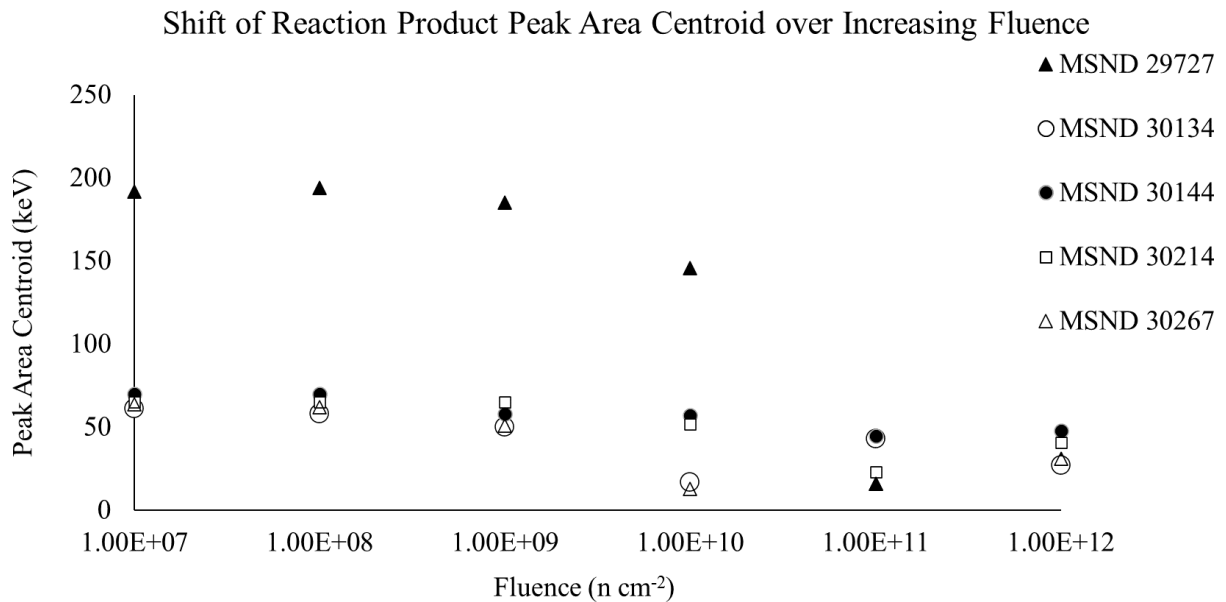


Figure 6-21 The shift of the reaction product spectrum over increasing fluence for each device is shown.

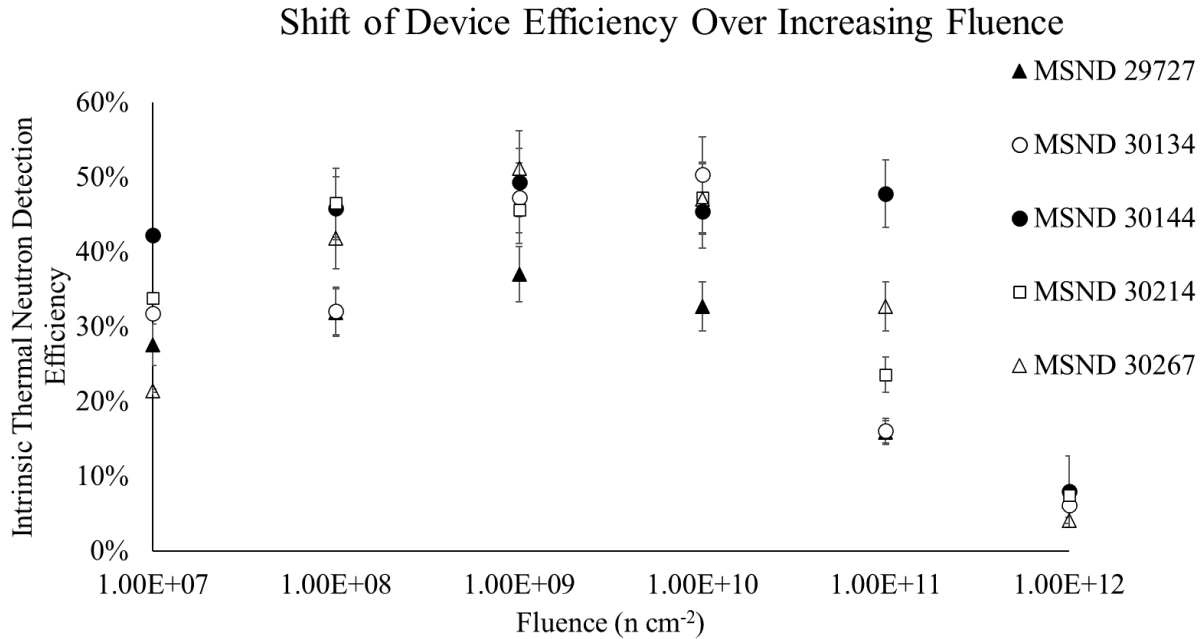


Figure 6-22 The change in the area of the reaction product spectrum is shown for each device.

Table 6-9 The spectral shift for each device over 10^7 to 10^{12} thermal neutron fluence exposures are represented by the peak area centroid movement.

Device 29727	Initial	1.00E+07	1.00E+08	1.00E+09	1.00E+10	1.00E+11	1.00E+12
Area	17589	15454	16118	19331	12678	7826	863
Area Error	132	124	127	139	125	116	51
Area Centroid ROI Net	187	192	194	185	146	16	30
Counts	12103	12340	14244	16557	14618	7085	2968
Efficiency	35%	28%	32%	37%	33%	16%	6.65%
Efficiency Error	2.70%	2.75%	3.18%	3.69%	3.26%	1.59%	0.67%
Device 30134	Initial	1.00E+07	1.00E+08	1.00E+09	1.00E+10	1.00E+11	1.00E+12
Area	7327	11646	2244	10118	14920	1253	301
Area Error	124	151	72	160	156	62	27
Area Centroid ROI Net	61	61	58	50	17	43	27
Counts	14216	14353	21100.5	22509	7193	2732	1782
Efficiency	32%	32%	47%	50%	16%	6%	3.99%

Efficiency Error	3.17%	3.20%	4.70%	5.01%	1.61%	0.62%	0.41%
Device 30144	Initial	1.00E+07	1.00E+08	1.00E+09	1.00E+10	1.00E+11	1.00E+12
Area	4672	10066	10942	11960	8531	10399	2561
Area Error	102	143	148	160	149	161	178
Area Centroid	71	70	70	58	57	45	48
ROI Net							
Counts	9635	18864	20475	22021	20291	21334	3553
Efficiency	22%	42%	46%	49%	45%	48%	7.96%
Efficiency Error	2.15%	4.20%	4.56%	4.90%	4.52%	4.75%	0.80%
Device 30214	Initial	1.00E+07	1.00E+08	1.00E+09	1.00E+10	1.00E+11	1.00E+12
Area	7908	10815	8998	10396	9514	3280	154
Area Error	108	149	151	151	154	76	22
Area Centroid	65	65	65	65	52	23	41
ROI Net							
Counts	11617	20797	20399	21118	10546	3359	1974
Efficiency	34%	47%	46%	47%	24%	8%	4.42%
Efficiency Error	2.59%	4.63%	4.54%	4.70%	2.35%	0.76%	0.45%
Device 30267	0	1.00E+07	1.00E+08	1.00E+09	1.00E+10	1.00E+11	1.00E+12
Area	4303	9885	12096	11626	21417	2351	228
Area Error	87	141	156	148	170	60	23
Area Centroid	63	64	62	51	13	16	31
ROI Net							
Counts	7361	18713	22846	21033	14625	1845	1560
Efficiency	21%	42%	51%	47%	33%	4%	3.49%
Efficiency Error	1.65%	4.17%	5.09%	4.69%	3.26%	0.42%	0.36%
Average Pre-Irradiation Efficiency			24.09%	Average Post-Irradiation Efficiency			4.42%
Average Pre-Irradiation Efficiency Error			0.35%	Average Post-Irradiation Efficiency Error			0.45%

The reverse leakage current for the devices also changed as a function of neutron fluence, increasing as the fluence was increased (see Table 6-10). From Table 6-10, device 30267 initially had the highest leakage current, indicating inferior performance from surface leakage. It also had lower ϵ_{in} for the initial pre-irradiation measurements.

Table 6-10 The leakage current measurement before the commencement of the experiments at -3V in reverse bias for each device is compared to the leakage current under the same measurement parameters after the irradiation exposures.

Device	Bias (V)	Leakage	
		Before (Amps)	After (Amps)
30267	-3	-1.03E-08	-4.06E-07
30134	-3	-8.6E-09	-3.96E-07
29727	-3	-5.0E-09	-3.66E-07
30144	-3	-9.3E-09	-7.75E-08
30214	-3	-7.9E-09	-4.07E-07

A comparison IV curve is shown in Figure 6-23 for Device 29727, which had the lowest initial leakage current and the second highest thermal neutron detection efficiency. Device 29727 concluded the testing with one of the best post irradiation leakage currents while maintaining one of the highest thermal neutron intrinsic efficiencies. It is however worth mentioning that the device, while operable, did have a noisier signal, as indicated by the shift in the resulting thermal neutron spectral results.

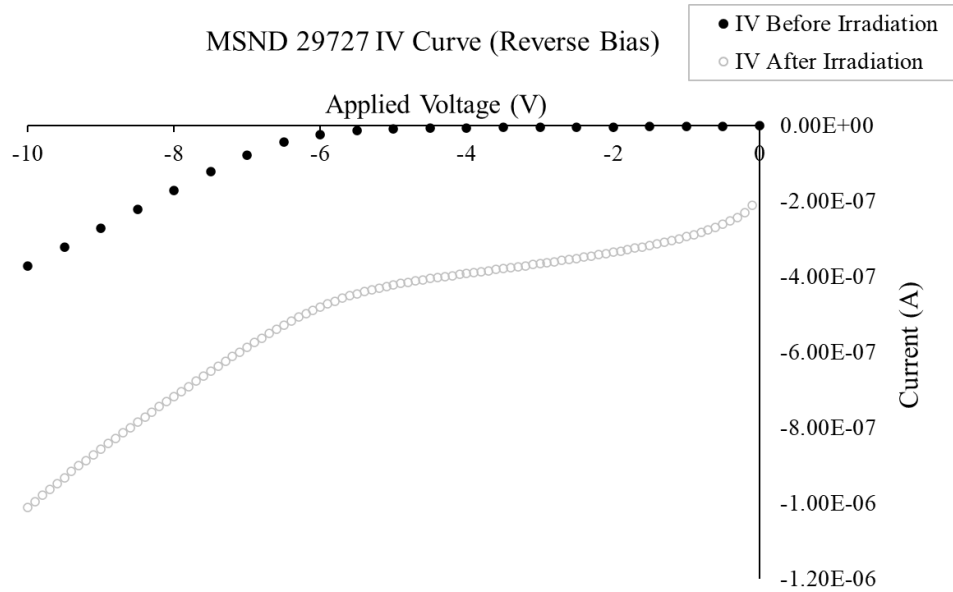


Figure 6-23 The IV comparing the before and after irradiation leakage current is compared for the device with the lowest leakage current (77nA) post exposure.

Furthermore, device 30144 which began as the second highest leakage current device, was able to maintain an operable leakage current of only 75 nA at -3V. When examining the spectral shift of device 30144 in Figure 6-19, the device's spectral performance did not deteriorate at the same rate as the other four devices, showing a bit more resilience. MSND 30144 ranked in the bottom two of the devices for efficiency. The apparent radiation hardness improvement is most likely due to a lower ${}^6\text{LiF}$ neutron conversion material packing density. Although the lower packing density resulted in lower relative ϵ_{tn} , the decrease in the number reaction products intersecting the Si diode reduced the total lattice damage.

Initially device 30214 was one of the the highest ϵ_{tn} ; however, by the conclusion of the experiment, the final efficiency was determined to be one of the worst of the group. The degradation of the performance further indicates that the high ${}^6\text{LiF}$ packing fraction that

increases ϵ_{tn} works to also reduce radiation hardness. As hypothesized, device 30214 likely experienced a greater number of reaction products traversing and damaging the silicon material. Additionally, as previously noted, IRIS port three, which housed device 30214 for the duration of the experiment was breached by water. Unfortunately, the total fast and thermal neutron fluence intersecting device 30214 was unknown, and there was quite possibly a slightly enhanced thermal neutron flux as fast neutrons lost energy in the IRIS tube 3. However, when examining the spectrum shift of device 30214 throughout the various fluence treatments, the shift follows a similar pattern as observed with the other devices.

In summary, the device degradation was apparent at fluences above 10^{10} cm^{-2} , while catastrophic damage occurred at fluences exceeding 10^{12} cm^{-2} . The average intrinsic thermal neutron detection efficiency of the five devices decreased from $24.09 \pm 0.35\%$ to $4.41 \pm 0.1\%$. MSNDs with the lowest initial ϵ_{tn} yielded the highest radiation hardness, a consequence of lower ^6LiF powder packing density and, therefore, lower neutron reaction product damage. The eventual higher damage of the MSNDs with the highest initial ϵ_{tn} indicates higher vulnerability to neutron damage than the lower efficiency counterparts. All MSNDs began the experiment with a leakage current less than 11 nA (at -3V reverse bias); however, at neutron fluences exceeding 10^{12} cm^{-2} , the average leakage current of all five devices exceeded 300 nA. Although performance was compromised, they did continue to operate beyond $10^{12} \text{ n cm}^{-2}$. Because the MSND devices did not show significant damage up to $10^{10} \text{ n cm}^{-2}$, they remain possible candidates for many applications where the expected thermal neutron exposure is lower than $10^{10} \text{ n cm}^{-2}$. The lifetime response for DS-MSND devices is anticipated to closely follow that of the MSND.

Further investigation however should be performed to understand the impact of the activation products resulting from the activation of the devices shield. While all activation products were gamma emitters (^{196}Au , $^{197\text{m}}\text{Au}$, ^{198}Au , ^{199}Au , ^{55}Fe , ^{54}Mn , ^{56}Mn , ^{59}Fe), the possible dark current contribution of those activation products should be further investigated. Furthermore, more work should also be performed to understand the impacts of time on the silicon to heal voids, interstitials, and anti-sites produced throughout the experiment. It is possible that leakage current at the time of the final efficiency evaluation was more severe than reported nearly two years later. Future work to evaluate the ability for MSND to anneal reaction product damage over time should would provide valuable insight to the operational lifespan of MSND devices.

7 Performance and Analysis of ARGANOT



A single ARGANOT module is shown operating in handheld mode.

“Priority efforts include [...] preventing illicit trafficking by closing key gaps in the radiation detection architecture through the provision of fixed and mobile detection equipment.”[154]

~ Bruce Held, National Nuclear Security Administration Administrator

The ARGANOT system, as shown in Figure 7-1, underwent extensive testing, including neutron response as a function of distance, angle, and movement as well as the gamma-ray response in terms of alerting, spectroscopy, and directionality.



Figure 7-1 The android phone and two modules are shown. The ARGANOT modules are available with and without labels.

7.1 Neutron Response

A single ARGANOT module was put in a vest and placed around an ANSI standard phantom. The acrylic phantom simulates the neutron moderating effects of a human body and the

center of mass if 1 m above the floor. An hour-long background was collected and may be viewed in Figure 7-2. Following the background, a 21.9-ng ^{252}Cf neutron source was placed inside a 9.7-cm diameter by 12.8-cm tall moderator cask composed of 0.5-cm thick lead, 1-cm thick steel, and 2.5-cm thick HDPE concentric cylinders as reflected in Figure 7-3.

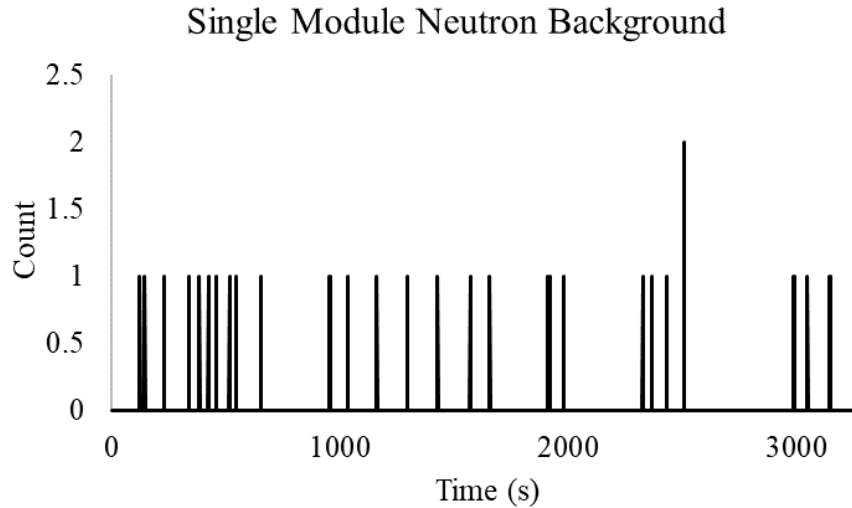


Figure 7-2 The single module neutron background response over an hour-long measurement is shown.

The moderated cast was placed 1 m off the ground and 0.5 m away from the face of the ARGANOT module face. All data was recorded, and the source was moved to 1.0, 1.5, and 2.0 meters. The moderator cast was removed, and all measures were repeated for a bare ^{252}Cf source. To characterize the effects of the phantom body moderation, the vest was removed from the phantom body. The vest populated with a single module was then suspended from the ceiling with a thin metal wire and a wire hanger with the center of the vest approximately 1 m above the ground. The bare and moderated ^{252}Cf response for the free-standing vest were then recorded at distances of 0.5, 1.0, 1.5, and 2.0 meters. The vest was then populated with eight modules and the response was recorded as before for the bare and moderated ^{252}Cf source in the free-standing position and when worn by the phantom. A background was collected for each measurement arrangement.



Figure 7-3 The Ansi Phantom used throughout testing along with the moderator is shown.

Following the completion of all ^{252}Cf measurements, the source was removed and an approximately 6.92 nCi moderated and bare AmBe source was then used to characterize the single module and four module populated ARGANOT system's response to a fast neutron spectrum on a phantom for distance of 0.5, 1.0, 1.5, and 2.0 meters. The count rate over distance for each measurement scenario was then plotted over distance for the single and ARGANOT system and compared.

7.1.1 Distance Response

The count rate over an hour long measurement period when the ARGANOT system was placed on the phantom when exposed to a moderated ^{252}Cf and moderated AmBe source at 0.5 meters may be viewed in Figure 7-4 and Figure 7-5.

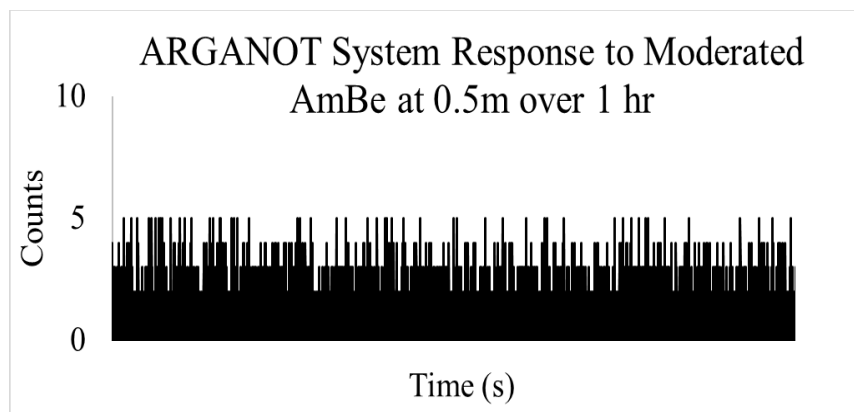


Figure 7-4 Four detectors were placed into the ARGANOT garment and placed on the phantom. The observed neutron count rate over the hour-long measurement when exposed to the moderated AmBe source is shown.

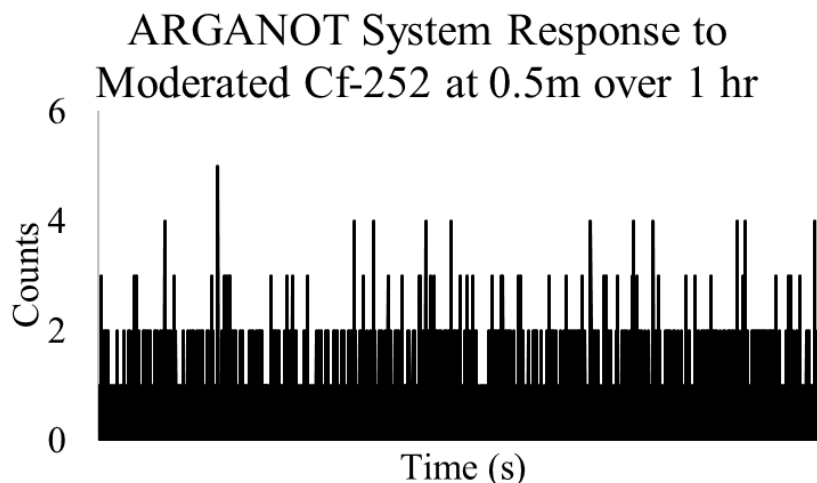


Figure 7-5 Eight detectors were placed into the ARGANOT garment and placed on the phantom. The observed neutron count rate over the hour-long measurement when exposed to the moderated ^{252}Cf source is shown.

The response as a function of distance for the single modules while on the phantom and off for both the bare and moderated ^{252}Cf source may be viewed in Figure 7-6. As expected, the count rate was significantly higher for the moderated ^{252}Cf source while on the phantom due to the moderation of the neutrons to thermal energies, which increased the probability of absorption within the neutron conversion material. When the response of the detector on the phantom was compared to the free standing vest suspended from the ceiling (see Figure 7-7), results indicated that the phantom significantly aids in moderating and scattering additional neutrons into the detector, increasing the ability to detect a source over a distance. As expected, as the distance

away from the source increased, the response decreased. The decrease may be attributed to the decrease in the solid angle as well as air attenuation. However, at 1.5 meters, the neutron response slightly increased, possibly due to the air attenuation and moderation of the epithermal and fast neutrons to thermal energies that are easily detectable by the MSND and DS-MSND devices. At 2 meters, the response again decreases, likely due to a large percentage of the neutrons being fully attenuated by the air by that point.

The intrinsic thermal neutron detection efficiency was then determined at over distance for each measurement scenario. The solid angle was determined at each point for a single ARGANOT module (11cm x 5cm x 1.5 cm) 1m off the floor as reflected in Figure 7-8 where the black point defines the location of the source, and the red rectangle represents the outline of the single DS-MSND populated ARGANOT module. The source was consistently treated as a point source and placed 0.8 m above the ground in line with the detector. Results of the intrinsic efficiency over distance are reported to be as high as $7.12\% \pm 0.58\%$ for a moderated ^{252}Cf when the module was on a user body at 1.5 m, which demonstrate an increasingly decent response as distance between the source and the face of the detector increases. As shown in Figure 7-9, an increase in intrinsic neutron detection efficiency as distance increases was observed, which was likely due to parallax effects and a decreased average neutron energy over increasing distance.

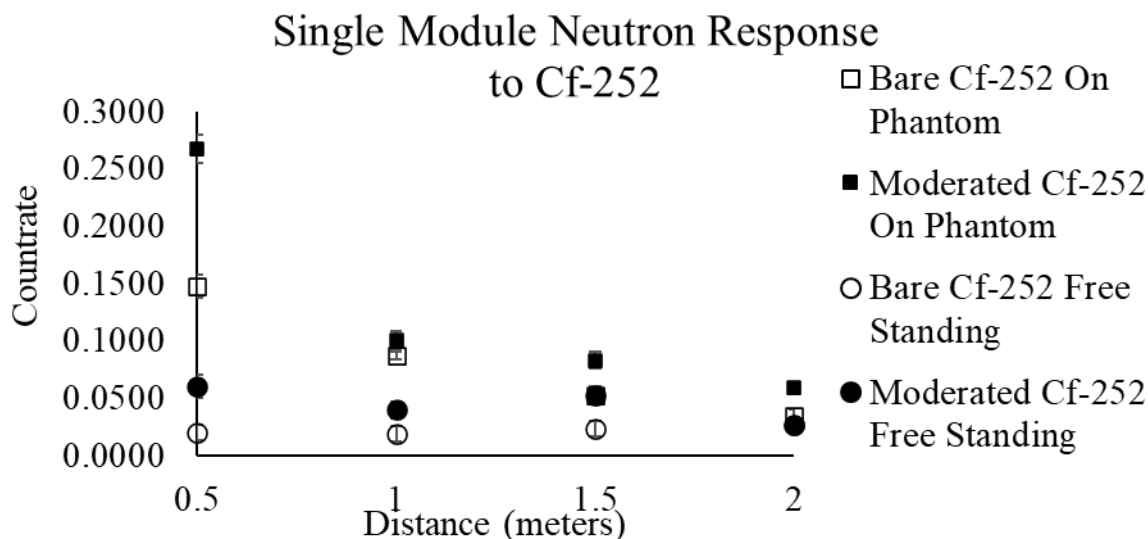


Figure 7-6 The Neutron response of a single module in response to a bare and moderated ^{252}Cf source both on and off the phantom is compared. All associated errors are included.

However, as reflected in Figure 7-6, once the source was 2.0 from the detector, the number of neutrons reaching the detector decreased and therefore the intrinsic efficiency at 2.0 also decreased.



Figure 7-7 The free standing ARGANOT system is pictured. The vest was populated with up to eight modules and exposed to a bare and moderated ^{252}Cf source.

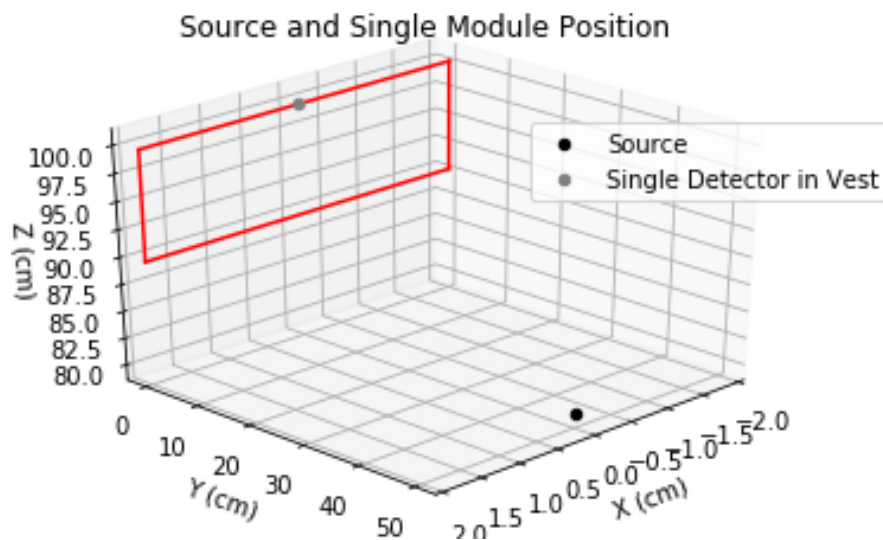


Figure 7-8 The visualization of the solid angle of the detector for a source at 0.5m is shown. As the distance increased, the solid angle of the detector decreased.

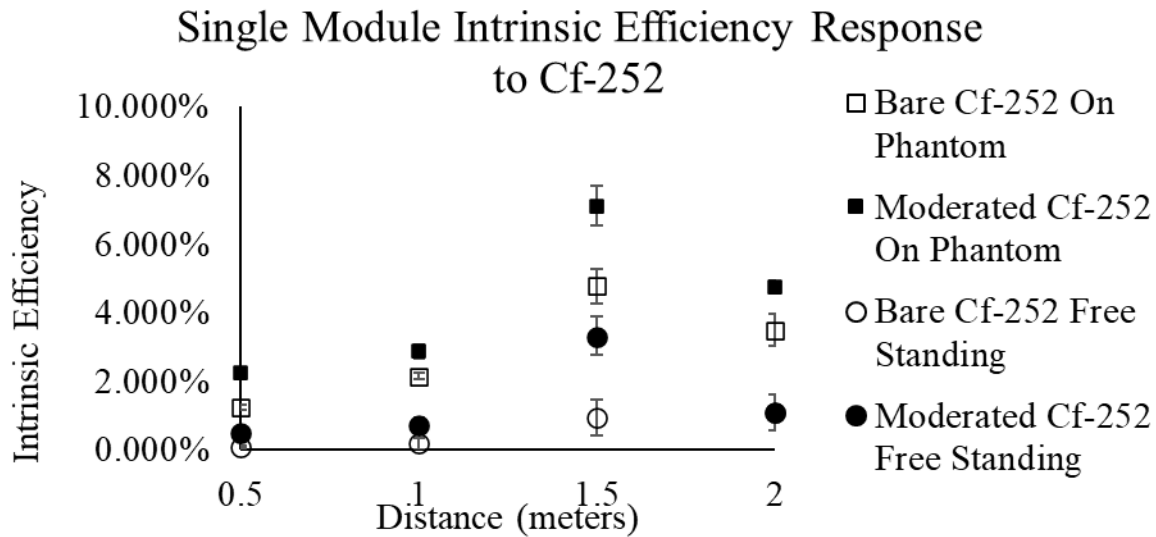


Figure 7-9 The intrinsic thermal neutron response of a single module in response to a bare and moderated ^{252}Cf source both on and off the phantom is compared. All associated errors are included.

When the ^{252}Cf response for the eight-module populated ARGANOT system was examined, a similar trend was observed for both the count rate over distance and the intrinsic efficiency over distance. The solid angle was determined with the entire chest of the system representing the area of interest and the source consistently represented as a point source. For the ARGANOT system count rate response over distance, as shown in Figure 7-11, minimal increase in count rate between 1.5 and 2.0 meters was observed as was for the single module. The steady decrease in count rate with increasing distance is likely due to the fact that modules were equally distributed on the phantom's front and back side, contributing to equivalent moderation of neutrons at both close and far distances.

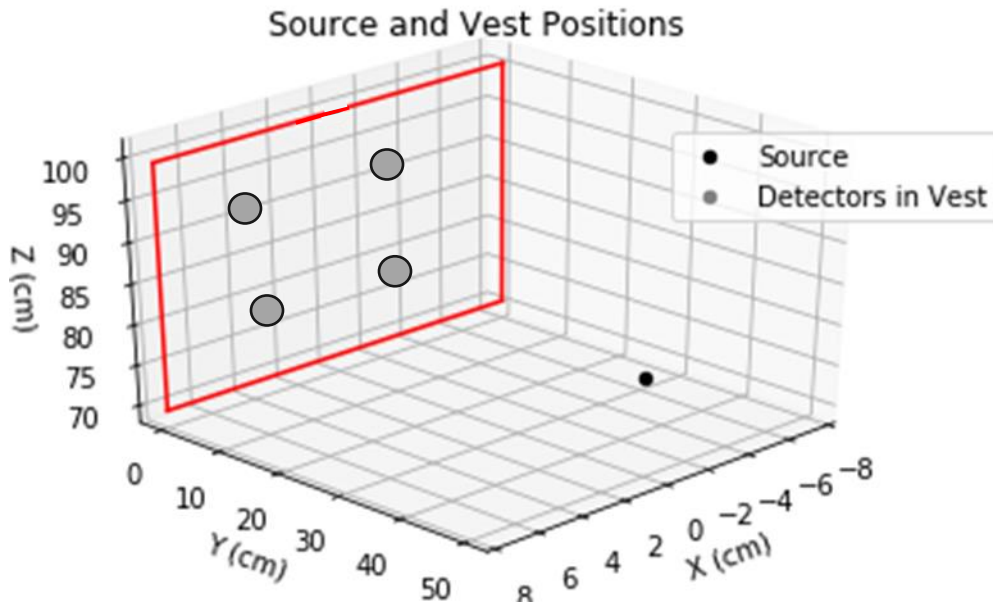


Figure 7-10 The solid angle of the ARGANOT system is visualized for a ^{252}Cf source at various distances.

The intrinsic efficiency of the system was then determined with the moderated ^{252}Cf when placed on the phantom as expected was reported in Figure 7-12 to remain between $1.093\% \pm 0.010\%$ and $1.578\% \pm 0.105\%$ through 0.5 m to 2.0 m as shown in Figure 7-12. The bare ^{252}Cf intrinsic efficiency of the ARGANOT system deviated minimally with an average intrinsic efficiency of $1.165\% \pm 0.052\%$ when worn on a body, indicating the system is just as effective in alerting to unmoderated SNM as it is to shielded or moderated SNM material.

When the system was free standing, the average efficiency of the moderated ^{252}Cf source was $0.575\% \pm 0.066\%$ while the average intrinsic efficiency for a bare ^{252}Cf source was $0.204\% \pm 0.052\%$, once again indicating that the ARGANOT modules are more sensitive when worn on a user body.

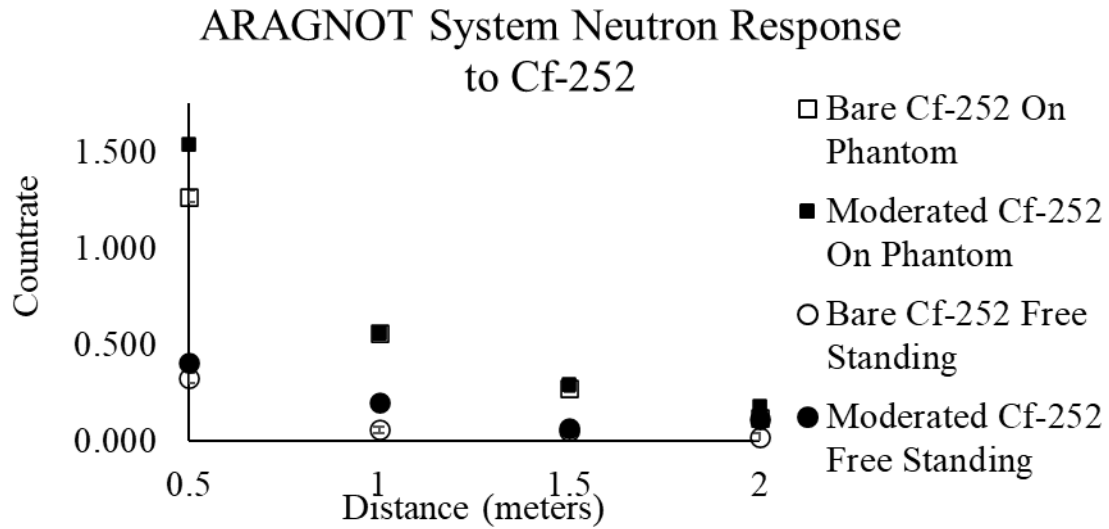


Figure 7-11 The AGRANOT system response to a ^{252}Cf in various detection scenarios are compared over increasing distance. All errors are included.

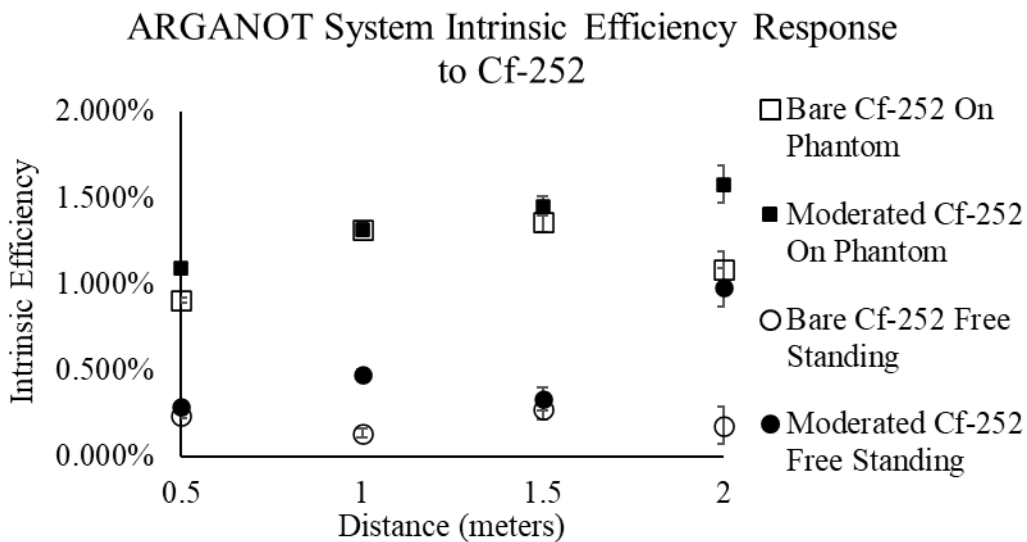


Figure 7-12 The AGRANOT intrinsic thermal neutron efficiency is reported for various detection scenarios over distance. All errors are included.

Table 7-1 The intrinsic efficiency and associated error for the various detection scenarios both on and off the phantoms is shown.

SYSTEM Phantom	Intrinsic Eff.	Intrinsic Eff. Error
Moderate Distance		
0.5	1.093%	0.010%
1	1.317%	0.036%
1.5	1.449%	0.056%
2	1.578%	0.105%
Average	1.359%	0.052%
Bare Distance		
0.5	0.903%	0.013%
1	1.314%	0.032%
1.5	1.359%	0.062%
2	1.086%	0.102%
Average	1.165%	0.052%
SYSTEM Free Standing	Intrinsic Eff.	Intrinsic Eff. Error
Moderate distance		
0.5	0.288%	0.009%
1	0.475%	0.031%
1.5	0.330%	0.065%
2	0.978%	0.113%
Average	0.575%	0.066%
Bare distance		
0.5	0.232%	0.010%
1	0.133%	0.028%
1.5	0.272%	0.062%
2	0.177%	0.107%
Average	0.204%	0.052%

When the same measurements with a single and four module populated ARGANOT system were then performed with a moderated and bare AmBe source on the phantom, results shown in Figure 7-13, again indicated that the moderated neutron source produced higher count rates than the bare source. The response to the moderated and bare sources drastically varied at a distance of 1.0 meter, with the moderated response reportedly three times higher than the bare response. The minimal bare response likely indicated that the energy of neutrons was not moderated to

thermal levels. It appears that the moderated response at 1 m was further reduced to thermal energies, increasing the likelihood of detection. At 2.0 m, the bare response appears to increase, perhaps due to increased air moderation of fast neutrons. Future work should examine the AmBe response at greater distances. As shown in Figure 7-14, the intrinsic efficiency of both the moderated and bare AmBe source at 1.0 meters increases. The bare efficiency at 2.0 meters increases, indicating the air moderation of the fast neutrons was significant at 2.0 meters.

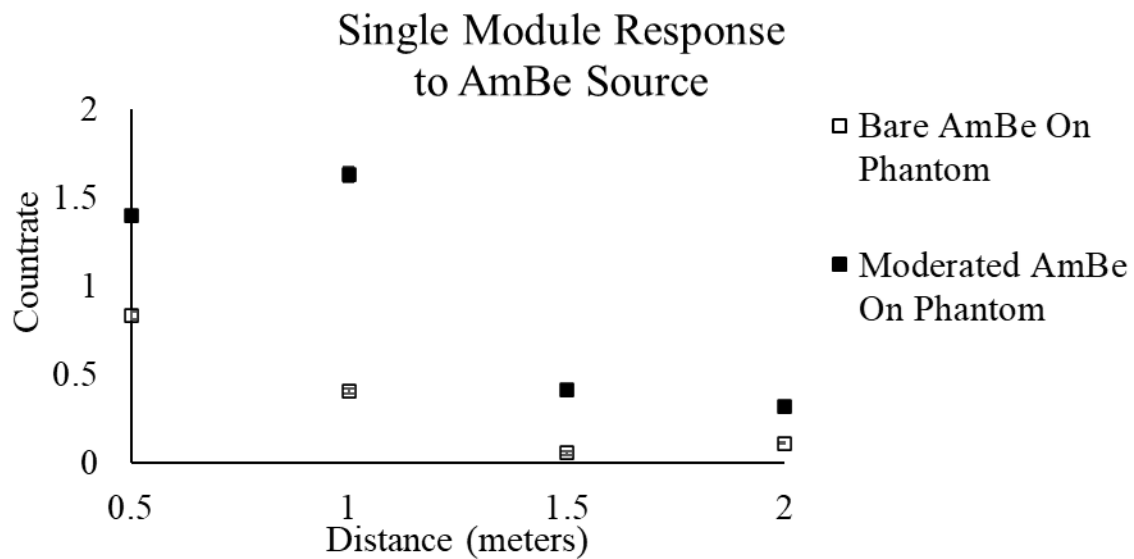


Figure 7-13 The single module response to an AmBe source is shown over a distance of 0.5 to 2.0 meters. All errors are included.

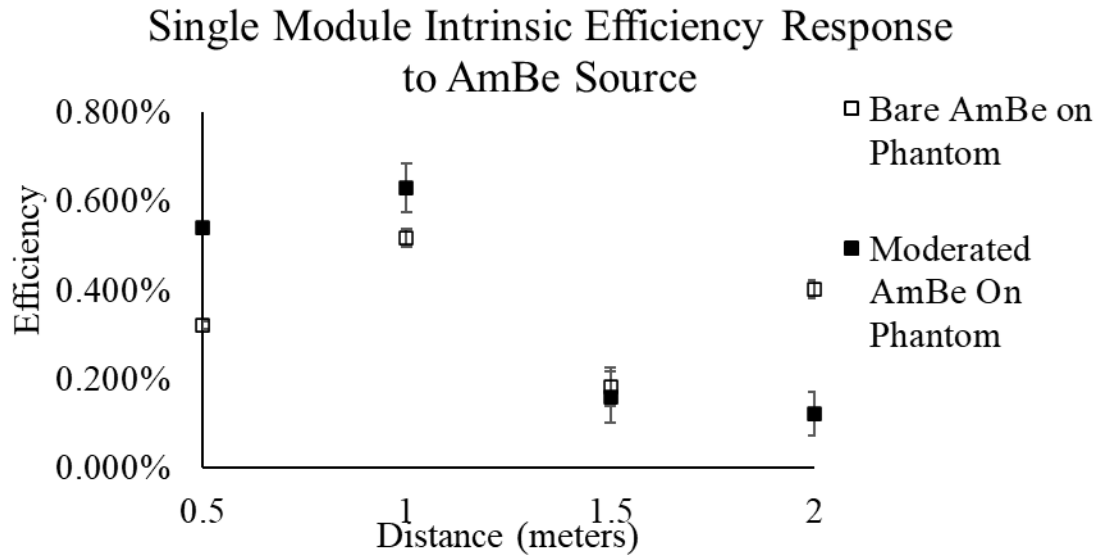


Figure 7-14 The intrinsic efficiency of a single ARGANOT module on the phantom is shown. All errors are included.

All results were in general agreement with previous work performed by Ochs et al. in which the intrinsic efficiency of a MSND populated was presented [155]. When comparing the response of the DS-MSND populated ARGANOT modules to the response of the MSND populated MND modules, the bare efficiency on the systems when worn by the phantom were quite similar. Additionally, the ARGANOT system's modules were only comprised of nine DS-MSND devices and eight modules across the vest, while the 16 MND modules, each contained 24 MSND devices. The ARGANOT system offers increased sensitivity, with half the number of modules.

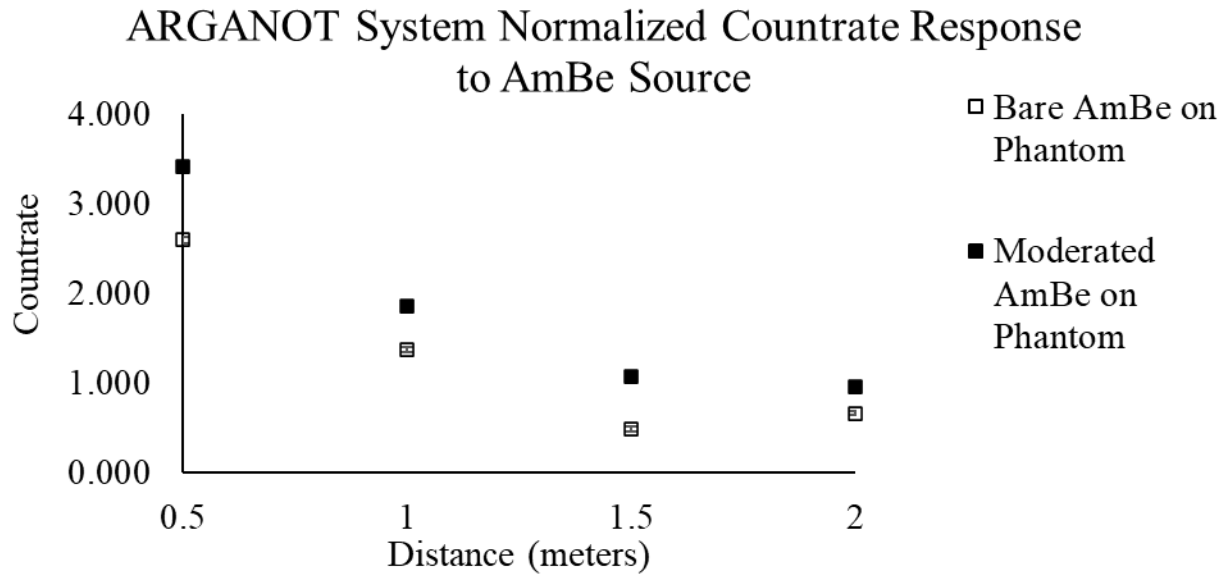


Figure 7-15 The four populated module response to an AmBe source is shown over a distance of 0.5 to 2.0 meters. All errors are included.

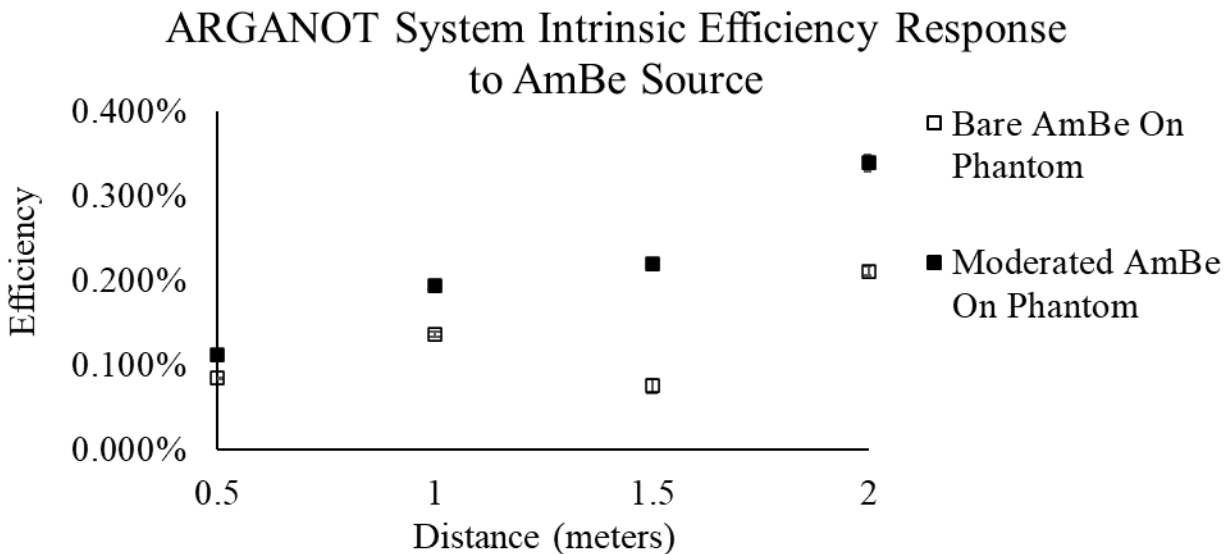


Figure 7-16 The four populated module response to an AmBe source is shown over a distance of 0.5 to 2.0 meters. All errors are included.

7.1.2 Angular Response

A single ARGANOT module was then placed at center mass on a phantom and a bare ^{252}Cf source was then placed 1 meter from the face of the detector. A measurement was then conducted for approximately 30 minutes. The center of mass of the phantom was then rotated

counterclockwise in 45-degree increment. A 30-minute measurement was collected at each angle. At the conclusion of the bare measurements, the source was then moderated, and the phantom was again rotated around the center of mass for 30-minute measurements for each angle increment. The full ARGANOT module was then populated with eight ARGANOT modules and 30-minute measurements were then collected at each angle for both a moderated and bare ^{252}Cf source.

The results were then plotted, and count rates were normalized to the count rate reported in position 0 degree where the source was placed directly in line with detector. The results of the individual module over the five increments of angles are displayed in Figure 7-17. The bare response at 45, 135, and 180 degrees better retain their response to the source than the moderated counterpart. When compared to the fully populated ARGANOT system displayed in Figure 7-18, the full system demonstrated a more equivalent response across all angles than with the single modules measurements due to the even distribution of detectors across the wearer's body. Within the ARGANOT results, the moderated response retained sensitivity better than that of the bare source but was inferior to the response of the detector to the bare source for angles of 135. Interestingly, the superior response of the system when placed at 135 and 180 degrees may be explained by the differing thermal neutron detection efficiencies of each module. The more efficient detector(s) may have been placed closest to the source, which contributed to the superior performance for phantom positions of 135 and 180. Therefore, if directionality for neutron emitting material was implemented, the directionality function would have to implement a correction factor to normalize the efficiency of each module to ensure that the detector(s) with the highest count rates represent the detectors closest to the source. Additionally, as reflected in Table 7-2 the DS-MSND response rate is superior to the MSND. Therefore, the modules populated with DS-MSND verse MSND should be appropriately weighted to ensure the directionality function is accurate.

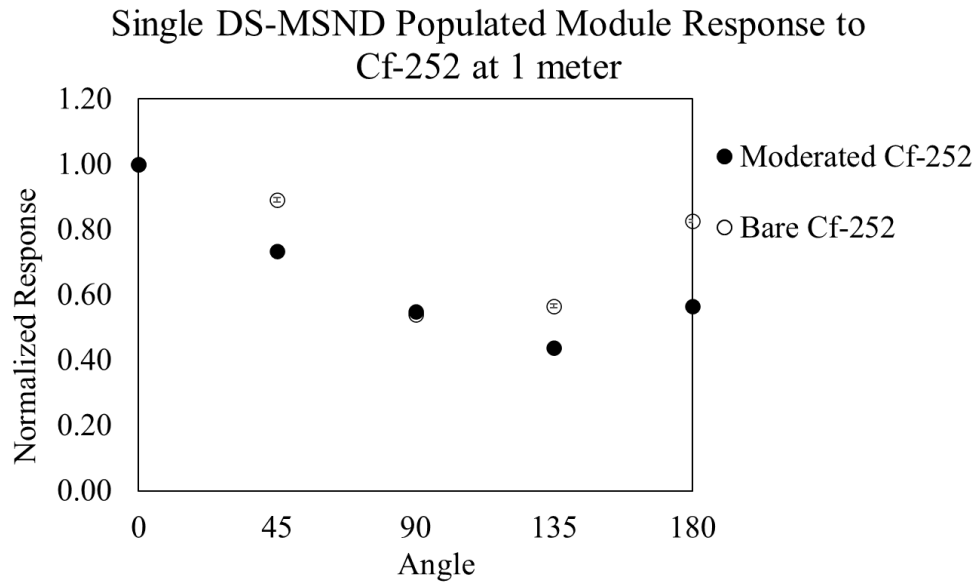


Figure 7-17 The angular response of a single module over 45-degree increments for a bare and moderated Cf-252 is shown.

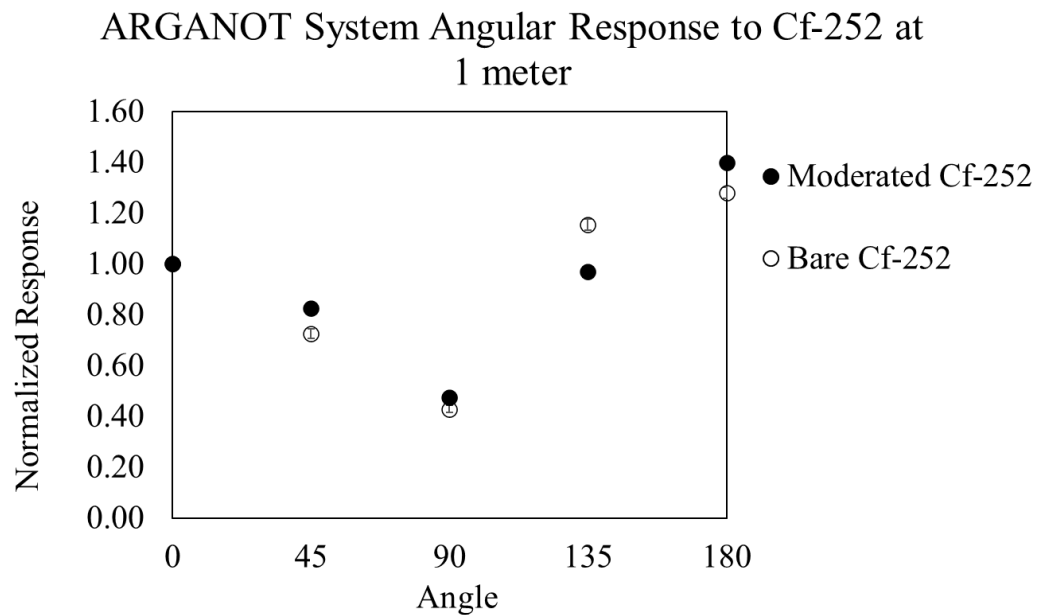


Figure 7-18 The angular response of a fully populated module over 45-degree increments for a bare and moderated ^{252}Cf is shown.

Table 7-2 A comparison of the MSND and DS-MSND response rate is shown or a moderated ^{252}Cf source.

Detector	Source	Duration (sec)	Neutron Counts	Net Count Rate (cps)
MSND	Cf-252	6,050	12,270	2.03
DSMSND	Cf-252	3,706	11,445	3.09
MSND	Background	55,850	1,080	0.02
DSMSND	Background	38,410	5,403	0.14

Throughout the ^{252}Cf testing, the $\text{SrI}_2(\text{Eu})$ gamma-ray detector on board recorded the gamma-ray spectrum emitted by the ^{252}Cf source. Because SNM can emit both gamma-ray and neutrons, the response of the DS-MSNDs to neutrons in the presence of a gamma-ray source was characterized.

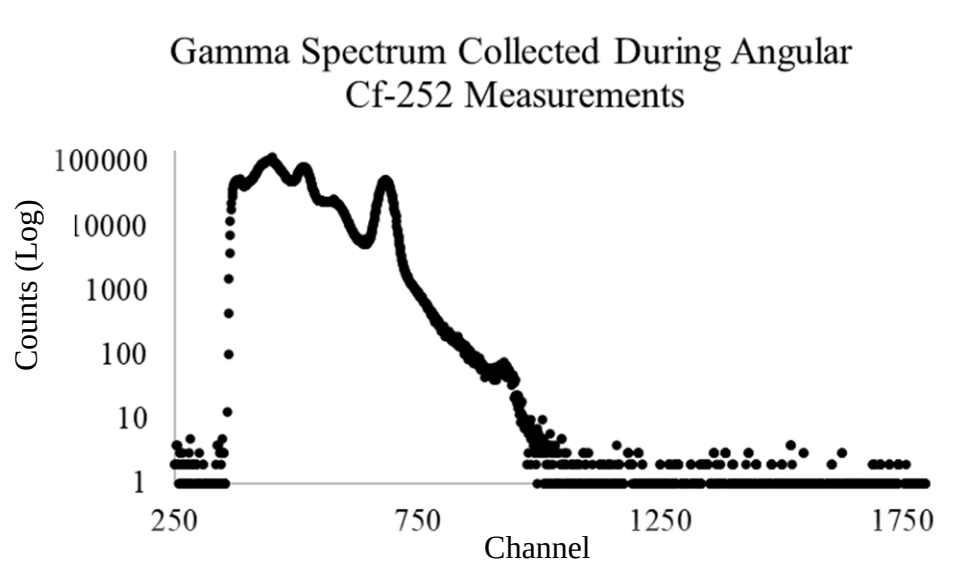


Figure 7-19 The spectrum collected throughout one of the angular measurements was collected.

7.1.3 Gamma-Ray Rejection

The gamma-ray rejection performance of ARGANOT was tested by exposing the modules to a high gamma-ray environment. The elevated gamma-ray environment did not stimulate false neutron counts in the absence of a neutron source and did not cause an increase in

neutron counts during exposure to a neutron source. The phone application shown in Figure 7-20 is shown alerting to an elevated gamma-ray count rate when exposed to a 0.8mCi ^{137}Cs source. The neutron count rate however remains steady, seemingly unaffected by the elevated gamma-rays.

To determine the Gamma-ray Rejection Ratio (GRR) of two modules DS-MSND populated ARGANOT module, the modules were placed in the user garment and placed on the center of an ANSI standard phantom. The modules were then connected to the android device and a 0.8mCi ^{137}Cs gamma-ray source was placed 18cm away from the face of the detector at equal heights off the ground, placing the detector face in a dose of 10mR/hr. The neutron data was collected for 90 minutes. The gamma-ray source was then removed, and a background measurement was collected. The intrinsic efficiency of the neutron detector when exposed to the gamma-ray source was determined, approximating the gamma-ray source as a point source. Detector modules were determined to have a poor intrinsic efficiency response to the gamma-ray source, indicating an average GRR of approximately 5.69E-05 at a dose rate of 10mR/hr. Table 7-3 shows the GRR values for both individual modules.

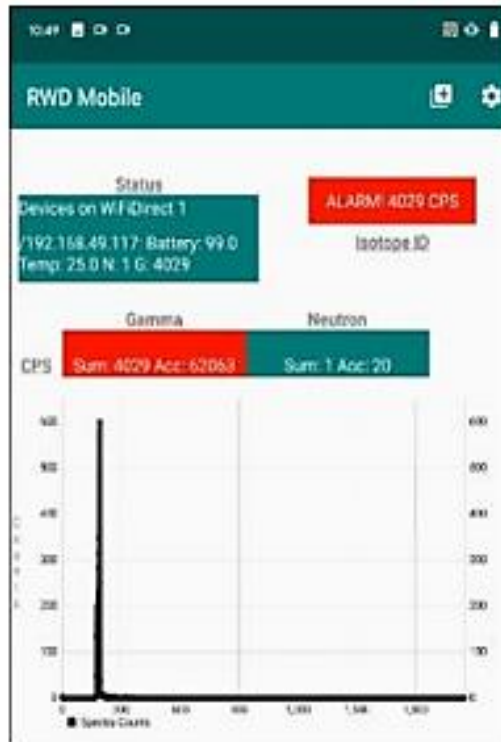


Figure 7-20 The Gamma-ray alarm in red is alerting and the gamma-ray spectrum begun to form. Notice the neutron count rate remains steady through the high gamma-ray exposure.

Table 7-3 The GRR for two modules at a dose rate of 10mR/hr is shown.

Module	GRR
3A	2.71E-05
3E	8.68E-05

Three modules were then placed on the chest of the phantom within the vest. The modules were again connected to the android device, and the ^{137}Cs source was again placed at a dose rate of 10 mR/hr. A moderated ^{252}Cf neutron source was then placed 2 m away. A 35-minute measurement was then performed. The gamma-ray source was then removed and a second 35-minute measurement with the neutron source only was then conducted. Then absolute efficiency of the ARGANOT neutron modules were then determined. As defined by Pacific Northwest Laboratory's Neutron (PNNL) Detector Gamma-Ray Insensitivity Criteria, the Gamma-ray Absolute Rejection Ratio in the presence of neutrons (GARRN) value of the three-module system was determined to be 1.022 ± 0.096 , well within the acceptable gamma-ray rejection ratio defined by PNNL [156].

7.2 Gamma-Ray Response

To allow the user to perform isotope identification, activity estimations, and dose estimations, GADRAS software was implemented for the ARGANOT system. To do so, a response function had to be built for the ARGANOT module gamma-ray detector within the GADRAS software. A response function was created for a single module both on a phantom and free standing. The room chosen for the calibration measurements location was selected based on uniform and low background as well as a consistent temperature. The room was cleared to ensure only necessary equipment remained in the room. Any remaining equipment was relocated to the walls to minimize any unwanted scattering from occurring. All radiological sources were removed from the room.

Emphasis was placed on minimizing scattering of photons. Therefore, low density and Z materials were selected in the testing configuration. For example, the assembly is hoisted 100 cm in the air by a simple cardboard box to further minimize any scattering events. Extreme caution was used to maintain a consistent measurement environment throughout testing. To generate training set to construct the response function with, the module was placed on the center of the phantom's chest, as shown in Figure 7-21. At the start of the calibration measurements, the initial background measurement was collected for ten minutes. Following the background measurements, a gamma-ray source from Table 7-4 was brought into the calibration room and placed 8cm from the face of the detector. A ten-minute foreground was then collected. At the conclusion of the measurement, the source was retired, and the next source was brought into the room and placed at the exact location as the previous. A ten-minute measurement was performed for all source in Table 7-4. At the conclusion of all foreground measurements, a secondary background was then collected.

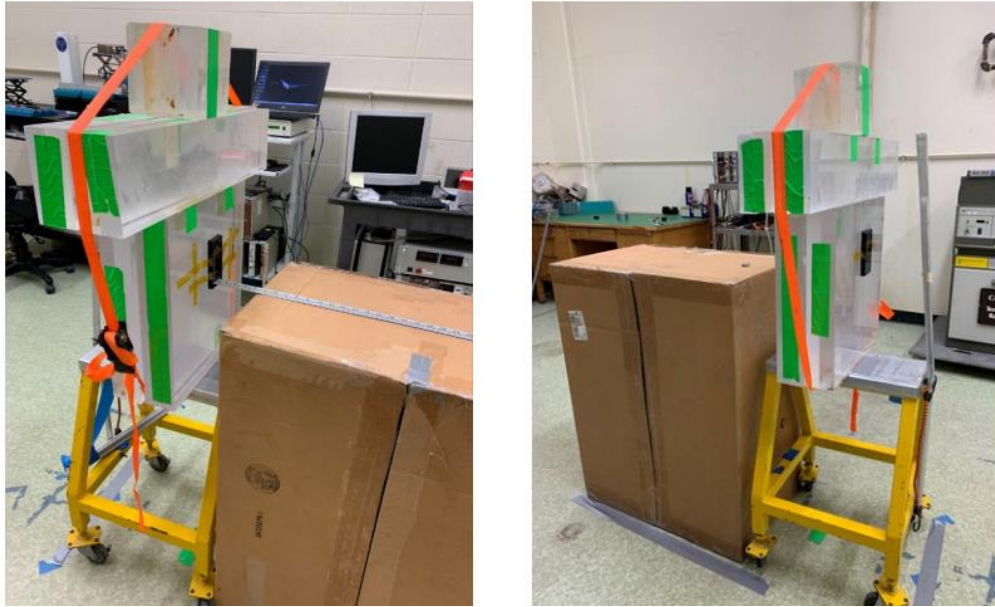


Figure 7-21 The ARGANOT module was tested on the phantom with the various sources. The measurements were being used to construct the response functions for the GADRAS identification and dose estimations. The module was on the phantom front (left) when the source was 20cm in front of while the right image shows the module on the phantom's back for the same testing scenario.

Following the measurements with the module on the phantom, a set of measurements using the same sources in Table 7-4 were then performed for a freestanding module as displayed in Figure 7-22. To hold both the module and source in place throughout all measurements, a low-density 3D printed holder was constructed with an aluminum rod connecting the two holders. The detector was suspended 100 cm above the ground while the source sat 20 cm away on the edge of the box. The yard stick pictured was removed during each measurement.

Following the collection of all measurements, the resulting spectrums were inputted into the GADRAS software as shown in Figure 7-23. The sources within the software were then created to simulate the source use in the empirical measurements. The factors within the response function were then altered as shown in Figure 7-24.

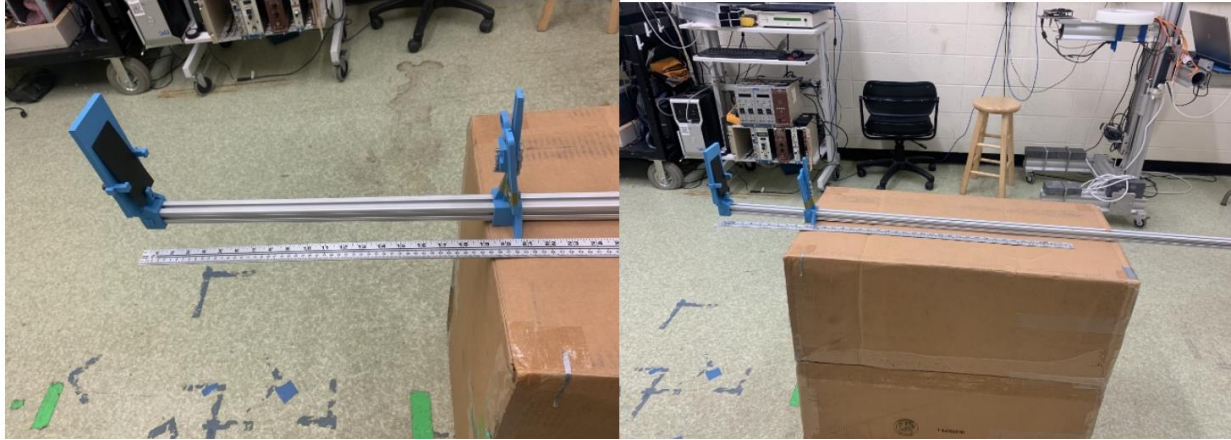


Figure 7-22 The testing assembly for the GADRAS calibration measurements is shown. Care was used throughout the measurements to ensure a consistent uniform testing environment. The yard stick shown in the images above was removed during the measurement to reduce scattering.

Table 7-4 The sources used throughout the calibration measurements are shown below.

Isotope	Half-life days	Lambda	Original activity μCi	Assay date	Date of measurement	Half-lives past	New activity μCi
Na-22	949.73	7.30E-04	9.902	3/1/2020	10/12/2021	0.621	6.5
Mn-54	312.3	2.22E-03	10.29	3/1/2020	10/12/2021	1.889	2.7
Co-57	271.8	2.55E-03	9.752	3/1/2020	10/12/2021	2.171	2.22
Cs-137	11205.5	6.19E-05	10.17	3/1/2020	10/12/2021	0.053	9.64
Cd-109	462.6	1.50E-03	10.36	3/1/2020	10/12/2021	1.275	4.13
Co-60	1924.645	3.60E-04	10.51	3/1/2020	10/12/2021	0.307	9.09

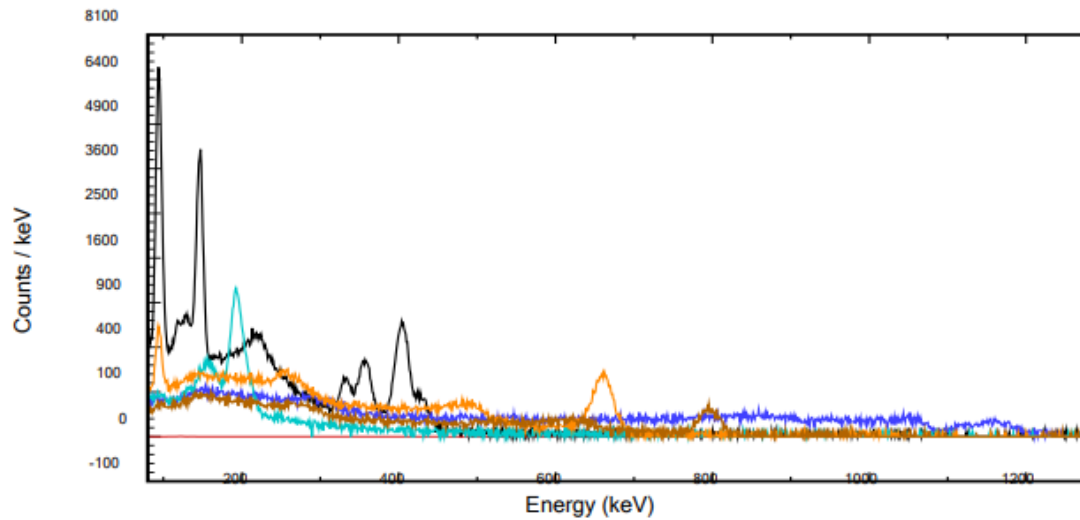


Figure 7-23 The spectrums collected with the ARGANOT modules was pulled into GADRAS.

Detector		Plot	Time History	Analyze	Model	Neutron	Inj
File Restore Points Help							
Detector Properties Type: Srl2 Efficiency (%): 0.38		Default ECal ☐ Prefer Ecal in File ☒ Always Use This Ecal			Environment On Ground		
Dimensions Setback (cm): 0.45 Length (cm): 0.6 Width (cm): 3 Height (cm): 1.3 Shape Factor: 100 Dead Layer (mm): 0 Scalar: 1		Order 0 in E: -524 Order 1 in E: 2933.21 Order 2 in E: 1960 Order 3 in E: 0 Low Energy: 80			Photon Scatter Clutter: 6.97 0 Degrees: 4 45 Degrees: 4.35 90 Degrees: 7.26 135 Degrees: 11 180 Degrees: 4.94 Rate @ E -> Edge: 3.26 Rate @ E -> 0: 0 Increase with E: 3.05 Attenuate: 2.22		
Peak Shape FWHM @ 0 (keV): 2.9 FWHM @ 661 (%): 3.42 FWHM Power: 0.6 Low-E Skew/Tail: 0 High-E Skew/Tail: 0 Low-E Skew Power: 0.5 High-E Skew Power: 0 Low-E Skew Extent: -2.48 High-E Skew Extent: -2.48		Inner Attenuator Atomic Number: 13 AD (g/cm2): 0.137 Porosity (%): 0			Advanced Scatter Air Pressure: 0.959 atm or Elevation (m):		
Lower Level Discriminator LLD (keV): 84 LLD Sharpness: 0		Outer Attenuator Atomic Number: 6 AD (g/cm2): 0.175 Porosity (%): 0			Neutron Scatter		
		Timing Compute Pileup: ☒ Pileup Rejector Present: ☐ Shape Time (μs): 20 Dead Time (μs): 25			Computation Options Weight Range for Fitting Lower Limit (keV): 20 Upper Limit (keV): 2039 Default # Channels: 2048 Template Error (%): 5 Assume MCA Bins from Inbin: ☐ Collapse Spectra to Rebin: ☐		
		Shield / Collimator: Fluorescence X-rays: Coincidence / Imaging: Miscellaneous:					

Figure 7-24 The Response Function for the module's using Srl₂(Eu) is shown.

Variable such as the deadtime of the ARGANOT gamma-ray detector were measured using the two-source method. Measurements revealed the $\text{SrI}_2(\text{Eu})$ detectors yield around 28 μs deadtime ($\pm 0.03 \mu\text{s}$). This is a tolerable and anticipated dead time for this detector type. When the comparison of real-life spectrum (solid line) to simulated spectrum (dotted line) was performed as shown in Figure 7, the source and the detector conditions were further altered so that the simulation for the given sources would emulate the empirical measurements.

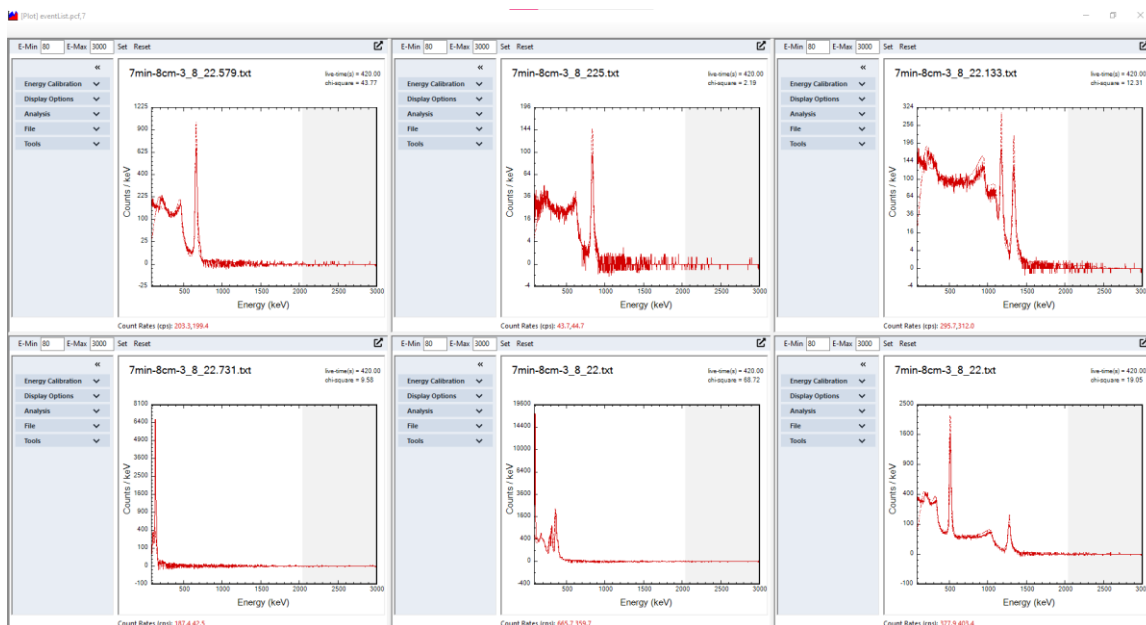


Figure 7-25 The empirical (solid line) and simulated spectrum (dotted line) for the given source conditions are shown. The overlay matches well, indicating a decent response function.

7.2.1 Isotope Identification

The GADRAS software was used to perform isotope identification and activity estimations on several sources. A Ba-133 source placed 8 cm away from the face of a single ARGANOT module and a measurement was taken for seven minutes. A background measurement of the same time followed. The spectra were then inputted into the GADRAS software for isotope identification and activity estimation. As shown in Figure 7-26, a list of considered isotopes and the suspected activity are listed. Ba-133 was however correctly selected with an estimated activity of 6.1 μCi . The known activity of the source at the time was 8.75 μCi , meaning the GADRAS software underestimated the activity by 2.65 μCi .

***** SOURCE RANKING BY SINGLE REGRESSION *****

detector name : RWDGenIV
 distance (cm) : 8.0
 foreground spectrum: eventList.pcf,6
 background spectrum: eventList.pcf,4
 collect date/time : 11-Mar-2022 01:56:18.00
 reduced chi-square : 81.514 weight range (keV): 19>2037 keV
 relative background: 1.000 background unc (%): 0.000
 template error (%) : 5
 gain shift (%) : 0.000 gain shift unc (%): 0.000

Source	Chisqr	Activity	Units	AN	AD	Gain(%)
Ba133	81.51	6.1 +/- 0.0	uCi	10.0 +/- 0.0	0.0 +/- 1.5	0.00
Co57	158.31	283.8 +/- 1.3	mCi	6.1 +/- 5.0	100.4 +/- 172	0.00
Cs137	159.51	266.0 +/- 1.3	Ci	4.0 +/- 5.6	255.8 +/- 408	0.00
Am241	164.55	43.9 +/- 0.2	mCi	29.1 +/- 69.	4.9 +/- 119	0.00
Ce139	173.33	77.8 +/- 0.4	MCi	7.1 +/- 43.	256.0 +/- 67.	0.00
Mn54	174.26	18.7 +/- 0.1	Ci	4.0 +/- 6.8	255.5 +/- 380	0.00
Na22	196.13	2.8 +/- 0.0	uCi	4.0 +/- 59.	52.2 +/- 296	0.00
Eu154	196.42	575.9 +/- 5.6	mCi	4.0 +/- 19.	254.3 +/- 999	0.00
Co60	196.59	154.2 +/- 1.5	mCi	4.0 +/- 21.	254.4 +/- 912	0.00
Eu152	197.44	488.6 +/- 5.0	mCi	4.0 +/- 77.	254.2 +/- 987	0.00
Th228	204.25	365.1 +/- 6.3	nCi	4.0 +/- 332	11.5 +/- 244	0.00
Ho166	205.13	10.6 +/- 0.2	uCi	10.0 +/- 0.0	0.0 +/- 88.	0.00
Y88	205.13	6.2 +/- 0.1	uCi	8.7 +/- 164	99.9 +/- 999	0.00

Ba133, 6.1uCi

Figure 7-26 Several isotopes were considered by the GADRAS software when a single regression was conducted. The software correctly selected the Ba-133 isotope and provided a reasonable activity estimate as highlighted by the red outline.

A vial of highly enriched uranium (HEU-93%) was then placed 3 inches above the ARGANOT detector. An 1800 second foreground and background measurement was collected and fed into the GADRAS software. The collected foreground may be viewed in Figure 7-27. Source identification and activity estimation results displayed in Figure 7-28 show the software correctly selected the isotope U-235 with an estimated mass of 273 mg. The true mass was approximately 57 mg.

A glass bottle of thorium nitrate was then placed 3 cm above the detector and a 1200 second live time spectrum was then collected. The spectrum shown in Figure 7-29 was fed into the GADRAS multiple regression analysis. Results shown in Figure 7-30, correctly identified the material as Th-232. The estimated mass was 2.61 grams, which decently reflects the true quantity of 2.2 grams.

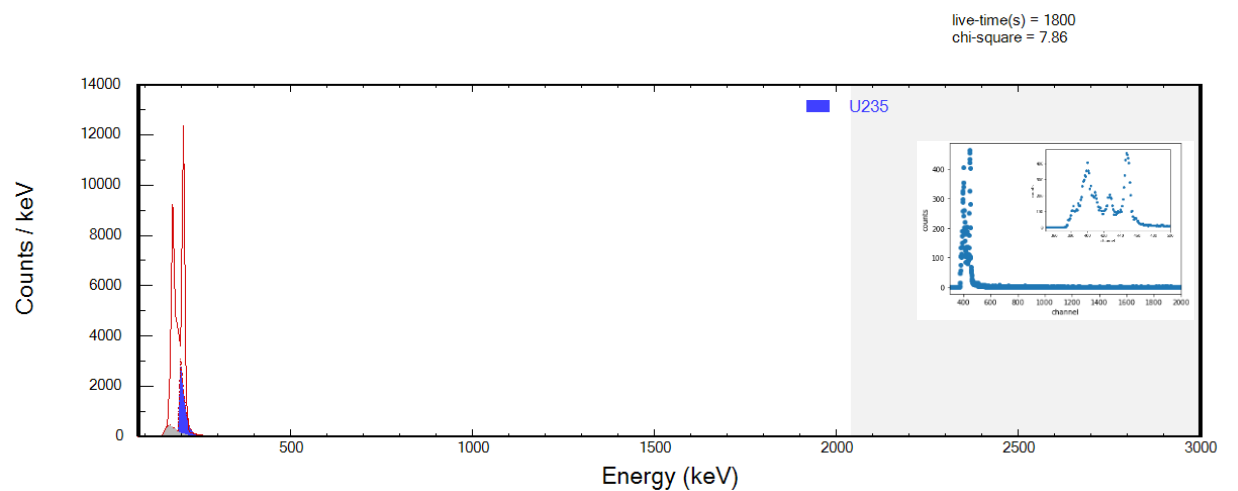


Figure 7-27 The gamma-ray spectrum collected for 93% enrich. U-235 is shown. The GADRAS software is shown fitting the peak to the suspected isotope.

Analysis Results		***** MULTIPLE LINEAR REGRESSION ANALYSIS *****				
File	Display	PlotOtherSolution				
Threat:	(SNM=H)		detector name	: RWDGenIV		
TTF,TTB:	1800, 600		distance (cm)	: 3.0		
Chi Square:	7.85878229141235		foreground spectrum:	eventList_Test.pcf,2		
Net Gammas:	154.09 cps		background spectrum:	eventList_Test.pcf,4		
Net Neutrons:	0 cps		collect date/time	: 14-Mar-2022 12:53:00.00		
Sigma Gammas:	1888.2		reduced chi-square	: 136.294	weight range (keV):	19>1203 keV
Sigma Neutrons:	0		relative background:	1.000	background unc (%)	: 0.000
SNM Prob:	(High)		template error (%)	: 5	gain shift (%)	: 0.000
Alarm Severity:	RED		gain shift (%)	: 0.000	gain shift unc (%)	: 0.000
Alarm Description:	Probable SNM		material 1			
Event Type:	SNM		shape	: point		
Isotopes:	U235 (H)		composition	: U235	100 %	
			Source	Activity(nCi)	Weight(gm)	AN AD
			U235	590. +/- 5.90	2.73E-01	88.6 +/- 16. 1.9 +/- 101
			U235,590.1nC(an=88.6,ad=1.9)			

Figure 7-28 The GADRAS Analysis for 93% enrich. U-235 correctly identified the material as SNM and estimated the mass to be 273 mg.

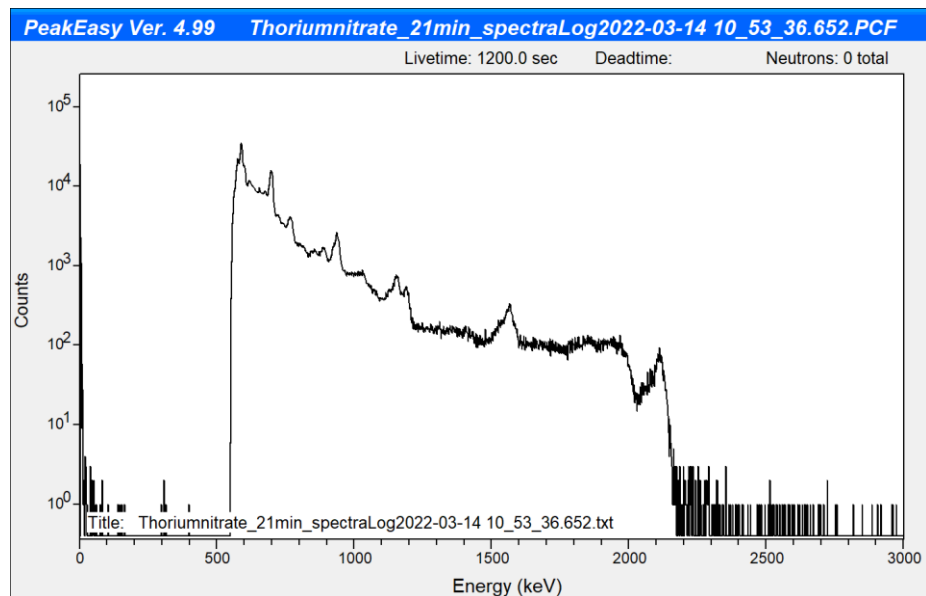


Figure 7-29 The Thorium Nitrate spectrum collected over the 120 second measurement is shown.

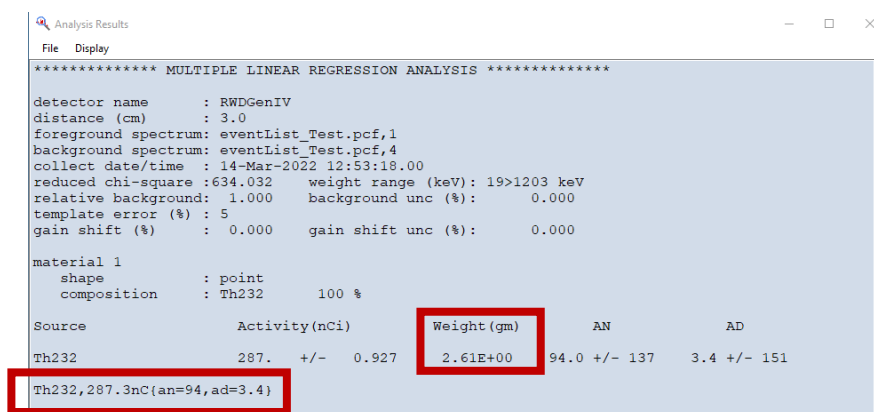


Figure 7-30 The GADRAS Analysis for the Thorium Nitrate housed in a glass container to shown. The software correctly identified the material and suggested an accurate mass.

7.3 Mobile Detection

The ability for the user to operate the system in the field by wearing it in a low-profile garment or flying it on a drone showcases the flexibility of the technology. Throughout testing, the system response to a moving source was demonstrated when worn by a user. Additional testing was performed to demonstrate the performance of a single module when in flight on a drone to search and locate radiological sources.

7.3.1 Moving Source

A series of tests were conducted in which a single DS-MSND populated module was exposed to a moving 6.92 nCi bare AmBe neutron source. As defined by ANSI N42.53, the source was attached to a rail and moved down the rail at rate of 1.2 m/sec. The module was placed at the center mass of the phantom centered 1 m from the track. Initially the source was shielded behind a high-density polyethylene board. Upon the start of the test the source was removed from its shielding until it reached the end of the track and was again shielded. The neutron counts over the course of the source's movement across the view of the detector were recorded. The same test was then conducted for a MSND populated module. Figure 7-31 displays the configuration of the Phantom wearing the vest in relation to the moving source.

Results shown in Figure 7-32 demonstrate that both modules were able to alert to the passing of the neutron source. The DS-MSND module had a much higher and more uniform performance when compared to the MSND module. The average intrinsic thermal neutron detection efficiency of the Single sided devices used in this module were approximately $23\% \pm 1\%$ whereas the average thermal neutron detection efficiency of the dual sided populated module using recent rather lower efficiency devices was approximately $33\% \pm 2\%$.



Figure 7-31 The phantom is shown wearing the ARGANOT populated garment as the moderated ^{252}Cf neutron source passes on the mail.

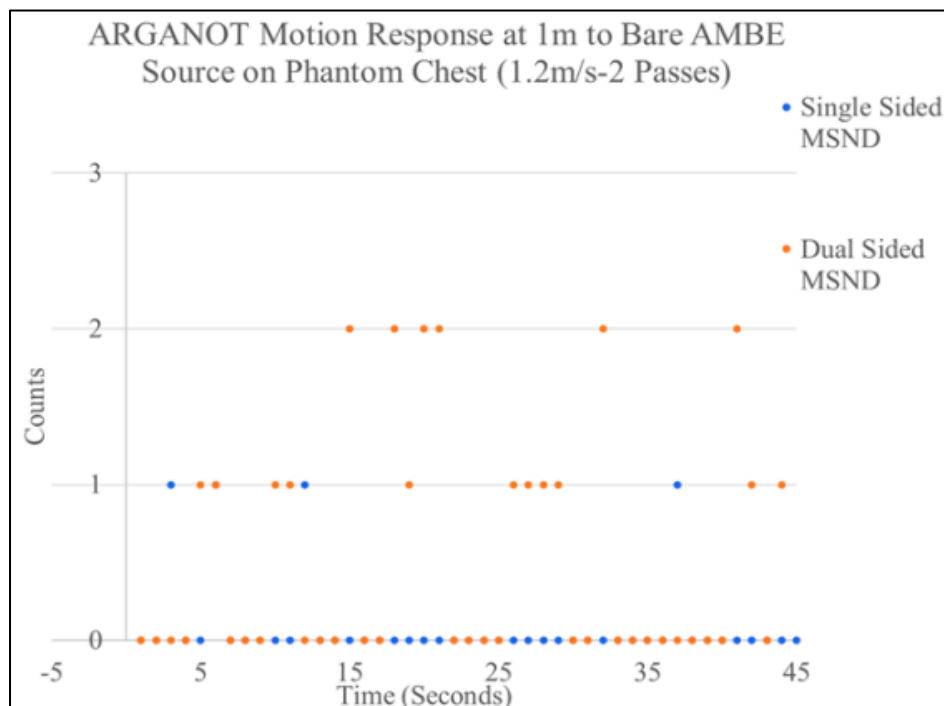


Figure 7-32 shows the plotted results of the motion testing for a single module populated with both MSND and DS-MSND devices.

7.3.2 Drone Flight

The performance of a single ARGANOT module when flown on a DGI drone was characterized. As shown in Figure 7-33, the module was attached to the bottom of the drone. The communication android device was attached to the top. The drone initially flew two background mission with an average height of 7.5ft and a speed of 6 mph and 8.5ft at a speed of 6.75mph, see Figure 7-34. The background was recorded and a 0.8mCi ^{137}Cs source was placed in the field 40 meters from a 21.9 ng ^{252}Cf neutron source. Two grid routes were run with average altitudes and speeds of 3.3ft and 5.2mph, 9.9ft and 7mph, and a direct flyover at an altitude of 8ft and a speed of 6.4mph. Throughout the measurements, both gamma-ray and neutron count rates were recorded along with the collected gamma-ray spectrum. The gamma-ray and neutron count rates and collected background spectrum for the 9.9ft and 7mph flight are displayed in Figure 7-35 and Figure 7-36.



Figure 7-33 The DGI drone the ARGANOT module was flown on is pictured.

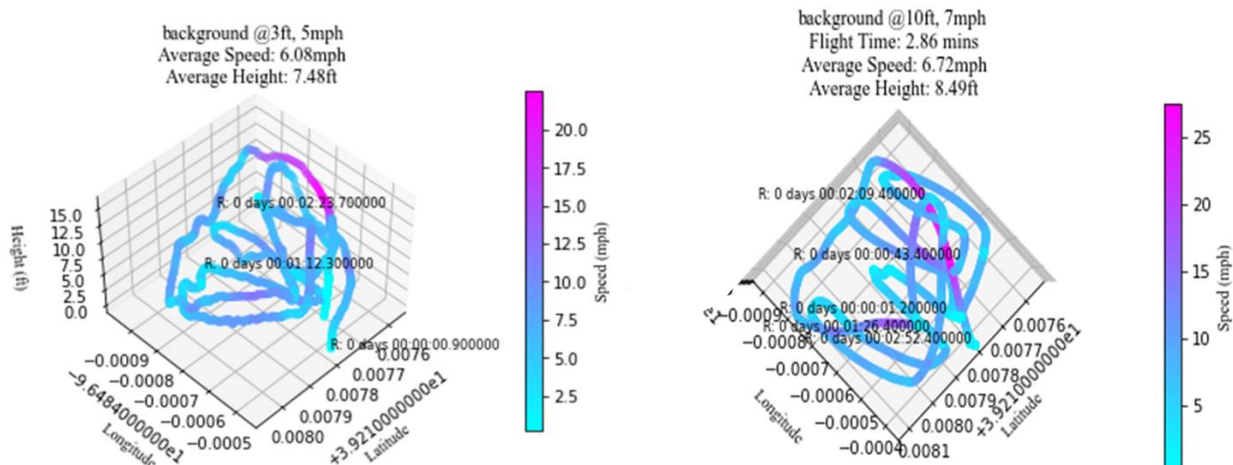


Figure 7-34 The background flight that was aimed to fly at 3ft high at 5mph (left) and 10ft, at 10mph (right). The recorded average speed and height were 6.1 mph and 7.5ft and 6.7mph and 8.5ft.

The gamma-ray and neutron count rates over the flight path are displayed for the 3.3ft and 5.2mph flight in Figure 7-37. The gamma-ray count rate over time and the detection probability is shown in Figure 7-38. The threshold as indicated with red represents the alerting threshold while individual counts are highlighted to represent whether they triggered the alarm or not. The threshold was determined to be >25 counts per second by setting the number of false alerts to one every eight hours, or once a working day. The neutron response throughout the 3.3ft and 5.2mph time of flight is displayed in Figure 7-39. The threshold based on the neutron background was set to >1 count per second. The gamma-ray and neutron counts over the 9.9ft and 7mph flight path is shown Figure 7-40. Within the flight path, the 0.8mCi ^{137}Cs gamma-ray source, a bare 29 ng ^{252}Cf neutron source, and 0.8uCi ^{137}Cs check source locations are all labels across the map. The color bar correlates to increased count rates, correlating with position of sources. The 0.8mCi source was located by the drone and circled. The neutron count rate over the time interval for the path is shown in Figure 7-41 and the gamma-ray count rate over the time interval for the path along with the gamma-ray detection probability is shown in Figure 7-42. Count rates represented on the flight path map correlate with the elevated count rates shown in the time interval. The gamma-ray threshold was set to >19 counts per second, and was triggered three times, correlating with passes by the gamma-ray source through the flight search.

The final flight flown was a direct fly over the ^{252}Cf and ^{137}Cs sources. The flight path and collected gamma-ray spectrum may be viewed in Figure 7-43. A clear ^{137}Cs 662 keV peak is

visible at channel 520. The flight was initiated near the 0.8mCi gamma-ray source, flew over top of the bare ^{252}Cf source, and returned to the operator's location, southeast of the gamma-ray source. The first 50 seconds of the flight count rate which includes the drone circling the ^{252}Cf source, is shown in Figure 7-44. Using the data and the background, the gamma-ray threshold was determined to be >19 counts per second, which was a false alert allowance of once per eight hours. Through the direct flyover flight, the module gamma-ray alerted four separate periods, with three correlating with the drone flying in the vicinity of the ^{137}Cs or ^{252}Cf sources. The outlier alert took place after the drone left the ^{137}Cs location and was in route to the ^{252}Cf source. One possible explanation for the alert was that drone was repositioned with the wind to be in an optimal detection position before returning to a forward moving position. The day of the testing had reported wind gusts of up to 15 mph and occasionally presented flight obstacles. Additionally, because the GPS location and time was not perfectly in sync with the ARGANOT system, it is possible that the reported GPS location and time stamps deviated slightly as the ARGANOT system only reported every second, whereas the drone system reported every tenth of a second.

Throughout the drone flights, using the determined threshold, all passes over the gamma-ray source triggered the alarm. Throughout all flights, the count rates clearly increased when approaching the vicinity of the source and gamma-ray spectrums were collected to indicate the presence of a ^{137}Cs source. Though, there are instances where the data point indicating a rise in count rate does not perfectly align with the position of the source on the map. As shown in gamma-ray plot in the left image of Figure 7-37, the source was reported to be located several feet before and several feet after the actual source location. The deviation largely results from changes in drone speed and the drone location updating every tenth of a second whereas the ARGANOT system updates per second. Despite a few minimal deviations in elevated count rates and reported gps location, results reveal that the ARGANOT modules perform well on drones at a variety of speeds and altitudes. Future work will include synchronizing the ARGANOT application location and time stamp with the drone to eliminate any deviation or lags. Additionally, future work will log gamma-ray spectrums per one second time interval to allow a time interval of spectrums to be collected as the drone approaches the source.

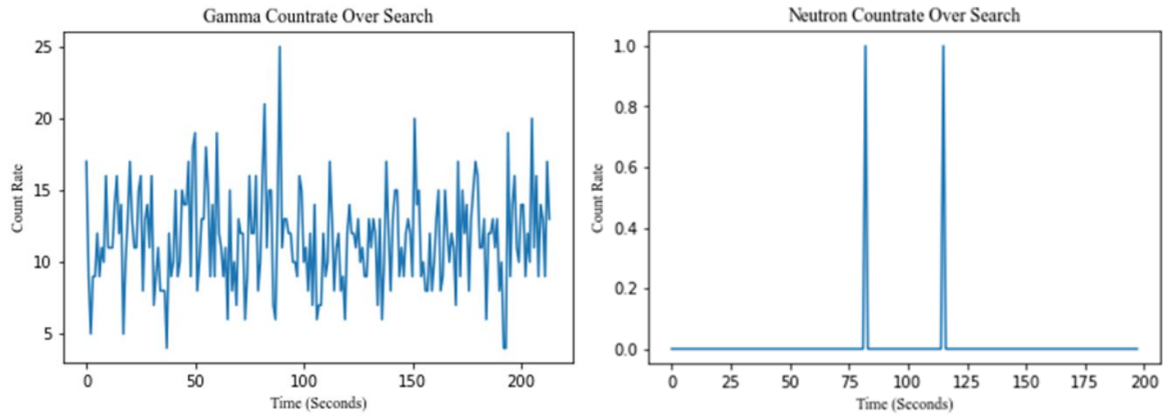


Figure 7-35 The gamma-ray count rate over the search (left) and the neutron count rate over the search (right) are shown for a drone grid search flight over the field at an average altitude of 3.3ft and 5.2mph.

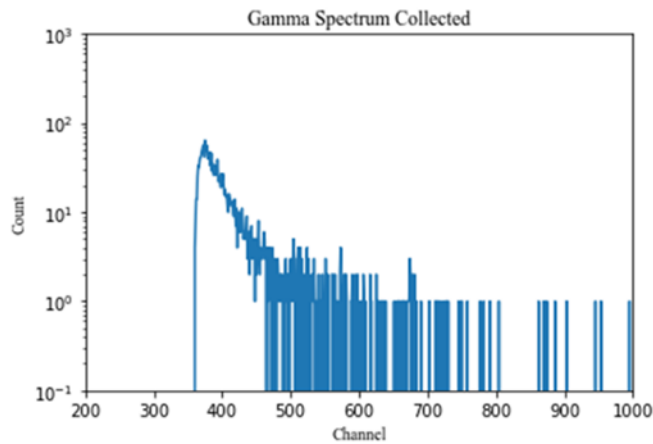


Figure 7-36 The gamma-ray spectrum for the 3.3ft and 5.2mph grid search of the field is shown. A ^{137}Cs peak began to form over the measurement.

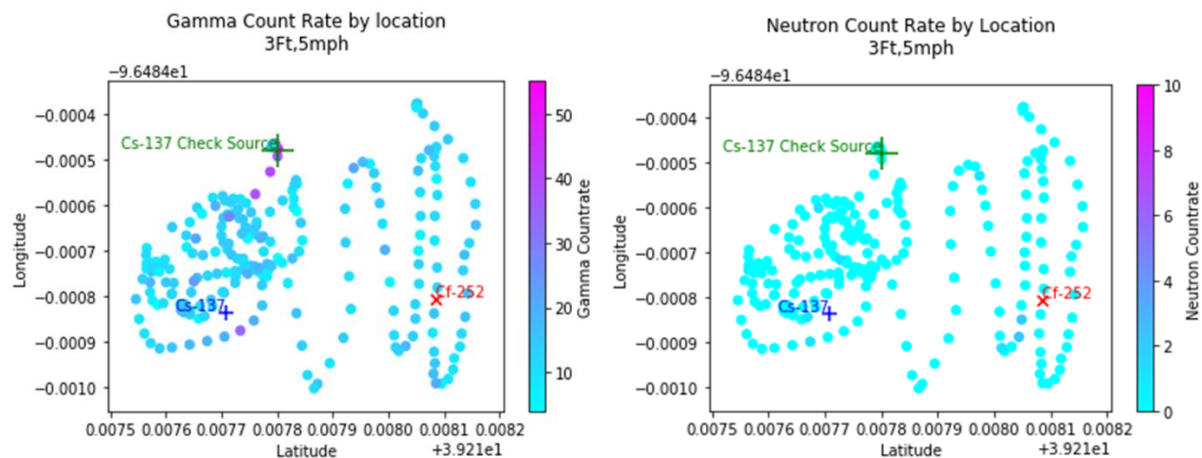


Figure 7-37 The gamma-ray (left) and neutron (right) countrates over the search are shown. The change in data point color to darker colors indicate the presenece of a source.

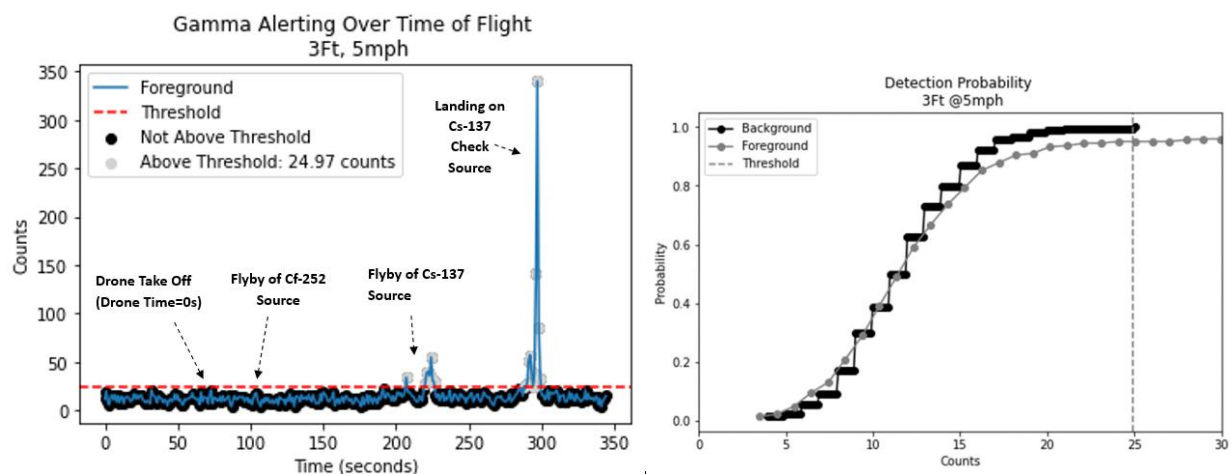


Figure 7-38 The threshold was determined using the probability distribution function (right) of both the gamma-ray foreground and the background measurement for 3ft 5mph flight. The count rate over the time of flight that triggered an alarm is shown on the left.

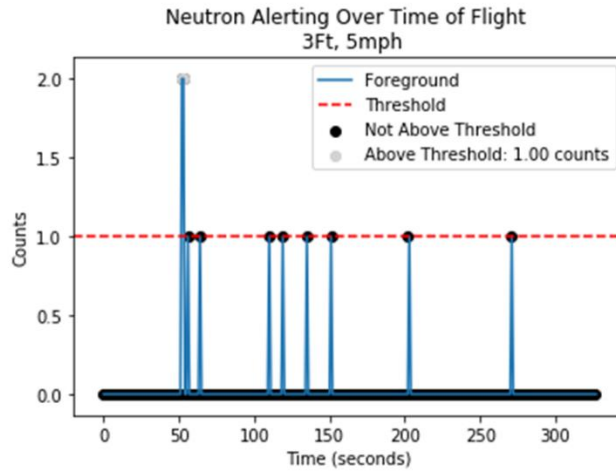


Figure 7-39 The neutron count rate over the flight and the triggered alarms are shown.

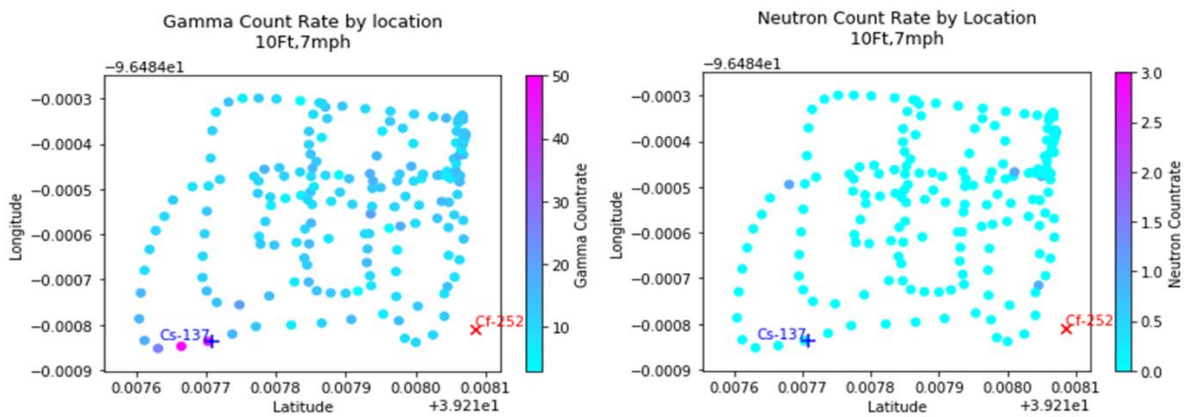


Figure 7-40 The gamma-ray(left) and neutron (right) count rates over the flight path for the 10ft 7mph flight are shown.

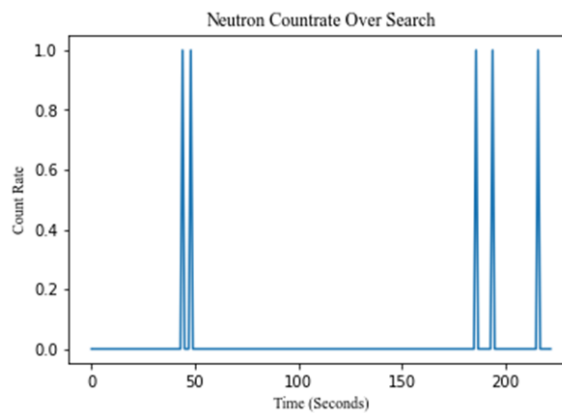


Figure 7-41 The neutron count rate over the flight and the triggered alarms are shown for the 10 ft 7mph flight.

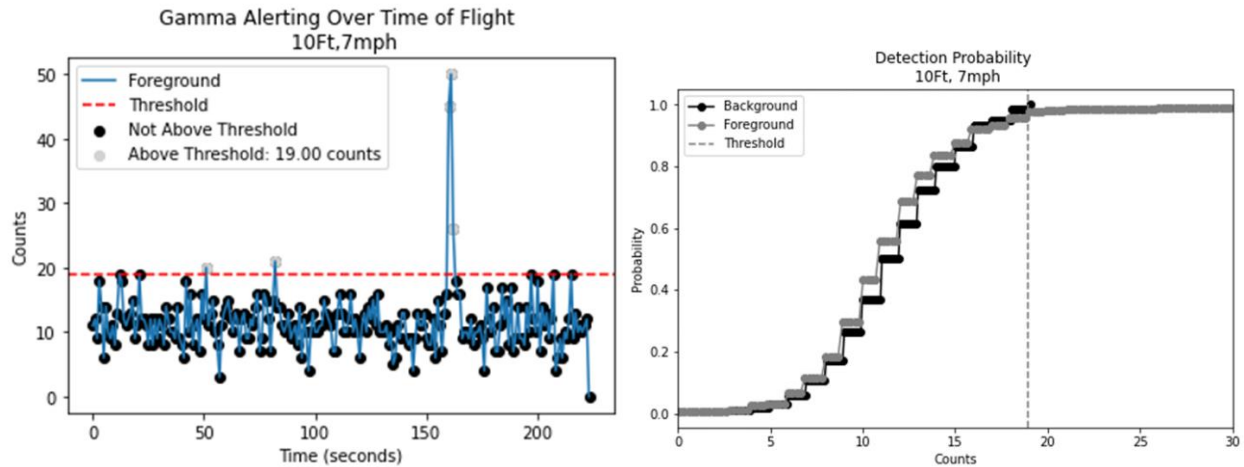


Figure 7-42 The threshold was determined using the probability distribution function (right) of both the gamma-ray foreground and the background measurement for 3ft 5mph flight. The count rate over the time of flight that triggered an alarm is shown on the left.

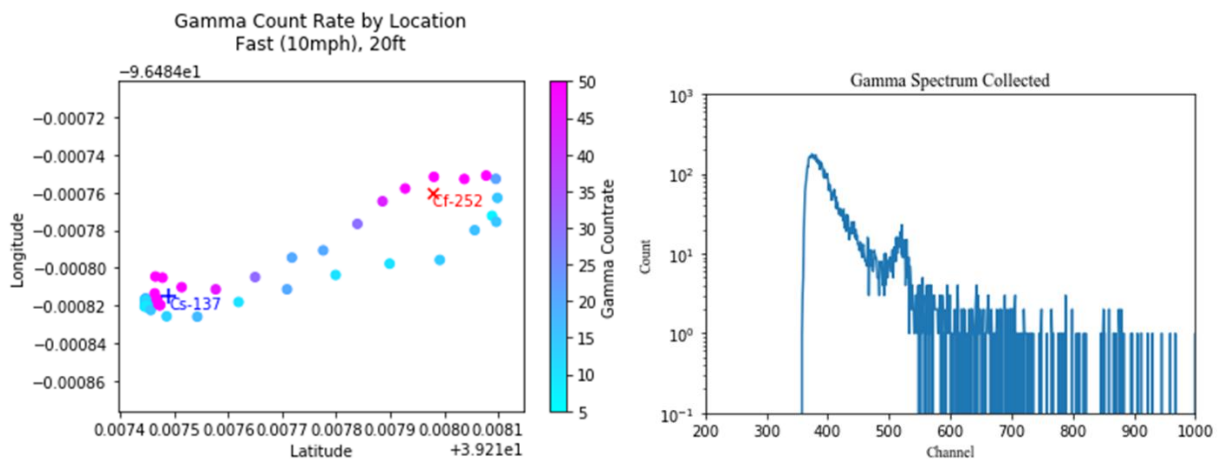


Figure 7-43 The gamma-ray count rates (left) and gamma-ray spectrum (right) collected for the direct flyover flight are shown. In the spectrum shown, the zero offset of the x axis is rather large, with channel 380 representing approximately 80 keV.

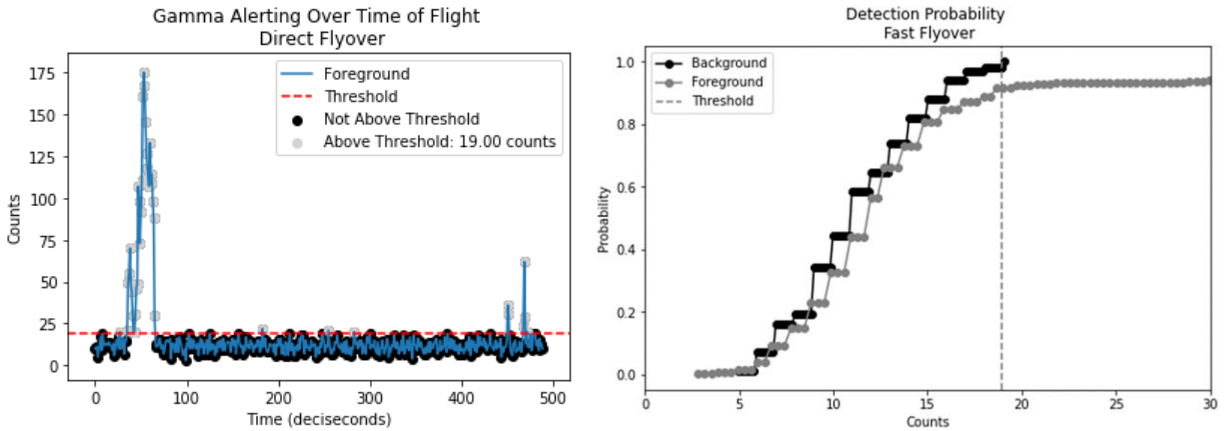


Figure 7-44 The threshold was determined using the probability distribution function (right) of both the gamma-ray foreground and the background measurement for the direct flyover flight. The count rate over the time of flight that triggered an alarm is shown on the left.

7.4 Directionality

There are many scenarios where the search application of the ARGANOT system would be valuable to a user. Whether a user needs to place a system roadside to indicate a source direction of travel in traffic, search a building thought to house nuclear material, or even map a nuclear disaster region, the ARGANOT is an optimal candidate.

When opening the ARGANOT application, if directionality features are desired, the set-up wizard provides detailed instructions to the user to ensure the position of each module is known. As shown in the right image of Figure 7-45, the identification number of each module appears on a specific location on the phantom's body. Within the directionality algorithm, the gamma-ray counts of each module is considered and the arrow shown on the left of Figure 7-45, tells the user the direction of the source. In the event that the deviation between count rates is not large enough to trigger a change of direction, the arrow remains at 0 degrees. While the ARGANOT system can operate as a single module or in a distrusted mode, if the user wishes to use the directionality feature, they must equip the vest with 2, 4, or 8 modules. The modules must be evenly placed across the front and back of the body to ensure each modules' reported count rate is evenly weighed within the algorithm. As the number of modules increases, the angle resolution increases. Presently, the ARGANOT system is only capable of applying the directionality algorithm in search of gamma-ray sources.

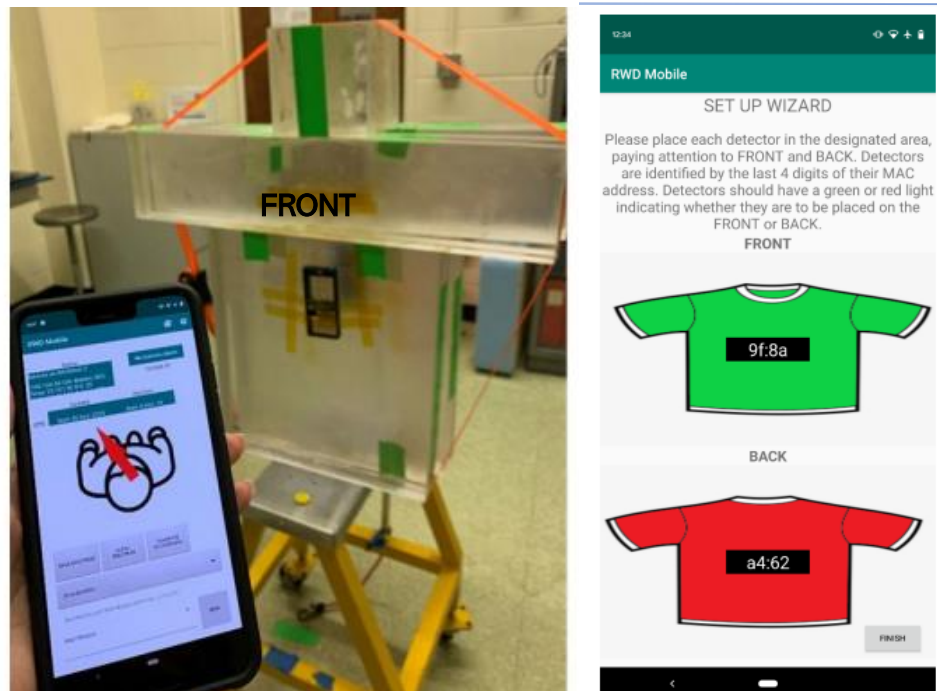


Figure 7-45 The directionality page (right) instructs the user where to place the modules on the body. The user then can be notified of the direction with the highest count rate and identify the source location (left).

7.4.1 Gamma Directionality

Four ARGANOT modules were placed on the phantom according to the ARGANOT android screen instructions. The phantom chest was then positioned to be parallel with the track, 2 ft from the rail track. A 0.83mCi source was placed on a rail to the left of the phantom, 1 meter above the ground as displayed in Figure 7-46. The source remained in place for approximately two minutes. The rail speed was then set to 1.2 ms^{-1} and the source moved down the rail. The rail movement was recorded as reflected in Figure 7-47 and the directionality red indicator arrow shown in the left image of Figure 7-48 began pointing towards the source to the left of the phantom. As the source moved across the chest of the phantom, the indicator at one point indicates the source was directly in front of the phantom, shown in the image on the right of Figure 7-48. The source at approximately second 1215 reached the end of the track and then returned back to the starting location, as reflected in Figure 7-47.

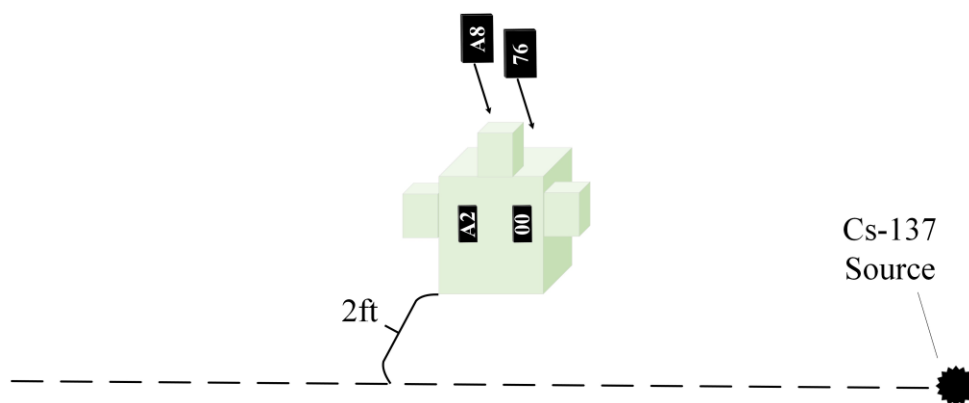


Figure 7-46 The phantom equipped with four ARGANOT module was placed parallel with the path of the source. The source passed across the face of the detector.

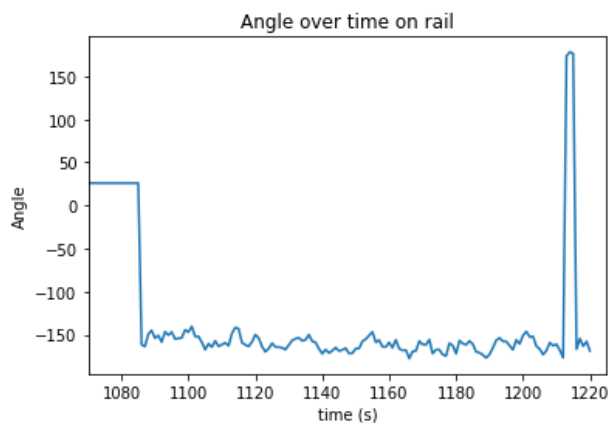


Figure 7-47 An ARGANOT system was located in the middle of a track, two feet from where a 0.83mCi ^{137}Cs source would pass. The ARGANOT system alerted to the source as it was initially placed, and as it passed by the phantom.



Figure 7-48 Two of the four ARGANOT modules distributed across the phantom's body are shown communicating livetime directionality information to the android advice. As shown on the left, the source is initially located to the left of the phantom. As the source approaches the phantom, the android device alerts the user that the source is in front of them.

The phantom was then repositioned for the chest to be perpendicular to the track as reflected in Figure 7-49. Within the configuration, detector modules A2 and 00 faced the source head on while modules A8 and 76 were located on the opposite side of the phantom.

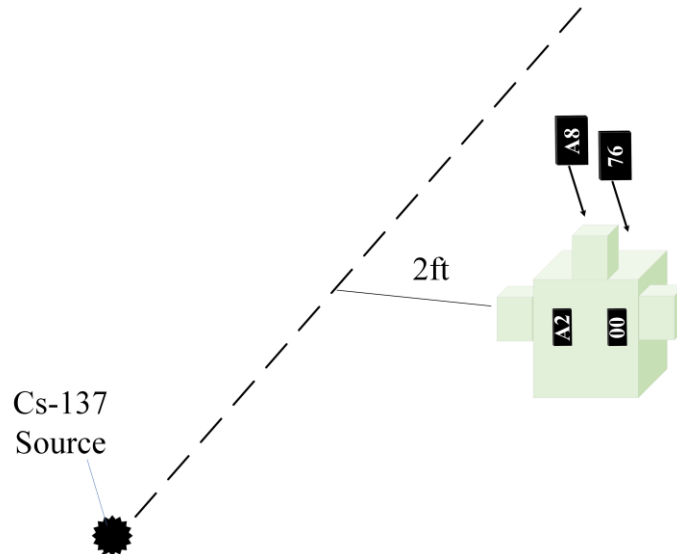


Figure 7-49 The path of the phantom was then placed perpendicular to the path of the source.

As plotted in Figure 7-50, the source initially was located behind the phantom. The user was alerted to the source location as shown in Figure 7-51. By second 250, the source began moving to the opposite side of the rail. Within the time plot of Figure, the source can then be seen returning to the starting location.

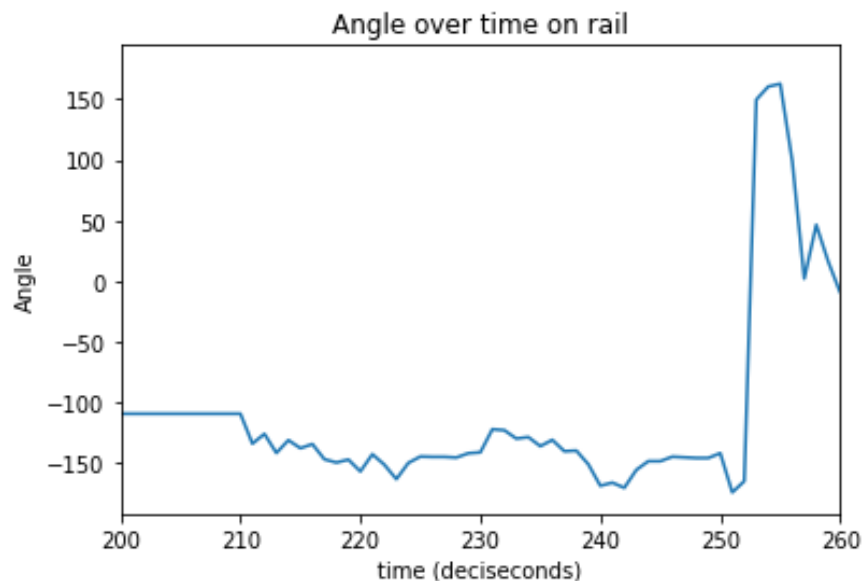


Figure 7-50 The location of the 0.83mCi ^{137}Cs source pass by the phantom placed perpendicular to the rail is shown. The source started to the left of the phantom and traveled to the right. The source then began its return to the starting position.

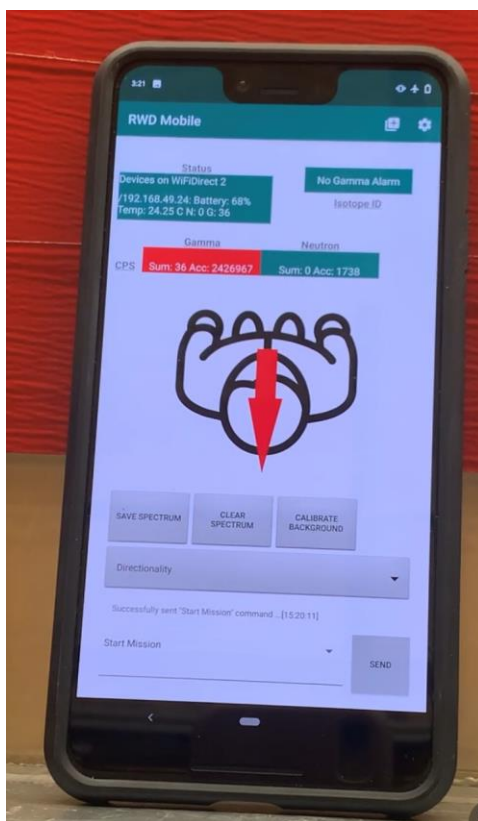


Figure 7-51 The direction arrow indicated that the source was on the rear of the user.

At the conclusion of the motion testing, the ^{137}Cs source was then placed to the left of the phantom, which was facing the rail. The four-module populated system then recorded the position of the stagnant source for over ten hours. Results in Figure 7-52 initially showed the source moving from the right to the left side of the track where the source remained for the continuation of the measurement. Throughout the ten-hour measurement, the directionality indicator consistently alerted the user to the source location directly to the left and no deviation was observed.

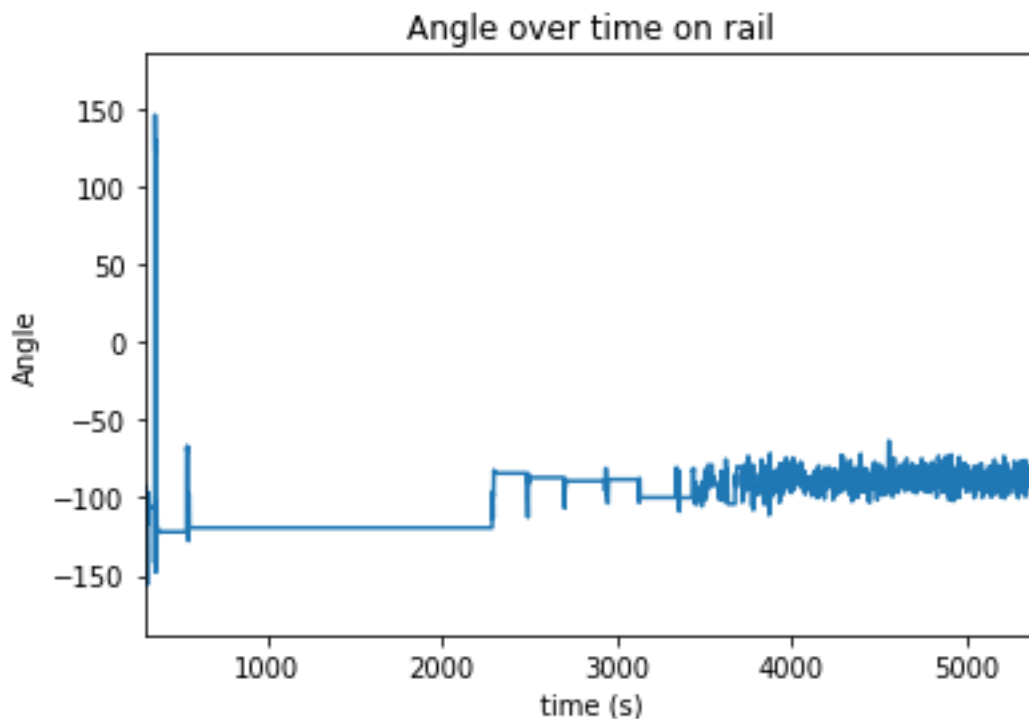


Figure 7-52 The source was placed for over ten hours to the left of the phantom when parallel to the track. The results over the ten hours period reveal the system consistently and accurately indicated the source position relative to the phantom.

7.5 Fast Neutron Detector Design Optimization

According to the Atomic Energy Act of 1954, SNM is plutonium, uranium-233, or uranium enriched in the isotopes uranium-233 or uranium-235, but does not include source material. The United State Nuclear Regulatory Commission (NRC) states that “Special nuclear material is only mildly radioactive, but it includes fissile isotopes — uranium-233, uranium-235,

and plutonium-239 — that, in concentrated form, could be used as the primary ingredients of nuclear explosives” [157]. The NRC goes on to warn that large quantities of SNM, known as strategic special nuclear material (SSNM), an nuclear explosive device can be made. Because Pu for example is a fast neutron emitter, it was also beneficial to begin to develop a neutron sensor that was sensitive to fast neutrons that could populate the ARGANOT modules.

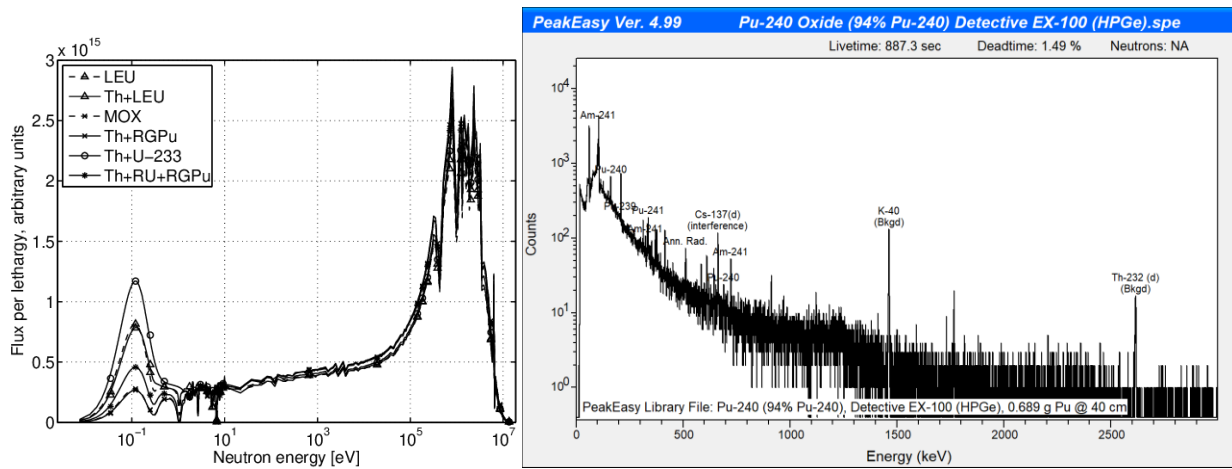


Figure 7-53 The neutron spectrum of several nuclear fuels which all contain components of SNM and the gamma spectrum of Pu-240 oxide (94% Pu-240) collected on a HPGe detector are both shown to demonstrate the expected SNM signatures [158].

To maintain the size and reproducibility of a fast neutron detector that could be implemented easily into the ARGANOT system, MSND were an optimal candidate for fast neutron detection when backfilled with a material sensitive to recoil protons produced from elastic scattering in the hydrogen. The scattering cross section for hydrogen is larger than the absorption cross section for thermal neutron conversion materials such as ^{10}B for neutron energies above 1 keV, but it is much lower than the absorption cross section for ^{10}B for neutron energies below 1 keV. Previous work by McGregor et al. has examined the performance of silicon based High-density polyethylene (HDPE) thin film devices with thickness of 125 μm and 875 μm when exposed to 14 MeV neutrons from a D–T reaction generator [159]. Spectra of both HDPE thicknesses when compared to the thin film device’s bare response indicated the presence of a fast neutron source, with 875 μm HDPE film producing a superior fast neutron intrinsic detection efficiency.

However, due to the difficulty in backfilling HDPE into the trenches of MSND type devices, paralyene was selected as an alternatively rich hydrogenous backfill material that could

easily melt into the trenches of the devices, making the backfill percentage 100% [160]. Therefore, to optimize the performance of the fast MSND, MCNP simulation were conducted for a number of geometric parameters as explained in Figure 7-54, such as the trench depth, the trench pitch, the trench width for standard size diode of 1 cm by 1 cm with a wafer thickness of 0.0625 cm. Specific parameters may be viewed in Table 7-5. Additionally, for each combination of trench dimensions, the LLD was 300, 800, or 1300 KeV.

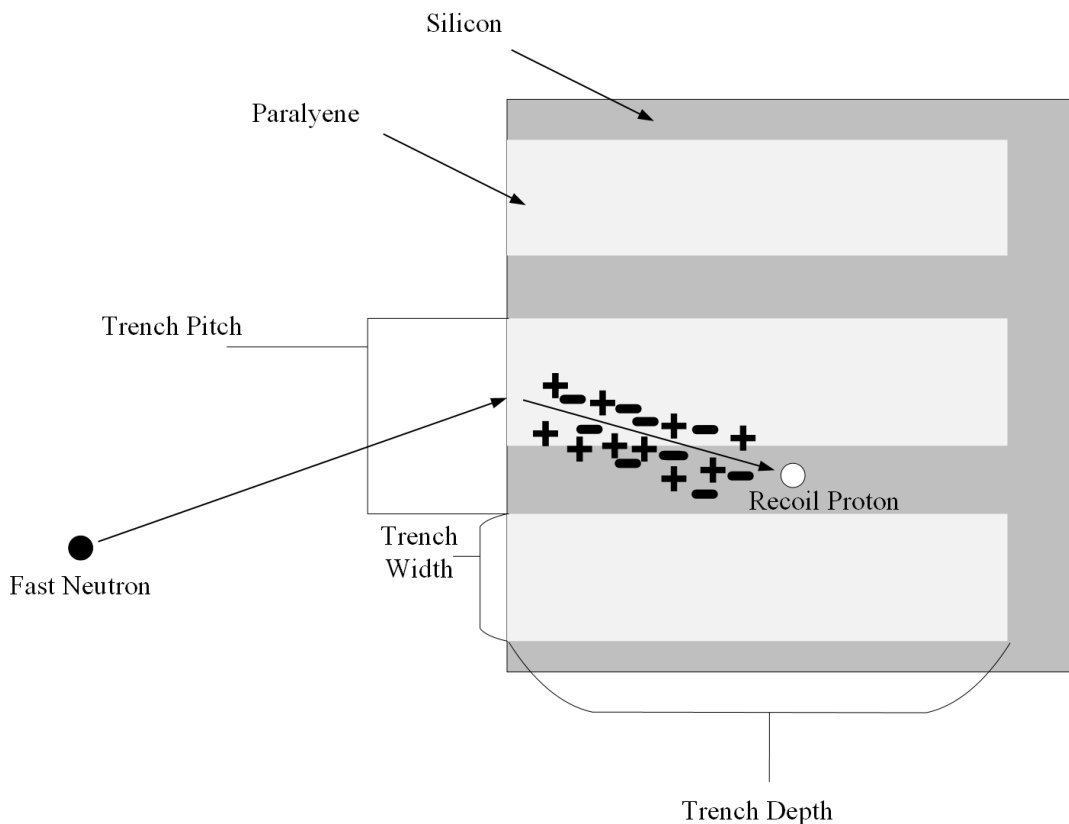


Figure 7-54 A fast neutron enters the trenches filled with paralyene, producing a recoil proton. The recoil proton then ionized the silicon material, producing a signal. The Trench pitch, width, and depth are all indicated.

Table 7-5 The various MSND device dimensions are shown. The wafer thickness was assumed to be constant, along with the diode length, width, and the backfill percentage.

Wafer Thickness (cm)	0.0625		
Trench Depth (cm)	0.02	0.04	
Pitch (cm)	0.002	0.005	0.01

Trench Width (cm)	0.0002	0.003	0.01
Backfill Packing Fraction	100%		
Diode Width (cm)	1		
Diode Length (cm)	1		
LLD (KeV)	0.3	0.8	1.3

The various simulations were performed for each combination of dimensions with the various LLDs for fast neutrons of energies shown in Table 7-6.

Table 7-6 The various energies of the fast neutrons in the simulation are shown in.

Energy (MeV)
0.4
0.6
1.7
2.4
3.4
4.9
7.0
10.0

Individual simulation results may be viewed in the Fast Neutron MSND Simulation Data Table of Appendix A. Using the legend of Figures 7-55, the plots of Figures 7-56 through Figure 7-61 demonstrate the isotropic efficiency of each pitch, trench width percentage, and lower-level discriminator. For each trench depth and pitch, a trench width percentage of 10% is represented by circular markers, a 50% trench width percentage is represented by triangular markers, and a 90% trench width percentage is represented by square markers. An LLD of 300 keV is represented by hollow markers, an LLD of 800 keV is represented by gray markers, and a 1300 keV marker is represented by a black marker.

KEY

- 0.1 Trench Width Percentage, 0.3 LLD
- 0.1 Trench Width Percentage, 0.8 LLD
- 0.1 Trench Width Percentage, 1.3 LLD
- △ 0.5 Trench Width Percentage, 0.3 LLD
- ▲ 0.5 Trench Width Percentage, 0.8 LLD
- ▲ 0.5 Trench Width Percentage, 1.3 LLD
- 0.9 Trench Width Percentage, 0.3 LLD
- 0.9 Trench Width Percentage, 0.8 LLD
- 0.9 Trench Width Percentage, 1.3 LLD

Figure 7-55 The legend indicating the simulated parameters for the data in the plots below is detailed. The LLD is represented in MeV.

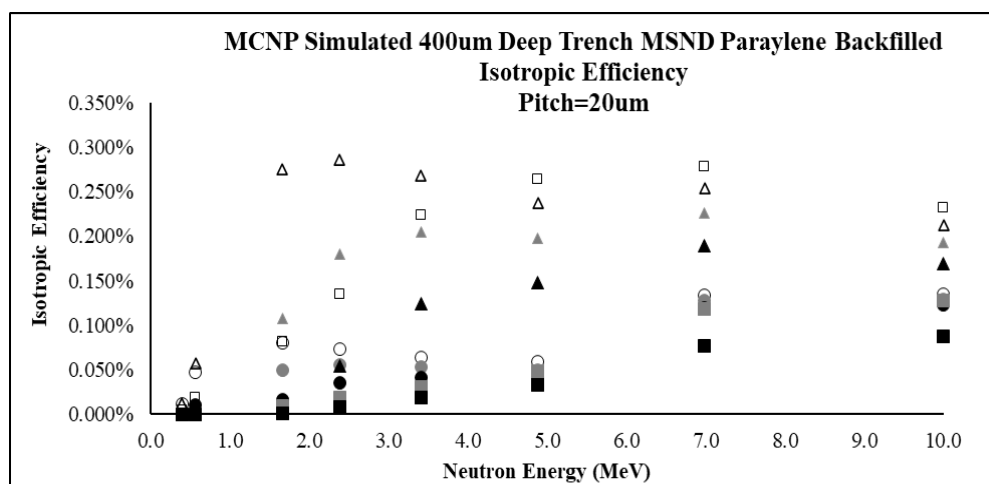


Figure 7-56 The simulated fast neutron response for a 400 μm deep trench and a pitch of 20 μm is shown.

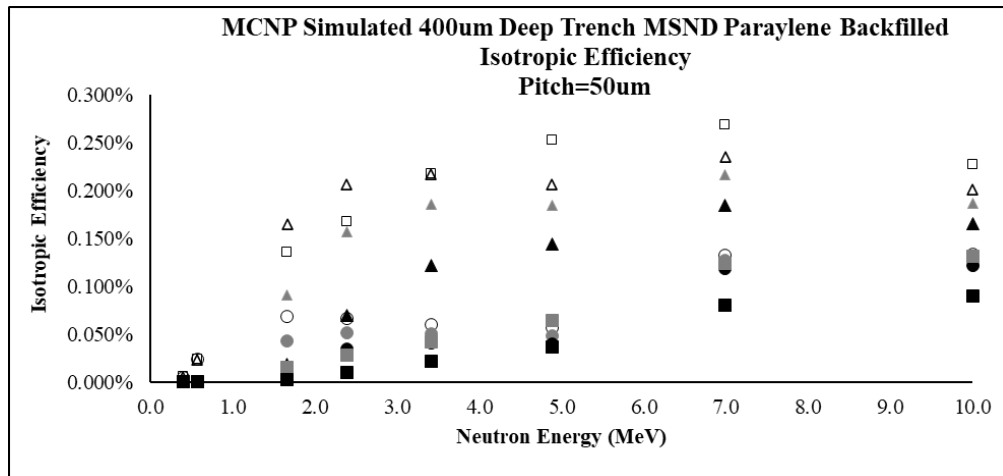


Figure 7-57 The simulated fast neutron response for a 400 μm deep trench and a pitch of 50 μm is shown.

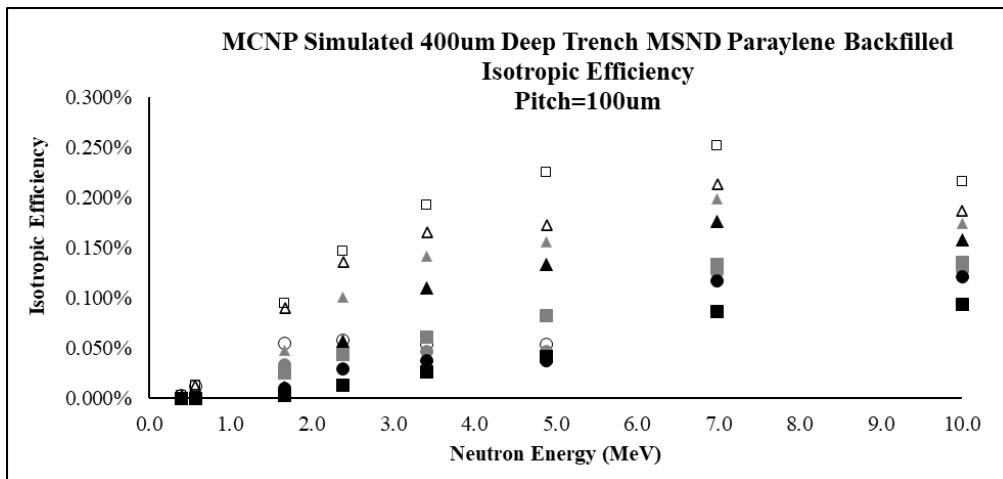


Figure 7-58 The simulated fast neutron response for a 400 μm deep trench and a pitch of 100 μm is shown.

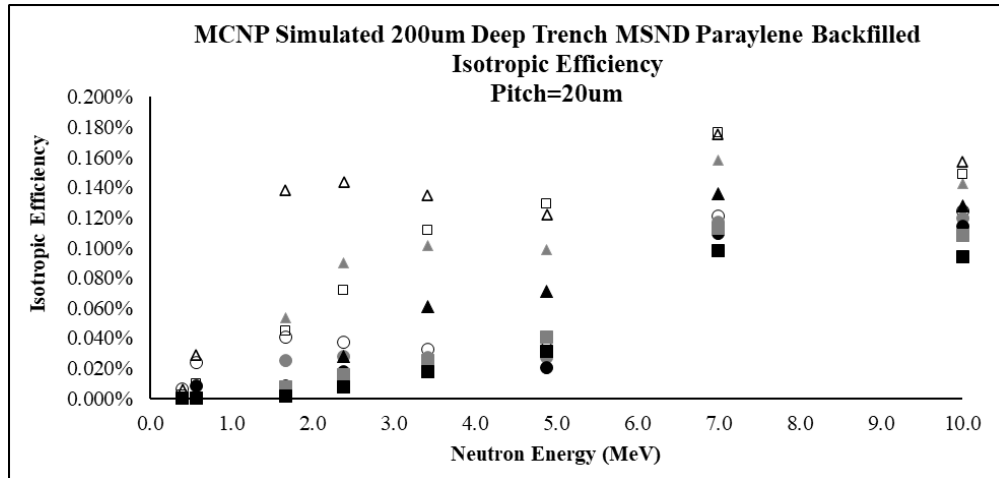


Figure 7-59 The simulated fast neutron response for a 200 μm deep trench and a pitch of 20 μm is shown.

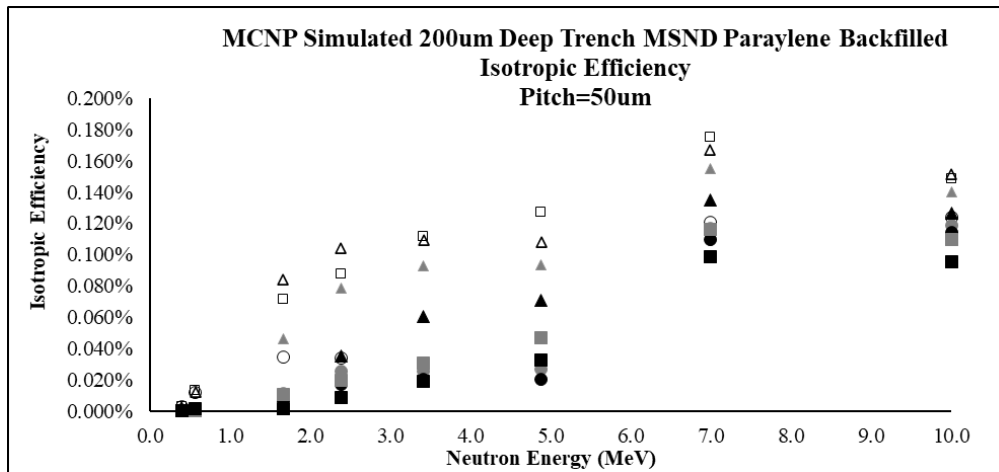


Figure 7-60 The simulated fast neutron response for a 200 μm deep trench and a pitch of 50 μm is shown.

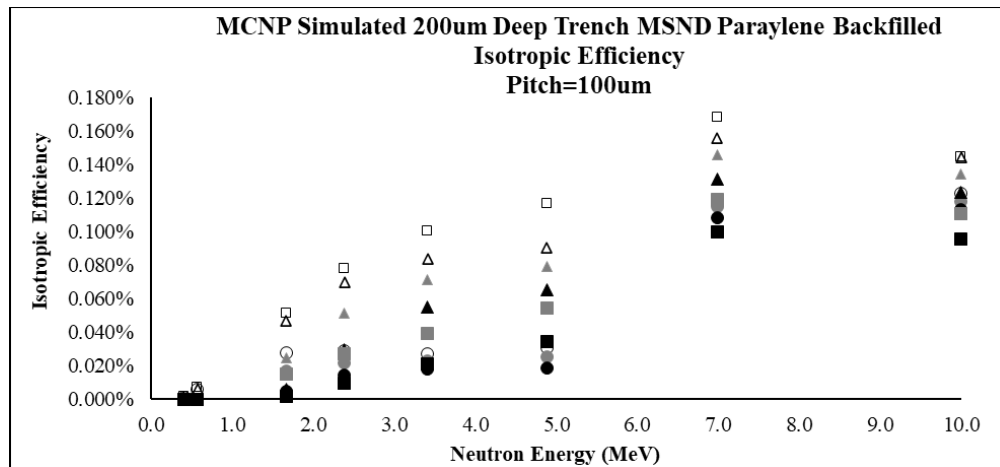


Figure 7-61 The simulated fast neutron response for a 200 μm deep trench and a pitch of 100 μm is shown.

When comparing the isotropic efficiencies, the 400 μm deep trenches consistently yielded higher efficiencies than the 200 μm trench depths. The 20 μm pitch also performed higher than larger pitched trenches, with a trench width percentage of 50% producing the highest efficiencies across all energies as demonstrated in Figure 7-56.

Therefore, the highest efficiency performing MSND backfilled with paraylene may be very close to existing MSND devices with a depth of 400 μm , though the pitch and trench width percentage must be adjusted from the existing design to optimize the fast neutron response performance.

7.6 Modeled SNM Detection Response

To understand the ARGANOT's ability to alert to SNM, GADRAS modeling was performed with the $\text{SrI}_2(\text{Eu})$ detector populating the ARGANOT module. Using the response function generated previously, the detector was placed more than 2 m away from the surface of aluminum structure that was 5 cm thick. The structure in this case was to stimulate a train car or truck. A mass of 200kg of U_8O_3 was then placed at the center of the aluminum structure. The modeled scenario is shown in Figure 7-62.

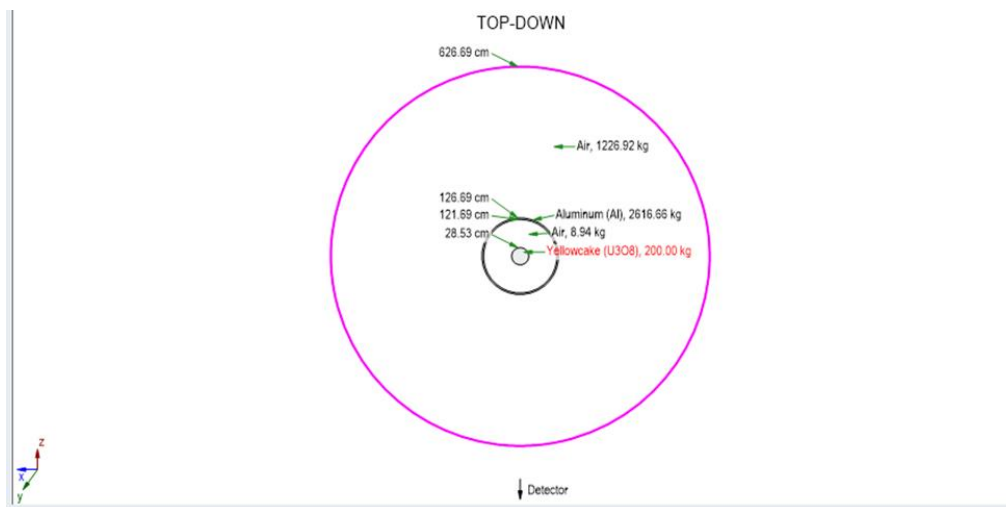


Figure 7-62 The GADRAS model in which a detector was placed over 2 m from an aluminum structure housing yellowcake Uranium is shown.

The resulting response is shown in Figure 7-63 for the 200 kg yellowcake was reported to have a k-eff value of 0.104 and a multiplication value of 1.11. The spectrum below demonstrates the intensity response in terms of gamma-ray counts per second over energy.

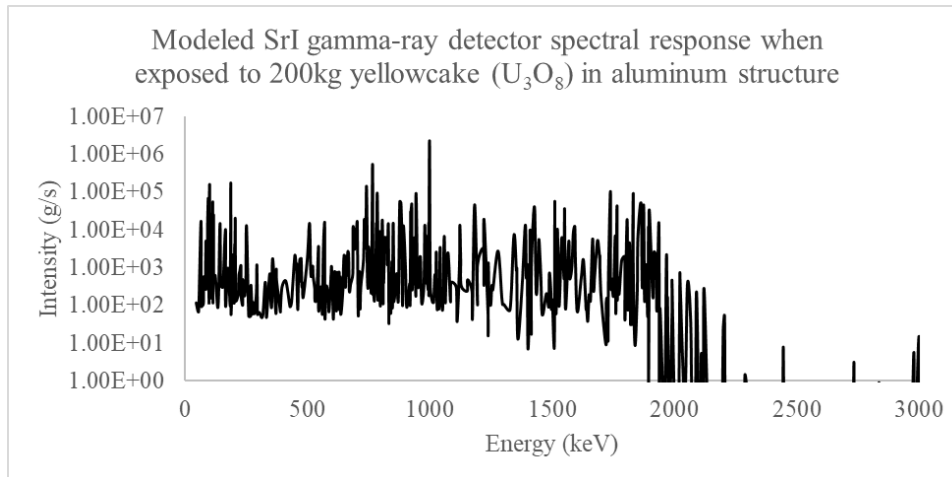


Figure 7-63 The GADRAS modeled response to SNM (200kg U_3O_8) when transported in an aluminum container.

Modeling results indicate an ability for the rather small $SrI_2(Eu)$ detector to still exhibit sensitivity to a SNM source when in the vicinity of the material in a transportation structure. This finding is especially encouraging considering in areas of the world where a collection roadside may be most valuable, larger sensors with higher efficiency may be impermissible. The ARGANOT while not as sensitive to such sources, may in fact be the only reasonable solution to

perform radiation measurement collections in areas of the world hostile to foreign radiation monitoring.

8 Conclusion



Developmental test flight of the US W88 Alt 370 submarine launched nuclear warhead. Image provided by the National Nuclear Security Administration.

“We are taking action to meet WMD challenges, and, as the nature of WMD threats is evolving, we know we have more work to do.” [161]

~ Jennifer C. Walsh, Undersecretary of Defense for Policy in 2021

8.1 Comparison to other Detection Systems

The ARGANOT system is quite competitive when compared to similar mobile detector systems presently available on the market. Perhaps the most promising He-3 replacement systems are $\text{LiLa}(\text{Br},\text{Cl})_6:\text{Ce}$ (CLLBC) and $\text{Cs}_2\text{LiYCl}_6:\text{Ce}$ (CLYC) based. The relatively new elpasolite scintillators are sensitive to dual gamma-ray and neutron and rely on pulse height discriminations to distinguish between events, which has proven difficult when implemented into a larger array, such as one needed in radiation detection in the field [162]. Additionally, the thermal neutron detection efficiency of CLYC:Ce is limited due to the competing neutron absorbers, Cl and Cs.

Furthermore, the sensors have largely been implemented as handheld instruments and have reported lengths as long as 9”, making it difficult to conceal the detectors [163]. The bulky rectangular metal exterior of the detector would be difficult to wear on the body and would face challenges with implementing a directionality feature. Lastly, the detector is rather obvious, whereas a single ARGANOT module closely resembles an external computer charger. The

ARGANOT modules are so inconspicuous that a single module conveniently fit into a common women clutch as shown below in Figure 8-1.

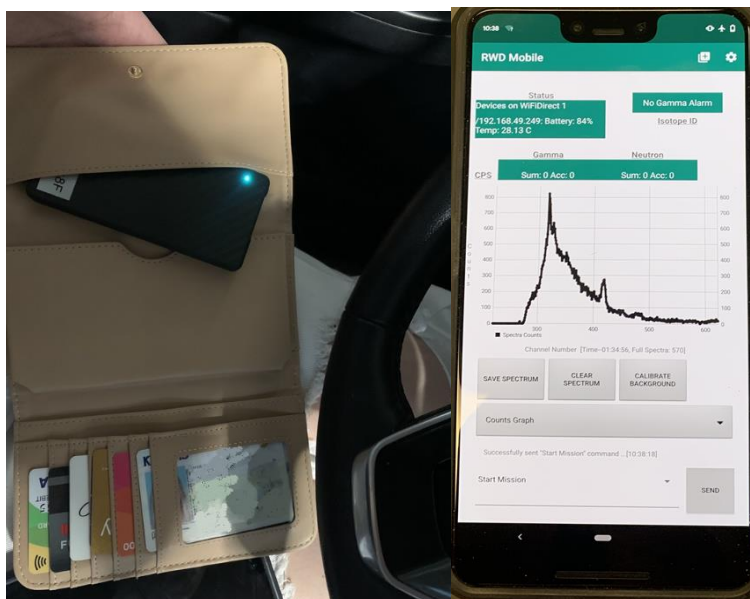


Figure 8-1 A real-life use is shown above in which the module was placed into a woman's clutch and walked through a common hardware store. Throughout the search of the store, all data was communicated to the phone app as shown to the right. The spectrum displayed on the android device was collected when the user was in the granite countertop aisle of the store.

Additionally, ARGANOT modules offer the user the decision to operate with a high efficiency CsI or high-resolution SrI(Eu) detector. As presented in Figure 8-2 below, when both detector types were both exposed 6 inches in distance to the same Cs-137 0.83 μ Ci source over a 15-minute period, the CsI produced spectrum had a significantly greater amount of counts in the full energy peak but at an energy resolution of 8.36%. The SrI ϵ spectrum produced a strong energy resolution of 3.2% at 662 keV.

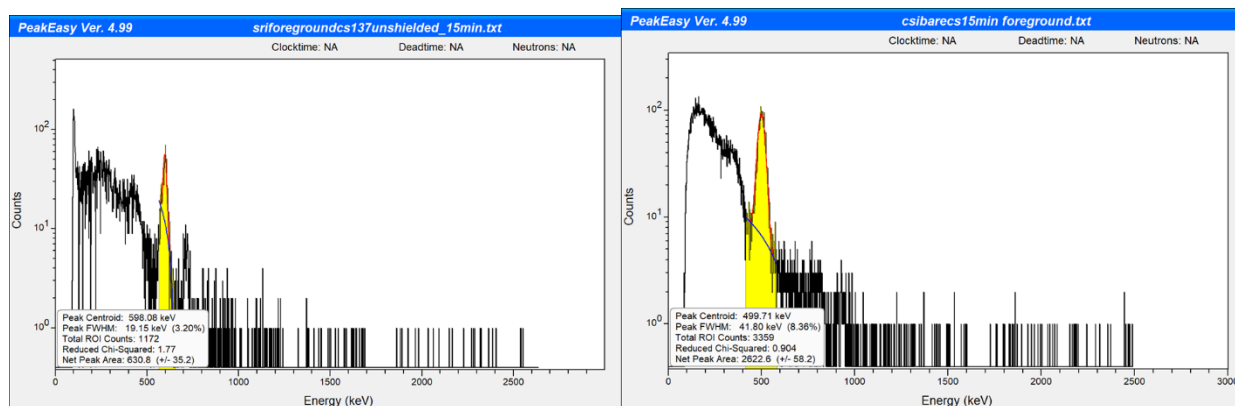


Figure 8-2 A direct comparison of the spectra produced from an ARGANOT module equipped with a CsI (right) and SrI₂(Eu) (left) gamma ray detector is shown. The 0.83 μ Ci source was placed 6 inches from the detector in both scenarios.

When directly compared to other competing wearable detectors such as the handheld D5 RIID radiation detector (see Figure 8-3) produced by Kromek operating with CLLBC, a dual neutron and gamma-ray detector, a single ARGANOT module outperforms the D5 RIID in terms of energy resolution particularly when considering weight. The reported energy resolution at 662 keV for the D5 RIID system weighing 1.5 lbs per instrument was reported to be 3.5%. A single module of the ARGANOT systems has achieved the same degree of energy resolution while weighing only 4 oz.

When comparing the ARGANOT system to the Kromek Backpack system weighing 11 lbs and using a ⁶LiF based neutron detector, the ARGANOT system once again boasts a superior response. As reported by Kromek, the backpack system while able to detect a moderated Cf neutron source with unreported conditions or efficiency, the system has yet to meet ANSI N42.53 § 6.4.3 concerning the device's ability to alert timely to a moving neutron source. The ARGANOT has demonstrated this ability and is able to also perform directionality to the moving source as demonstrated in chapter 7.



Figure 8-3 The Kromek handheld sensor is shown. The sensor is much bulkier and more difficult to operate covertly than the ARGANOT modules. Image retrieved from Kromek [164].

8.2 Conclusion and Summary of work

In today's global security world, the stakes to alert to, search for, and identify radioactive material has increased as both China and Russia grow their nuclear stockpiles and aggressive posture, unrest in the middle east continues, and US borders endure an influx of migrants. Tradition mobile radiation detector systems are equipped with He-3 gas detectors that are often bulky, expensive, and offer limited information to the user. The need for a low-cost, high-efficiency, mobile detection system sensitive to thermal neutron and gamma-rays has persisted. Single-sided MSNDs were developed as an early compact, low-power, inexpensive, mass-producible, and rugged solution offering moderate intrinsic thermal-neutron detection efficiency ($\sim 30\%$) [165]. While the single-sided MSNDs could be stacked to increase the neutron detection efficiency to up to 42%, aligning the devices was unreliable. Dual Sided-MSNDs were then developed and coupled with superior LiF backfill densities and increased trench depth etching processes on 1.5-mm thick silicon wafers, to achieve record-breaking intrinsic thermal-neutron detection efficiencies of $69.3 \pm 1.5\%$. In the present work, an alternative doping scheme known as the pvp-n-dot for DS-MSNDs was implemented. The pvp-n-dot doping profile was designed to bias across the p-type doped trenches and an n-type doped ring on the perimeter of the trenches. The top side of the device is read out through the traditional center bond pad, while the back side of the device is read out through the bottom contact. The devices produced with the

pvp-n-dot doping scheme achieved an average intrinsic thermal neutron detection efficiency of approximately 24%, a high of $64\% \pm 2.5\%$, and GRR of $5.69\text{E-}05$ at a gamma-ray dose rate of 10mR/hr. To accompany the thermal neutron detectors in the mobile radiation detection system known as the ARGANOT (Area Radiation Gamma and Neutron Origin Telemetry), $\text{SrI}_2(\text{Eu})$ and later $\text{CsI}(\text{Tl})$ scintillator using SiPMs were used to alert to gamma-rays and supply the user with gamma-ray spectrums. To ensure good energy resolution of the spectrum regardless of environmental conditions, a temperature-correcting algorithm was implemented for each module to maintain the energy calibration. Using Gamma-ray Detector Response and Analysis Software (GADRAS), directionality, source identification, and activity estimation were implemented.

The ARGANOT underwent extensive testing as a single module and as a full system worn in a garment on and off a phantom body. Results indicate that the ARGANOT optimal performance was best achieved when worn by a user but could alert to sources when operating in a standalone configuration or when flown on an Unmanned Aerial Vehicle.

The cost-effective ARGANOT system presents many benefits to a user such as its high efficiency, low profile, and discrete appearance; the ability to operate as a stand-alone module in a passive monitoring situation or as a directionality capable vest; and the ability to instantly provide the user information about the environment through tools such as GADRAS.

Future models of the ARGANOT may incorporate fast neutron detectors to allow the user sensitivity to fast neutron sources when sufficient moderation is not available. Additionally, the ARGANOT system may be able to take advantage of the directional-dependent kinematics of elastic scattering from HDPE-coated detectors [166].

Appendix A

Fast Neutron MSND Simulation Data

<i>Trench Depth 200um</i>									
Pitch (cm)	0.002								
Trench Width %	0.1								
LLD (MeV)	0.3	0.8	1.3	0.3	0.8	1.3	0.3	0.8	1.3
Energy (MeV)									
0.4	0.006%	0.000%	0.000%	0.006%	0.000%	0.000%	0.002%	0.000%	0.000%
0.6	0.024%	0.000%	0.008%	0.029%	0.000%	0.000%	0.010%	0.000%	0.000%
1.7	0.041%	0.025%	0.008%	0.138%	0.054%	0.004%	0.045%	0.008%	0.001%
2.4	0.037%	0.028%	0.018%	0.144%	0.090%	0.028%	0.072%	0.015%	0.008%
3.4	0.032%	0.027%	0.021%	0.135%	0.101%	0.061%	0.111%	0.025%	0.018%
4.9	0.034%	0.027%	0.020%	0.122%	0.098%	0.071%	0.129%	0.041%	0.031%
7.0	0.121%	0.117%	0.110%	0.175%	0.158%	0.136%	0.176%	0.113%	0.098%
10.0	0.125%	0.119%	0.114%	0.157%	0.142%	0.127%	0.149%	0.108%	0.094%
Pitch (cm)	0.005								
Trench Width %	0.1								
LLD (MeV)	0.3	0.8	1.3	0.3	0.8	1.3	0.3	0.8	1.3
Energy (MeV)									
0.4	0.002%	0.000%	0.000%	0.003%	0.000%	0.000%	0.003%	0.000%	0.000%
0.6	0.012%	0.000%	0.000%	0.012%	0.000%	0.000%	0.013%	0.000%	0.002%
1.7	0.034%	0.011%	0.007%	0.084%	0.046%	0.009%	0.071%	0.010%	0.002%
2.4	0.034%	0.025%	0.017%	0.104%	0.078%	0.035%	0.088%	0.019%	0.008%
3.4	0.031%	0.026%	0.020%	0.109%	0.092%	0.060%	0.112%	0.030%	0.019%
4.9	0.033%	0.027%	0.020%	0.108%	0.094%	0.070%	0.127%	0.046%	0.032%
7.0	0.121%	0.117%	0.110%	0.167%	0.155%	0.135%	0.175%	0.116%	0.099%
10.0	0.124%	0.119%	0.114%	0.152%	0.140%	0.126%	0.148%	0.110%	0.095%
Pitch (cm)	0.01								
Trench Width %	0.1								
LLD (MeV)	0.3	0.8	1.3	0.3	0.8	1.3	0.3	0.8	1.3
Energy (MeV)									
0.4	0.001%	0.000%	0.000%	0.001%	0.000%	0.000%	0.001%	0.000%	0.000%
0.6	0.006%	0.000%	0.000%	0.006%	0.000%	0.000%	0.007%	0.000%	0.000%
1.7	0.028%	0.017%	0.005%	0.047%	0.025%	0.006%	0.051%	0.015%	0.002%
2.4	0.029%	0.022%	0.014%	0.070%	0.051%	0.029%	0.078%	0.027%	0.010%
3.4	0.027%	0.023%	0.018%	0.084%	0.071%	0.055%	0.101%	0.039%	0.021%
4.9	0.031%	0.025%	0.019%	0.090%	0.079%	0.065%	0.117%	0.054%	0.034%
7.0	0.119%	0.115%	0.108%	0.156%	0.146%	0.131%	0.168%	0.119%	0.100%
10.0	0.123%	0.118%	0.113%	0.144%	0.134%	0.123%	0.145%	0.111%	0.096%

Trench Depth 400um

Pitch	0.002								
Trench Width %	0.1			0.5			0.9		
LLD (MeV)	0.3	0.8	1.3	0.3	0.8	1.3	0.3	0.8	1.3
Energy (MeV)									
0.4	0.012%	0.000%	0.000%	0.012%	0.000%	0.000%	0.004%	0.000%	0.000%
0.6	0.048%	0.000%	0.010%	0.058%	0.000%	0.000%	0.019%	0.000%	0.000%
1.7	0.080%	0.050%	0.016%	0.275%	0.107%	0.008%	0.082%	0.010%	0.002%
2.4	0.073%	0.056%	0.036%	0.286%	0.180%	0.054%	0.135%	0.019%	0.008%
3.4	0.064%	0.053%	0.042%	0.268%	0.205%	0.124%	0.223%	0.030%	0.019%
4.9	0.059%	0.050%	0.040%	0.237%	0.197%	0.147%	0.263%	0.049%	0.033%
7.0	0.134%	0.128%	0.119%	0.254%	0.226%	0.189%	0.278%	0.118%	0.077%
10.0	0.135%	0.128%	0.123%	0.212%	0.193%	0.169%	0.232%	0.128%	0.087%
Pitch	0.005								
Trench Width %	0.1								
LLD (MeV)	0.3	0.8	1.3	0.3	0.8	1.3	0.3	0.8	1.3
Energy (MeV)									
0.4	0.005%	0.000%	0.000%	0.005%	0.000%	0.000%	0.005%	0.000%	0.000%
0.6	0.023%	0.000%	0.000%	0.023%	0.000%	0.000%	0.024%	0.000%	0.000%
1.7	0.068%	0.042%	0.015%	0.165%	0.091%	0.018%	0.136%	0.015%	0.002%
2.4	0.066%	0.051%	0.034%	0.207%	0.157%	0.069%	0.167%	0.028%	0.010%
3.4	0.059%	0.050%	0.040%	0.217%	0.185%	0.121%	0.218%	0.042%	0.021%
4.9	0.057%	0.048%	0.039%	0.206%	0.185%	0.144%	0.252%	0.064%	0.036%
7.0	0.132%	0.127%	0.118%	0.235%	0.217%	0.184%	0.269%	0.124%	0.080%
10.0	0.133%	0.128%	0.122%	0.201%	0.187%	0.166%	0.227%	0.131%	0.089%
Pitch	0.01								
Trench Width %	0.1								
LLD (MeV)	0.3	0.8	1.3	0.3	0.8	1.3	0.3	0.8	1.3
Energy (MeV)									
0.4	0.002%	0.000%	0.000%	0.002%	0.000%	0.000%	0.003%	0.000%	0.000%
0.6	0.011%	0.000%	0.000%	0.012%	0.000%	0.000%	0.013%	0.000%	0.000%
1.7	0.055%	0.033%	0.010%	0.090%	0.047%	0.011%	0.094%	0.025%	0.003%
2.4	0.057%	0.044%	0.029%	0.136%	0.100%	0.057%	0.146%	0.043%	0.013%
3.4	0.054%	0.046%	0.037%	0.165%	0.141%	0.110%	0.192%	0.061%	0.026%
4.9	0.053%	0.046%	0.037%	0.172%	0.155%	0.133%	0.224%	0.083%	0.042%
7.0	0.130%	0.125%	0.117%	0.213%	0.198%	0.176%	0.251%	0.133%	0.086%
10.0	0.132%	0.127%	0.121%	0.187%	0.174%	0.158%	0.216%	0.135%	0.093%

Appendix B

SrI split crystal

C This code simulates the cracked SrI response in MCNP

C

=====
=

C -----CELL CARDS-----

C

=====
=

C SrI Detector

10 3 -3.667 -10 IMP:P=1 IMP:E=0 \$ crystal 1

12 3 -3.667 -12 IMP:P=1 IMP:E=0 \$ crystal 2

11 1 -0.001205 -11 10 12 IMP:P=1 IMP:E=0 \$ inside the cask

13 4 -2.33 -13 11 10 12 IMP:P=1 IMP:E=1 \$ SiPM 1

14 4 -2.33 -14 11 10 12 13 IMP:P=1 IMP:E=1 \$ SiPM 2

C Environment

30 1 -0.001205 -30 10 12 11 13 14 IMP:P=1 IMP:E=0 \$ Air outside detector

31 0 30 IMP:P=0 IMP:E=0 \$ Killzone

C

=====
=

C -----SURFACE CARDS-----

C

=====
=

C ----- SrI Detector -----

C SrI crystal

10 RCC 0 0 0.0000 0 0 1.2 3 \$ SrI crystal 1

11 RCC 0 0 1.2 0 0 1.21 3 \$killzone crack

12 RCC 0 0 1.21 0 0 1.27 3 \$ SrI crystal 2

C -----SiPM-----

13 RCC 0 0 1.29 0 0 1.25 3 \$SiPM1

14 RCC 0 0 0 0 0 -0.07 3 \$SiPM2

C ----- Environent-----

C Kill zone

30 SO 110 \$ Killzone

C

=====

=

C -----DATA CARDS-----

C

=====

=

C

=====

=

C PHYSICS/CUT OFF

C

=====

=

MODE P E

nps 1E10

=====

=

C MATERIALS

C

=====

=

C --- Materials (weight fraction)---

M1 \$ Air, rho = 0.001205 (Gas)

6000 -0.000124

7014 -0.755268

8016 -0.231781

18040 -0.012827

M2 \$ Aluminum 6061-O, rho = 2.70. (Metal)

12000 0.011162

13027 0.977325

14000 0.005796

22000 0.000499

24000 0.001017

25055 0.000435

26000 0.001987

29000 0.001174

30000 0.000606

M3 \$ NaI sub in for SrI, RHO = 3.66

11000 -0.153373 \$ Na

53000 -0.846627 \$ Iodine

M4 \$ Si, RHO=2.33

14000 -1.000000

M5 \$Lead,RHO = 11.35 (Metal)"CLEAN" Pb

82000 -1.000000

C

=====

=

C SOURCE - Point Isotropic 661.7 keV gamma-ray source (¹³⁷ Cs)

C

=====

=

SDEF POS 20 20 0.775 ERG=0.6617 PAR=2

C SDEF ERG=1.00 PAR=2 POS=d5

C

=====

=

C TALLY

C

=====

=

C --- SPECTRUM -----

F8:E (10)(12) \$(13)(14)

E8 0 2048i 3

FT8 GEB -0.0077604 0.0976772 0.0156470 \$ NaI

C --- EFFICIENCY -----

F18:E (13)(14) \$ (10)(12)

E18 0 0.050 3

Python Set Flight Alert

This code helped plot and set the alerting level for the drone light.

Robyn Hutchins, KSU, 2023

```
#####  
#####  
  
## import pandas as pd  
  
## import scipy.stats as stats  
  
## import matplotlib.pyplot as plt  
  
## import numpy as np  
  
#  
  
## # Read the Excel file and specify the sheet names  
  
## file_path = "5012023flightdata.xlsx"  
  
## sheet_name_background = "background @10ft, 7mph"  
  
## sheet_name_foreground = "Fast (10mph), 20ft"  
  
#  
  
## # Read the data for background  
  
## df_background = pd.read_excel(file_path, sheet_name=sheet_name_background)  
  
## df_background = df_background.dropna(subset=['N']) # Drop rows with NaN values in "M"  
column  
  
## background_values = df_background['N']  
  
#  
  
## # Calculate the CDF for background  
  
## cdfBG = stats.cumfreq(background_values, numbins=len(background_values))  
  
#  
  
## # Read the data for foreground (cts)  
  
## df_foreground = pd.read_excel(file_path, sheet_name=sheet_name_foreground)  
  
## cts_values = df_foreground['N'][100:] # Select values from position 100 onwards  
  
#  
  
## # Set the FAR value  
  
## FAR = 1 / (8 * 3600)  
  
#
```

```

## # Plot the foreground M column
## plt.plot(cts_values, label='Foreground N')
#
## # Find interval value
## ind = np.argmax(cdfBG.cumcount / len(background_values) >= (1 - FAR))
#
## # Find the threshold value
## Threshold = cts_values[ind]
#
## # Highlight data points above the threshold in red
## plt.scatter(np.arange(len(cts_values)), cts_values, c='black', label='Data Points')
## plt.scatter(np.arange(len(cts_values))[cts_values > Threshold], cts_values[cts_values >
Threshold], c='red', label='Above Threshold')
#
## # Set labels and title
## plt.xlabel('Time (seconds)')
## plt.ylabel('Counts')
## plt.title('Gamma Alerting Over Time of Flight \n 3Ft, 5mph')
#
## # Display the legend with threshold value
## legend_text = f"Threshold: {Threshold:.2f}"
## plt.legend(['Foreground', 'Below Alert Threshold', 'Above Alert Threshold', legend_text])
#
## # Display the plot
## plt.show()
#
## print(f"The threshold value is: {Threshold}")
#
#####
#####

```

```

#import pandas as pd
#import scipy.stats as stats
#import matplotlib.pyplot as plt
#import numpy as np
#
## Read the Excel file and specify the sheet names
#file_path = "5012023flightdata.xlsx"
#sheet_name_background = "background @3ft, 5mph"
#sheet_name_foreground = "3Ft,5mph"
#
## Read the data for background
#df_background = pd.read_excel(file_path, sheet_name=sheet_name_background)
#df_background = df_background.dropna(subset=['N']) # Drop rows with NaN values in "M"
column
#background_values = df_background['N']
#
## Calculate the CDF for background
#cdfBG = stats.cumfreq(background_values, numbins=len(background_values))
#
## Read the data for foreground (cts)
#df_foreground = pd.read_excel(file_path, sheet_name=sheet_name_foreground)
#cts_values = df_foreground['N']
#
## Set the FAR value
#FAR = 1 / (8 * 3600)
#
## Calculate the threshold value based on FAR
#threshold_quantile = np.quantile(background_values, 1 - FAR)
#
## Plot the foreground M column

```

```

plt.plot(cts_values, label='Foreground N')
#
## Plot the threshold line
plt.axhline(y=threshold_quantile, color='r', linestyle='--', label='Threshold')
#
## Highlight data points above the threshold in red
plt.scatter(np.arange(len(cts_values)), cts_values, c='black', label='Data Points')
plt.scatter(np.arange(len(cts_values))[cts_values > threshold_quantile], cts_values[cts_values >
threshold_quantile], c='lightgray', label='Not Above Threshold')
#
## Set labels and title
plt.xlabel('Time (seconds)')
plt.ylabel('Counts')
plt.title('Neutron Alerting Over Time of Flight \n 3Ft, 5mph')
#
## Display the legend with threshold value
legend_text = f"Above Threshold: {threshold_quantile:.2f} counts"
plt.legend(['Foreground', 'Threshold', 'Not Above Threshold', legend_text])
#
## Display the plot
plt.show()
#
#print(f"The threshold value is: {threshold_quantile}")
#####visalization#####
import pandas as pd
import scipy.stats as stats
import matplotlib.pyplot as plt
import numpy as np
#
## Read the Excel file and specify the sheet names

```

```

#file_path = "5012023flightdata.xlsx"
#sheet_name_background = "Background over Fast fly loc"
#sheet_name_foreground = "Fast (10mph), 20ft"
#
## Read the data for background
#df_background = pd.read_excel(file_path, sheet_name=sheet_name_background)
#df_background = df_background.dropna(subset=['N']) # Drop rows with NaN values in "M"
column
#background_values = df_background['N']
#
## Filter out values greater than 500
#background_values = background_values[background_values <= 500]
#
## Calculate the CDF for background
#cdfBG = stats.cumfreq(background_values, numbins=len(background_values))
#
## Read the data for foreground (cts)
#df_foreground = pd.read_excel(file_path, sheet_name=sheet_name_foreground)
#cts_values = df_foreground['N']
#
## Filter out values greater than 500
#cts_values = cts_values[cts_values <= 500]
#
## Set the FAR value
#FAR = 1 / (8 * 3600)
#
## Calculate the threshold value based on FAR
#threshold_quantile = np.quantile(background_values, 1 - FAR)
#
## Plot the foreground M column

```

```

#plt.plot(cts_values, label='Foreground N')
#
## Plot the threshold line
#plt.axhline(y=threshold_quantile, color='r', linestyle='--', label='Threshold')
#
## Highlight data points above the threshold in red
#plt.scatter(np.arange(len(cts_values)), cts_values, c='black', label='Data Points')
#plt.scatter(np.arange(len(cts_values))[cts_values > threshold_quantile], cts_values[cts_values >
threshold_quantile], c='lightgray', label='Not Above Threshold')
#
## Set labels and title
#plt.xlabel('Time (seconds)')
#plt.ylabel('Counts')
#plt.title('Neutron Alerting Over Time of Flight \n 3Ft, 5mph')
#
## Display the legend with threshold value
#legend_text = f"Above Threshold: {threshold_quantile:.2f} counts"
#plt.legend(['Foreground', 'Threshold', 'Not Above Threshold', legend_text])
#
## Display the plot
#plt.show()
#
#
# # Plot CDF of foreground and background
##plt.figure(figsize=(10, 5))
##plt.plot(cdf_background.lowerlimit + np.linspace(0, cdf_background.binsize *
cdf_background.cumcount.size, cdf_background.cumcount.size), cdf_background.cumcount /
cdf_background.cumcount.max(), '-o', label='Background')
##plt.plot(cdf_foreground.lowerlimit + np.linspace(0, cdf_foreground.binsize *
cdf_foreground.cumcount.size, cdf_foreground.cumcount.size), cdf_foreground.cumcount /
cdf_foreground.cumcount.max(), '-o', label='Foreground')

```

```

##plt.axvline(x=threshold, color='r', linestyle='-o-', label='Threshold')
## # Set labels and title
##plt.xlabel('Counts')
##plt.ylabel('Detection Probability')
##plt.title('CDF of Foreground and Background Counts')
##plt.legend()
#
# # Display the plot
#plt.show()
#print(Threshold)
#####excludes neutron noise#####
import pandas as pd
import scipy.stats as stats
import matplotlib.pyplot as plt
import numpy as np

# Read the Excel file and specify the sheet names
file_path = "5012023flightdata.xlsx"
sheet_name_background = "Background over Fast fly loc"
sheet_name_foreground = "Fast (10mph), 20ft"

# Read the data for background
df_background = pd.read_excel(file_path, sheet_name=sheet_name_background)
df_background = df_background.dropna(subset=['M']) # Drop rows with NaN values in "M"
column
background_values = df_background['M']

# Filter out values greater than 500
background_values = background_values[background_values <= 500]

```

```

# Calculate the CDF for background
cdfBG = stats.cumfreq(background_values, numbins=len(background_values))

# Read the data for foreground (cts)
df_foreground = pd.read_excel(file_path, sheet_name=sheet_name_foreground)
cts_values = df_foreground['M']

# Filter out values greater than 500
cts_values = cts_values[cts_values <= 500]

# Set the FAR value
FAR = 1 / (18000 * 3600)

# Calculate the threshold value based on FAR
threshold_quantile = np.quantile(background_values, 1 - FAR)

# Plot the foreground M column
plt.plot(cts_values, label='Foreground M')

# Plot the threshold line
plt.axhline(y=threshold_quantile, color='r', linestyle='--', label='Threshold')

# Highlight data points above the threshold in red
plt.scatter(np.arange(len(cts_values)), cts_values, c='black', label='Data Points')
plt.scatter(np.arange(len(cts_values))[cts_values > threshold_quantile], cts_values[cts_values >
threshold_quantile], c='lightgray', label='Not Above Threshold')

# Set labels and title
plt.xlabel('Time (deciseconds)')
plt.ylabel('Counts')

```

```
plt.title('Gamma Alerting Over Time of Flight \n Direct Flyover')

# Display the legend with threshold value
legend_text = f"Above Threshold: {threshold_quantile:.2f} counts"
plt.legend(['Foreground', 'Threshold', 'Not Above Threshold', legend_text])

# Display the plot
plt.show()
```

Fast Neutron Simulations

```
#####  
#####  
  
# The python script will calculate isotropic th. n. det. eff.  
  
# 1. Define MSND Geometry  
# 2. Define Source Geometry (thermal, adjustable incidence angle)  
# 3. Generate MCNP input file DSMSND_ang.i  
# 4. Run MCNP simulation  
# 5. Read MCNP output, extract efficiency, write to DSMSND_ang_result.txt  
# 6. Repeat  
  
#  
# Robyn Hutchins, KSU 2022  
# Based on Taylor Ochs, Radiation Detection Technologies, 2021  
# Based on Taylor Ochs, Kansas State University, 2018  
# Based on Ryan G. Fronk, KSU, 2014  
  
#####  
#####  
  
import os    # Allows for terminal commands  
import shutil # Allows for file/directory manipulation  
import numpy as np  
import pathlib  
import re  
  
  
os.system('cls')    # not sure what this does  
print("Running")  
  
  
#####  
#####  
  
# Functions
```

```
#####
####

# def inputfile(pitch,trench_width,wafer_thickness,trench_depth,Backfill_Density,
LLD,E,i_file_name): # Write the Input File

# # Writes the input file based on what is sent to it"
# file = ["LiF MSND-ST Isotropic Response"]
# file.append("C
=====")
# file.append("C -----CELL CARD-----")
# file.append("C
=====")
# file.append("C DSMSND")
# file.append(" 10 2 -2.329 -10 u=1 $ Front Sidewall ")
# file.append(" 11 1 -1.11 10 u=1 $ Front Trench, PN ")
# file.append(" 12 0 -11 lat=1 fill=1 u=2 $ Front Unit Cell ")
# file.append(" 13 0 -12 fill=2 trcl=1 $ Front Diode ")
# file.append(" 14 2 -2.329 -13 12 trcl=1 $ Middle of Diode ")
# file.append("C World")
# file.append(" 100 0 #13 #14 -100 $ VOID IN WORLD ")
# file.append(" 101 0 100 $ OUTSIDE WORLD ")
# file.append(" ")
# file.append("C
=====")
# file.append("C -----SURFACE CARDS-----")
# file.append("C
=====")
# file.append("C DSMSND")
# # Inject data into input file.
# file.append(" 10 RPP 0.0000 {0} -0.5 0.5 -{1} {1} $ Front Sidewall" \
# .format( str(trench_depth), "%.5f" % ((pitch - trench_width)/2.0) ) )
# file.append(" 11 RPP 0.0000 {0} -0.5 0.5 -{1} {1} $ Front Unit Cell" \
```

```

#         .format( str(trench_depth), "%.5f" % (pitch/2.0) ) )
# file.append(" 12 RPP 0.0000 {0} -0.5 0.5 -0.5 0.5 $ Front Diode" \
#         .format( str(trench_depth) ) )
# file.append(" 13 RPP {0} {1} -0.5 0.5 -0.5 0.5 $ Middle of Diode" \
#         .format( str(trench_depth), "%.5f" % (wafer_thickness) ) )
# file.append(" 14 RPP -0.3255 0.3255 -0.5005 0.5005 -0.5005 0.5005 $ f1")
# # Done injecting data into input file.
# file.append("C
# file.append(" 100 so 5 $ universe
# file.append("
# file.append("C
=====
=)
# file.append("C -----DATA CARDS-----
# file.append("C
=====
=)
# file.append("C
# file.append("C
=====
=)
# file.append("C PHYSICS/CUT OFF
# file.append("C
=====
=)
# file.append("TR1 -{0} 0 0 1 0 0 0 1 0 0 0 1 1" \
#         .format( str(wafer_thickness/2.0)))
# file.append("MODE N H
# file.append("IMP:N 1 5R 0
# file.append("IMP:H 1 4R 0 0
# file.append("nps 1E10
# file.append("PHYS:N 6J 1 $ NCIA, 5 use NCIA not Table
# file.append("CUT:N 2J 0

```

```

# file.append("CUT:H J 0.01 0 $ 10keV Energy Cut-off          ")
# file.append("C
=====
=")
# file.append("C MATERIALS                                     ")
# file.append("C
=====
=")
# file.append("M1 $ Paralyne N, C16H16 RHO=1.11              ")
# file.append("    1001.70c 16  $ 1-H                          ")
# file.append("    6000.70c 16  $ C                                ")
# file.append("C                                                ")
# file.append("M2 $ NATURAL SILICON, RHO = 2.3290                      ")
# file.append("    14028.70c -9.2223E-01 $ 28-Si                          ")
# file.append("    14029.70c -4.6850E-02 $ 29-Si                          ")
# file.append("    14030.70c -3.0920E-02 $ 30-Si                          ")
# file.append("C                                                ")
# file.append("C
=====
=")
# file.append("C SOURCE - COLL. 0.5cm DISK 0.0253eV ALONG THE X-AXIS.
")
# file.append("C
=====
=")
# file.append("SDEF POS= 0 0 0 PAR= N ERG={0} RAD=d1 CEL=100"
#         .format(str(E)))
# file.append("SI1 0 4                $ 4 cm radius (8cm diameter )    ")
# file.append("SP1 -21 2              $ spherical sample                ")
# file.append("C
=====
=")
# file.append("C TALLY                                     ")

```

```

# file.append("C
=====
=")
# file.append("C --- SPECTRUM -----")
# file.append("F8:H (10 14)")
# file.append("E8 0 500i 10")
# file.append("C --- EFFICIENCY -----")
# file.append("F18:H (10 14)")
# file.append("E18 0 {0} 5")
# .format(str(LLD)))
# file.append("F1:N 13012.2 14013.1 12.5 12.6 10.3 10.4 T")
# file.append("*C1 90 0 T")
## file.append("F2:N 13012.2 14013.1 12.5 12.6 10.3 10.4 T")
## file.append("*C2 90 0 T")
## file.append("SD2 1 1 0.0625 0.0625 0.0625 0.0625 2.25")
# file.append("PRDMP 1E10 1E10")
# file.append("C
=====
=")
# file.append("C Problem Stuff")
# file.append("C
=====
=")
# file.append("TF18 J J J J J 3 J $ Select bin I want most precision on ")
# file.append("C STOP F18 0.01 $ Runs until 0.01 relative error achieved ")

# MCNP_input_file = open(i_file_name,'w') # Creates DSMSND.i input file irl
# for line in file: # Writes to input file.
#     MCNP_input_file.write("%s \n" % line)
# MCNP_input_file.close() # All done!

# return;

```

```
#####
#####

# def mcnp(i_file_name, o_file_name): # Runs MCNP

#   bat_file=["@ REM - Sets the MCNP Environment Variables for use in standard"]
#   bat_file.append("@ REM - command prompt. This will help with MatLab functionality.")
#   bat_file.append("@ set MCNPPATH=D:\\C drive\\MY_MCNP\\MCNP_CODE\\bin")
#change this PART ROBYN dont change bin
#   bat_file.append("@ PATH %MCNPPATH%;%PATH%")
#   bat_file.append("@ set DISPLAY=:0.0")
#   bat_file.append("@ set DATAPATH=D:\\C drive\\MY_MCNP\\MCNP_DATA") #Robyn
also chnage this...not mcnp data
#   bat_file.append("")
#   bat_file.append("mcnp6.exe i={0} o={1} " \
#                   .format(str(i_file_name),str(o_file_name)))
#   bat_file.append("erase run*")

#   BFile = open('MCNP.bat', 'w')
#   for line in bat_file:
#       BFile.write("%s\n" % line)
#   BFile.close()

#   # Runs batch file
#   os.system('MCNP.bat')

#   return;

#
#####
#####

def output(td,uc,twuc,pf,LLD,E,o_file_name):
```

```

# Output traditional Eff (counts per source particle)
# Calculate and output iso_eff

output = open(o_file_name) # Opens output file.


# Grab Data
temp_f8 = []
temp_f1 = []
parsing_f8 = False
parsing_f1 = False
for line in output:    # Starts going through the output file.

    if parsing_f8:
        temp_f8.append(line)
    if line.startswith( '1tally      18' ): # Looking for the efficiency tally.
        parsing_f8 = True
    elif line.startswith( '      total' ):
        parsing_f8 = False

    if parsing_f1:
        temp_f1.append(line)
    if line.startswith( '1tally      1      n' ): # Looking for the efficiency tally.
        parsing_f1 = True
    elif line.startswith( ' =====' ):
        parsing_f1 = False

output.close() # Finished with output file.

```

```

# Parse F8 tally
for line in temp_f8:
    if line.startswith(' 5.0000E+00'):
        f8_eff = float(line.split()[1])
        f8_rel_err = float(line.split()[2])

# Parse F1 tally
union = False
angle = False
for line in temp_f1:
    if re.match(' \d',line) and union == True and angle == True:
        f1_tot = float(line.split()[0])
        f1_rel_err = float(line.split()[1])
    if line.startswith(' surface union total'):
        union = True
    if union == True and line.startswith(' angle bin: 180.0      to 0.90000E+02 degrees'):
        angle = True
    else:
        angle = False

# Calculate iso Eff and output
isotropic_eff = f8_eff/f1_tot
output_line = '%s %s %s %s %s %s %s %s %s %s %s %s \n'%
(td,uc,twuc,pf,LLD,E,f8_eff,f8_rel_err,f1_tot,f1_rel_err,isotropic_eff)
with open("Efficiency.o", "a") as myfile:
    myfile.write(output_line)

```

```

#
#####
#####

# DSMSND Dimensions  Units in cm
waferthickness_val = [0.0625]
Trench_depth = [0.0200,0.0400] # Trench Depth
Pitch_val = [0.0020,0.0050,0.0100] # Unit Cell
Width[0.0020,0.0025,0.0030,0.0035,0.0040,0.0050,0.0100]
Trench_width_percentage_val = [0.10,0.50,0.90] #[0.10,0.20,0.30,0.40,0.50,0.60,0.70,0.80,0.90]
Backfill_packingfraction_val = [1.0]
diode_width = 1.0
diode_length = 1.0
LLD_List = [0.3,0.8,1.3] #[0.3,0.5,0.8,1.0,1.3]

# setting up file names
i_file_name = 'ST_Offset_PN_SS_iso.i' # You can change this
o_file_name = 'ST_Offset_PN_SS_iso.o' # You can change this too

#efficiency = open('Efficiency.o', 'a')
#efficiency.write("\n Trench_Depth  Pitch  Trench_Width/UC  LiF_packFrac  LLD
F8_Efficiency  Rel_Error  F1_Efficiency  Rel_Error  Iso_efficiency \n")

# Incidence Angles setup
ang_res = 2.5 # Units degrees
ang_min = 0.; # Minimum angle in degrees
ang_max = 1. # Maximum angle in degrees
ang_radius = 5. # Radius in cm of source arc path
pi = 3.14

```

```
B_Density_Cryst = 2.206 # g/cm3 for 95% enriched
LiF_Density_Cryst = 2.546 # g/cm3 for 95% enriched
```

```
Energies = np.logspace(-0.4,1,10)
```

```
print(Energies)
```

```
# f = open("C:/Users/rhutc/OneDrive/Documents/work/Simulations/neutron2/startcodes.txt",
"w")
```

```
# for E in Energies:
```

```
#     for LLD in LLD_List:
```

```
#         for td in Trench_depth:
```

```
#             for uc in Pitch_val:
```

```
#                 for tw_uc_ratio in Trench_width_percentage_val:
```

```
#                     tw = uc * tw_uc_ratio
```

```
#                     for pf in Backfill_packingfraction_val:
```

```
#                         rho = pf * LiF_Density_Cryst
```

```
#                 # IF statement here for LLD and energy
```

```
#                 if E < LLD:
```

```
#                     pass
```

```
#                 # Use functions
```

```
#                 inputfile(uc,tw,waferthickness_val[0],td,rho,LLD,E,i_file_name) # Make input
file
```

```
#                 mcnp(i_file_name,o_file_name) # Runs MCNP simulation
```

```
#                 output(td,uc,tw_uc_ratio,pf,LLD,E,o_file_name) # Pull out the stuff i want
and make output files
```

```

#                #print(td,uc,tw_uc_ratio,pf,LLD,i_file_name)
#                # Check to see if output folder exists, if not make one
#                pathlib.Path('./iput').mkdir(parents=True, exist_ok=True) #Robyn change this
back to ouputs

#                # Move and rename output file
# #                shutil.move(o_file_name, './output/'+o_file_name
+'_{0}_{1}_{2}_{3}_{4}_{5}.o' \
# #                .format(str(td*1e3),str(uc*1e3),str(tw_uc_ratio),str(pf),str(LLD),str(E)))

#                shutil.move(i_file_name,'./iput/'+i_file_name + '_{0}_{1}_{2}_{3}_{4}_{5}.i'
#Robyn change this back to outputs too
#                .format(str(td*1e3),str(uc*1e3),str(tw_uc_ratio),str(pf),str(LLD),str(E)))
#                #print('mpirun --mca btl ^openib
#                /homes/bahadori/MY_MCNP/MCNP_CODE/MCNP620/bin/mcnp6.mpi i='+i_file_name +
#                '_{0}_{1}_{2}_{3}_{4}_{5}.inp' #Robyn change this back to outputs too
#                #.format(str(td*1e3),str(uc*1e3),str(tw_uc_ratio),str(pf),str(LLD),str(E)) + '
#                o='+i_file_name + '_{0}_{1}_{2}_{3}_{4}_{5}'
#                .format(str(td*1e3),str(uc*1e3),str(tw_uc_ratio),str(pf),str(LLD),str(E))+'.out')
#                f.write(('mpirun --mca btl ^openib
#                //homes/bahadori/MY_MCNP/MCNP_CODE/MCNP620/bin/mcnp6.mpi i='+i_file_name +
#                '_{0}_{1}_{2}_{3}_{4}_{5}.i' #Robyn change this back to outputs too
#                .format(str(td*1e3),str(uc*1e3),str(tw_uc_ratio),str(pf),str(LLD),str(E)) + '
#                o='+i_file_name + '_{0}_{1}_{2}_{3}_{4}_{5}'
#                .format(str(td*1e3),str(uc*1e3),str(tw_uc_ratio),str(pf),str(LLD),str(E))+ '.o' \
#                + ' RUNTPE='+i_file_name + '_{0}_{1}_{2}_{3}_{4}_{5}'
#                .format(str(td*1e3),str(uc*1e3),str(tw_uc_ratio),str(pf),str(LLD),str(E)) + '\n\n'))
# #spectrum.save('Spectrum_Out.xls')
# #efficiency.close() # Close efficinecy file
# f.close()
# print("Fin.")

```

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