

Characterization of a Proof-of-Concept Imager to Safeguard Spent Nuclear Fuel Using Passive Fast Neutron Emission Tomography

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Abstract

A primary part of nuclear nonproliferation and nuclear security principles involves verification of spent fuel to ensure that no material diversion has occurred. As a result, various verification technologies have been developed to detect when irradiated fuel pins have gone outside of regulatory control. One proposed method of spent nuclear fuel verification includes using a fast neutron emission tomography system for spent nuclear fuel. This system may allow for imaging of individual fuel pins and quantifications of neutron source strength. A proof-of-concept fast neutron emission tomography system for verification of spent nuclear fuel was constructed at Oak Ridge National Laboratory (ORNL). An analytic neutron response model used in maximum likelihood expectation maximization (MLEM) reconstructions was created based on the system's geometry and results from Monte Carlo radiation transport simulations. In order to ensure accurate reconstructions of spent fuel, a series of measurements were performed to characterize the as-built imager. Results from this data were incorporated into the analytic neutron response model to match the analytic responses to the real response of the system. Characterization measurements were made to improve neutron detector efficiency values and estimate collimator penetration of neutrons and collimator slit scattering values. Using the data collected, these values were incorporated into the analytic response model and new reconstructions of mock fuel pins were created.

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Chapter 1. Introduction

1.1 Safeguards Context Background

From its inception, the International Atomic Energy Agency (IAEA) was created with the purpose of promoting peaceful applications of atomic energy and with the power to verify the peaceful utilization [1]. As early as 1946, discussions took place amongst the United Nations security council powers and Canada to establish a committee which would control the information and technology surrounding nuclear applications. By 1953 when President Eisenhower spoke in front of the UN General Assembly, he proposed an organization which would promote the peaceful applications of nuclear power by sharing nuclear material and nuclear energy technology with other countries. However, he also stipulated that this international organization would need to have enough power to control how to prevent any country from transitioning to non-peaceful applications [1]. By 1968, the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) specified that participating countries would work with the IAEA to prevent the diversion of nuclear technology from peaceful purposes to nuclear weapons or nuclear explosive devices [2]. The signing states agreed to enact safeguards measures which allow the IAEA to establish proof that nuclear technologies are not being used for weapons purposes. This includes preventing material created during the nuclear fuel cycle from being diverted for weapons purposes. The nuclear fuel cycle is the process which extends from when uranium is mined from the earth, through the processing and enrichment of nuclear fuel, until the uranium is removed from a nuclear reactor as spent nuclear fuel and put into long term storage. At every stage along this process, the IAEA needs to be sure that no material or technologies were diverted for weapons purposes.

Since the ratification of the NPT, there have been many discussions as to how the IAEA can prove nuclear materials and technology are only being used for peaceful purposes [2]. One of the most basic methods of implementing nuclear safeguards comes from material control and accountability. By taking into account the initial quantity of the material present in a nuclear facility and monitoring the quantity of the same, material diversions can be detected as missing material [2,3]. This method of the verification of nuclear material quantities can be applied to a number of types of nuclear facilities at different stages during the nuclear fuel cycle, including

reactor facilities and spent nuclear fuel storage sites where spent nuclear fuel (SNF) is handled and stored [3]. Because SNF holds the potential to be diverted for non-peaceful purposes, it is particularly important to establish verification methods at these locations to track the amount of nuclear material present. As a result, the IAEA has prioritized developing methods for the verification of nuclear material within spent nuclear fuel assemblies [3].

When determining the standards for verification of SNF, i.e. how confident the IAEA is that a material is or is not present, there are two main situations in which the IAEA assumes material diversion of spent nuclear fuel may occur: gross defects and partial defects [3]. Gross defects are the diversion of full spent fuel assemblies, while partial defects involve the diversion of single or multiple fuel pins within an assembly. The IAEA's verification standard is that they must be able to detect partial defects where more than 50% of the spent fuel pins have been diverted [3]. However, during operation of nuclear power plants, operators are permitted to replace single fuel pins during reactor refueling outages if the fuel pins experience leakage. Fuel pin leakage happens when the casing around spent nuclear fuel ruptures and small quantities of nuclear fuel escape the casing and move into the reactor coolant [4]. This does not require replacement of a whole new spent fuel assembly for the ruptured fuel pin, rather just the affected pin can be replaced. IAEA's verification standard of only detecting the diversion of 50% or more of the fuel pins in a fuel assembly makes it difficult to determine if a fuel rod or several fuel rods have been removed and diverted for proliferation purposes, or if a fuel rod has been replaced in a refueling outage due to leakage issues [3, 5].

Due to the radioactive nature of SNF, radiation detection methods have been used to non-destructively determine the presence of nuclear material in spent nuclear fuel. However, initial detection methods tended to look at the fuel assembly as a whole and check for gross defects as opposed to checking for partial defects [5]. For this reason, creating a verification tool with single fuel pin resolution as well as source age determination has value in detecting individual fuel pin diversion. When focusing on detection of individual spent fuel pins, tomographic methods of verification have been proposed and developed since 1991 [5]. These methods potentially allow for image reconstruction of spent nuclear fuel with information about the strength of the radioactive sources within the fuel rod.

In addition to meeting the IAEA's safeguards verification standards, tomographic

verification systems also support the general mission of the IAEA to develop surveillance techniques for nuclear materials and to support the pursuit of continuity of knowledge over nuclear materials and facilities [3,5]. Continuity of knowledge (CoK) is an important goal of safeguards nuclear material accountability, which says that it should always be known where nuclear material is and what sort of activities it is going through [6]. In order to ensure that CoK is accurate, tools must be developed to examine nuclear materials. Tomographic verification methods for spent nuclear fuel verification support the continuity of knowledge by allowing inspectors to verify the quantity and strength of nuclear material [6].

1.2 Computed Tomography

Computed tomography (CT) has been researched for medical applications since the 1960's with x-rays being the most common type of CT [7]. In all CT, cross sectional images are created using projected data taken from many angles around the object of interest. In x-ray tomography an external source is used to produce the radiation signatures detected. Using an external source of radiation signatures and measuring how much attenuation occurs through the object of interest is referred to as transmission computed tomography [8]. The other type of tomography is emission computed tomography where the signatures detected originate from the object of interest. Emission computed tomography has been a research topic for decades now, with Positron Emission Tomography (PET) first taking place in 1977 [7, 8].

In gamma emission CT an object is divided into separate lines of response, as demonstrated by the parallel lines in front of each rotating gamma camera in Figure 1, and multiple views, illustrated by the different rotating gamma camera placements [8]. The data from all lines of response at one view is a 1-D projection at different gamma camera angles. Using the measurements from each of the projection angles, an emission computed tomography algorithm, or ECT algorithm, can combine the projections to form a reconstruction image of the object of interest [8]. The same principles of emission tomography apply whether gammas, neutrons or another radiation signature is used. For passive neutron emission tomography, the lines of response are isolated using collimation and the detectors used in the system are neutron detectors as opposed to gamma cameras [7, 8].

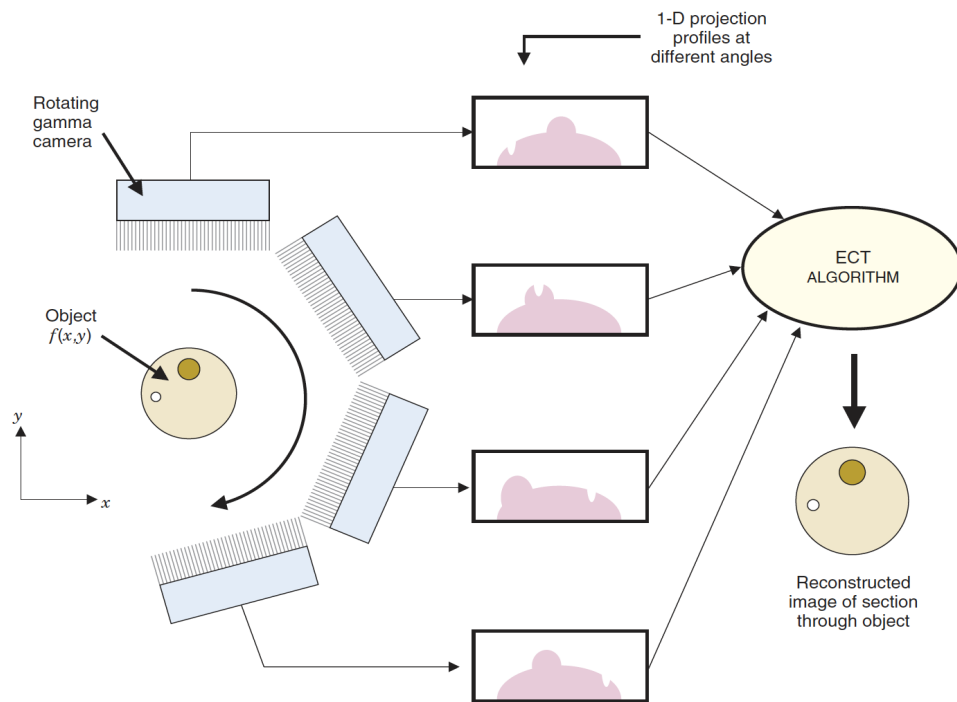


Figure 1. Emission computed tomography flowchart [8].

The projection data gathered in this system can be represented by a sinogram. A sinogram compiles this data by mapping the projections into a graph. Figure 2 shows a mapping of the projections into a sinogram [8]. Each measurement in the system is taken by collecting data for each line of response with the detector at a particular position. By creating a graph where the line of response and the detector position are the x and y axes, respectively, a color spectrum can be used to represent the relative intensity detected at each line of response. All projections can be added to the sinogram.

1.3 Iterative Image Reconstruction Algorithm

Computed tomography image reconstruction algorithms use the projection data coming from a detection system and combine it to create image reconstructions. The most basic image reconstruction algorithm divides all signal data in one perspective equally amongst the pixels examined by that perspective [8]. This same method is repeated for all perspectives. Figure 3 shows the accumulations of these perspectives into an image [8]. By combining enough perspectives, the location with higher source intensity becomes more obvious as these partial divisions pile up [8]. This represents the source distribution, but basic back projection attributes some source strength to pixels where no source is present. In cases where there are multiple sources, where the source is spread out, or where the contrast between the source and the background is low, this initial back projection does not create sufficiently accurate images to pinpoint the source location [8].

In cases where simple back projections are insufficient to create an accurate image, iterative methods can be used to improve the image quality. Figure 4 diagrams the logic behind an iterative conversion method [8]. An initial image estimate is made and then forward projected. In cases where simple back projections are insufficient to create an accurate image, iterative methods can be used to improve the image quality. An initial image estimate is made and then forward projected. Forward projection is the inverse of back projection. Instead of starting with a series of projections and then layering to form an image, forward projection starts with an image and then works to calculate projections that would layer to form the image. After forward projecting the image estimate, the sinogram created from measured data can be compared to the

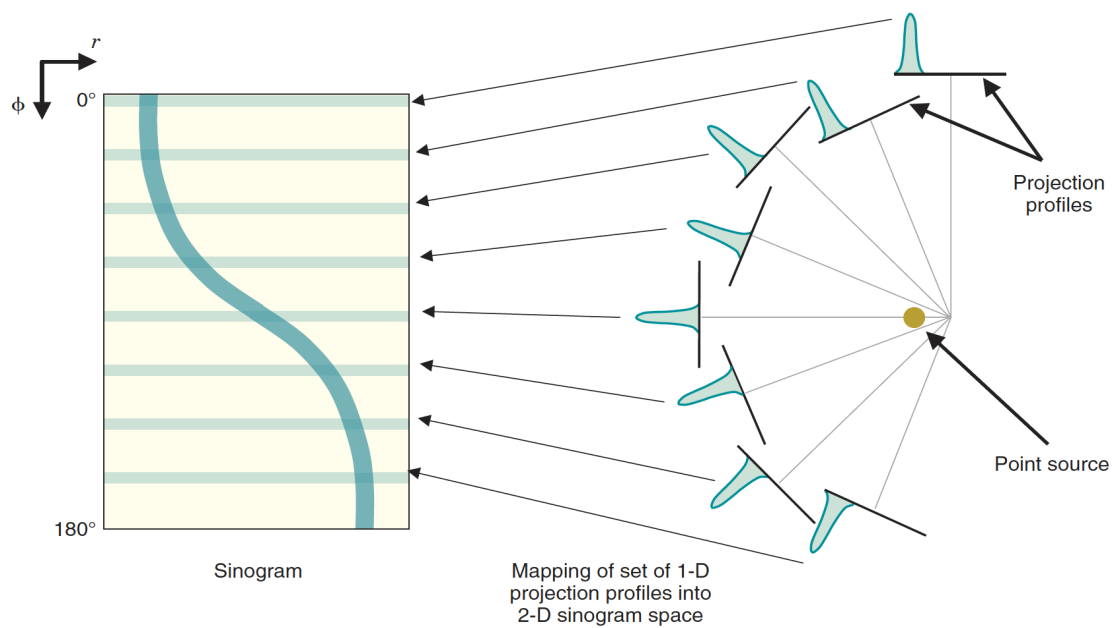


Figure 2. Creation of a sinogram from a set of projections [8].

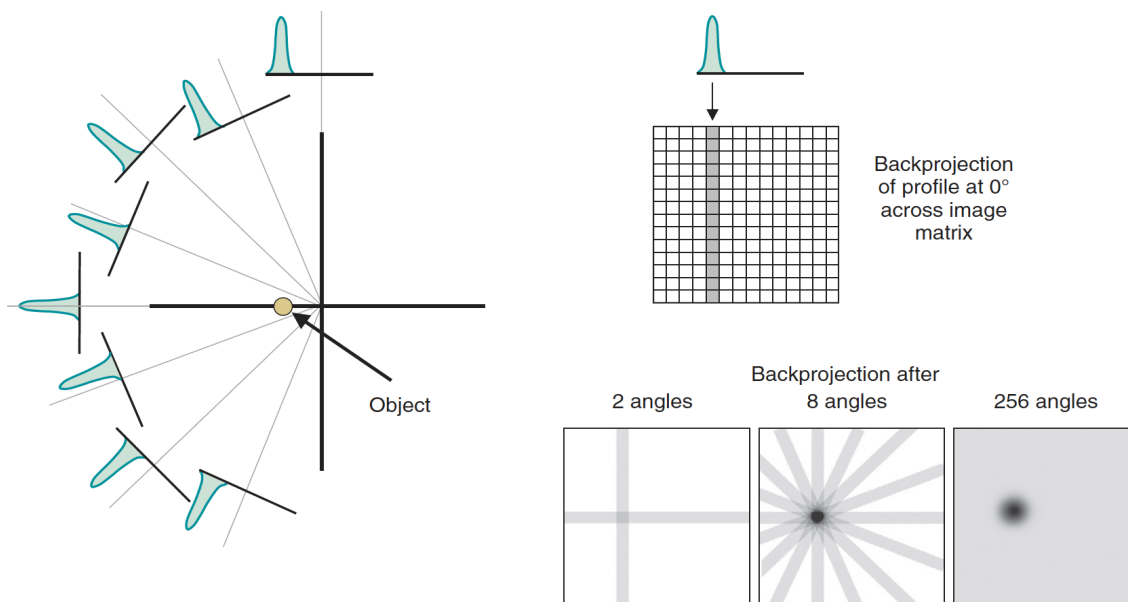


Figure 3. Creation of a back projection image from 1-D projections [8].

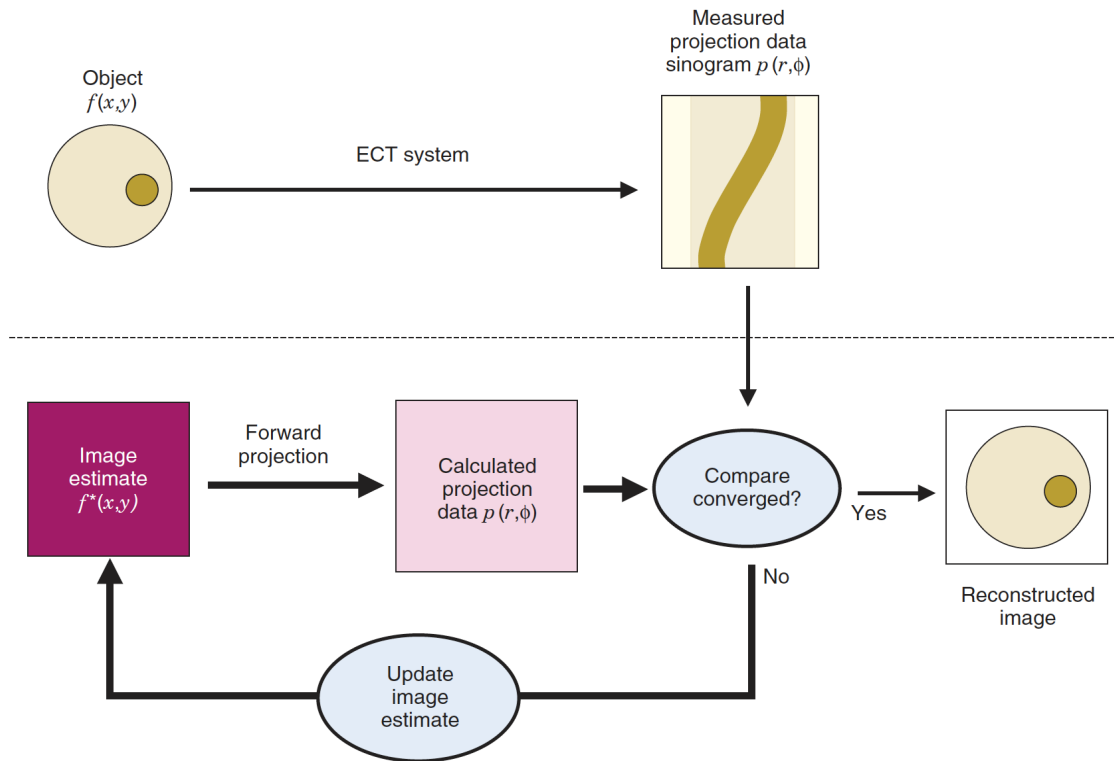


Figure 4. Logic flow chart for iterative reconstructions [8].

calculated sinogram created by the forward projection process. The difference between these can then be used to modify the forward projection values. The updated forward projection can be back projected to create an updated image estimate. This process then repeats, iteratively changing the image estimate forward projection until the image converges. Convergence is when there is no difference between the estimated forward projection and the measured sinogram or until the difference is small enough that the reconstructed image adequately represents the original object. Determining whether the reconstructed image adequately represents the original object is dependent on the purpose for creating the image and can be different for every iterative reconstruction process.

1.4 Improvements to Iterative Image Reconstruction Algorithm

Basic iterative reconstruction methods may take many iterations to converge. Iterative reconstruction methods can be modified in ways to force the reconstructions to converge faster while still maintaining accuracy. One common way of speeding up the iterative convergence is to incorporate prior information about the system, such as knowledge of the object shape [8,9]. An algorithm may also “force” the reconstructed image to be nonnegative by setting any pixels which go negative to zero. Both methods can be used in an algorithm to increase the convergence rate [8].

Statistical considerations can also be applied to iterative methods to determine the most likely source distribution in the object and converge images faster. Maximum likelihood expectation maximization (MLEM) is an iterative algorithm which assigns a higher statistical weighting factor to portions of projections with greater detected signal as opposed to equally distributing the signal across all pixels viewed in the projection like a simple back projection would [8, 9]. MLEM algorithms have been widely used in medical tomography applications for decades and are a well-established method of creating measurement-informed reconstruction images [9].

Mathematically, the initial iteration can be represented as:

$$Ax = y \quad \text{(Equation 1)}$$

$$y_i = \sum_j a_{ij}x_j \quad \text{(Equation 2)}$$

where x is an image whose pixel values, represented by x_{ij} correspond to source intensity at a particular location. The matrix A of a_{ij} represents the statistical probability that radiation emitted from the i -th pixel will be detected in the j -th projection. The variable y is the measured projection data at a particular location, i.e., the measured sinogram. Each successive iteration can be represented as:

$$x_j^{(k+1)} = x_j^{(k)} \frac{1}{\sum_i a_{ij}} \sum_i a_{ij} \left(\frac{y_i}{\sum_h a_{ih} x_h^{(k)}} \right) \quad (\text{Equation 3})$$

where k is the current iteration [9]. The term in parentheses in the denominator represents a summation over all image pixels which must be computed prior to the summation over the j -th projection. To avoid confusion, it is given an index of h instead of i .

1.5 Fork Detector Irradiated Fuel Measurement

One of the most used detector designs for safeguarding spent nuclear fuel is the fork detector [10, 11]. The fork detector was built in response to an IAEA call for a transportable detector which could verify spent fuel assemblies and ensure that they have not been modified since their removal from the reactor core [10, 11]. The fork detector has a U-shaped detector head which is designed to fit around a spent fuel assembly so that the assembly is between the tines of the detector. An example of a fork detector is pictured in Figure 5 [12]. Fork detectors are designed with two fission chambers and an ionization chamber in each of the arms of the fork. One of the fission chambers is embedded in a high-density polyethylene moderator surrounded by a layer of cadmium, which allows it to detect epithermal and fast neutrons [13]. The other fission chamber is unmoderated allowing for the detection of thermal neutrons. The fission chambers used together allows for the passive detection of total neutrons coming from spent nuclear fuel. The ionization chamber detects the gross gamma flux measurement coming off spent nuclear fuel [12]. Figure 6 shows a diagram of the fork detector around a PWR fuel assembly as well as a cross section of one of the tines in the detector, showing the placement of the fission chambers and ionization chamber [13].

The ratio of the neutron counts from spent nuclear fuel to the gamma counts raised to the 4th power is a constant value that is independent of burnup [14]. The fork detector relies on this



Figure 5. A fork detector used to verify spent nuclear fuel [12].

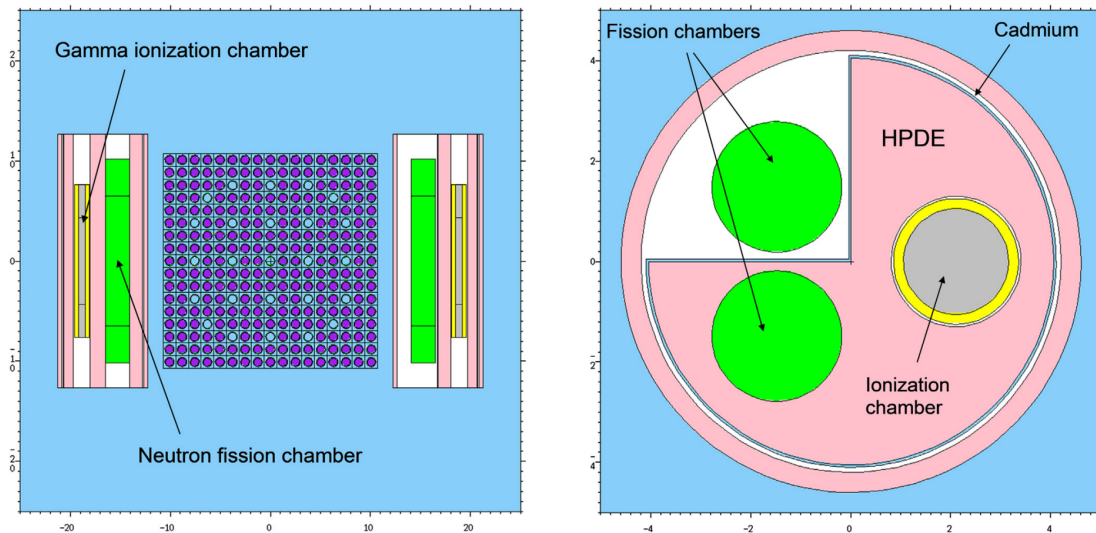


Figure 6. Diagram of the fork detector around a PWR fuel assembly (left) and a cross section of one of the detector tines (right) [13].

constant to verify spent nuclear fuel because the neutron and gamma counts from untampered spent nuclear fuel will result in that constant value. If fuel has been tampered with and more than 50% of spent nuclear fuel rods have been removed from the assembly, the ratio of gamma and neutron counts will no longer match the appropriate constant [14]. Measurements made with the fork detector evaluate the ratio for various fuel configurations and were proven to be able to tell if 50% of fuel pins were removed [10, 11, 14]. While the detector is easy to deploy and use, it is still limited in its sensitivity. The fork detector cannot tell if only a few spent fuel pins in an assembly have been removed [13, 14]. In order to get better sensitivity than 50% of the fuel pins missing, additional verification tools were requested by the IAEA.

1.6 Muon Tomography for Spent Nuclear Fuel

One currently researched tomography-based method for imaging spent nuclear fuel is muon tomography. Muon tomography uses cosmic ray muons to which are charged particles created either directly or indirectly in the Earth's atmosphere through interactions between atoms in the atmosphere and cosmic radiation [15]. As these muons pass through an object, interaction between the muons and the atoms in the object cause the muons to scatter in the object and lose energy. Since muons are charged particles, their primary interactions through materials come from Coulomb scattering. The scattering angle between the original muon direction and new direction after interaction with an atom in the object of interest is proportional to the Z value of the atoms in the object. By measuring muon flux before interacting with the object and measuring the muon flux at scattered angles after passing through an object of interest, the Z value of the object can be calculated. Muon tomography uses these principles to create images of objects from these calculated Z values [15]. These muons intersect the earth at a rate of 1 per cm^2 per min [15, 16]. Muons approaching earth's surface do so at an average energy of 4 GeV making them highly penetrating [15].

A detector system for imaging spent fuel casks with cosmic ray muons was designed and constructed at Los Alamos National Laboratory (LANL) [16]. Two drift tube tracking detectors were designed to be placed on either side of the object of interest in order to record incoming and outgoing muon trajectories. The individual drift tubes were 1.2 m long aluminum tubes with an

outer diameter of 5.08 cm and an inner diameter of 5 cm. The tubes were filled with a 47.5% Ar, 42.5% CF₄, 7.5% C₂H₆, 2.5% He gas mixture with a gold-plated tungsten anode wire running down the center of each tube [16]. When muons enter each tube, they ionize electrons in the fill gas. The ionized electrons travel to the anode wire where they are multiplied and collected as signal. The individual tubes can measure the radial distance from the anode wire where muon interactions occurred [16]. In order to resolve the ambiguity with the radial direction, a second layer of tubes are needed. Each layer of tubes is offset by 90 degrees from one another, as shown in Figure 7, to allow for multi-dimensional particle tracking [16]. Each detector consists of 288 drift tubes arranged in alternating layers of tube orientation. Figure 8 shows the two drift tube tracking detectors around a spent fuel cask [16]. The drift tube detectors must be placed at different elevations along the spent fuel cask due to the directionality of cosmic ray muons. The flux of cosmic ray muons is described by $\cos^2\theta_z$, where θ_z is the angle from the zenith. In order to detect a higher muons flux, one of the detectors was placed 1.2 m higher than the other detector [16]. This set up was tested over 200 hours and proven to create images which would be able to detect missing whole spent nuclear fuel assemblies within a cask [15, 16, 17].

Additional algorithms for muon tomography have been implemented to try to create full spent fuel cask images with better resolution or which require less detection time [17, 18, 19, 20]. However, there have been attempts to develop individual fuel pin imaging with cosmic ray muons [15]. Simulations have been performed which indicate that it would be possible to use muon tomography to detect individual spent fuel pins within a LWR assembly within a spent fuel cask [15]. If the planar detectors could be placed directly above and below a spent fuel cask, simulation indicated that individual spent fuel pins could be imaged given measurement times on the order of 2-3 years [15]. This has not been tested using measured data yet.

1.7 Passive Gamma Emission Tomography

Passive gamma emission tomography (PGET) has been under research and development for imaging spent fuel assemblies with single fuel pin resolution since the 1980s [5]. The PGET method relies primarily on the detection of high energy gamma rays from spent fuel fission products. The most important gamma-ray emissions for spent fuel after a cooling time of 5 years

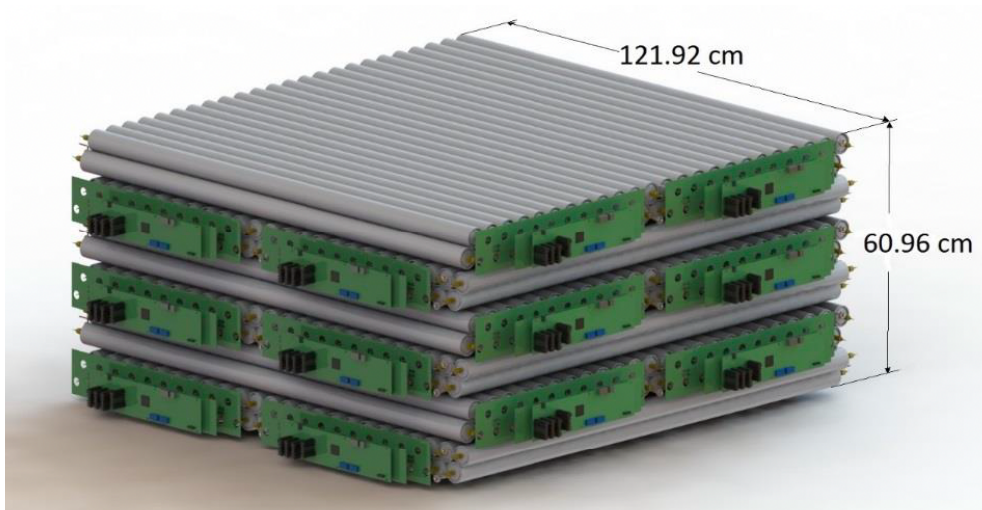


Figure 7. Diagram of a single planar drift tube tracking detector for muon tomography of spent nuclear fuel [16].



Figure 8. Image of two planar drift tube detectors for muon tomography surrounding a spent fuel cask [16].

are listed in Table 1 [21]. The Cs and Eu content in spent fuel is considered the most useful for PGET [21]. Figure 9 shows the contribution to the total gamma ray activity for various gamma emitters in spent fuel at a burnup of 12.18 GWd/tU [22]. Beyond 2 years of cooling time, the ^{137}Cs gamma ray begins to dominate the gamma activity, meaning that the ^{137}Cs gamma ray becomes roughly proportional to burnup and other short lived isotopes decay away [21]. This allows for easier detection and isolation of the 662 keV gamma ray coming from ^{137}Cs which allows for better image resolution [22]. Theoretically, PGET can be performed beyond one year of cooling; however, high gamma doses, like those coming off spent nuclear fuel which has been cooled for less than 2 years, have the potential to damage gamma ray detectors and introduce artifacts into the image reconstruction. In practice it has been recommended to use PGET after at least 2 years of cooling time.

A prototype for PGET was designed and implemented in response to an IAEA call for a tomographic verification method [23]. Figure 10 (left) shows a diagram of the prototype detector around a fuel assembly, while Figure 10 (right) shows the cross section of the detectors [23]. The prototype detector consists of 2 detector arrays which rotate 360 degrees around the spent fuel assembly. Each of the detector arrays consists of a multi slit collimator made of tungsten followed by 104 CdZnTe crystal detectors [23]. The detectors rotate in steps around the assembly, sampling every 4 mm. The positions of the arrays are 2 mm apart such that the total system resolution is 2 mm [23, 24]. The PGET detector is intended to be deployed underwater in a spent fuel pool so the array heads as well as all electronics are sealed and waterproof. The inspection volume is 32.5 cm across, which is large enough to image a PWR 17x17 fuel assembly [23].

Despite having the largest contribution to the activity of spent fuel, ^{137}Cs does not have the highest gamma energy, which is a drawback for tomography due to self-shielding in spent fuel [25, 26]. From the center of a fuel assembly to the outer edge, 662 keV ^{137}Cs gamma rays can experience large attenuation by a factor on the order of 100 [27]. The attenuation coefficient of a 662 keV gamma through nuclear fuel is 1.324 cm^{-1} meaning that the average distance that a 662 keV gamma can travel through the fuel assembly without an interaction is 0.755 cm [27]. Collimators are used in a PGET system to attenuate low energy gamma rays and define the lines of response for the gamma detector used. PGET has been used to image spent nuclear fuel

Table 1. Most important gamma-ray emissions from fission products [21].

Isotope	Half-life (yrs)	Gamma Ray energy (keV)	Branching Ratio (%)
^{134}Cs	2.07	604.7	97.6
		795.8	85.4
^{137}Cs	30.08	661.1	85.1
^{144}Ce - ^{144}Pr	0.78	1489.2	0.3
		2185.6	0.7
^{154}Eu	8.60	996.3	10.3
		1004.8	17.4
		1274.4	35.5

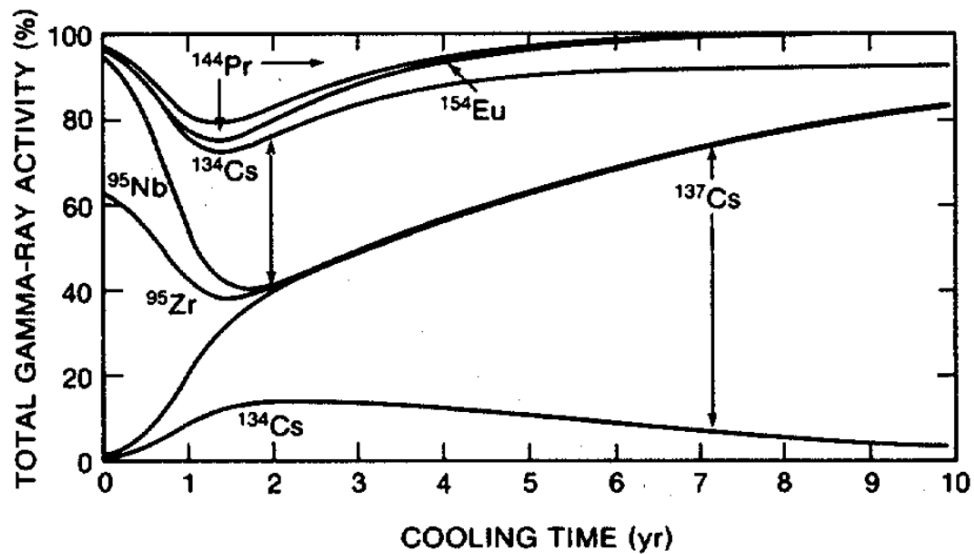


Figure 9. Isotope contributions to the total gamma ray activity over 10 years [22].

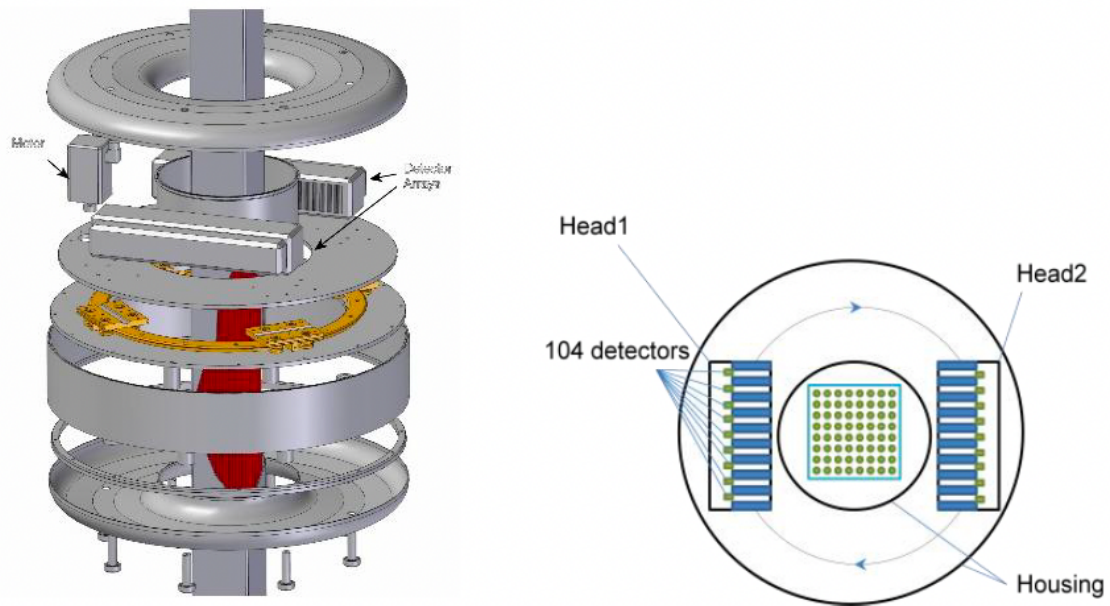


Figure 10. Diagram showing the PGET device around a spent fuel assembly (left) as well as a cross section of the device (right) [23]

assemblies with cooling times as long as 24.5 years [21, 27].

In order to compensate for the larger attenuation factor for 662 keV ^{137}Cs , PGET reconstructions began incorporating counts from gamma rays with energies over 700 keV [21]. Higher energy gamma rays have smaller attenuation factors across the width of the fuel assembly and can be used along with the 662 keV ^{137}Cs counts to accurately image the center of fuel assemblies. The 1274.4 keV gamma from ^{154}Eu has an attenuation coefficient of 0.689 cm^{-1} in uranium oxide meaning it can travel 1.452 cm on average without an interaction in the fuel [21]. This is nearly twice as far as the 662 keV ^{137}Cs gamma. In comparison, a standard 17 x 17 PWR fuel assembly is 24 cm by 24 cm in dimension. The drawback to using higher energy gamma rays like ^{154}Eu is the small contribution of these gammas to the total gamma activity. This makes getting sufficient statistics to create a reconstruction a longer process than using a more active gamma emitter. After reconstruction methods for PGET began to include more advanced attenuation responses, such as including additional high energy gamma rays, it was determined that attenuation would no longer be a limiting factor in the use of PGET [21, 22]. PGET has been successfully shown to reconstruct measured data from BWR and VVER-440 spent fuel assemblies while detecting single pin diversions [28, 29, 30]. BWR fuel assemblies typically consist of 4 bundles of fuel rods arranged around a cruciform center control rod with each bundle arranged into a square lattice of fuel rods. The number of fuel rods in each lattice varies between different reactors, but the BWR fuel assembly bundles proven to be imaged successfully with PGET have 4 lattices of either 8 by 8 or 10 by 10 fuel rods [23]. BWR 8 by 8 assemblies typically consist of 63 fuel pins with a fuel pin pitch of 1.628 cm and a bundle pitch of 13.024 cm making them smaller and less dense than other commonly used reactor assembly configurations [31]. The larger space between fuel pins makes BWR 8 by 8 fuel bundles easier to image than more dense bundles. VVER-440 spent fuel assemblies are hexagonal in shape with 126 fuel rods. The total assembly has a pitch of 14.70 cm and the fuel pin pitch is 1.22 cm [32]. The VVER-440 assembly is a very dense fuel pin configuration, but with only 126 fuel rods per bundle, the assembly is smaller [32]. In comparison, the PWR 17 by 17 fuel assembly has a fuel pin pitch of 1.2932 cm and a total assembly pitch of 21.98 cm [31]. The BWR assembly bundle represents an easier fuel assembly to image given the fewer fuel pins and larger pitch. The VVER-440 is more difficult to image given the fuel pin density, but it is still a relatively small

assembly. PWR reactors are the most abundant reactor type in the world but since the 17 by 17 fuel assembly is both quite dense and consists of many more fuel rods, the PWR assembly represents a very difficult assembly to image [33]. PGET has been used to reconstruct simulated PWR 17 by 17 spent fuel assembly data but has not been tested on measured PWR data [27].

1.8 Fast Neutron Emission Tomography for Spent Nuclear Fuel

Spent nuclear fuel has a high neutron emission rate on the order of 10^7 or 10^8 n/s. Many of these neutrons come from transuranic isotopes that are formed during reactor operation. As uranium isotopes capture neutrons, built-up quantities of transuranic materials emit neutrons during their own decay processes. Specifically, the main sources of neutrons in fuel assemblies are ^{244}Cm , ^{242}Cm , ^{240}Pu , ^{238}Pu , and ^{242}Pu . The emission rates of these neutron emitters during a 10-year cooling period of a fuel assembly with a 31.5GWd/tU burnup are shown in Figure 11 [5]. Traditionally, measuring neutrons from spent fuel has meant measuring the ^{239}Pu content in the fuel. However, as demonstrated in Figure 11, the Pu content saturates with increased burnup or exposure. This means that detector systems which use this technique for spent fuel verification would be minimally sensitive to rod replacements made after the first cycle in a reactor.

After approximately 20 months, ^{244}Cm neutron emission dominates all neutron emission through the end of the 10-year observation time. The isotope ^{244}Cm has a half-life of 18.2 years and a neutron production rate of 1.64×10^7 neutrons/s/gram with much of this neutron emission coming from spontaneous fission [5]. This is lower than the 3.35×10^9 gammas/s/gram from ^{137}Cs alone [5]. Unlike the Pu content, ^{244}Cm does not saturate as a function of burnup, so using this as the signature of choice would enable an imager to be sensitive to rod replacements made after the first cycle. For this reason, the neutrons from the spontaneous fission of the ^{244}Cm isotopes built up in the SNF will be the primary neutron signature detected by a fast neutron emission tomography (FNET) system.

While the highly penetrating nature of fast neutrons makes them prone to escaping spent nuclear fuel assemblies, the same nature also makes detecting the fast neutrons difficult. Any tomography system using fast neutrons needs to modulate the neutron flux using some sort of aperture to detect fast neutrons. Previous work at ORNL demonstrated the feasibility of using a

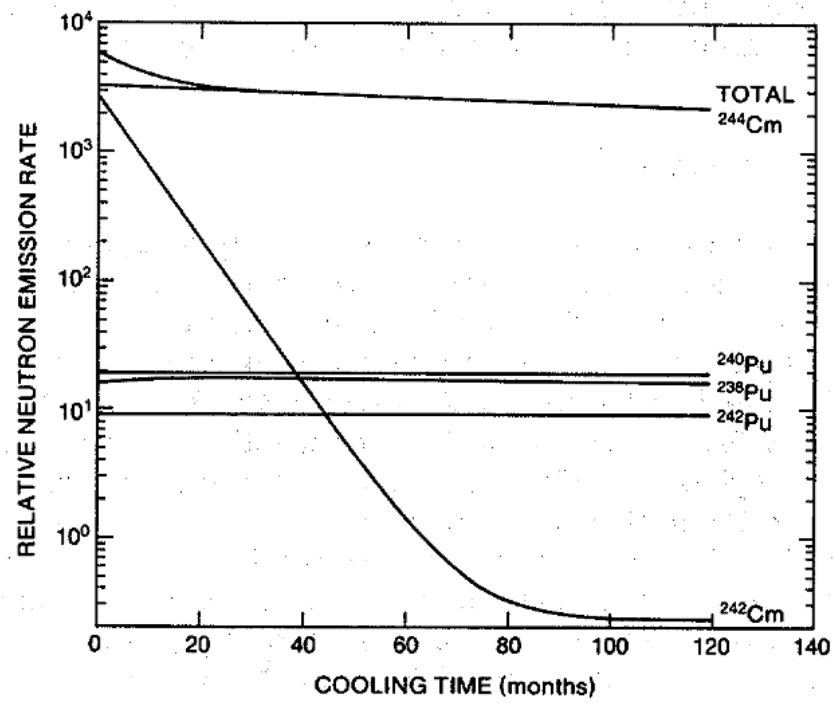


Figure 11. Relative neutron emission rates per isotope from a PWR spent fuel assembly at a burnup of 31.5 GWd/tU over the course of 10 years [5].

fast neutron emission tomography system to generate images which could confirm the presence of spent nuclear fuel [30]. In order to modulate the neutron flux, a two-part collimator was designed which combined radial slits and a pinhole aperture as shown in Figure 12 (left) [30, 34]. The collimator was made of high-density polyethylene. The “thick pinhole” portion of the collimator faces the neutron source and serves to reduce neutron flux which would otherwise interact with the neutron detectors. The radial slit collimator portion centers all the collimator slits on the pinhole opening and ensures that the majority of neutrons detected are coming from the neutron source itself rather than from any scattering which may occur elsewhere in the room or system. The detector system consists of an array of pixelated liquid scintillator imaging detectors which are arranged into five columns, each containing three detectors. These columns are placed in an arc configuration. The constructed prototype imager system is shown in Figure 12 (right) [30, 34]. Initial measurements of five ^{252}Cf sources took place at ORNL. The proof-of-concept imager was successfully used to image the five sources and could resolve the source positions at distances as small as 1.5 cm apart [30].

Following the ^{252}Cf source measurements, the imager was then used to create images of a soup can containing shortened MOX fuel rods [30]. These measurements were performed at the Zero Power Physics Reactor facility at Idaho National Laboratory. In order to image fuel, the imager was modified to shield the detector array from gamma-rays coming from the MOX fuel. The MOX array consisted of 32 fuel rod locations arranged as seen in Figure 13 [30]. Of the 32 positions, 31 of these were filled with 6-inch-long MOX fuel rods which each contained 3.66 g of ^{240}Pu . Each rods produced approximately 3,700 neutrons per second with the neutrons coming primarily from spontaneous fission of Pu. The last position was filled with a depleted uranium rod. The imager system was successfully proven to be able to image the neutron sources within the MOX fuel can and it was possible to determine the location where the depleted uranium rod was located as well. However, the imager design would need to be modified in order to be used on spent fuel. The gamma-ray fields coming from spent fuel assemblies would be too high for the segmented scintillator detector and would overwhelm the detector [30, 34, 35].

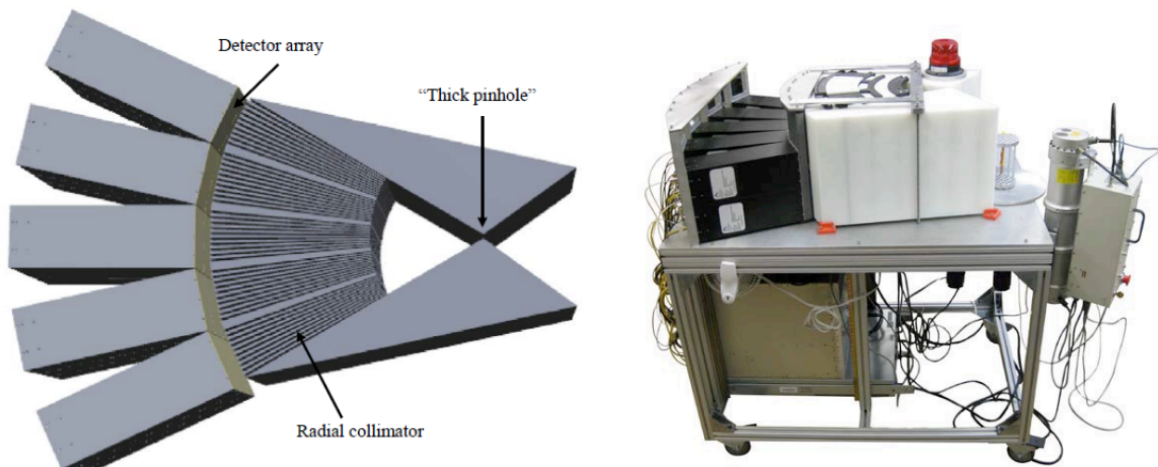


Figure 12. Diagram of a fast neutron imager (left) and the constructed prototype imager (right) [30, 34].



Figure 13. MOX fuel can at INL with 32 available positions for fuel rods [30].

1.9 Fast Neutron Emission Tomography Compared to Other Methods

Comparing FNET to the previously discussed spent nuclear fuel verification detector systems shows the gaps in technology that a FNET system would fill. While the fork detector system is portable, quick, and easy to use, the fork detector system only yields information about whether a spent fuel assembly has been tampered with. The system provides no further information about the neutron emission or gamma emission rates of the spent nuclear fuel, how the fuel assembly has been modified if it has been, or the extent to which it has been modified. Muon tomography has been proven to successfully identify the complete removal of a whole spent fuel assembly in a storage cask. While simulation suggests that muon tomography can be used to image missing individual spent fuel rods in an assembly, this has not been proven to work with measured data yet. Simulation suggests that this process will take years of measurement to generate an image which would identify missing fuel rods. PGET allows for successful imaging of BWR and VVER assemblies using measurements that take several hours to collect, rather than several days. PGET has been demonstrated to detect missing individual spent fuel pins or places where fuel pins have been replaced by dummy rods or fresh fuel rods. However, PGET but does not yield information about allowable fuel rod replacements or burnup for each spent fuel pin individually. Fast neutron emission tomography has the potential to yield reconstructed images of spent nuclear fuel which are sensitive to missing rods as well as rod replacements. Additionally, FNET has the potential to quickly measure spent nuclear fuel assemblies over several hours and perform burnup estimation on individual fuel rods. For these reasons, FNET has the potential to meet the criteria of an imaging modality which would be able to verify spent nuclear fuel in accordance with IAEA verification standards and go beyond these standards to yield more information about spent fuel and support the continuity of knowledge over special nuclear materials. For this reason, this document details the characterization of a proof-of-concept FNET detector system for spent nuclear fuel.

1.10 Statement of Original Contributions

This dissertation follows dissertation work performed by Dr. Anagha Iyengar in her thesis titled, “The Design of an Imager to Safeguard Spent Fuel Using Passive Fast Neutron Emission Tomography” [34]. Using the radiation transport and the analytic code models developed in that

thesis as a basis, this document discusses further work on the development of a fast neutron emission tomography imager for spent nuclear fuel.

Original contributions for this project begin in Section 2.7.6 where simulated MNCP6 radiation transport data was used to fit various constants which define the individual components of the neutron response model for the imager design using a 96-slit collimator. This was the final part of determining an analytic response code for the imager geometry described in Sections 2.8 and 2.9, the design characteristic conclusions from Dr. Iyengar's dissertation work.

Due to outside considerations, the imager model was not able to be constructed according to Dr. Iyengar's design. This work details the changes that were made to the imager during fabrication. In addition to running radiation transport simulations to determine coefficients for the analytic neutron response model as described in the previous work, novel contributions to the project include adapting the radiation transport and analytic models to match the geometric design of the as-built detector system.

Following the fabrication of the 6 detector modules, imager efficiency testing for each individual detector module was performed. In this experiment, the intrinsic, geometric, and absolute efficiencies for each detector module were calculated. In addition, the high voltage and trigger threshold voltages for each detector were determined and a value for detector operation compared.

Other novel contributions include characterization of the as-built imager system using measured data and incorporation of that data into an updated analytic neutron response code. Following individual detector module analysis, the total imager system was constructed onsite at ORNL with the help of moving and rigging experts. Initial measurements were taken in a preliminary measurement campaign which demonstrated the system's ability to detect neutrons and reconstruct measured data with the adapted analytic neutron response code. Further measurements characterizing the imager were later performed to inform the analytic neutron response code and account for changes between the ideal imager and the as-built imager. These characterizing measurements include determining the relative efficiency of the individual detectors, and additional changes to collimator penetration. The document details the measurement campaign plans, execution and data collection. Novel contributions include the incorporation of measured data into the analytic response code for improved response

simulations. Additionally, novel contributions include the measurements of neutron sources at each position in the field of view with and without attenuation rods present and the reconstruction and analysis of these sources.

Chapter 2. NET System Design Prior to This Work

2.1 Initial System Design Requirements

Fast neutrons are highly penetrating. With a mean free path of 1.905 cm in fuel, 1 MeV neutrons readily escape from standard PWR fuel rods which only have a 0.9652 cm diameter [31, 32]. Unfortunately, the long mean free path of fast neutrons in borated high-density polyethylene, 4.630 cm at 1 MeV, also makes fast neutrons difficult to collimate and measure with fine spatial resolution [36]. In addition, it is difficult to measure the neutron source strength compared to the overwhelming gamma ray emissions from spent fuel.

Creating a fast neutron detector which can image spent nuclear fuel means balancing several characteristics: high detection efficiency, high spatial resolution, and the ability to minimize gamma exposure while maintaining a size roughly equivalent to a spent fuel cask. The compact size of the imager is important so as not to disrupt operations at any facility in which verification measurements are taking place.

The neutron source strength, compared to the gamma strength, in spent fuel means that the number of neutrons which are present should be detected with as much efficiency as possible. Ideally, the detectors would record all incident neutrons. More realistically, the detectors should be able to record at least 1/3 of the incident neutrons. Designs which place large detectors close to the fuel and using wide collimator slits would increase the neutron efficiency.

Due to the small sizes of both the fuel pins (0.9652 cm) and the gaps between them (0.328 cm), the system spatial resolution needs to be less than 3 millimeters in order to define and image individual pins [34]. Spatial resolution is how finely the imager can pinpoint the location of a source. A spatial resolution less than 3 mm would need lines of response to be approximately the same width as the fuel pin gap and consequently, collimator slits comparable in size to the fuel pin gap. Increased amounts of neutron moderator define these lines of response but decrease the efficient use of the neutrons.

The gamma-ray dose to the detectors must also be properly managed to avoid gamma pile up in the detectors. Gamma dose can be reduced with dense collimator materials, but these are not very good for neutron moderation and collimation because neutron attenuation in dense materials is low. Additionally, the neutron detectors chosen must be somewhat gamma blind,

which means using a neutron detector where the neutron energy loss happens in a moderator rather than in the active detector volume. The drawback to moderated detectors is that they are larger than non-moderated detectors.

Despite the challenges associated with an NET imager system, previous work determined that the use of a novel collimator concept appears to make it possible to construct an imager having sufficient resolution to identify individual fuel pins while also keeping the imager sufficiently compact, efficient, and radiation resistant to be practical [34]. This novel collimator is referred to as a “parallel-slit ring collimator” [34, 35].

2.2 Neutron Detector Design

The neutron detectors for the imaging system were chosen with gamma-sensitivity in mind. All neutron detectors are gamma sensitive to some degree. Gammas from spent fuel impart energy to electrons in the detector fill gas via Compton scattering interactions. The resulting high energy electron proceeds to ionize more electrons along its path. The electronic pulses caused by these gamma-produced electrons are comparable to the pulses created by neutron interactions with electrons in the fill gas. Due to the higher gamma emissions from SNF than neutron emissions, there is a potential for many more gamma pulses to be recorded than true neutron pulses. If many pulses from both neutron and gamma interactions are registered close together within the resolving time of the detector electronics, they can pileup and result in false counts. The neutron detectors must be chosen to favor neutron absorption and limit gamma absorption. This can be done by changing neutron energies to favor absorption and by choosing materials which are highly neutron absorbing. Table 2 shows the gamma and neutron interaction probabilities for thermal and fast neutrons in common neutron detector materials. Thermal neutrons can be absorbed with a higher probability than gamma-rays. However, for fast neutrons, like those coming from spent nuclear fuel, the magnitude of the gamma-ray interaction probabilities is on the same order as the neutron interaction probabilities.

Due to their commercial availability, the detectors used in this NET system are boron-coated straw detectors embedded in a high-density polyethylene moderator (HDPE). These detectors are commercially available from Proportional Technologies Inc. (PTI). One straw

Table 2. Neutron and gamma-ray interaction probability for common neutron detectors [34].

	Interaction Probability	
Thermal Detectors	Thermal Neutron	1-MeV Gamma-ray
^3He (2.5 cm diam., 4 atm)	0.77	0.0001
Ar (2.5 cm diam., 2 atm)	0.0	0.0005
BF_3 (5 cm diam., 0.66 atm)	0.29	0.0006
Al tube wall (0.8 mm)	0.0	0.014
	Interaction Probability	
Fast Detectors	1-MeV Neutron	1-MeV Gamma-ray
^4He (5 cm diam., 18 atm)	0.01	0.001
Al tube wall (0.8 mm)	0.0	0.014
Scintillator (5.0 cm)	0.78	0.26

detector consists of a 4 mm diameter tube of aluminum lined with a boron-carbide-coated, copper straw along the outside diameter, as shown in Figure 14 [34]. Table 3 shows the main components of the boron carbide straw as well as their dimensions [34, 37]. An anode wire runs through the center of the tube, separated from the straw by a fill gas. Fast neutrons are thermalized in the HDPE moderator and then captured by the ^{10}B in the B_4C layer. The high energy neutrons (0.84 MeV or 1.47 MeV) interact with the boron producing secondary particles via the $^{10}\text{B} + n \rightarrow ^7\text{Li} + \alpha$ reaction. The resulting alpha particle and ^7Li isotopes are detected when they escape the B_4C layer, enter the detector fill-gas, and proceed to ionize the gas atoms as they travel. The straw detector then acts as a proportional counter. The electrons created in the gas drift toward the thin anode wire down the center of the straw and away from the tube wall which acts as a cathode [34, 35]. When the detector threshold is exceeded, a signal is generated.

These detectors exhibit particularly good gamma blindness for a number of reasons. The first reason is the small detector fill gas volume means that there is a limited space in which any particle may interact and create a signal. Figure 15 shows the interaction probability for gamma rays and neutrons in ^{10}B [36]. The probability of gamma interactions in boron is much lower than the ^{10}B interactions with thermal neutrons. Additionally, the detector consists mostly of an inert moderator which decreases the number of gamma ray interactions. As instrumented, the boron straw detectors can tolerate gamma fields up to approximately 500 R/h [34, 37].

A single detector module for this system consists of several of the boron carbide straws arranged in rows then embedded in a HDPE matrix. For this particular system, one way to minimize the detector module's gamma sensitivity is to limit the active volume of the detector. This means choosing a straw arrangement geometry with the least number of straws and largest amount of moderator. The boron straw detectors were arranged such that a full imager would use 2304 of these straws. The final imager design has 12 detector modules, each containing 24 rows of 8 boron straw detectors that are read out together. Figure 16 shows a drawing of the straw detector arrangement of one of these detector modules [34]. All together the detectors consist of 288 rows of straw detectors embedded in HDPE, with each giving a readout in the full detector imager. A single detector module consists of 24 rows of straws in HDPE, an air gap which surrounds the HDPE matrix containing the straws, aluminum walls separating rows of detectors,

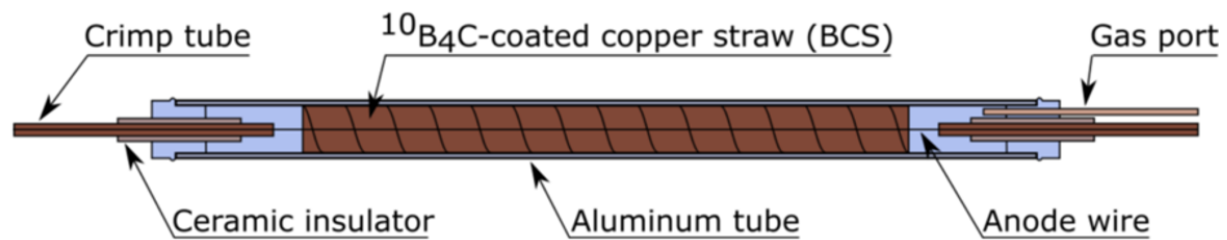


Figure 14. Drawing of a Boron-Coated Straw detector [34].

Table 3. Main components of a boron carbide straw and respective dimensions and material parameters [34].

Component	Material	Radius (cm)	Thickness (cm)	Density (g/cm ³)
Fill gas	90% Ar, 10% CO ₂	0.22150	-	1.20×10 ⁻³
Boron Carbide	B ₄ C with ¹⁰ B enriched to 96%	0.22164	1.4×10 ⁻⁴	2.38
Coating tube	Copper	0.22434	2.7×10 ⁻³	8.96
Detector Wall	Aluminum	0.27666	5.2×10 ⁻²	2.70

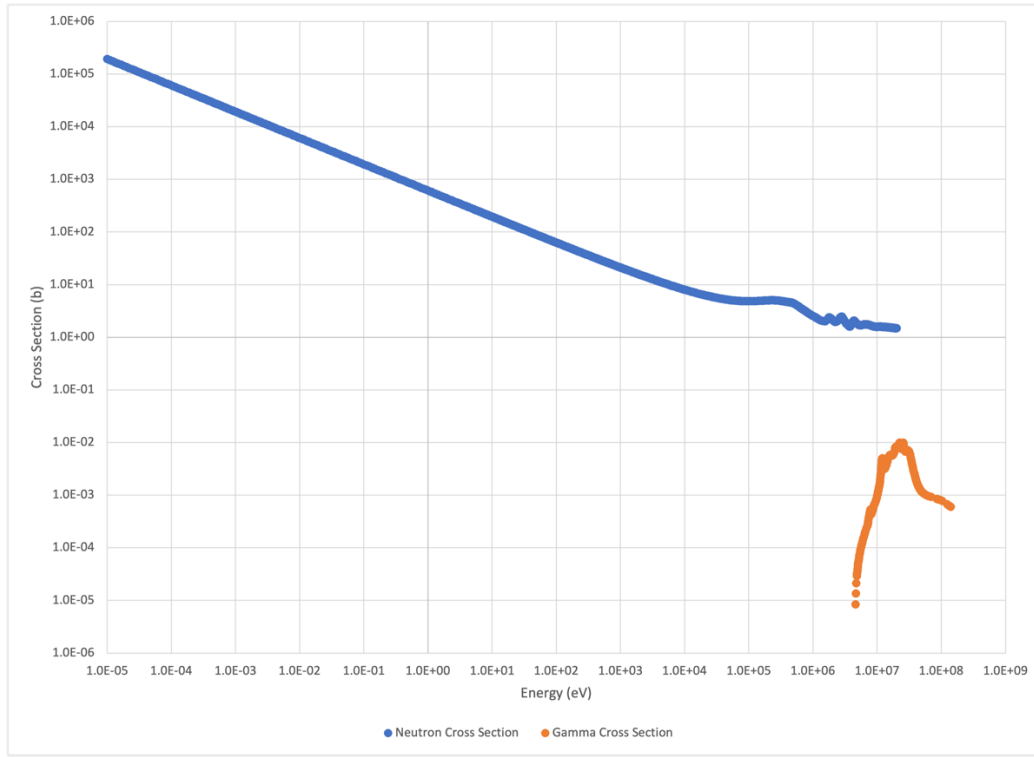


Figure 15. Interaction probability as a function of energy for neutrons (blue) and gammas (orange) in ^{10}B [36].

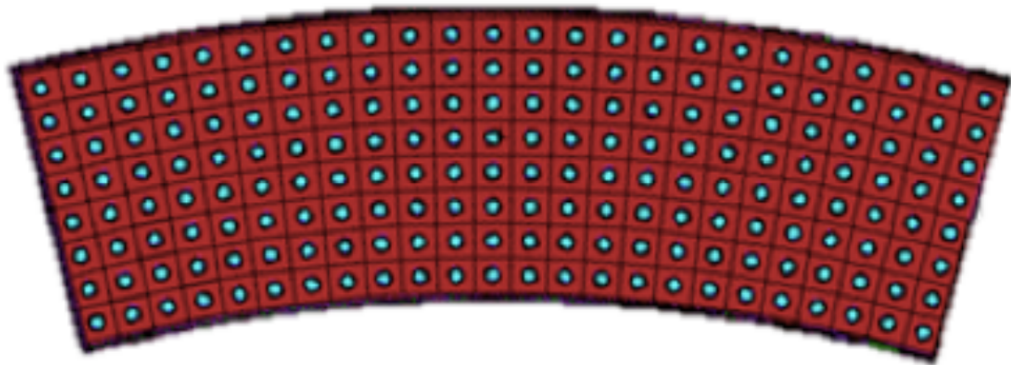


Figure 16. MCNP model of a single detector module in the final geometry [34].

and Cd layers on the inner and outer radii of the module to remove thermal neutrons. Each module is 113 cm tall, 100 cm of which consists of boron coated straw detectors, and the remaining 13 cm are composed of electronics and casing around the detectors.

2.3 Parallel Slit Ring Collimator Concept

A novel collimator was designed based on a parallel slit collimator but rotated into an annulus. Figure 17 shows the difference between a conventional parallel slit detector and the modified parallel slit collimator [34]. Rotating the slits in the collimator allows a greater distance between each detector element and slit than would otherwise be required using a conventional parallel slit collimator. Rotating the position of each slit through a known angle achieved equal detector spacing around the outside of the collimator annulus. This maintains the function of the collimator and the lines of response along particular chords across the central volume of the annulus, also known as the inspection volume or the field of view (FOV), are maintained, as in the parallel slit collimator. The increased distance between detector elements reduces the neutron scattering which may occur between detectors, increasing spatial resolution. Because all projection angles around the fuel assembly will need to be imaged by rotating the collimator, the rotated slits still create the same line of response as originally. This makes Figure 17(e) equivalent to Figure 17(a) upon a full rotation of the collimator.

The number of slits in the collimator was chosen to be 96. Due to the angular rotation that transformed a parallel slit collimator to a modified parallel slit ring collimator, the slit openings around the outer edge of the collimator were evenly spaced. Using 96 collimator slits meant that each slit would point to every third detector. The detectors were each placed at an angle of 1.5 degrees around the collimator meaning the outer slit edges were arranged every 4.5 degrees. This would create 96 lines of response across the field of view for each projection.

2.4 Parallel Slit Ring Collimator Parameters

After deciding to use the modified parallel slit ring collimator, a number of sensitivities studies were performed to determine the final design of the collimator. First, non-variable parameters for the imager design were identified. The imager design was constrained by a non-variable size. The inner radius of the imager is 17.667 cm, which was calculated to accommodate

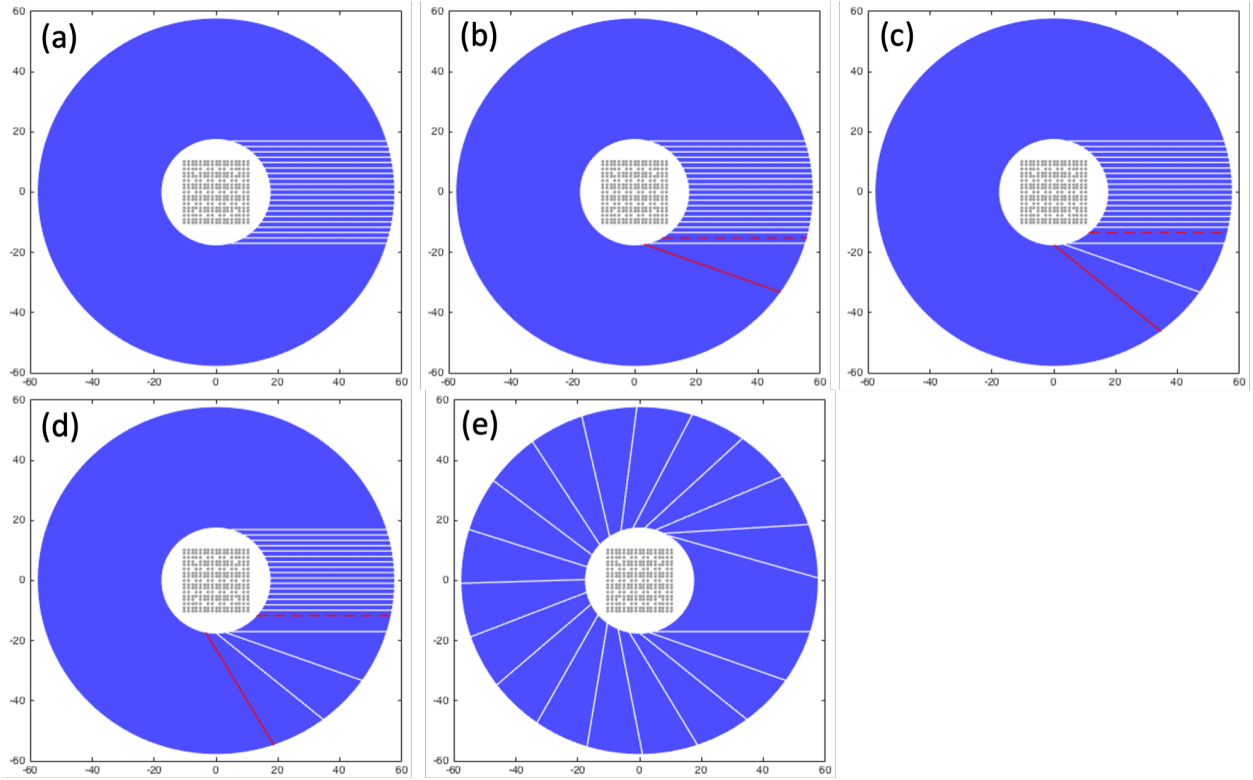


Figure 17. The evolution of the design of (a) a traditional parallel slit collimator through (b-d) the slit rotations into (e) the modified parallel slit ring collimator [34].

an assembly 1 cm greater than the half diagonal of a standard PWR 17×17 fuel assembly (24 cm) [34].

PWR reactors are the most common types of nuclear power reactors in the world with 304 PWR reactors in operation which use this fuel assembly size [38]. Given that the number of PWR reactors greatly surpasses the number of any other type of reactor, the NET system was designed around a typical PWR 17×17 fuel assembly. Additional typical reactor fuel assembly dimensions are included in Table 4. For square fuel bundles, demarcated by an asterisk, the half diagonal length of the fuel assembly is used. For cylindrical fuel assemblies the half diagonal length is equivalent to their radius. Out of the 438 reactors in existence, 435 of them (99.3%) have fuel assemblies or bundles small enough to fit in this imager inner radius.

The total outer radius limit of the imager was selected in order to be reasonably handled by a spent fuel facility. This means that the imager must be able to fit in a spent fuel cask that operators at spent fuel storage facilities would typically handle. The baseline imager was designed so the device would fit within 48 cm internal radius of the GNS Castor series of transport casks because this cask type is commonly used for international fuel shipments. The collimator height was approximately 1 m. This height increased the measurement efficiency but also limited image reconstruction uncertainty due to spent fuel warping. Spent fuel assemblies can warp along their 4 m length by as much as 1 cm. Constraining the imager to 1 m in length minimizes the amount of uncertainty in the fuel length location to 0.25 cm or less. This uncertainty length is comparable to the space between each fuel pin. Shielding made up of borated polyethylene and stainless steel also cover the top and bottom of the imager to protect the detectors from radiation coming from the fuel assembly extending out on either side of the imager.

Following the size constraints, parameters of the imager were identified as variable: stainless steel thickness, borated polyethylene thickness, and slit width. The collimator is made up of two materials – an inner ring of stainless steel and an outer ring of borated polyethylene - with a number of slits to define the lines of response. The borated polyethylene in the collimator attenuates high energy neutrons to create the distinct lines of response. The stainless-steel portions provide structural support as well as attenuate out a portion of the gamma emissions from the spent fuel [34]. MCNP simulations were conducted to determine the thicknesses of the

Table 4. Reactor types, the number in existence, and their typical assembly sizes.

Reactor Type	Number [33]	Typical Fuel Assembly Half Diagonal Lengths (cm)
Pressurized Water Reactor*	304	16.667
Boiling Water Reactor*	61	15.26 [38]
Pressurized Heavy Water Reactor	48	5.1 [39]
Advanced Gas-Cooled Reactor	11	13.175 [40]
Light Water Graphite Reactor	11	~4 [41]

borated polyethylene and stainless steel as well as the slit widths.

Gamma dose rate and neutron detection efficiency simulations were run in MCNP. The initial MCNP simulations determined that the minimum quantity of borated polyethylene for neutron collimation was 30 cm. From this, eleven different combinations of stainless steel and borated polyethylene thicknesses were analyzed. Table 5 shows the thicknesses viewed. Slit width was also a variable parameter for the image [34]. In addition to the 11 different combinations of collimator component thicknesses, 32 different slit widths were simulated. These included 9 different parallel slit widths with dimensions of 1 mm to 5 mm in 0.5mm intervals, and 23 tapered slit combinations with inner dimensions of 1 mm, 2 mm, 2.5 mm, and 3 mm and then proceeding to an outer dimension of 5 mm in 0.5 mm intervals.

2.5 *Gamma Dose*

The gamma dose rate of a worst-case scenario gamma source was calculated using MCNP. The gamma source was assumed to be a fuel with a high burnup of 40 Gwd/MTU and a short cooling time of one year. This gamma source was chosen as a “worst case” scenario to yield a conservative estimation for the gamma dose expected. This gamma source yields an emission rate of 4.03×10^{16} photons/second. Figure 18 shows the gamma dose rate deposited in the detectors with the gamma emission rate divided equally across all fuel pins in the fuel assembly. The collimator used in this simulation had parallel slit widths 3 mm wide consisting of 10 cm of stainless steel followed by 30 cm of borated polyethylene. The gamma dose rate to any of the detectors does not exceed 500 R/h with this collimator which was determined to be the tolerable dose rate in the detectors. This is due to the low-probability, large amplitude interactions of single gamma rays in the boron-straw neutron detectors [34].

2.6 *Imager Specific Signal-to-Noise Ratio*

After determining how these parameters could be modified, the conditions on which parameter performance is judged were identified. The collimator needed sufficient spatial resolution to isolate the fuel pins and fuel pin gap. Figure 19 shows the position of a single pin and the fuel pin gap halo which isolated the pin from others around it. To quantify this spatial

Table 5. Collimator component variations for comparisons [34].

Collimator Component Thickness (cm)			Outer Borated Polyethylene Shielding (cm)
Stainless Steel	Borated Polyethylene	Total	
5	35	40	5.822
10	30	40	5.822
6	36	42	3.822
5	38	43	2.822
6	37	43	2.822
7	36	43	2.822
5	40	45	0.822
6	39	45	0.822
7	38	45	0.822
8	37	45	0.822
10	35	45	0.822

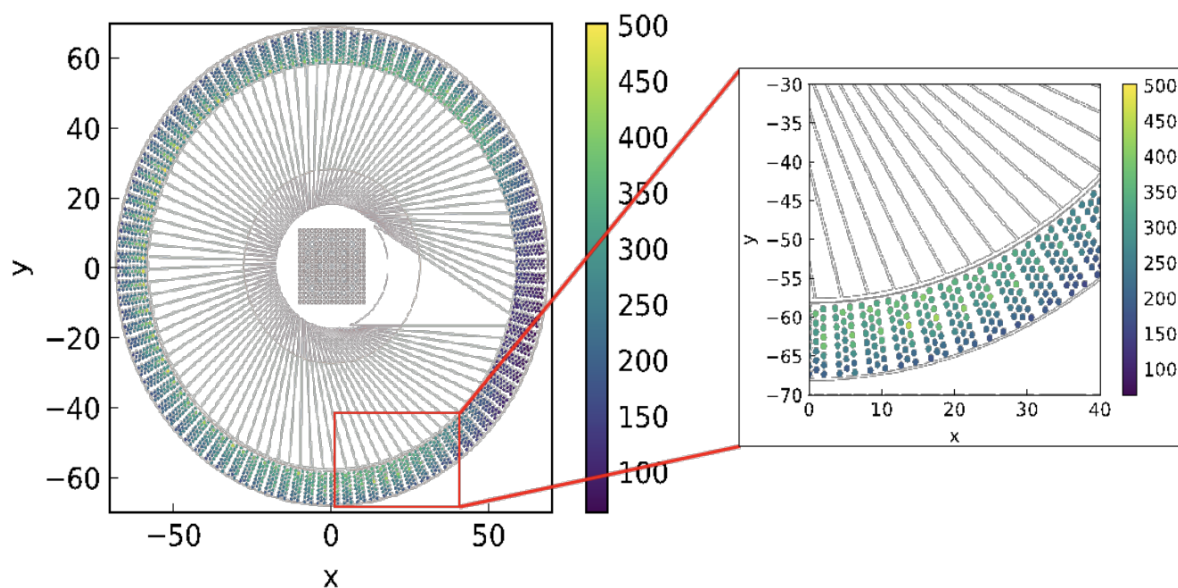


Figure 18. Gamma dose rates in R/hr for all detectors using a gamma source with an emission rate of 4.03×10^{16} photons/second [34].

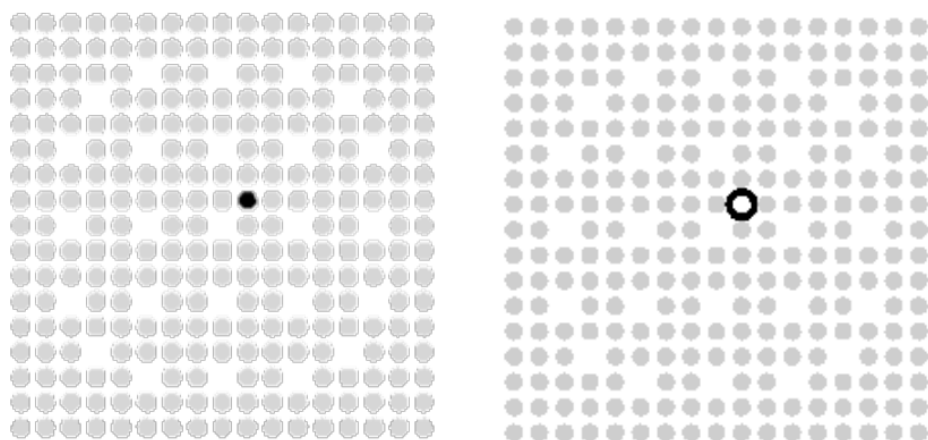


Figure 19. Image showing position of single pin (left) and the associated halo (right) [34].

resolution, a signal-to-noise ratio (SNR) was previously defined in Dr. Iyengar's dissertation [34]. The signal is the magnitude of the difference between the calculated response of the imager to a single pin and the calculated response from the imager to the halo surrounding that pin. This can also be described as the difference in neutron counts attributed to the fuel pin and the neutron counts attributed to the space surrounding the pin. The noise is approximated by the square root of the neutron counts for the full assembly in the region of interest for the particular fuel pin. The signal-to-noise ratio can be written as:

$$SNR = \frac{|R_p - R_h|}{\sqrt{R_a}} \quad (\text{Equation 3})$$

where R_p represents the response to the fuel pin, R_h is the response to the halo, and R_a is the response to the whole assembly.

In general, to best reconstruct an image of the fuel pins, the activity associated with each fuel pin needs to be attributed to the pin itself rather than the inter-pin space. This maximizes the numerator value in Equation 3. The denominator in Equation 3 is also the uncertainty or standard deviation of the neutron measurement in that particular region. The best imager would maximize the SNR to ensure that the detected activity from the fuel pins is larger than the uncertainty in the detector measurements, while maintaining a tolerable gamma ray dose of 500 R/h or less to the detectors.

2.7 Analytic Neutron Response

Maximizing the SNR requires knowing the neutron response on the millimeter length scale for the entire FOV for each projection angle. Generating responses for all points in the field of view for all the different potential collimator configurations becomes computationally expensive very quickly. To reduce the number of simulations, an approximate imager response was developed based on analytical models of the neutron response components. The analytic model for neutron response was based on the expected physical processes, then empirically fit to agree with data from a limited number of MCNP simulations. It is important to note that the response needed to capture spatial variations that could otherwise be attributed to differences in neutron intensity. Small inaccuracies in the response could impact the attribution of neutron

intensity to a specific fuel pin because a single fuel pin corresponds to less than 0.4% of the total intensity.

2.7.1 Detection Efficiency

The detection efficiency variation across individual rows of detectors is also taken into account. Each row of 8 detectors has a slight difference in detection efficiency due to small variations in the boron carbide layers in individual straws from manufacturing. Small differences in the gain of each detector row also contribute to efficiency variations. Figure 20 shows simulated absolute neutron detection efficiency across all 288 rows of straw detectors when a line source of ^{244}Cm is placed at the center of the imager inspection volume and no collimator present [34]. PTI, the manufacturer for the boron coated straw detectors, has reported that the detector straws have an intrinsic efficiency of 54% [34, 37]. The simulated efficiency also shows 12 larger dips in the number of neutron counts detected. Due to the design of the detector modules, there is a gap in the HDPE moderator at the edges of each detector module. The Al casing that surrounds the outside of each detector module creates a moderator gap when each of the 12 individual detectors are placed next to one another. The gap creates a decrease in detector efficiency because the Al does not moderate neutrons as well as HDPE. The efficiency loss at the edges of the detector modules is about 8% compared to the average efficiency for detector straw rows in the center of each module.

2.7.2 Inter-detector Scattering

Inter-detector scattering was also considered wherein neutrons traverse the collimator slit and then scatter into a neighboring detector to generate signal. Inter-detector scattering limits the spatial resolution of the detector. Neutrons that traverse the collimator slits scatter in the moderator around the boron coated straws but due to this scattering they may be detected in a row of straw detectors that is farther from the slit in which they traveled. This can lead to misattribution of neutron paths when the neutron is detected in a row of straw detectors corresponding to a different slit than the one it actually traveled down. Inter-detector scattering is the physical process that results in a characteristic spreading of the detector response which is conventionally known as the point spread function (PSF).

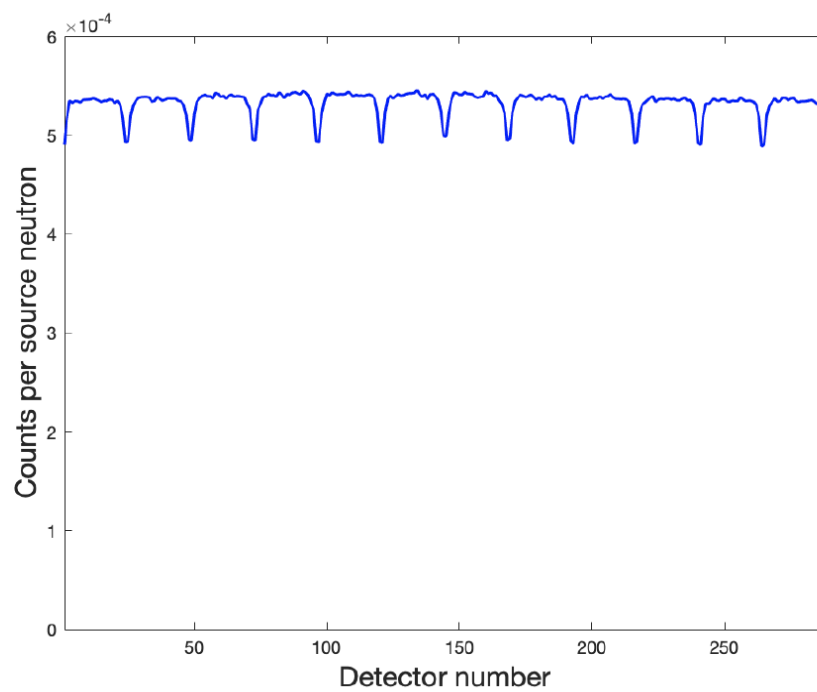


Figure 20. Detector efficiencies by detector row number [34].

The PSF was characterized by simulating a beam of neutrons with the energy spectrum of ^{244}Cm scattering down individual detector slits. The neutron capture rate in each detector was recorded. From this data, a double Gaussian fit for the PSF was created and can be written as:

$$PSF = A(\theta) \left(\exp\left(-\frac{(x-x_0(\theta))^2}{2\sigma(\theta)^2}\right) + \frac{1}{1.75} \exp\left(-\frac{(x-x_0(\theta))^2}{2(1.75\sigma(\theta)^2)}\right) \right) \quad (\text{Equation 4})$$

where, A , x_0 and σ represent the amplitude, offset and width of the distribution, respectively. The value x represents the distance (in number of detectors) between the detector at the outer diameter of the slit and the detector where the response peaks. The quantities A , x_0 and σ are written as a function of the angle θ between the slit and the unit vector normal to the detector surface. Note that the distance between neighboring detectors at the edge of detector modules must be reduced to 0.8716 to make the parameters a relatively smooth function of the angle θ . This shorter distance reflects the reduced amount of HDPE moderator at the edges of each detector module and the decreased distance reflects an effective distance in HDPE moderator. The fit values for the PSF are tabulated in Table 6. Figure 21 shows an example PSF for a particular slit of the collimator where the angle θ is zero [34]. In this instance the slit is normal to the detector surface. Figure 21(a) shows a linear scale of the simulated neutron captures (black) across all 288 detectors as well as the double Gaussian PSF fit (red). Figure 21(b) shows the logarithmic scale of the same counts and fit equation [34].

2.7.3 Collimator Penetration

Due to the quantity of slits made into the collimator there are paths through which the effective density of the collimator is much lower than if the same collimator thickness were completely made of borated polyethylene. Neutrons coming from a spent fuel assembly could potentially travel through the collimator in these effectively thinner areas without being attenuated. If they are moderated and absorbed in the boron straws, this can lead to misattribution of neutron source signal to a path along a collimator slit when the neutron actually travelled directly through the collimator. The portion of the neutron counts which would penetrate the collimator in this way was simulated using a line source with the neutron spectrum of ^{244}Cm which was centered in the inspection volume. Then the number of neutrons detected was compared by taking the negative natural logarithm of the number of neutron counts detected for each detector with and without the collimator present, which can be written as:

Table 6. Estimated fit function for the point spread function [34].

Parameter	Estimated Function
$A(\theta)$	$7.3569 * 10^{-2} + 2.7192 * 10^{-4} * \theta$
$\sigma(\theta)$	$2.6312 + 1.221 * 10^{-2} * \theta$
$x_0(\theta)$	$1.5664 * 10^{-4} + 5.9349 * 10^{-2} * \theta$

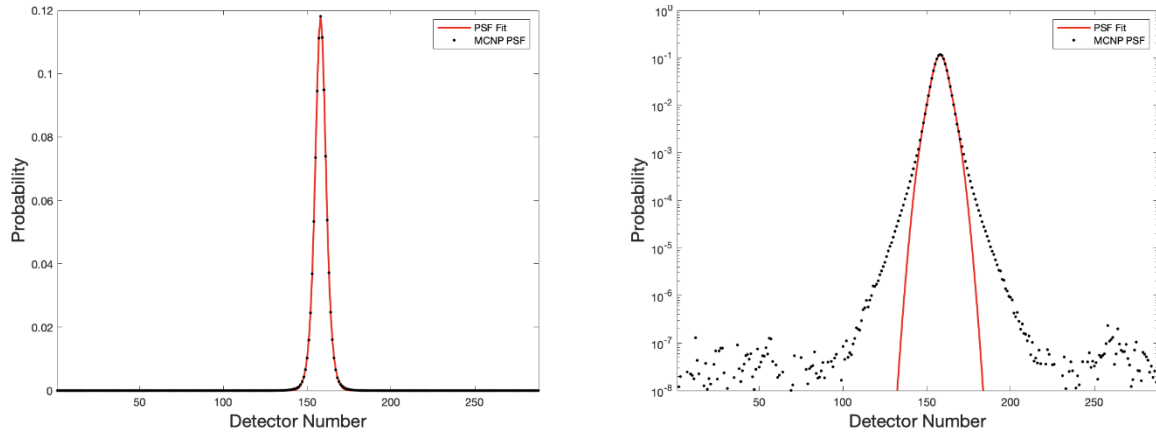


Figure 21. MCNP Simulated point spread function (black) and analytically fit PSF (red) for slit 49 in linear scale (left) and logarithmic scale (right) [34].

$$t_{eff} = -\ln\left(\frac{C_{coll}}{C_{nocoll}}\right) \quad (\text{Equation 5})$$

where t_{eff} is the effective collimator thickness, C_{coll} is the number of counts with a collimator present and C_{nocoll} is the number of counts with no collimator present.

Figure 22 shows the calculated neutron effective thicknesses for each detector row number (black) [34]. The difference in the effective thickness changes for each detector row because the density of the slits in the collimator changes for each path as the slits are unevenly distributed around the collimator. The collimator penetration would vary depending on the source location through the inspection volume. In order to generalize the collimator penetration values, an analytic fit to the data was created. The results of the fit for a line source centered in the inspection volume can be seen in Figure 22 in red [34].

Figure 23 shows the relative distances and angles from any detector to an arbitrary source point in the inspection volume [34]. The path through the borated polyethylene portion of the collimator has a length of $t + \Delta t$, where t is the straight-line path from the detector to the collimator inner edge and $t + \Delta t$ represents the increase of the path length from the detector to the collimator edge in the direction of the source position. The effective collimator thickness values for all detectors given any source position can be written as:

$$\Delta t_{eff} = t_{eff} \frac{\Delta t}{t} \quad (\text{Equation 6})$$

where Δt_{eff} is the effective collimator thicknesses given any source location, and t_{eff} is the initial effective collimator thickness values with a source at the center of the FOV. This is multiplied by the ratio of the difference in collimator path length from director to the source point, Δt , to the direct path length through the collimator, t . This ratio can be written as:

$$\frac{\Delta t}{t} = \frac{R_{out}\cos(\theta) - \sqrt{r_{in}^2 - R_{out}^2\sin^2(\theta)}}{R_{out} - r_{in}} - 1 \quad (\text{Equation 7})$$

where t is the shortest collimator path length, θ is the angle between t and $t + \Delta t$, R_{out} defines the radius from the center of the inspection volume to the detector edge and r_{in} defines the radius of the inspection volume. Then, the collimator penetration for any point in the FOV can be estimated as:

$$P = A_{cp} \left(\frac{R_{out}}{R}\right) \exp(-\Delta t_{eff}) \exp(-t_{eff}) \quad (\text{Equation 8})$$

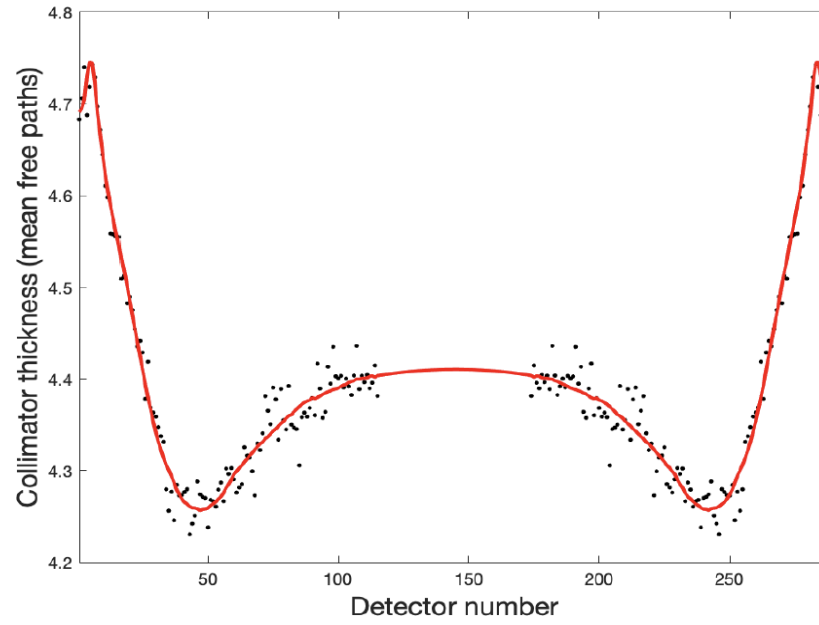


Figure 22. Effective collimator thickness for each row with a line source at the center of the FOV [34].

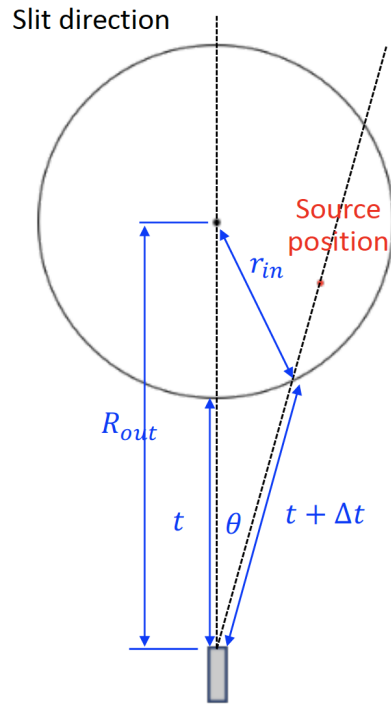


Figure 23. Drawing representing the distances used when calculating collimator penetration [34].

where P is the collimator penetration, and the factor A_{cp} is to fit radiation transport simulated data. Given that $A_{cp} \exp(-t_{eff})$ yields the value of collimator penetration with the source in the middle of the inspection volume, the factor $(R_{out}/R) \exp(-\Delta t_{eff})$ corrects for different source–detector distances and paths through the collimator.

2.7.1 Slit Scattering

One portion of the neutron response is slit scattering. This refers to a neutron that may be incident on a region of the collimator close to a slit opening, but not directly in the path of the slit, which scatters one or more times and is then directed unimpeded down the collimator slit.

Figure 24 shows a drawing of a slit scattering example [34]. The probability of slit scattering can be written as:

$$S = A_s \frac{t}{s} \frac{r_0}{r+2} \left\{ \begin{array}{l} \cos\theta, \text{ if } \cos\theta > \cos(\theta_1)\cos(\theta_2) \\ \cos(\theta_1)\cos(\theta_2), \text{ otherwise} \end{array} \right\} \quad (\text{Equation 9})$$

where A_s is a factor determined by radiation transport simulations, r is the distance from the source to the slit opening, r_0 is the distance from the center of the FOV to the slit opening, and the angle between r and r_0 is θ_1 . The length and direction of the collimator slit is s and the shortest distance from the detector and the collimator edge is defined as t . The angle between the slit direction and the source location is θ , and θ_2 is the angle between the FOV center and the slit direction.

2.7.2 Direct and Edge Transmission

The direct and edge responses of the collimator quantify the transmission of neutrons unimpeded down the collimator slits. The direct transmission simulated the amount of signal from neutrons that traveled to the detectors unimpeded by a collimator slit. The edge transmission represents the number of neutrons which travel down collimator slits but where the collimator wall partially or fully occludes the slit. An example of this would be neutrons traveling through the partially occluded slit wall and into one of the collimator slits, and then continuing down the slit to the detectors.

In order to simulate the direct and edge components of the response, a ^{244}Cm line source was placed in a simulated collimator with a single slit in it. The line source was moved

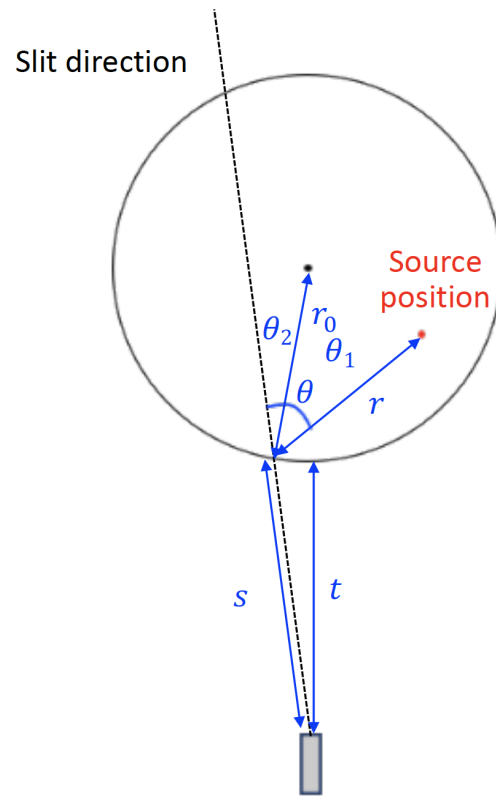


Figure 24. Slit scattering schematic [34].

perpendicularly to the slit direction at 3 different distances from the slit opening. These distances were 5.04, 10.08, and 20.16 cm. Figure 25 shows a drawing depicting the collimator with a single slit of length t with an outer opening of w , and a perpendicular line at distance d from the slit opening along which line sources were moved [34]. The distance that the source was moved from the centerline of the slit is w_s . For the simulations associated with each distance from the slit opening, 89 different source positions were simulated spanning the distance from the slit centerline of $w_s = -3.5$ cm to $+3.5$ cm in 0.1 cm increments and then extending to $w_s = -8$ cm to $+8$ cm from the slit centerline in 0.5 cm increments. Figure 26(a) shows the results from these simulations.

The distance past the slit edge can be linearly scaled as:

$$x = a_0 \frac{w}{t} d + a_1 \frac{w}{t} \quad (\text{Equation 10})$$

where d is the distance, $a_0 = 1.1896$, and $a_1 = 3.7057$. The reduction in relative intensities are well aligned at each distance, as shown in Figure 26(b) [34]. This means the slit edge region responses can be modeled as a factor f , written as:

$$f = 1 - \frac{x}{a_0 \frac{w}{t} d + a_1 \frac{w}{t}} \quad (\text{Equation 11})$$

The direct transmission response, where the source has unimpeded view of the entire back opening w , should scale as one over the distance from the source to the detector. However, because the slit opening at the inner diameter of the collimator is smaller than that at the outer edge of the collimator, this response becomes more complicated. The different solid angles created by the inner and outer collimator slit widths are taken into account by allowing an exponent for the inverse distance from the detector that can be greater than one. The total direct and edge components can be combined and written as:

$$D = A_d \left(\frac{R_{out}}{d+t} \right)^\alpha f_1 f_2 \quad (\text{Equation 12})$$

where R_{out} is the outer collimator radius, t is the collimator thickness, and d is the distance from the inner slit opening to the source. The α factor was determined by evaluating radiation transport simulations and finding that an alpha factor of 1.43 used as an exponent makes the equation best match the simulated data. The factor A_d is chosen to match radiation transport simulation data. The f_1 and f_2 factors are calculated for each slit edge according to Equation 11.

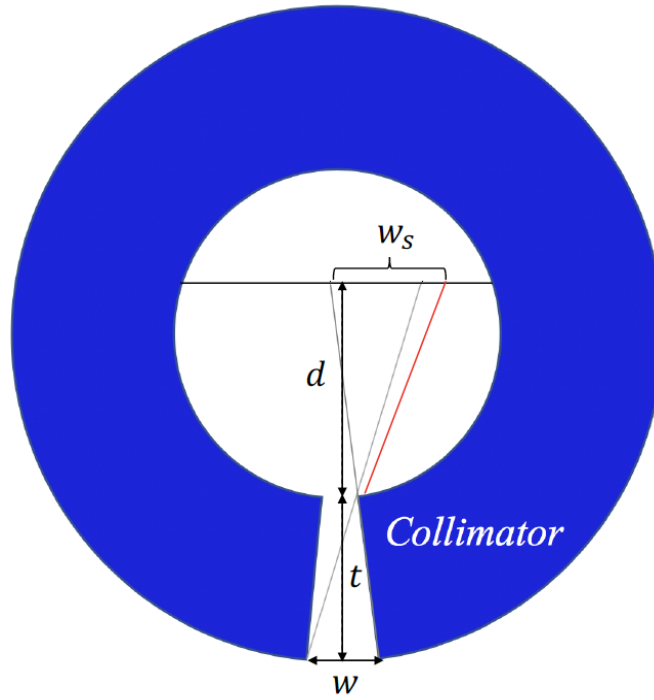


Figure 25. Single slit collimator drawing for simulating edge and direct scattering [34].

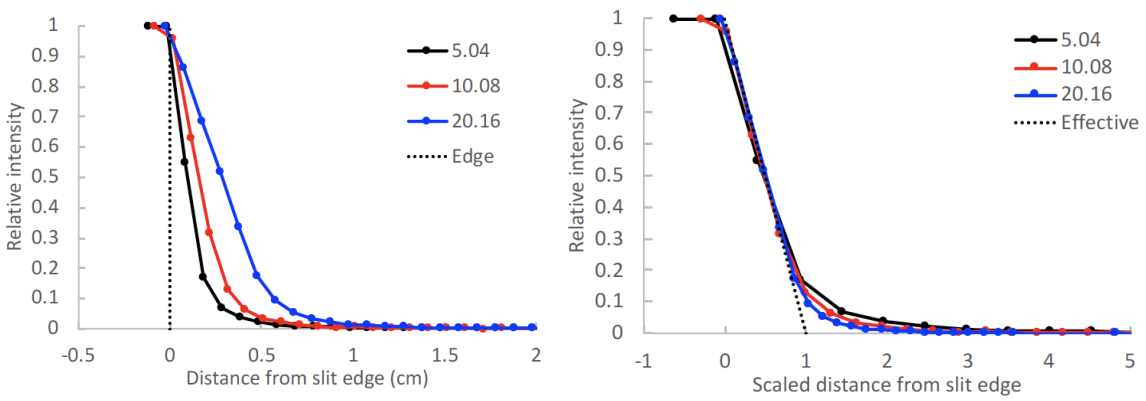


Figure 26. The relative neutron intensity as a function of distance from a slit edge (left) and relative intensity using a scaled distance from the slit edge (right) [34].

2.7.3 Total Neutron Response

The total neutron response can be determined by combining all components of the response. The total response can be written as:

$$T = \epsilon_{rel} \circ (PSF * D + PSF * S + P) \quad (\text{Equation 13})$$

where total neutron response is represented by T and was calculated by taking the sum of the direct and edge transmission, the collimator penetration and the slit scattering components. The direct and edge transmission as well as the slit scattering components were operated on by the PSF, then all components are operated on by the relative detector efficiency, ϵ_{rel} . In the equation the “ $\circ$ ” operator refers to element-wise multiplication whereas “ $*$ ” refers to matrix multiplication. The expression D is interpreted as a 96×1 matrix (one value for each slit), and the PSF as a 288×96 matrix (all detector values for each slit) that, when matrix-multiplied by D , yields a 288×1 matrix (one value for each detector). Likewise, ϵ_{ref} is a 288×1 matrix of relative efficiency values (one for each detector). This matrix is used to multiply each contribution elementwise to arrive at the total response. Similarly, S is defined as a 96×1 matrix, and P was defined as a 288×1 matrix. The total analytical neutron response for one projection is a 288×1 matrix.

As previously stated, the values of the constants A_d , A_{ss} , and A_{cp} , were fit from radiation transport simulation data. Radiation transport simulations were performed for a ^{244}Cm line source placed at positions along the central axis of slit 49 at distances between 1 and 35 cm from the slit opening in 1 cm increments. Similar simulations were performed for a collimator having only slit 49. The positions of the 49th slit and the 35 points where the source is positioned are shown in Figure 27 for (left) a collimator with a single slit and (right) a collimator with 96 slits.

The single slit collimator simulations were used to generate data with very minimal contributions from the collimator penetration response while the full collimator simulations were used to examine all contributions to the neutron response. Using the data from the 35 simulations, values for A_d , A_{ss} , and A_{cp} were chosen to match the simulated data. The best fit values were $A_d = 3.390725 \times 10^{-4}$, $A_{ss} = 4.553 \times 10^{-6}$, and $A_{cp} = 4.363601 \times 10^{-4}$. Figure 28 shows the comparison between the neutron response generated by MCNP radiation transport code and the analytic response code for all detectors when the source is placed at the 35 different locations

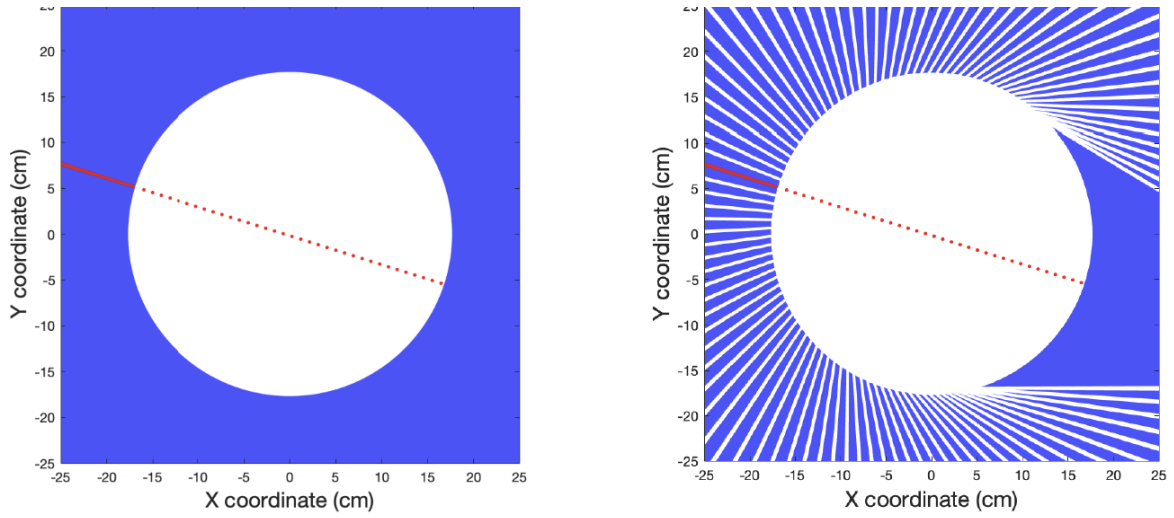


Figure 27. Neutron source positions at every 1 cm from the collimator edge with a single slit collimator (left) and a full 96 slit collimator (right).

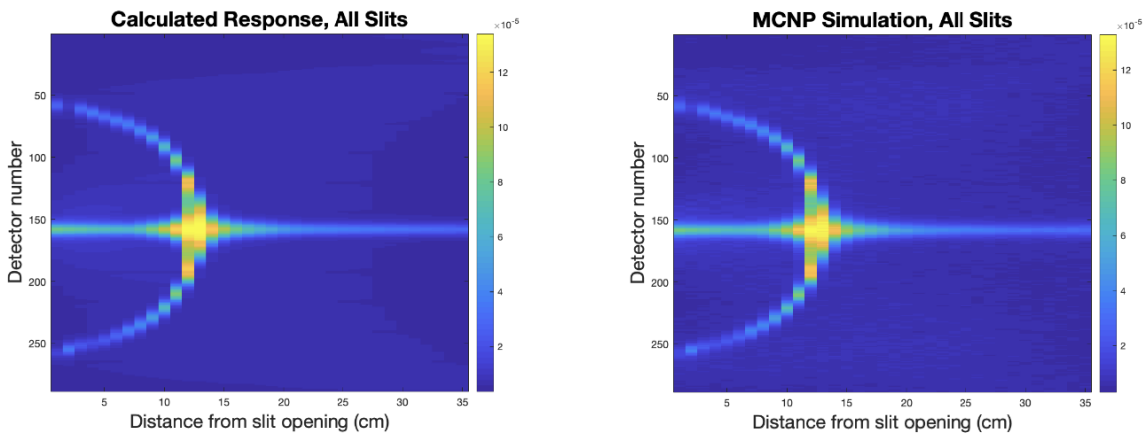


Figure 28. Calculated neutron response for all 36 source positions using the analytic model (left) and the simulated neutron response from MCNP (right) [34].

shown in Figure 27 [34]. The calculation visually matches.

2.8 *Final Parallel Slit Ring Collimator Characteristics*

Ultimately, using various simulations, an optimized version of the parallel slit ring collimator was created. The estimated neutron response for each of the configurations described in Section 1.7.4 was calculated, including the results with various slit widths and slit tapers. Then the responses were used to calculate the SNR for each configuration. It was determined that the values for all configurations that have the same slit dimensions were approximately the same, but the SNR does increase with increasing quantities of borated polyethylene in the collimator.

Neutron response simulations indicate that the collimator spatial resolution is effectively limited by inter-detector scattering (PSF), not slit width. This means wider slits can be used to increase neutron efficiency and more stainless steel can be added to the collimator design to attenuate gamma rays that would otherwise deliver too much gamma dose for wider slits to be used. While steel poorly modulates neutrons, it effectively controls gamma dose. Simulations also found that the performance of collimator configurations varies only weakly with the collimator thickness. As a result, the more compact designs are preferred.

The SNR is plotted as a function of gamma dose in the detectors in Figure 29 (left) for all configurations which have a more compact design with a thickness less than or equal to 45 cm [34]. Using 500 R/hr as a cut off gamma dose, many design configurations are eliminated. The blue dots in Figure 29 (left) represent one of the most compact collimator configurations with a total thickness of 40 cm: 30 cm of borated polyethylene and 10 cm of stainless steel [34]. Each blue dot represents a different slit width tested using this configuration and the majority of the slit widths used with this collimator configuration are below the gamma dose threshold. The best configuration would maximize the SNR while staying below the gamma dose rate threshold of 500 R/hr.

Using tapered slits can maximize performance from a compact design because they provide a combination of dose control and efficiency. The narrowness of the slit in the steel region attenuates more gammas and preserves the areal density of the collimator while the wider portions of the slit in the borated polyethylene region preserve the collimator efficiency.

Figure 29 (right) shows the SNR and gamma dose rates for different slit tapers using a

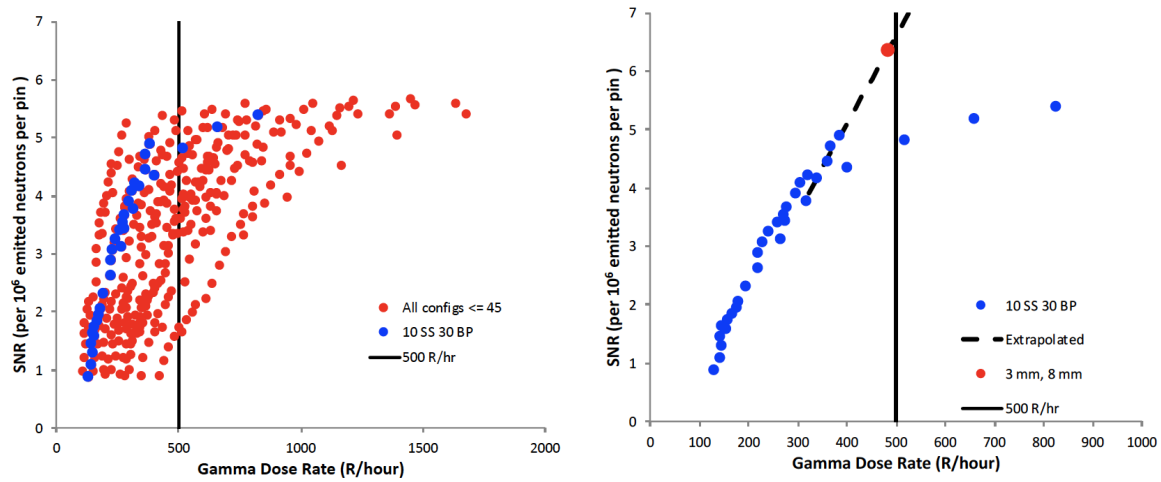


Figure 29. The signal to noise ratio as a function of gamma dose rate for all collimator material configurations and slit widths (left) and for a collimator configuration with 10 cm of stainless steel and 30 cm of borated polyethylene and all slit widths (right) [34].

collimator with 10 cm of stainless steel and 30 cm of borated polyethylene [34]. After examining the SNR and gamma dose rates from a variety of different slit tapers, a collimator composed of a 10 cm stainless steel section and a 30 cm borated polyethylene section with tapered slits with a width of 3 mm at the inner collimator radius and a width of 8 mm at the outer collimator radius produces a near-optimal value of SNR with a compact overall size. The SNR value for this configuration is marked by a red point on Figure 29 (right) [34].

2.9 Total NET System Design

The final collimator and detector system design is comprised of several different pieces. Figure 30 identifies each of these components. The imager FOV has a radius of 17.85 cm to allow for measurements of spent fuel assemblies. The collimator surrounding this field of view consists of an initial 10 cm of stainless steel followed by 29.22 cm of borated polyethylene that is 5% by weight boron. The slits were tapered with a width of 3 mm at the inner collimator radius and a width of 8 mm at the outer collimator radius. Ninety-six slits were incorporated into this collimator design.

Surrounding the collimator are 12 neutron detector modules. The detector modules sit 0.5 cm behind the collimator to allow the collimator to rotate without friction between the detector modules and collimator. More borated polyethylene, 5 cm in thickness, was added behind each of the detector modules to prevent background neutrons from scattering into the system. This gives a total imager diameter of approximately 146 cm. Table 7 contains information about the dimensions of the final collimator design [34].

While the detector system is intended to be used for fast neutrons, both fast and thermal neutrons are emitted from spent nuclear fuel. Ensuring that the majority of the neutron flux is primarily fast neutrons is done with material selection. The mean free paths (mfp) in air of fast neutrons and thermal neutrons are 2.51×10^4 cm and 1.83×10^4 cm, respectively. Meaning that both neutrons have the potential to travel in air to the detectors. The borated HDPE used in the collimator attenuates many of the thermal neutrons since the mean free path of thermal neutrons in borated high-density polyethylene is 0.421 cm. Given that the collimator consists of approximately 30 cm of HDPE, which is much larger than the mfp, the collimator acts as very effective shielding against thermal neutrons. The fast neutron mean free path in HPDE is 6.75

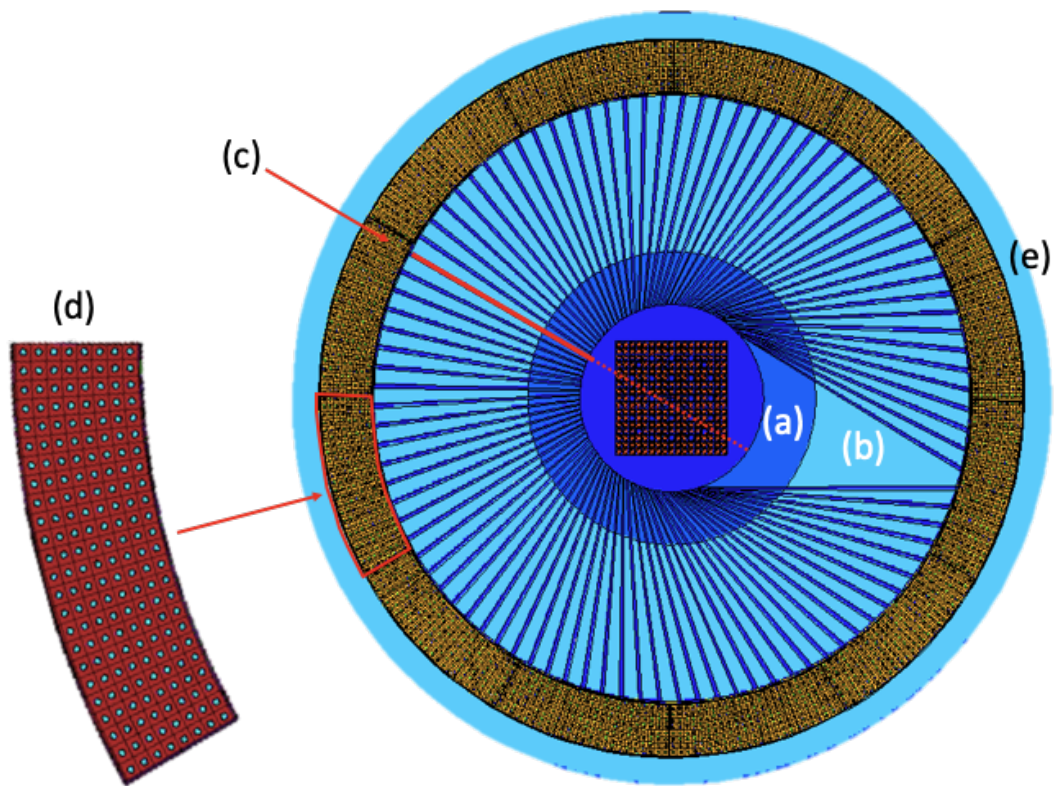


Figure 30. The collimator consists of 10 cm of stainless steel (a), 30 cm of borated polyethylene (b), with 96 slits in the collimator (c). The detector ring is 12 detector modules (d) with a 5 cm thick ring of HDPE surrounding the detectors (e) [34].

Table 7. Imager components and dimensions for the final imager design [34]

Imager Component	Dimension (cm)
Overall imager height	100.00
Fuel assembly cavity radius	17.677
Stainless steel thickness	10.000
Borated polyethylene thickness	29.523
Air gap between collimator and detector	0.3000
Inner slit width	0.3000
Outer slit width	0.8000
Al wall	0.0635
Cd layer	0.0406
Detector HDPE Thickness	9.9187

cm. The collimator is 4.44 times thicker than the mfp of the fast neutrons in borated HDPE. Many of the fast neutrons which enter the collimator will have interactions, but some will still travel through the collimator, and some may have multiple collisions and exit the collimator. The number of neutrons which make it through the collimator will need to be accounted for. There is also a 0.0406 thick layer of Cd lining the front of the detector between the collimator and the detectors and a second layer behind the detectors. The mfp of thermal neutrons in Cd is 0.00879 cm so any thermal neutrons which happen to make it to the cadmium layer will likely be attenuated before reaching the detectors. The mfp of fast neutrons in Cd is 3.91 cm, meaning most fast neutrons will pass through without an interaction in the Cd. The material selection ensures that a majority of the neutron flux that reaches the detectors are fast neutrons.

2.10 Analytic Response Code

The as-built imager geometry was recreated in MATLAB. This also included a way to rotate the imager a set amount to generate different perspectives. The geometry of the system was then combined with the neutron response model outlined in Section 2.7 and the equations coded in MATLAB to create a full analytic response model. Using the geometry of the system and a knowledge of a source strength, the neutron response for every millimeter in the FOV was created and stored in saved variables. This method was chosen because it is less computationally expensive than running 10^7 radiation transport simulations to produce results for every square millimeter in the inspection volume. It would also be more accurate than creating a grid of neutron responses spaced at every centimeter across the inspection volume and interpolating the results to get responses at every millimeter.

2.11 Simulated Reconstructions

A maximum likelihood expectation maximization (MLEM) reconstruction code was implemented in MATLAB to create forward and back projected images. The neutron response code was used to create forward projections, then neutron counts data from experiment or simulation was used as the measured data sinogram to compare to the simulated forward projections. A correction factor was multiplied to the analytic forward response and back

projected from there. The iterations repeated using the analytic neutron response model to generate and weight forward projections until the image began to converge.

The MLEM reconstruction starts by comparing each position and detector from the measured sinogram to an initial forward projection which is created by finding the analytic code response at that same position and detector and multiplying it by an initial guess. The initial guess for the reconstruction code is 1 for every detector in every position. A ratio of the differences between the two is created by dividing the measured data points by the forward projection. This ratio is then multiplied by the forward projection to give a corrected forward projection which accounts for the differences from the measured data. The corrected forward projection is saved to a variable labeled “correction” which tracks the changes with each iteration. The correction variable is multiplied by the initial guess to give a new guess for the next iteration. The next iteration continues using the new guess. Over time ratio between the corrected forward projection and the measured data becomes smaller as the guesses become better and closer to the measured data. Eventually, enough iterations take place to make the difference between the guess and the measured data negligibly different and at this point the data has converged, and an image is generated.

Chapter 3. Detector Construction

3.1 Proof-of-Concept Imager Design Changes

When constructing the imaging system, cost became a limiting factor. As a result, changes were made to the proof-of-concept imager system. Initially the collimator was designed such that it would be equivalent to having 96 parallel slits examining 96 different chords across the inspection volume [34]. These 96 slits would point towards an outer annulus of 288 detectors such that each slit pointed to every third detector. Arranging the slits in this way added a level of symmetry in the lines of response. The same line of response could be evaluated from different angles by two different slits. This gave 48 individual lines of response across the field of view.

Due to cost considerations changes had to be made in the prototype detector. Instead of the 96 slits in the collimator as originally planned, the number of slits was reduced to 72. This represents one slit for every 4 detectors rather than 1 slit for every 3 detectors. One of the concerns with decreasing the number of slits is the impact this reduction would have on the resolution. If the same symmetry from the 96 slit model were kept in the 72-slit model, the 72 slit version would only have 36 distinct lines of response. This would not be enough lines of response for sufficient spatial resolution across the field of view. Instead, fifty percent of the slits were offset by half-a-slit spacing, creating 72 separate lines of response.

In addition to changing the collimator, the number of detector modules themselves also had to be reduced. Rather than the planned 12 detector modules, the number of modules was dropped to 6 modules. Rather than a full ring of detector modules, every other detector position was replaced with a detector spacer made of aluminum and polyethylene. This held 6 of the detectors in alternating locations around the ring. To make a full measurement as though 288 detectors were used, the detector ring could be rotated by 1 detector module spacing, or, equivalently, by 30 degrees. Figure 31 shows how the detectors would be placed such that a full detector ring could be mocked up [42]. The downside of this change was that each measurement would need to be taken twice, once in each position, to get the full view.

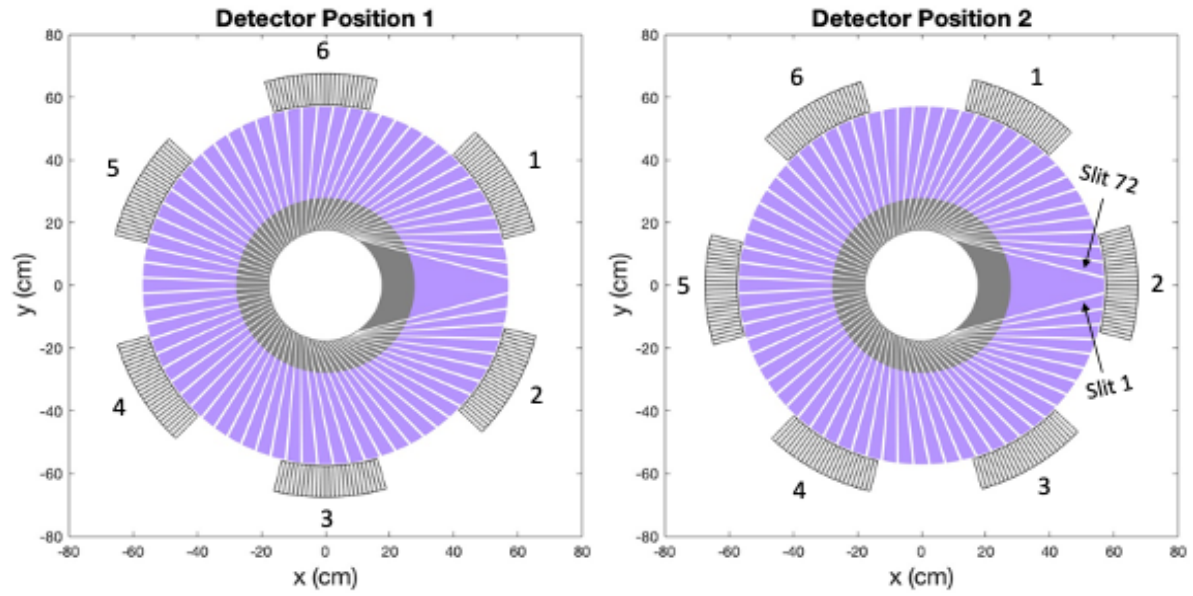


Figure 31. The two detector ring positions used to represent a full ring of 12 detector modules [42].

3.2 *Proof-of-Concept Imager Construction*

3.2.1 *Collimator Construction*

After design, the collimator was constructed by Agile Technologies. The collimator consisted of 72 slits which were designed with an inner diameter of 0.3 cm and an outer diameter of 0.8 cm. The 72 slits were arranged to cover the full 34 cm diameter of the FOV and view different chords across this region. Because of this each slit has different lengths and opening angles around the collimator. To accurately recreate this design, the collimator was built by assembling wedge-shaped pieces into the cylindrical collimator. When creating the collimator, instead of taking solid pieces of HDPE and stainless steel and then carving out gaps for the collimator slits, individual wedges composed of stainless steel and HDPE were constructed and combined. Figure 32 shows the stainless-steel wedge pieces (left) and the composite collimator wedges (right) [42]. These wedge-shaped pieces were cut to the size of the collimator material between the collimator slits. Each wedge was a composite consisting of a 10 cm thick stainless-steel portion which was screwed into a 29.22 cm thick HDPE collimator piece. In order to hold and arrange these composite wedges in the correct place, two guides were created, one for each the top and bottom of the collimator. Figure 33(a) shows the collimator guides with no wedges in place. These guides were created by taking rings of composite stainless steel and HDPE and machining slits into each one, shown in Figure 33(b) [42]. Then the collimator wedge pieces were slid into position using the guide slits. The final collimator with wedges installed is shown in Figure 33(c) [42].

3.2.2 *Detector Module Construction*

The 6 detector modules were constructed as designed with 24 rows of 8 boron coated straw detectors with each embedded in HDPE. Each row of eight straws is read out by a single, low power electronic front-end readout board. Images of a detector module are shown in Figure 34 [42]. The readout electronics are implemented by a fast current preamplifier as a front-end element, followed by another amplification stage, a passive, 6-pole, low-pass filter with a 56 MHz bandwidth, and a driver [42]. The resulting signal triggers a comparator which compares the signal to a common threshold value across the entire module. Any signal greater than this

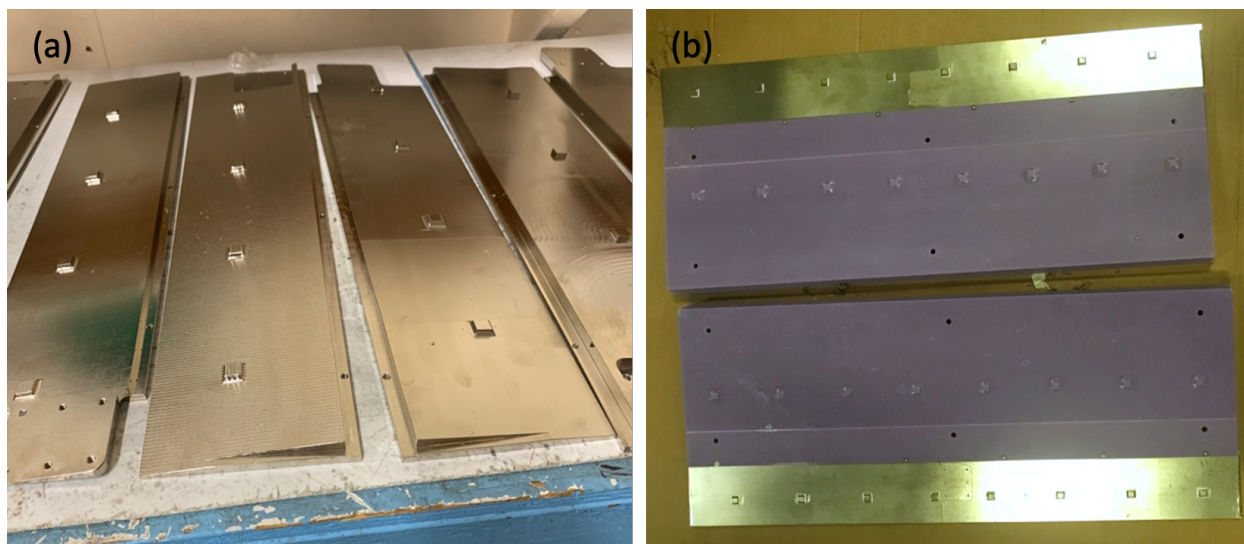


Figure 32. Composite wedges of the collimator [42].

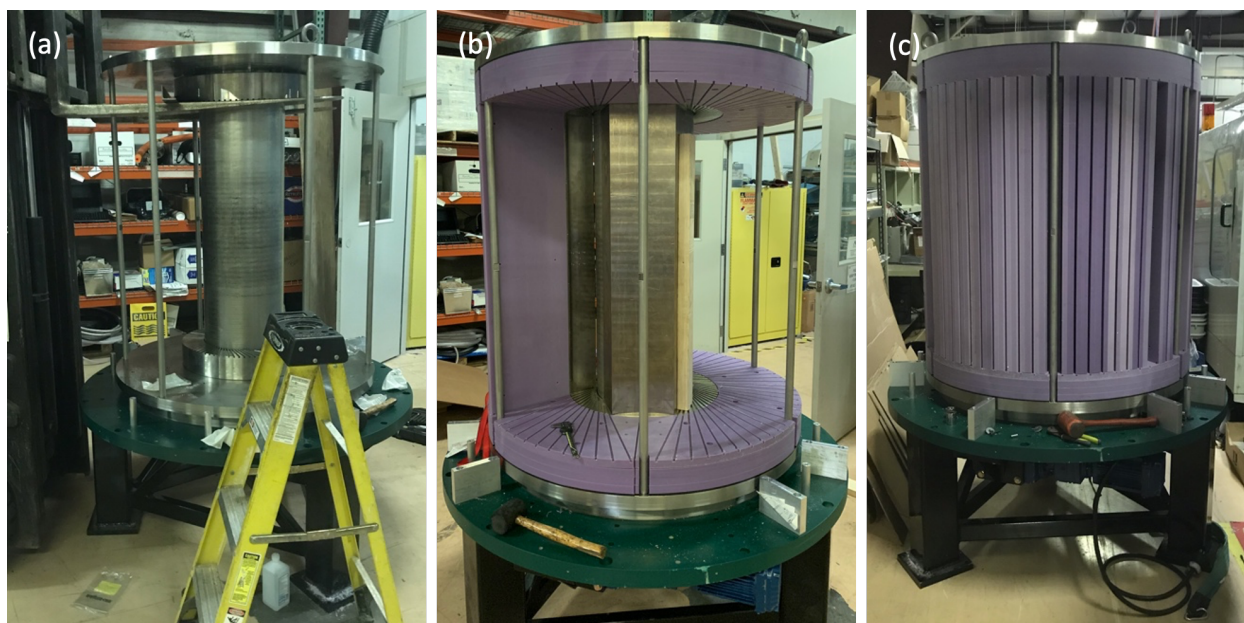


Figure 33. Assembly of the collimator [42].

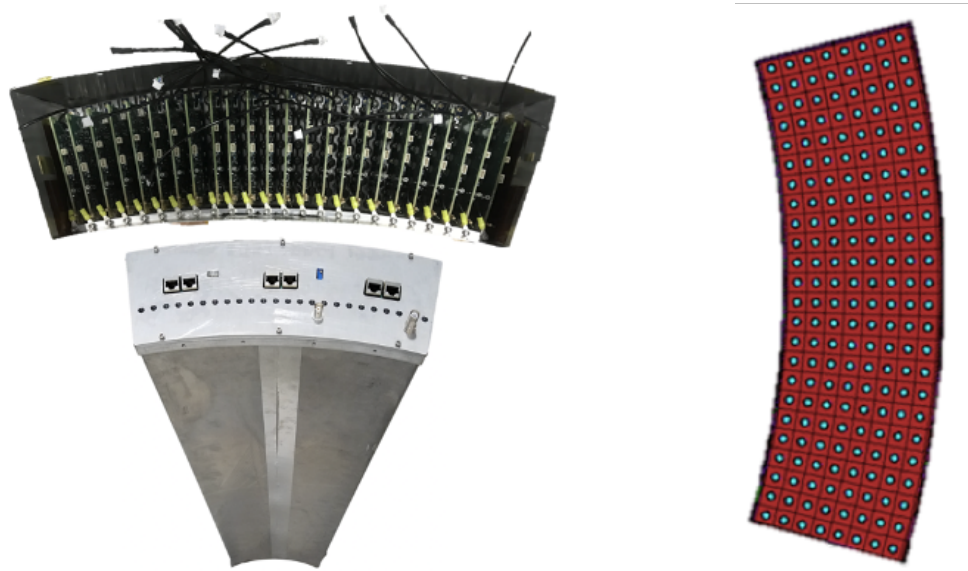


Figure 34. Detector module processing board (top left), photo (bottom left), and drawing of straw detectors embedded in HDPE (right) [42].

threshold is counted as an event. The event signal is then sent to a processing board which spans the entire detector module top surface. This board then buffers the signal and directs it into the output signal. This processing board also controls the detector voltage biasing value of ± 6.5 V. The output signals are provided for further processing to the back end and data acquisition (DAQ) electronics via Cat6e shielded cables.

The detector channels are numbered from 1 to 24. Six RJ-45 connectors cover an entire board and each carry four signals from the detector to the DAQ system [42]. The resulting current required by a single detector module is 1.38 A from the positive bias and about 0.46 A from the negative bias voltage.

3.2.3 *Full Imager Assembly*

Figure 35 shows a labelled drawing of the full assembly [42]. The imager was constructed onsite at ORNL. First, an assembly and support stand were set up with a variable speed drive motor and pinion mounted to the stand. This would provide the detector base. A stainless-steel base plate was mounted on top of the support stand as a stationary plate with mounted large diameter high-capacity slewing gear bearings, Kaydon Model #HS6-37N1Z. A secondary base plate was added on top of the bearings allowing the plate to rotate on the gear driven support bearings [42]. The collimator was mounted onto the upper plate. The collimator is rotated by a helical-bevel gearmotor and motion control inverter drive, SEW Eurodrive Model KA67R37** and MC07B0015-2B1** [42]. The motor and inverter drive together allow the collimator to rotate at various speeds between 0 and approximately 2 rpm. The power supply for the motor was installed on the leg of the imager support stand in a National Electrical Manufacturers Association rated electrical enclosure [42].

An additional stainless-steel annulus was added to the system around the second base plate to mount the detectors to the imager. This annulus was also placed on support bearings; however, these were not gear driven. This allowed the collimator and the annulus to be rotated independently of one another. A layer of 1" thick borated polyethylene was added above the annulus plate to prevent neutrons from scattering into the detectors from below. The six detector modules were mounted into this stainless steel and HDPE layer with detector spacers between each module. A 5 cm layer of HDPE was added behind each detector module to prevent

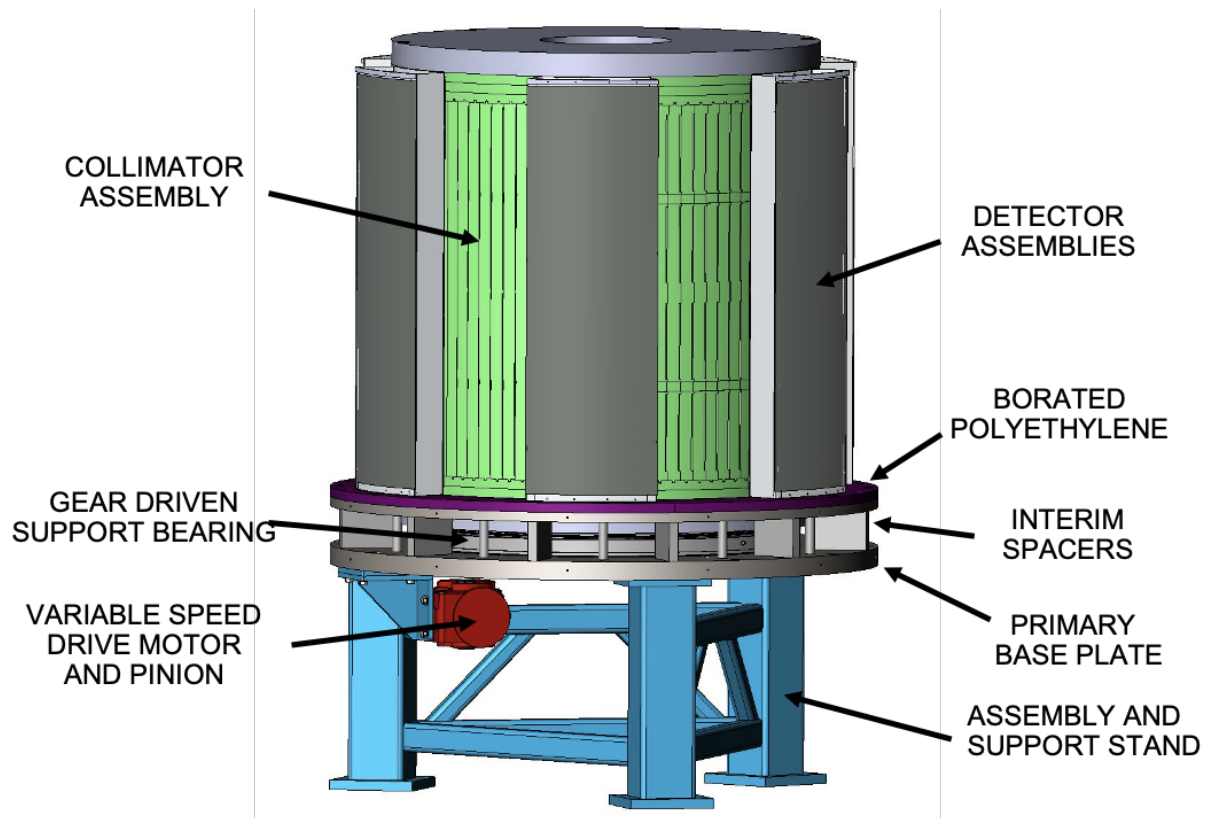


Figure 35. Drawing of the full imager assembly [42].

background neutrons from scattering into the system. Figure 36 shows a top-down view of the as-built fixture placed within the imager. A mock fuel assembly fixture was added to the system to allow sources and mock fuel rods to be accurately placed within the imager FOV. This fixture was created to hold mock fuel pins in an arrangement modeled on a PWR fuel assembly. The fixture consists of two aluminum grids with 17x17 holes of diameter 0.9652 cm spaced with their centers 1.27 cm apart. Four stainless steel rods span 94 cm and connect the upper grid and lower grid to ensure the fuel pins remain vertically aligned.

Two different types of rods were procured in order to populate this grid fixture, the first being a neutron source rod. A thin-walled aluminum tube with an outer diameter of 0.9525 cm acted as the source rod. An elastomer stop was placed 25 cm within the tube from the bottom of the rod. This allowed a thin, cylindrical source to be added into the tube from the top of the rod where it would sit atop the stopper. This placed the source at a position 3 cm below the centerline of the neutron detectors. This source rod could be moved from one assembly grid location to another by removing the rod and placing it into the next position of interest. Figure 37 shows the rod and how the rod is placed in the assembly grid.

The second set of rods, this time made of stainless steel, was procured to mimic the attenuation that would come from other fuel rods in the assembly. These 121.9 cm long, 0.9525 cm diameter, stainless-steel rods would be added to all positions in the grid fixture other than the position occupied by the neutron source to mock the neutron absorption from the rest of the fuel assembly. Figure 38 shows a drawing of the fixture populated with rods and positioned within the imager (a), as well as a cutaway of the fixture within the imager (b), and a rendering of just the fixture itself (c) [42]. By taking separate neutron measurements with the neutron source rod in different locations, a full fuel assembly's worth of data can be considered when all the data is summed.

3.3 Data Acquisition System

The DAQ system uses a CAEN V2495 Programmable Logic Unit with three added A395A digital input boards and FW2495SC FPGA firmware in order to be used as a multi-event scalar counter. Using the three additional boards, 160 independent counting channels are available to record data from the detectors. Given that only 6 of the detector modules were made,

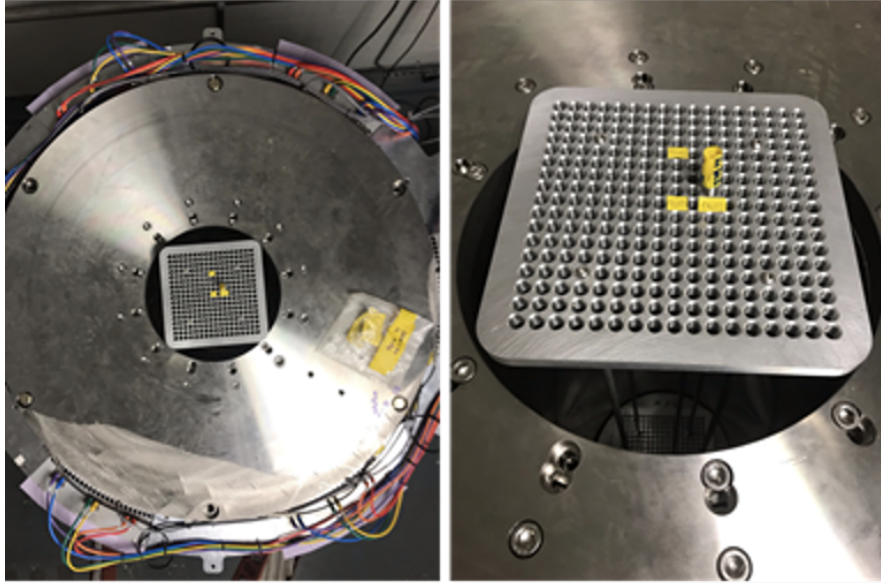


Figure 36. The grid fixture installed in the inspections volume.

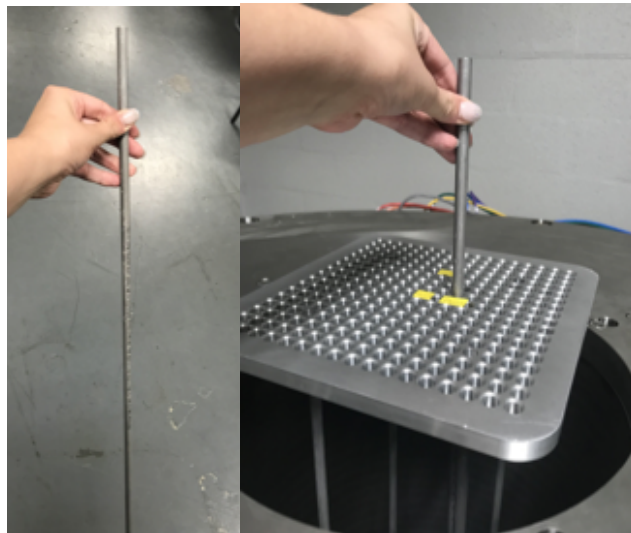


Figure 37. Neutron source rod by itself (left) and in the grid fixture (right).

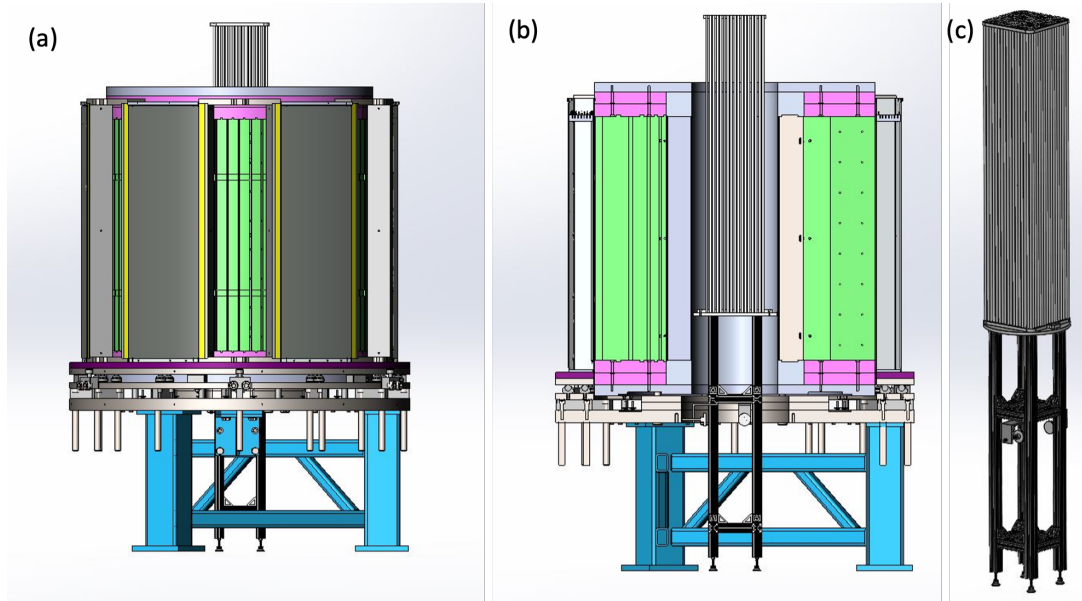


Figure 38. A drawing of the grid fixture within the full imager (a), a cutaway of the fixture's placement within the full imager (b), and an image of the fixture itself (c) [42].

with 24 data readouts each, only 144 of these 160 channels are used for signal. The remaining 16 channels allow the addition of a clock pulser and IR optical switches which encode the positions of the collimator as it rotates. The V2495 is then read by a custom software program created by Cadre5 which accesses the detector information through a SIS3153 VME bus and allows for system control through a custom interface. The interface allows for control of the two CAEN V6533P six-channel, positive polarity, 4 kV, 3 mA detector module power supplies as well. A photograph showing the V2495 logic unit, two V6533P HV modules, and VME bridge is shown in Figure 39.

The V2495 modules accept input on proprietary CAEN A967 flat ribbon cables. A custom interface module is located within a rack enclosure to convert the detector signals from Cat6e signaling to the CAEN flat ribbon cabling. The back of the module has 48 RJ-45 cable inputs across 4 PC boards upon which the incoming signal is buffered. At this point the signal is driven into the CAEN flat ribbon cables which are arranged on the front of this module. The CAEN ribbon cables consist of two 34 pin channels which merge into a 68-pin connector that is directly connected to the V2495 module and the accompanying A395A input boards. Each of the CAEN cables can carry up to 34 signals; however, only 32 of these signals are used on each. Due to the mechanical characteristics of the CAEN flat-ribbon cable, a straight-through mapping of the channel numbers is not possible.

In order to accommodate these cables, the 34 pin connectors are attached to the interface module starting from the right side of the module. This can be seen as the black module in Figure 40 with 12 connections. The first connector will plug to the lower rightmost mating connector on the panel. The second will plug under this connector. The next cable will plug in the center row according to the same modality, and so on. These connectors are assigned letters A-F to differentiate to which connector a reference is being made. The lower rightmost mating connector is labeled A, while the lower center connector is B and the leftmost lower connector is labeled C. The upper rightmost connector is labeled D, the center is labeled E, and the upper leftmost is labeled F. Consequently, the Cat6e cable on the opposite side of the module must be mapped accordingly.

Due to the design of the CAEN DAQ, the connector labeled C is designated as a set of 32 output channels. For this reason, no RJ-45 cables are connected to that connector. The back of



Figure 39. The data acquisition system.

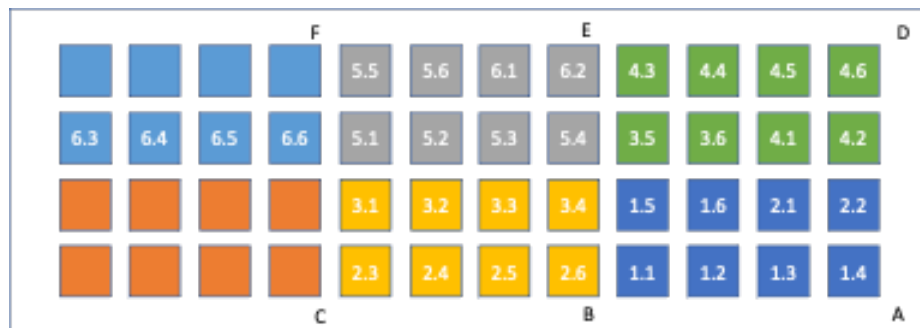


Figure 40. Wiring diagram for the translator.

the CAEN module has 48 RJ-45 inputs arranged in a grid of 4 rows by 12 columns. Within the grid, each of the 6 connectors has 8 inputs arranged in a grid of 2 rows by 4 columns, as shown in Figure 33. To read signals out from the detector modules, the RJ-45 inputs for the first detector are connected to Connector A. Beginning with the 2x4 grid starting in the lower rightmost corner of the module, the first RJ-45 cable is plugged into the lower left input of the 8 inputs in this group. The connections continue in this row of 4 from right to left before jumping to the upper row of 4 inputs and continuing left to right. The connections then go to the lower center 2x4 grid of inputs, corresponding to Connector B, and continue in the same pattern of connecting to the lower row, going from left to right, then the upper row left to right. Each RJ-45 input from each detector module is mapped in this pattern to the alphabetically advancing connector, skipping connector C.

The interface module requires approximately 8 A from a 3.3 V supply. To power the entire system, a Keysight N6700C modular rack power system was chosen. A single rack (N6700C) with two N6742B channels (8V, 12.5A for the detector module) and one N6731B channel (5V, 10A for the interface box) powers all 6 detector modules as well as the interface module.

Chapter 4. Detector Module Testing

Prior to assembling the full imager the detector modules were tested to determine the optimal threshold values for the detector modules. An initial experiment was conducted to determine the efficiency of these detector segments and compare the intrinsic efficiency of the detector modules to the manufacturer purported value. Various trigger thresholds and bias voltages were analyzed to see the impact on efficiency and compared to the manufacturer values.

4.1 Experimental Set-up

In this experiment, a ^{252}Cf neutron source was used to calculate the detector module efficiencies. Calculating the efficiency requires a reliance on a well-known source. In order to later determine the intrinsic efficiency of the detectors, the ^{252}Cf source used was first counted in the Californium shuffler at ORNL. The detection efficiency of the Californium shuffler is well known to be 16.76% with a 2% relative uncertainty, this was verified on June 6, 2019. Table 8 shows the measured values for the ^{252}Cf source used. The source had an original activity of 9.96×10^{-4} Ci at the date of creation on 3/13/07. At its last assay on 9/17/19, the source was measured at an activity of 3.71×10^{-5} Ci. The neutron yield of 99236 ± 1985 n/s was used in this experiment.

4.2 Experimental Procedure

For the purpose of the experiment, the detectors were arranged in a semicircle such that half of the ring of detector modules was created, as seen in Figure 41. The ^{252}Cf source can also be seen located at the center of where the fuel assembly would be, 57.2 cm from the inside of the Cd layer on the detectors and at a height of 56 cm from the base of the detectors. The source was placed on a ring stand and held in this location for the duration of the measurements. No collimator was used in this experiment. The detector modules were numbered from 1 to 6 from right to left around the semicircle. Detector module 1 had previously been damaged and was not used in the experiment except to moderate neutrons for better efficiency detection in module 2. Each of detectors 2-6 were counted individually to determine efficiency. A series of measurements were made with each detector module to better characterize the detector and these

Table 8. Source information and final yield values from the Californium Shuffler.

10/29/19	Single		Doubles	
Gross	16636.458	6.480	2258.183	16.148
Background	4.423	0.095	0.337	0.030
Net	16632.035	6.481	2257.806	16.148
6/6/19				
Gross	18291.538	4.58	3737.404	10.59
Bkg	4.618	0.11	0.524	0.10
Net	18286.92	5458	3736.88	10.60

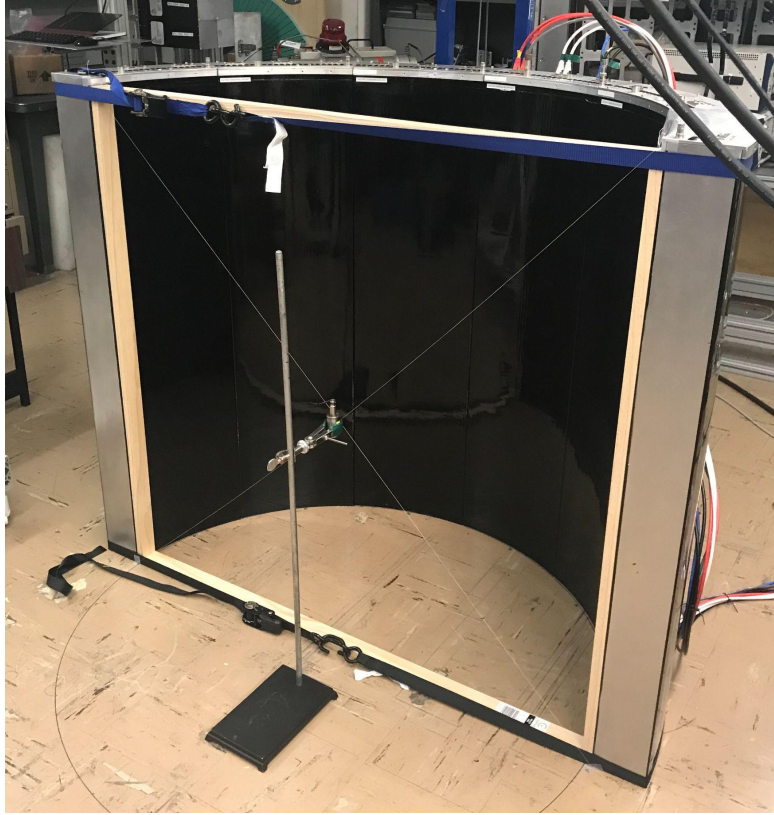


Figure 41. Experimental set-up for module efficiency testing.

measurements are summarized in Table 9.

Each detector module had a variable threshold that could be set to a specific voltage. Incoming signals that generate voltages above that trigger threshold would register in the detector as a neutron count. For each detector the lower threshold for detector triggering was changed over a series of values from 0 to 175 mV. When examining the detector modules, it was found that they had a slight offset built into the threshold, which allowed for voltages less than 0 mV to be applied as the threshold for triggering. This can be attributed to a calibration error in determining thresholds for the detectors and is not a negative polarity voltage applied to the detector threshold. The negative thresholds referred to in this paper represent the offset due to calibration error and do not represent negative polarity voltages. In addition to the measurements described in Table 9. These negative threshold values were recorded to observe the extent of the detector threshold offset. The threshold was lowered until it reached a voltage at which the detector was consistently triggering on noise rather than counts. A representative measurement was made on detector 4 to get this value and the measurements made are described in Table 10. A final comparison was made by changing the bias high voltage on the detector and observing the corresponding change in efficiency. Table 11 depicts the measurements that were taken to determine a representative efficiency curve for the change in bias voltage.

4.3 Discussion of Results

Count rates seen using the source described above were typically between 28 and 33 counts per second in each of the 24 detector channels for detectors 2-5, as seen in Figure 42. Detector 6 had a much lower count rate due to its placement around the semicircle of detectors. The steep decrease in the count rate of the detector channels in module 6 is a result of its placement in the arrangement of the detector modules. The lack of moderation from a module to the left of the detector reduces the count rate significantly as no neutrons are being moderated and scattered into the left side of the detector. In a full imager assembly, this would not be an issue as a full ring of detectors would leave no detector module with an unmoderated side. With the system that was used in this experiment, this is important to note while examining efficiencies. This was not the case for module 2 which saw no drop in the count rate because

Table 9. Standard measurements made on each detector module 2-6.

Run Number	Lower Threshold (mV)	Bias Voltage (V)	Time (s)
Background	25	700	1800
1	25	700	1200
2	0	700	600
3	25	700	600
4	50	700	600
5	75	700	600
6	100	700	600
7	125	700	600
8	150	700	600
9	175	700	600

Table 10. Negative threshold voltages to see noise triggering.

Lower Threshold Voltage (mV)	Bias Voltage (V)	Time (s)
0	700	30
-5	700	30
-7.5	700	30
-8	700	30
-9	700	30
-10	700	30

Table 11. Bias Voltage changes made on detector module 2.

Lower Threshold Voltage (mV)	Bias Voltage (V)	Time (s)
25	600	600
25	700	600
25	800	600
25	900	600

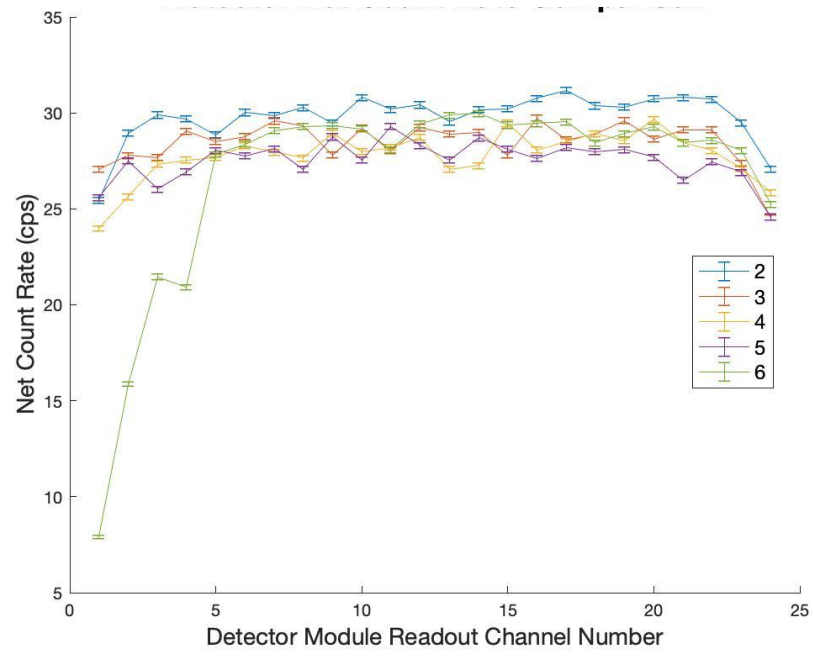


Figure 42. Net detector count rates for each detector module measured.

detector module 1 was in place in the semicircle, moderating and scattering neutrons into detector 2. As previously mentioned, detector module 1 was not operational at the time; however, had it been operational, a drop in the count rate along the right side of the detector in channels 19-24 would likely have been seen. It is expected that the drop would look symmetrical to the drop in channels 1-5 for detector module 6. The error bars in Figure 42 come from the statistical uncertainty i.e. the standard deviation of the counts.

4.3.1 Absolute Efficiency

The absolute efficiency of the detectors was determined from the count rate and knowledge of the source used. The neutron yield of the source described in Section 4.1 was used in this calculation. The calculation of the absolute efficiency for each detector module readout [43] can be described by the equation:

$$\epsilon_{absolute} = \frac{\# \text{ of neutrons detected}}{\# \text{ of neutrons emitted by source}} = \frac{\text{net count rate}}{\text{neutron source yield}} \quad (\text{Equation 14})$$

By averaging the absolute efficiency for each channel, an average absolute efficiency for each detector module can be found, as shown in Figure 43.

4.3.2 Geometric Efficiency

The geometric efficiency of the detector depends on the detector shape and design. The geometric efficiency represents the fraction of the emitted source neutrons incident upon the detector module [43] and can be written as:

$$\epsilon_{geometric} = \frac{\Omega}{4\pi} \quad (\text{Equation 14})$$

The solid angle of the detectors in this configuration can be defined as:

$$\Omega = 4\pi(1 - \cos(\tan^{-1} \frac{h}{d})) \quad (\text{Equation 15})$$

This is equivalent to taking the solid angle occupied by the top of the ring of detectors and the solid angle formed by the bottom of the ring of detectors and subtracting that value from 4π . Figure 44 is a diagram depicting the solid angles unoccupied by the detector ring and helps to illustrate Equation 15. The blue cylinder represents the ring detectors. The cones outlined in black indicate the unoccupied solid angles into which neutrons can be emitted by the source located at the connection of the two cones. The solid angle of the ring detector can then be

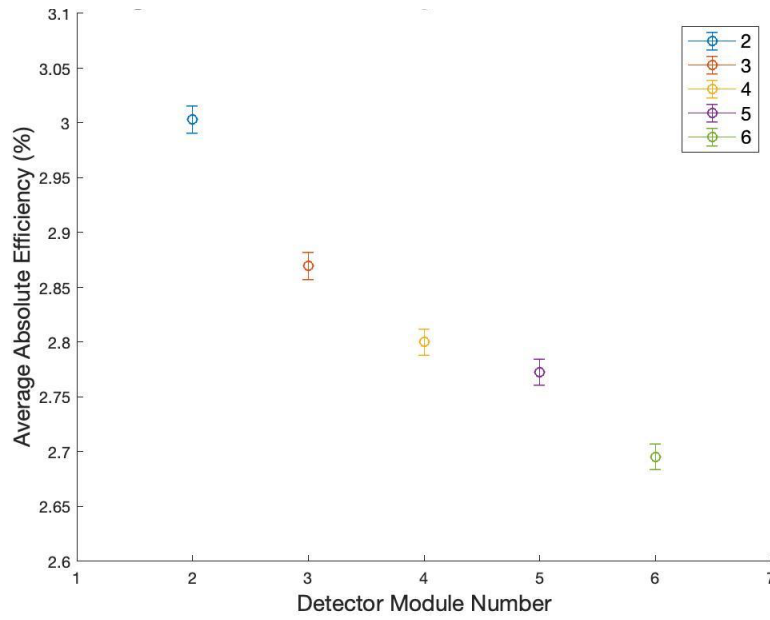


Figure 43. Average absolute efficiency for each detector module with a threshold value of 25 mV and a bias voltage of 700 V.

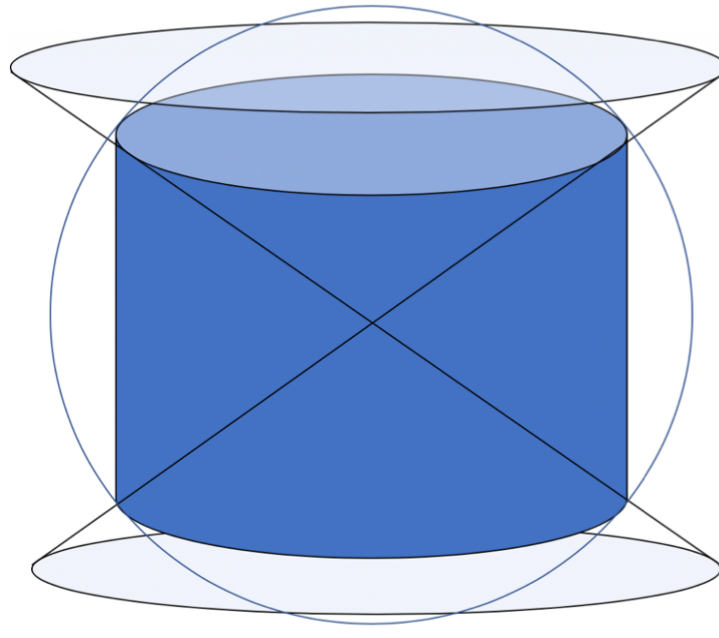


Figure 44. Schematic of the calculation of the solid angle for the space which the detector does not occupy.

defined as the total directions in which neutrons can be emitted minus the directions in which there is no detector. In this Equation 15, h represents the half height of the detector and d is the perpendicular distance from the source to the detector face.

Using these equations, the value for the geometric efficiency of one detector module can be determined to be 65.95%. Dividing the geometric efficiency for the full ring of detector modules by the number of detector modules (12), yields a geometric efficiency of 5.50% for a single detector module.

4.3.3 Intrinsic Efficiency

The intrinsic efficiency is a measure of how many of the neutrons incident on the detector are actually detected and can be written as:

$$\epsilon_{intrinsic} = \frac{\# \text{ of neutrons detected}}{\# \text{ of neutrons incident on detector}} = \frac{\epsilon_{absolute}}{\epsilon_{geometric}} \quad (\text{Equation 16})$$

Figure 45 shows the intrinsic efficiencies calculated for each detector module. The published value from Proportional Technologies Inc. is 54% intrinsic efficiency [37]. For detector modules 2-5 which have proper neutron moderation, the difference in their intrinsic efficiency values from the PTI reported value is less than 3%. Detector module 6, with insufficient moderation, has a value which differs from the PTI value by less than 5%.

The agreement between the intrinsic efficiencies found experimentally and the PTI value demonstrates that the efficiency is still comparable in the detector modules. Such detection efficiency meets the requirements to allow the proposed neutron emission tomography system to detect and image neutrons with sufficient accuracy to verify spent fuel.

4.3.4 Changing Threshold Trigger Voltage

Changing the threshold voltage for triggering the detector modules results in different efficiencies. Lowering the threshold needed to trigger the detector means that more events are being counted. To see the extent to which adjusting the threshold changed the efficiencies, intrinsic efficiencies for eight different positive threshold voltages and five negative threshold voltages were calculated. Detector module 4 was used to determine a representative detector module response to these changing thresholds, as shown in Figure 46. The 25 mV threshold used in the calculation of the intrinsic efficiencies shown in Figure 43, was an appropriate value to use

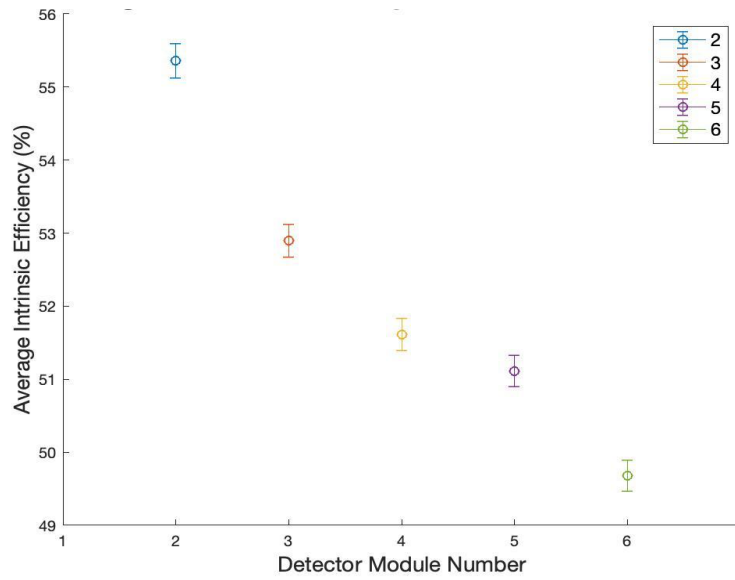


Figure 45. Intrinsic efficiency for each detector module with a threshold value of 25 mV and a bias voltage of 700 V.

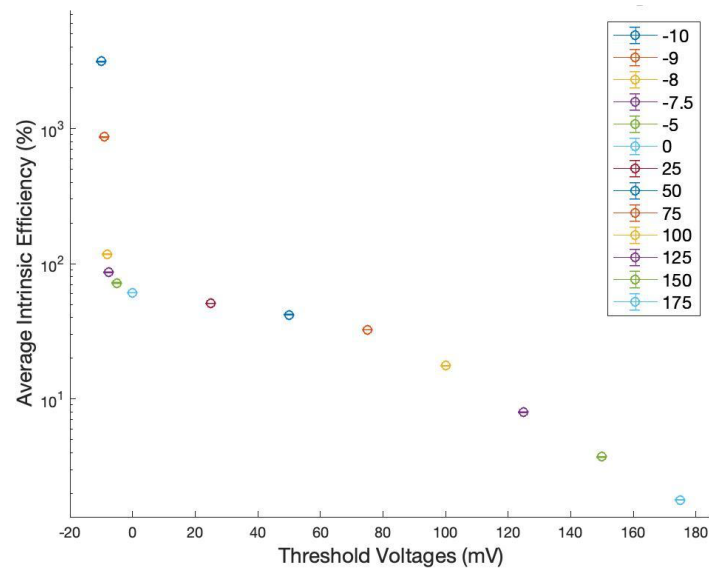


Figure 46. Intrinsic efficiency for detector module 4 with varying lower threshold voltages and a bias voltage of 700V.

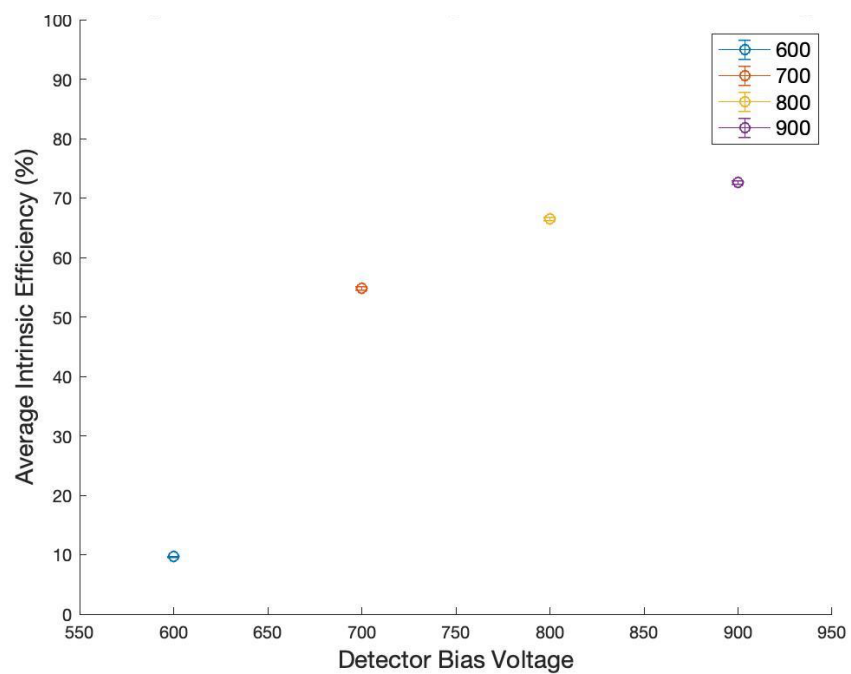


Figure 47. Intrinsic efficiencies for one detector with a threshold value of 25 mV and a varying bias voltage.

given that it yields an efficiency which is most comparable to the PTI reported value.

4.3.1 Changing Bias Voltage

Changing the bias voltage applied to the detector also changes the efficiencies. Figure 47 shows calculated intrinsic efficiencies for four different bias voltages. PTI has suggested that very high bias voltages show worse discrimination between neutron and photon interactions in the detector. This would imply that a lower bias voltage should be used. However, there is a significant difference in efficiencies between 600 V and 700 V. PTI recommended using a 700 V bias voltage for the detector modules because the efficiency was still relatively high at that value, but the photon and neutron discrimination would be much better than at higher voltages. Using a 700 V bias voltage yields results most comparable to the PTI intrinsic efficiency value for the boron coated straw detectors and all efficiencies in Figure 45 were made using this 700 V value.

4.4 Final Threshold Choices

After examining intrinsic efficiencies calculated from various trigger voltages and varying bias voltages, the intrinsic efficiency calculated under the conditions of 700 V bias voltage and a threshold trigger of 25 mV aligns with manufacturer reported value for intrinsic efficiency. Placing the boron coated straw detectors into this detector module configuration does not have a significant impact on intrinsic efficiency. The theoretical estimations for the accuracy of Neutron Emission Tomography were based on maintaining this intrinsic efficiency. The NET system, built to the designed geometry, maintains sufficient absolute detection efficiency to accurately image the fuel for verification purposes. This will lead to improved detection over present GET and support the IAEA's key mission to prevent the proliferation of nuclear weapons.

Chapter 5. Initial Measurements

A series of initial measurements were taken using the imager system. The imager system was set up as described in Section 3. The fully assembled imager and DAQ are shown in Figure 48. First, a background measurement was taken with no source present. The detectors were turned on and the measurement was taken for 58.9 minutes. These values were then used to subtract off room background from further measurements.

In order to represent the ^{244}Cm spontaneous fission source, two ^{252}Cf sources were used instead. Two ^{252}Cf sources, Cf-252-5442 and Cf-252-5443, had emission rates estimated to be 61090 ± 1222 and 6690 ± 134 neutrons per second at the time the measurements were taken, for a total of 67780 neutrons per second. The ^{252}Cf source was placed into the neutron source rod with the stopper in place at 25 cm from the bottom of the source rod. After accounting for the 1.5 cm height of the stopper, this allowed the sources to be positioned along the centerline of the imager's inspection volume. The rod was then placed into the fuel assembly mock grid. The rod was used to take measurements at grid positions corresponding to (0,0), (0,4), (2,0), (1,3), and (2,1), as can be seen in Figure 49. These positions were chosen because they allowed for examination of different source spacing. Using these, the detector could be tested to see if the reconstruction has sufficient spatial resolution to differentiate between sources located 1, $\sqrt{2}$, 2, and 4 fuel pin distances apart.

The collimator was rotated at a constant angular speed while recording data for one full rotation of the collimator. The neutron counts were then binned by collimator position and collimator positions were determined by the time it took to make one full rotation at a constant speed. For this experiment the number of data bins used, i.e., the number of projections, was equivalent to the number of slits in the collimator. The neutron counts per slit were then organized by detector number, 1 through 288, and projection number. This created a sinogram as it is typically referred to in conventional computed tomography. Details of the measurements are summarized in Table 12.

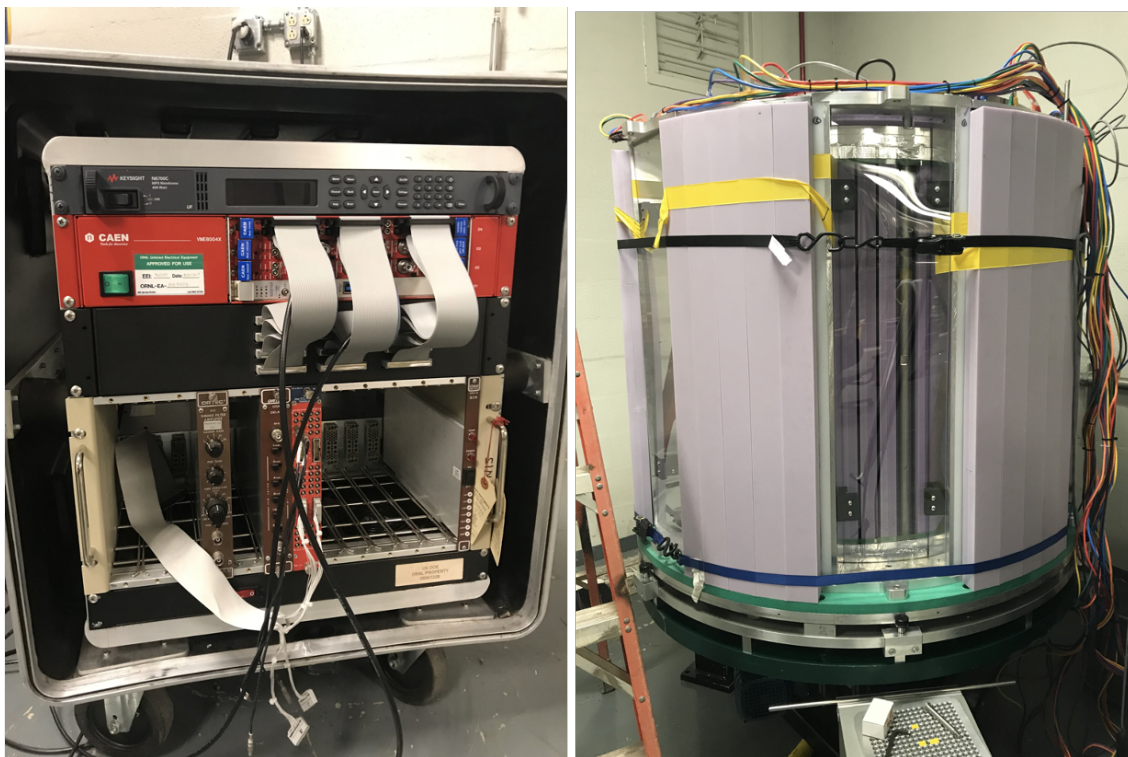


Figure 48. The DAQ and fully assembled imager system.

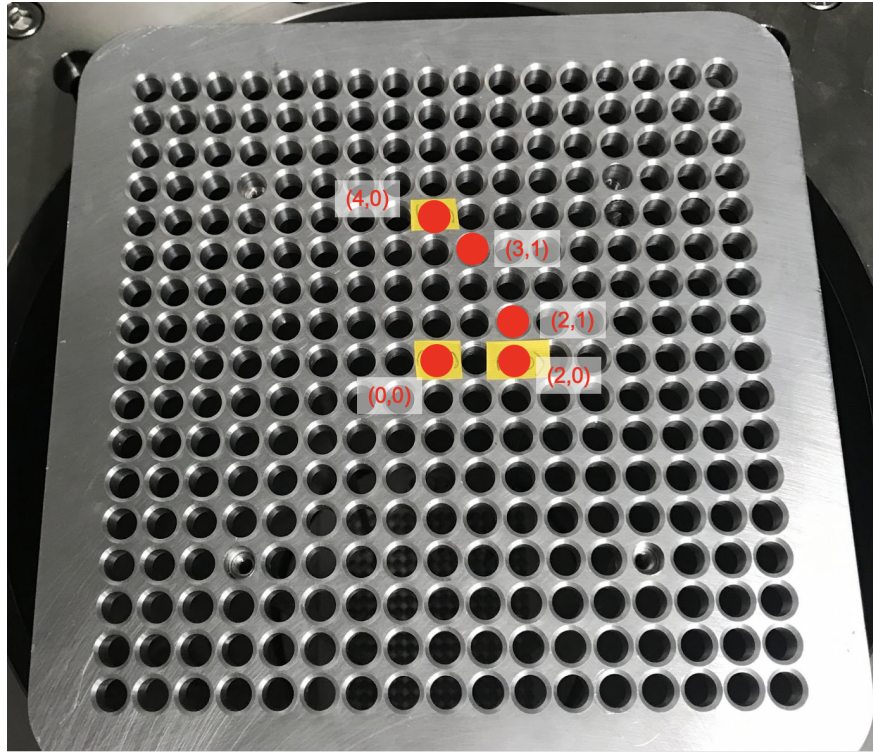


Figure 49. The grid with red labels on each fuel pin where a source measurement was taken.

Table 12. Summary of initial measurements performed.

Source slot	Date of Measurement	Measurement time (s)	Emitted neutrons per projection	Detected neutrons per projection
(0,0)	6/28/21	4520	4.26×10^6	7571
(0,4)	7/1/21	4641	4.37×10^6	7460
(2,0)	7/1/21	4518	4.25×10^6	7454
(1,3)	7/12/21	4593	4.32×10^6	7450
(2,1)	7/2/21	4606	4.34×10^6	7485

Chapter 6. Analytical Image Reconstruction Method

6.1 72 Slit Model Initial Changes

To make any potential reconstructions, the analytic response model must be adapted to match the as-built detector; meaning that the response model needed to be updated to account for 72 collimator slits instead of 96. The analytic model was first updated to include the as-built dimensions for the field of view and collimator thicknesses. Following that update, the code was changed to account for the spacing of the new slit numbers, spreading them around the collimator annulus. Next the number of positions at which the detectors were read out was changed from 96 to 72 as well, extending the rotational distance between each position. Instead of one slit for every 3 detectors where the arc length between slit openings at the detectors was previously 3.74 cm, now the arc length between slit openings is 5 cm because there is one slit for every 4 detectors. These were the minimum changes needed to update the model to a 72 slit collimator version.

6.2 Experimental Data Reconstructions

6.2.1 Single Source Reconstruction at the Origin

The first set of data to be examined was that of a single source placed at the center of the fuel assembly grid, position (0,0). After binning the data by detector number and projection, a sinogram was created for the experimental data, as seen in Figure 50(a). The data was then loaded into the reconstruction code and a forward projection of the data was created and can be viewed in Figure 50(b). The main detector signal can be seen as two lines of color moving diagonally through the images. Smaller horizontal lines, which occur at the edges of each detector module, can also be seen in these images for a total of 12 such lines in the images. These lines represent the change in detector efficiency along the edge of the detectors. Up until this point, this value has only been estimated, but moving forward the relative efficiency of each detector will be experimentally determined to properly account for the changes in efficiency for each detector. The data was back projected to generate an image. This process of forward and back projection was repeated 1000 times to generate images of the source. Figure 51(a) shows

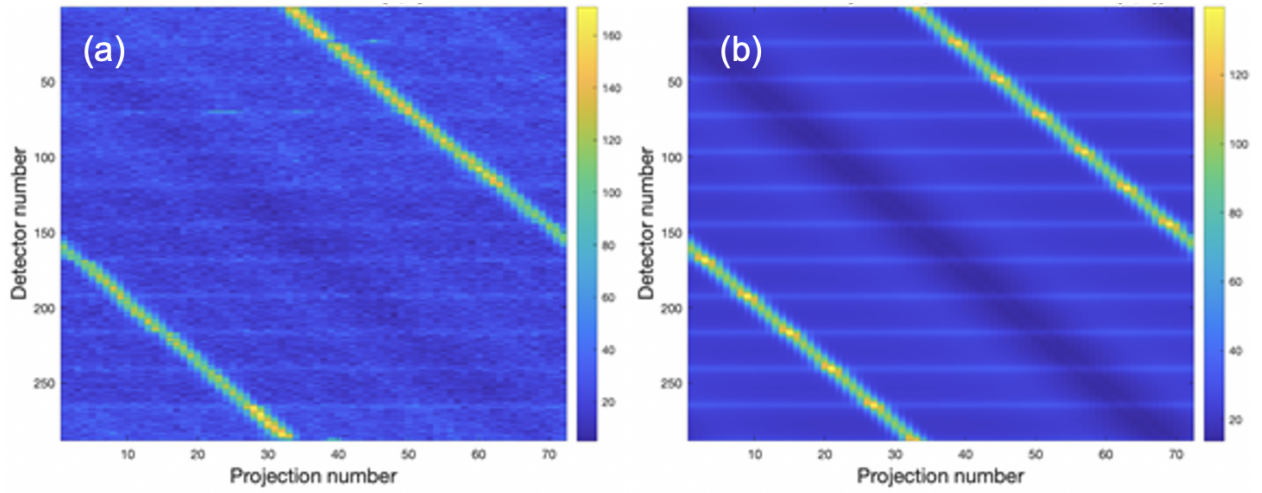


Figure 50. The sinogram generated from the experimental data with a source at (0,0) (a) and the corresponding forward projection (b).

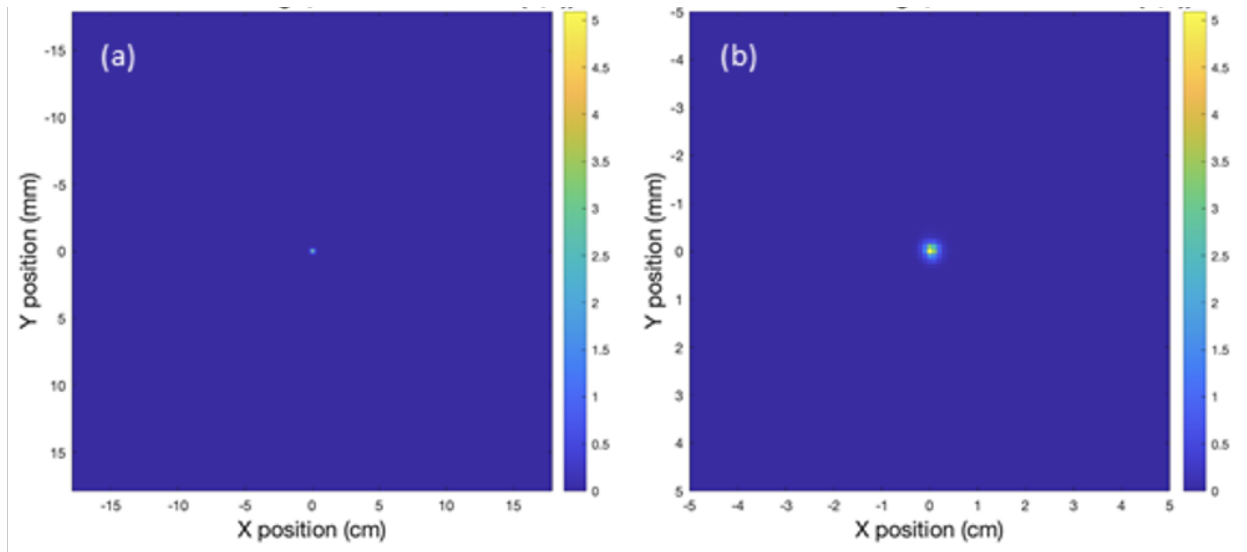


Figure 51. The reconstructed image of a single source at (0,0) (a) and a magnified version of the same image (b).

the reconstructed image of the source at (0,0) and Figure 51(b) shows a close-up image of the source. The source point is shown to be in the approximate center of the inspection volume, as it was placed in the experimental setup. No major distortions of the source shape were immediately seen upon examining the image.

6.2.2 Five Source Reconstruction

Following the reconstruction of a source point at (0,0), the individual measurements for (0,0), (0,4), (2,0), (1,3), and (2,1) were summed to represent a measurement containing five sources. Again, the data was binned by the number of detectors as well as the number of perspectives, and a sinogram was created showing data from all 5 sources. The difference between this sinogram and that of 1 source positioned at (0,0) is immediately obvious. Instead of 2 lines diagonally crossing the sinogram, more lines representing sources are present. These source lines cross over one another repeatedly and make for a more complicated sinogram than with only one source present. The sinogram data is then forward projected and the sinogram created from experimental data as well as the forward projection of the data are shown in Figure 52. After 1000 iterations of forward and back projecting an image was generated which showed all 5 sources in the volume of inspection.

Of particular interest is the image convergence, the point where the spatial resolution is sufficient such that it is obvious that each source is distinct from the others. Figure 53 shows the difference between an image generated with only 100 iterations and an image generated with 1000 iterations. In the 1000 iteration image the sources at (2,1) and (2,0) are clearly defined as separate sources. With only 100 iterations the edges between these two sources are so blurred that the separation between these two is not distinct.

This algorithm demonstrates sufficient spatial resolution to resolve individual fuel pins. However, the image does show some issues with the spatial reconstruction. Every source except the one at the origin is oblong-shaped, rather than circular, indicating issues with the simulated response. The intensities of each source should be similar, but the image shows a much greater intensity for the source at the center. The integrated intensity of the source at (0,0) is 3.85×10^6 emitted neutrons while the integrated intensity of the source at position (0,4) is only 2.28×10^6 emitted neutrons. This is a 59% difference in the emitted neutron intensity when the difference

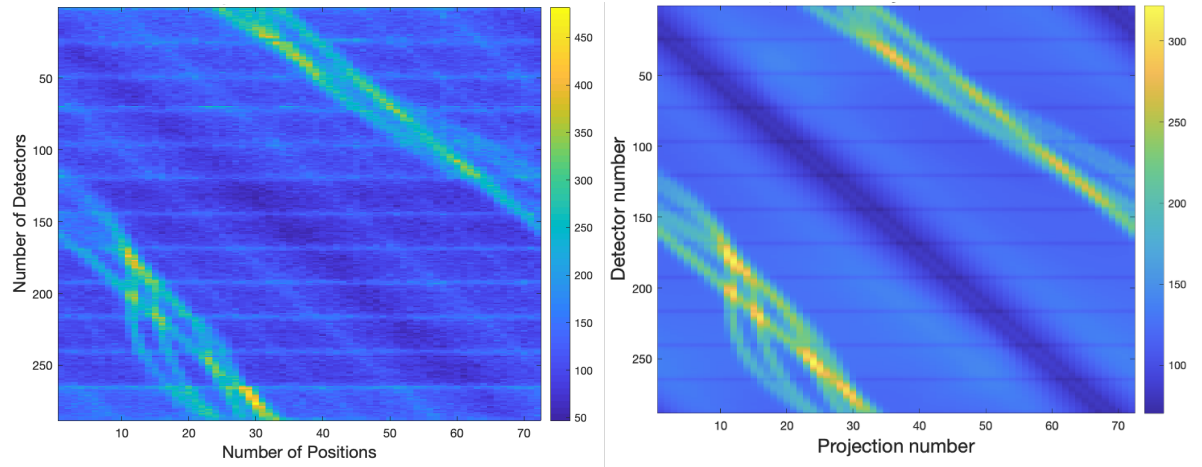


Figure 52. The experimentally generated sinogram for 5 sources (a) and the forward projection for 5 sources (b).

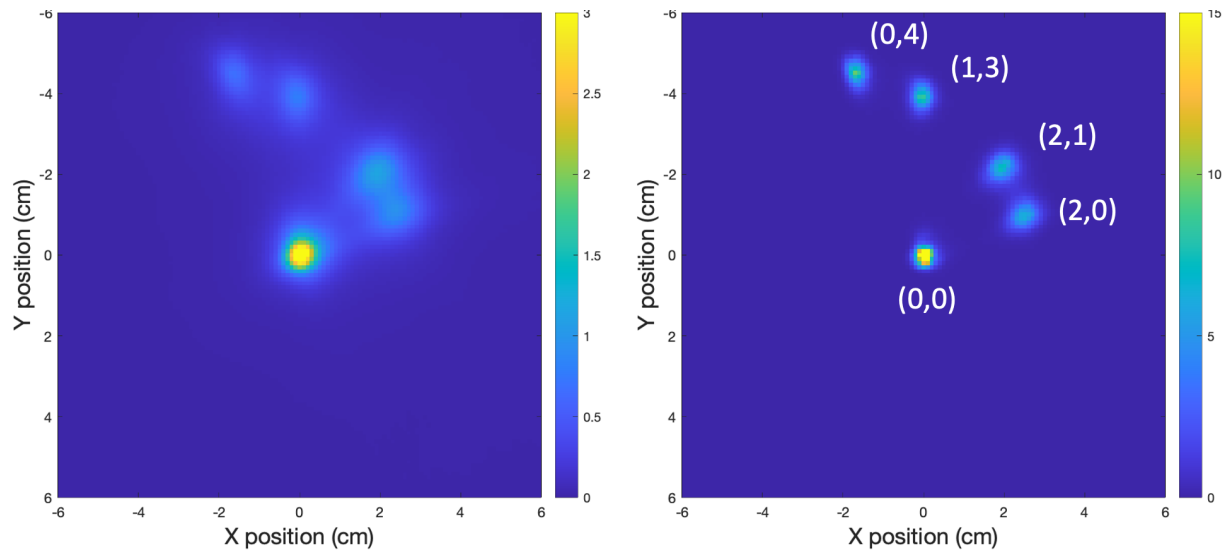


Figure 53. The difference in images created when (a) 100 iterations are used versus (b) when 1000 iterations are used.

emitted neutrons should be negligible. If all intensities of the sources are summed then the total source intensity in the image is accurate, but the reconstruction code is assigning too much of the neutron source to the source at position (0,0).

6.2.3 *Horizontal Sinogram Lines*

Following the examination of the sinograms generated by the experimental data, one difference seen between the forward projection and the experimental data occurs at the detector module boundaries. Looking at Figure 52(a) there are horizontal lines present in the sinogram which are lighter than the expected scattering contribution. In Figure 52(b) these same lines are shown but as lower than the background scatter contribution. The difference in these images is a result of neutron scattering out of the top and bottom of the imager. These neutrons bounce off the concrete floor and walls of the laboratory in which the imager is housed and are reflected back towards the detector modules. This type of scattering back into detectors from the environment is referred to as room return.

Since the decision was made to have only 6 of 12 detector modules built, these scattered neutrons interact with the sides of the detector modules. If 12 modules had been built, they would be abutting one another in the detector module ring. The sides of each of these modules would be directly next to the side of another module. Without all 12 modules, the sides of these detector modules are exposed. The neutrons coming from room return are now incident on the exposed detector module sides. These extra neutrons are recorded by the modules the same as source neutrons. These neutrons appear as “extra source” at the edges of the modules.

If all 12 detector modules had been present, there would be fewer counts at those locations. The aluminum casing between detector module edges takes up space that would otherwise have been occupied by more HDPE if the detector ring was one continuous annulus. Without the HDPE in those places, fewer fast neutrons are moderated to thermal temperatures, meaning fewer neutrons are captured in the boron coated straws at the edges of the detector modules because fewer neutrons have the correct energies to be absorbed by the straws. The forward projection in Figure 52(b) shows this impact. When looking at the sinogram, measures will have to be taken to properly account for the sinogram horizontal lines.

Chapter 7. Image Reconstruction Improvements

After determining that the imager would have sufficient spatial resolution to differentiate between individual pins, interest turns to reconstructing a mock full fuel assembly to complete the proof-of-concept demonstration. When adding more than 5 sources to the reconstruction, the distortion in the fuel pin reconstruction and the misattribution of the source intensity toward the center of the FOV becomes problematic. Prior to trying to make a full fuel assembly measurement and reconstruction, the issues seen in the reconstruction of 5 sources must first be addressed.

The analytic neutron response model does not match the as-built detector system. In order to make the previous reconstructions certain aspects of the analytic neutron model were updated to match the as-built system while others retained aspects from the original 96 slit model. The analytic response code was updated with the new 72 slit geometry as well as an updated 72 slit collimator thickness and relative efficiency. However, the neutron response model does not account for all the differences that may be seen in neutron response due to changes in the as-built detector. These changes include the impact of the missing detector modules and impact of any structural supports added during construction. The relative efficiency of each detector readout is going to be different from an idealized detector seen in the simulated model. The collimator thicknesses are going to change as well due to construction difficulties. Instead of performing many simulations to match the as-built detector module, measurements using the as-built detector system were taken in order to have data to fit the neutron response model to the real imager.

7.1 Simulation Verification

In hoping to improve the response for the analytic model of the 72-slit collimator, a few changes had to be made within the code to better reflect the as-built imager.

7.1.1 Edge and Direct Scattering Rotation Effects

One of the things that had to be considered when examining the differences between the ideal detector and the as-built is how the as-built detector continuously rotates. Initially, the analytic code set the number of positions at which the imager evaluated the source to the same as

the number of collimator slits, 72. When the analytic code calculates the position at which it evaluates the system it goes directly to that exact position and effectively gets a “point-and-shoot” measurement. In this way the positions where the analytic code calculates the lines of response are in very distinct positions with 5-degree rotations in between these positions. This is different from the way in which the as-built imager operates. The imager is constantly rotating and taking data and that data is binned into sections approximating the imager positions. Due to the constant rotation of the imager, there is a difference between the measurement data that the imager records across the 5 degrees of rotation and attributes to one position and the lines of response data that the analytic code creates for the one position that is 5 degrees different from the previous position. In order to try to remedy this issue, the position sampling for the analytic code was increased. Instead of sampling 72 times for 72 positions, the analytic code sampled the data 288 times or 4 times as often as previously. This means that the analytic code now samples at every 1.25 degree. Due to memory constraints with the code used to create the analytic response and reconstruct measured data, the 288 positions needed to be paired down to get an averaged 72 positions. Of these 288 positions, 4 of them were averaged together at a time to get one position.

7.1.2 Effective Collimator Thickness Change

After incorporating the new changes into the detector module, certain other characteristics had to be adapted to ensure the model was representative of the constructed system. Using an MCNP6 simulation model, a new effective collimator thickness could be created in the same manner as described in Section 2.7.3, but for a system with 72 collimator slits, instead of the original 96. First, a simulation placing a source at the center of the field of view was run, followed by a simulation of the same source at the center, but with no collimator present. Examining the ratio between these two simulations yielded the mean free paths through the collimator that neutrons would travel to be detected in each detector. This data was then fit to two curves using a polynomial fit to get a smoothed version of the data which was then incorporated into the analytic response code. The effective collimator thickness is used to simulate the collimator penetration values of neutrons as well as the scattering of neutrons into the detectors through collimator slits.

7.2 Measurement Verification

To improve the neutron response to match the as-built imager, a series of verification measurements were performed. These measurements will help further characterize the as-built system and improve image reconstruction by defining parts of the neutron response used in the reconstruction code in terms of measured values instead of simulated values.

7.2.1 Neutron Source

The neutron source used in this experiment was measured originally on Sept. 9 2021. The source was 17.256 wt% Cf-249, 18.815 wt% Cf-250, 8.133 wt% Cf-251, and 54.76 wt% Cf-252. The original activity of this source was 9.29×10^{-4} Ci or 34373000 Bq. The measurements were taken from August 4, 2022 through August 25, 2022 so the date of August 1, 2022 was used to represent the approximate neutron emission during this first measurement campaign. The value for neutron emission rate used in this experiment was 3.173×10^6 n/s. It was decided to set the collimator's rotation speed to 1 revolution/minute.

7.2.2 Efficiency Verification

The first measurement was an efficiency calibration for the detectors. Currently, the analytical model uses a calculated estimate for all 288 detector response efficiencies. In order to get a more accurate calibration the neutron source was attached to the inner edge of the collimator, aligned with the slit which is directly perpendicular to the inspection volume. This gave the neutrons a direct path to the detector opposite that slit. The collimator was then rotated at a constant rate, dosing each detector evenly. The efficiency of each detector was calculated, and this value used in the source reconstruction. The source location along the collimator wall and its rotation are displayed in Figure 54.

The efficiency of the system was measured as described and the results were compiled into an initial sinogram. This sinogram is shown in Figure 55. Immediately portions of the sinogram stand out. There are six separate horizontal regions in the sinogram which are emphasized by how drastically the source strength intensity and background change across these regions. As demonstrated in Chapter 4, the neutron detector modules do not have the same

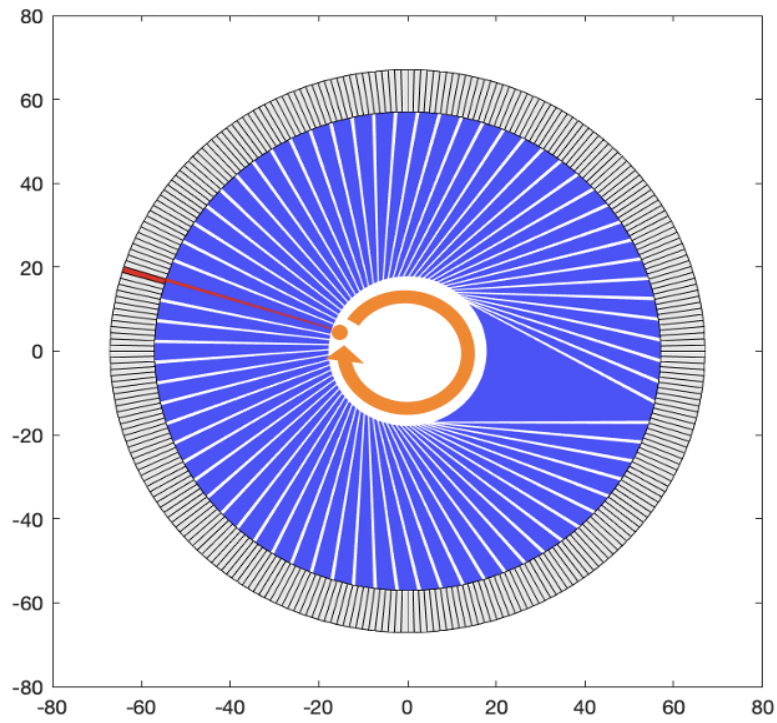


Figure 54. Efficiency measurement diagram.

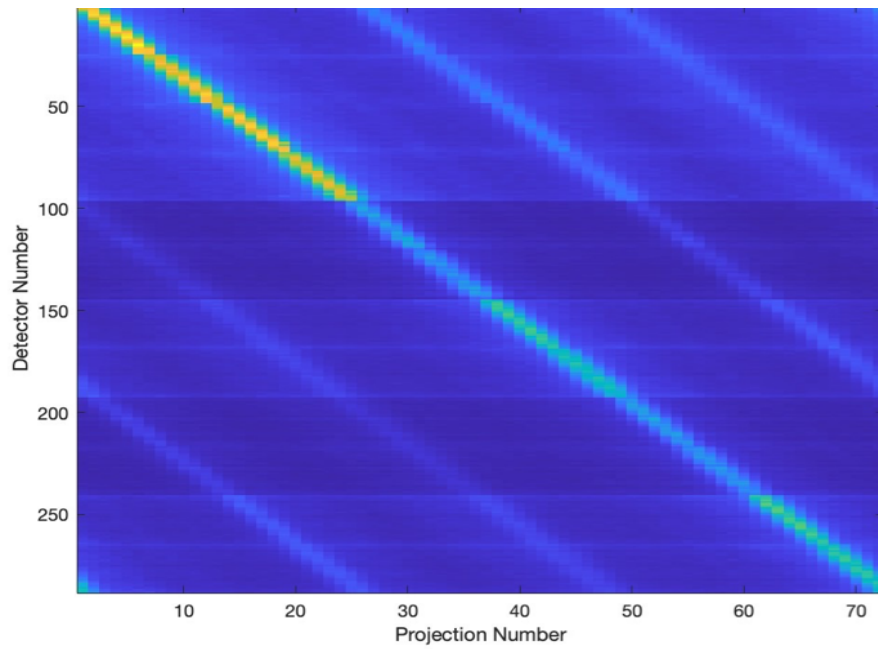


Figure 55. Uncorrected efficiency sinogram

efficiency across than when set to the same high voltage and threshold. In order to compare gains of each detector individually, a correction factor needed to be applied across each module. The correction factor was determined by dividing the average counts across all modules and dividing the average counts across each module. This resulted in a correction factor for each detector module. Figure 56 shows the sinogram after correcting the modules.

To get the relative efficiency values, the source intensity at each position was first calculated and averaged over all positions. The background counts at each position were also determined and subtracted from the total sinogram. This resulted in relative efficiency values which looked like those in blue in Figure 57. The red values are the calculated relative efficiency from simulation. That these measured values are almost the inverse of the simulated values pointed at an issue with the measurement. When only 6 detector modules were created instead of 12, aluminum detector spacers were used to hold the place of the missing detector modules. Without these detector modules in place, the edges of the detector modules which should be abutting are exposed. This allows neutrons to bounce off the walls in the room and scatter into the sides of the detector modules. This phenomenon of neutrons scattering off the surrounding environment and bouncing into the system is referred to as room return. Figure 58 shows how the neutrons escape from the exposed top and bottom of the imager and bounce around in the room, ultimately some of them scattering into the detector modules. These neutrons coming from room-return are creating the peaks in the relative efficiency.

In order to get the true relative efficiency, the neutrons from room return must be subtracted from the system. First the average signal values are calculated for each detector; this includes the average values at which each detector was examining the source, as well as when it was not. Following this, the average background values were calculated by determining the average counts per detector for the detectors which are not examining the source. This included the average values for the room return. By subtracting the average background from the average signal values, the average source counts could be determined. These average source counts were then used to determine the relative efficiency values. Figure 59 shows the difference between the average signal values, average background values and the average source counts. Figure 60 shows the corrected relative efficiency values coming from measured data versus the simulated relative efficiency values. In this case there is much better agreement between these values than

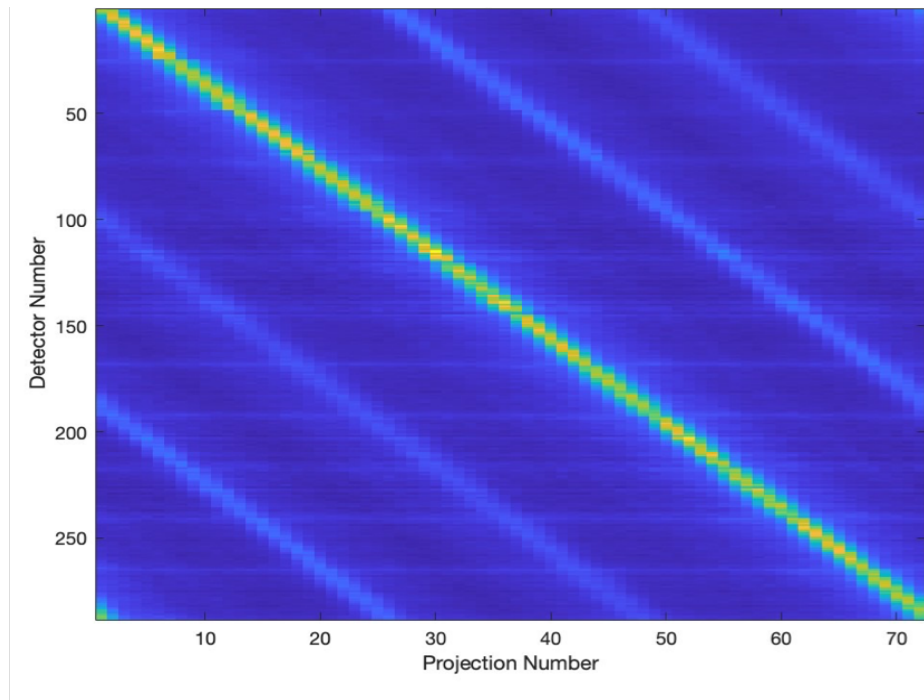


Figure 56. Module efficiency corrected sinogram.

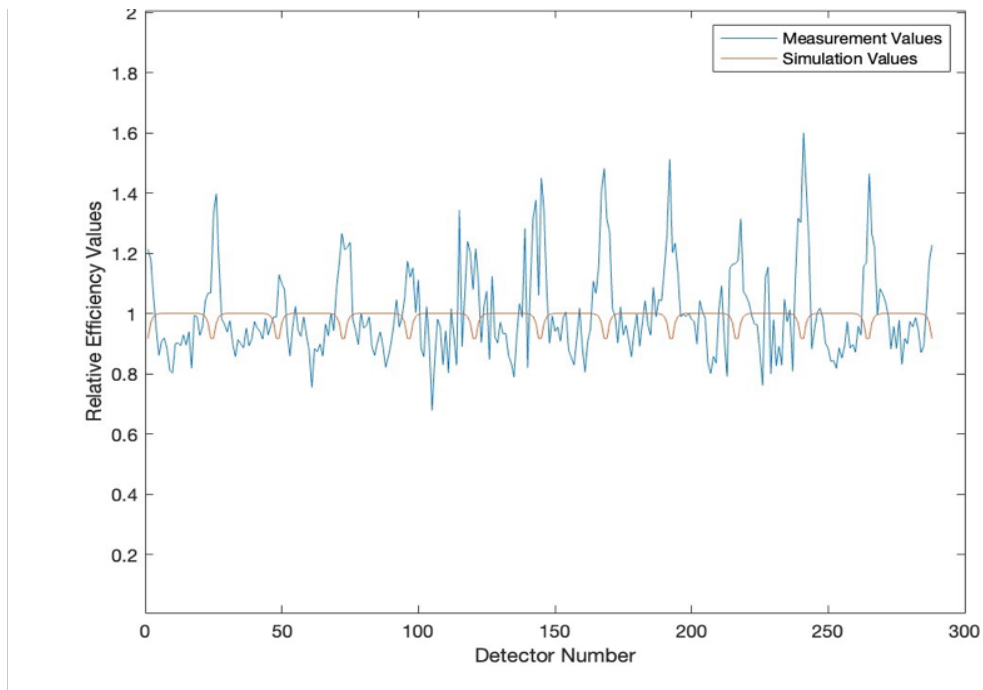


Figure 57. Relative efficiency values from simulation and measurement.

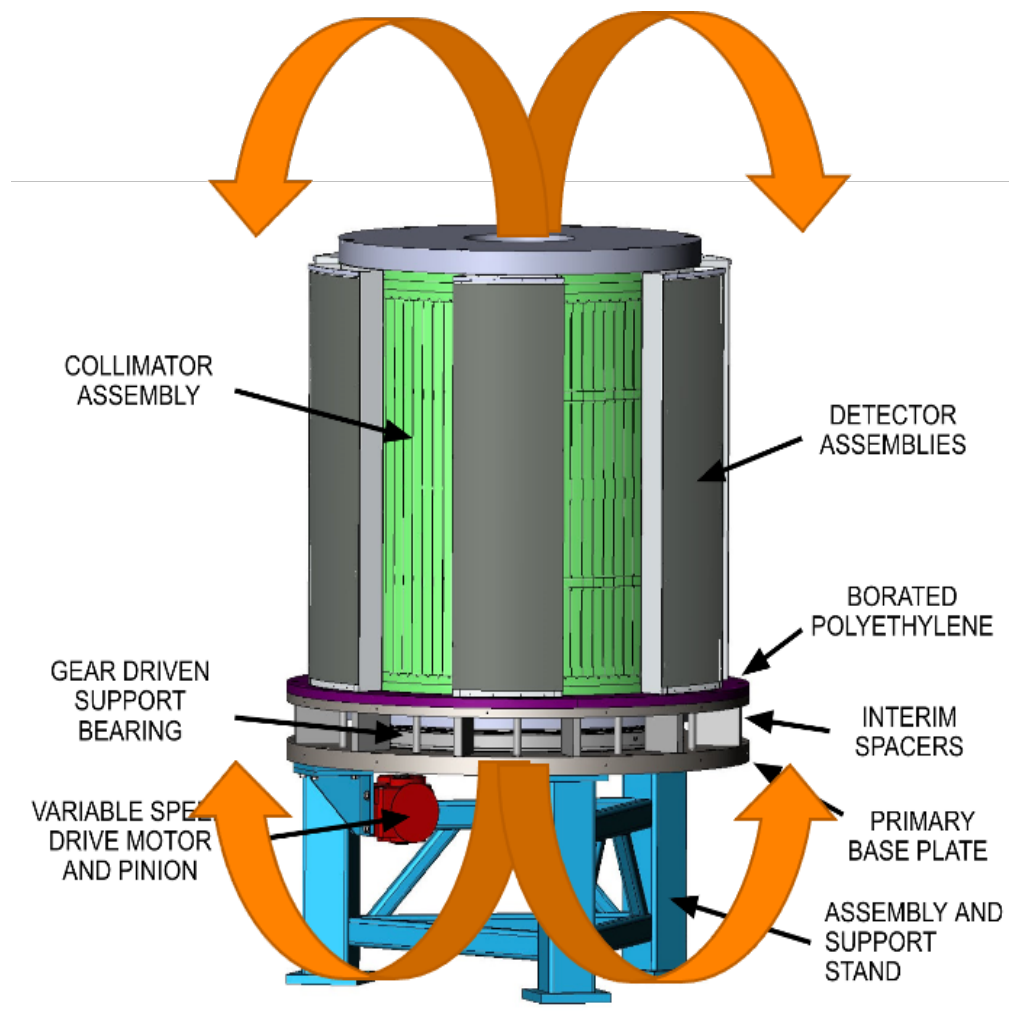


Figure 58. Room return diagram.

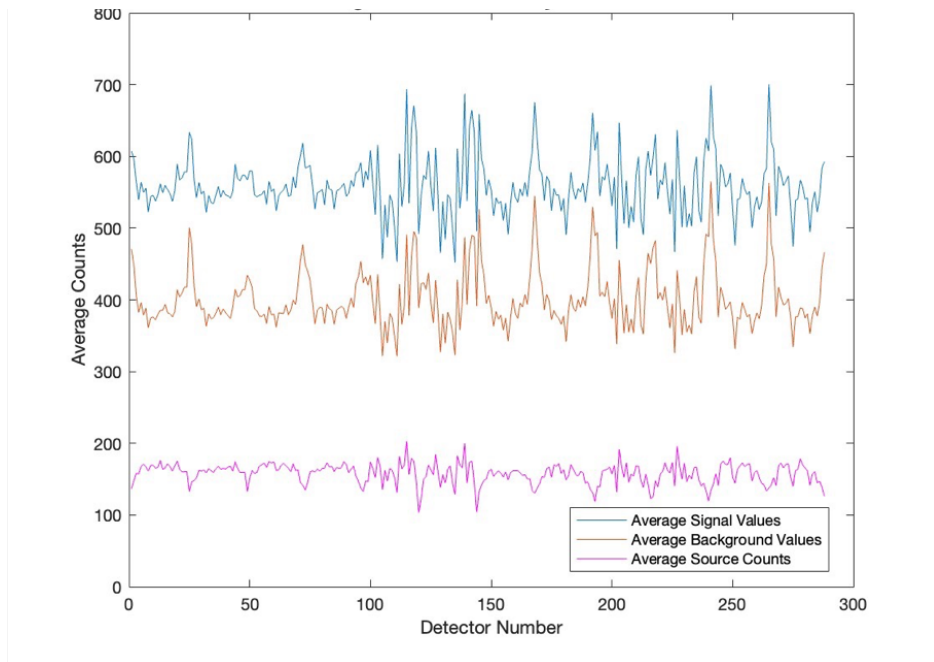


Figure 59. Average counts in efficiency measurement.

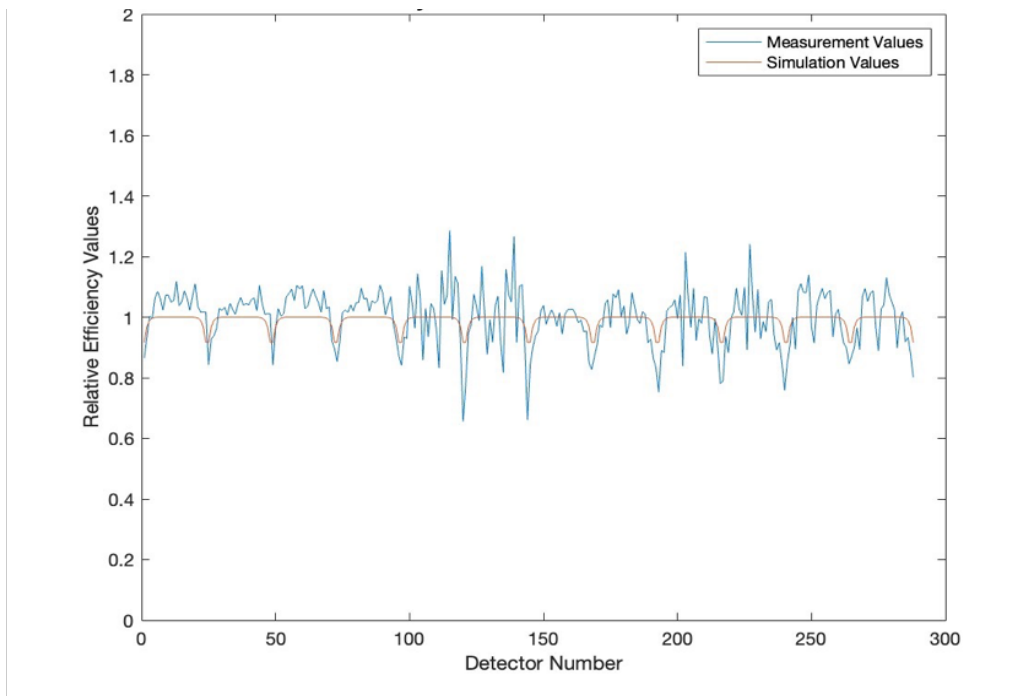


Figure 60. Relative efficiency values from simulation and measurement.

in Figure 57. The new relative efficiency values still have more variation across the 24 rows of detectors than in the simulated version, but this variation makes sense as each of the detectors has its own gain which must be accounted for. In order to apply this new efficiency in such a way that no new high frequency issues are introduced, a cubic polynomial was fit to the relative efficiency of each detector module. This resulted in smoother relative efficiency values as can be seen in Figure 61. These smoothed relative efficiency values are applied to the analysis of all further measurements.

7.2.3 *Measurement-Guided Effective Thickness Changes*

An initial measurement was taken with a source placed at the center of the inspection volume, also referred to as the origin point or fuel pin position (0, 0). For this measurement the source is placed in the center of the reconstruction volume and data taken with the collimator rotating. Data was recorded for an hour at detector position 1 and an hour at detector position 2. The sinogram created from the data origin measurement, pictured in Figure 62, immediately showed some interesting things occurring in the sinogram, one of which being the echoes of the source along various diagonal lines. These additional diagonal lines are located at 13 slits and 25 slits away from slit 37 where the source is being detected. Originally it was considered that there was a possibility that these slits were wider than the surrounding collimator slits. In reality, after examining how the imager was constructed, these locations are actually where there were stainless steel tie rods placed to support the imager. In constructing the imager, 6 of these 1" tie rods were evenly spaced around the imager to support the top and bottom of the collimator. At these places the collimator is physically shorter than in the other locations. Because there is physically less collimator in these places, there is less neutron attenuation happening around these slits. As a result, more neutrons are detected in the detectors surrounding these tie rods, leading to the diagonal lines.

In order to account for these lines, the total measured neutrons were projected into the slit axis as shown in Figure 63. At this point a fit was created to the peaks at 13, 25, 37, 49, and 61. The fitted values were used as a multiplier of the effective thickness. In these locations there was a different effective collimator thickness than originally expected, and this was a constant in the collimator, meaning the correction had to be applied before the lines of response were created.

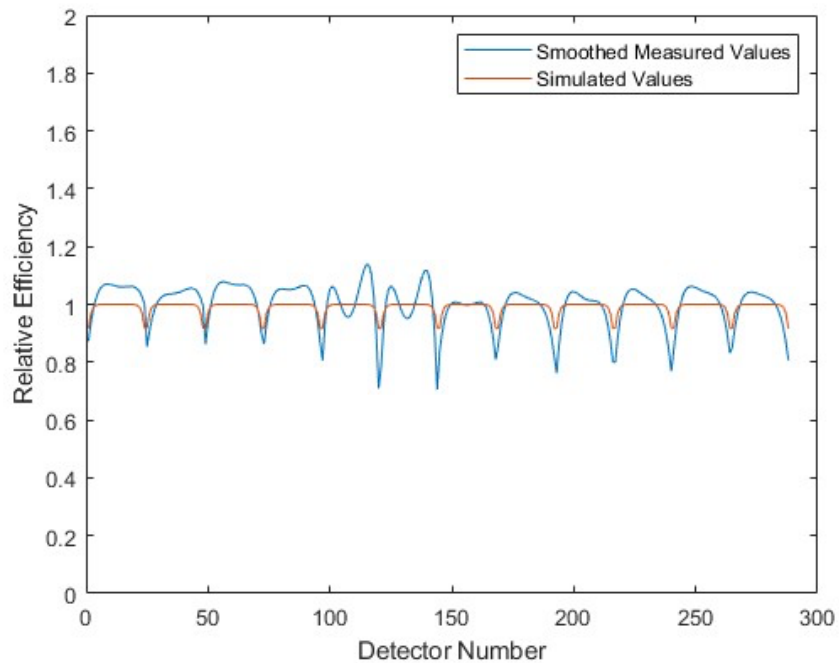


Figure 61: Smoothed relative efficiency values coming from measurements.

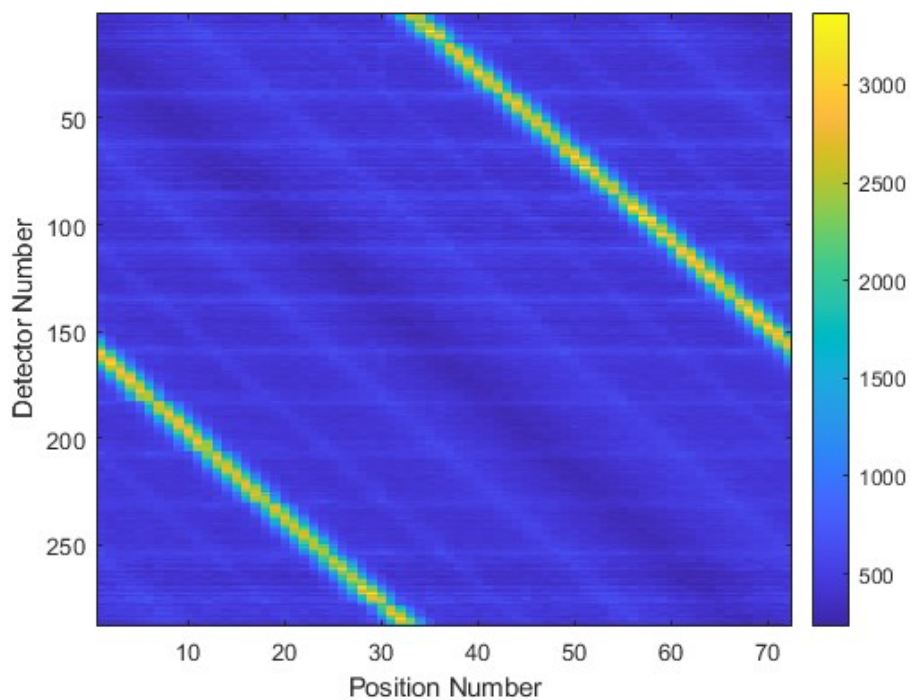


Figure 62. Sinogram created from the measured data with a point at position (0,0).

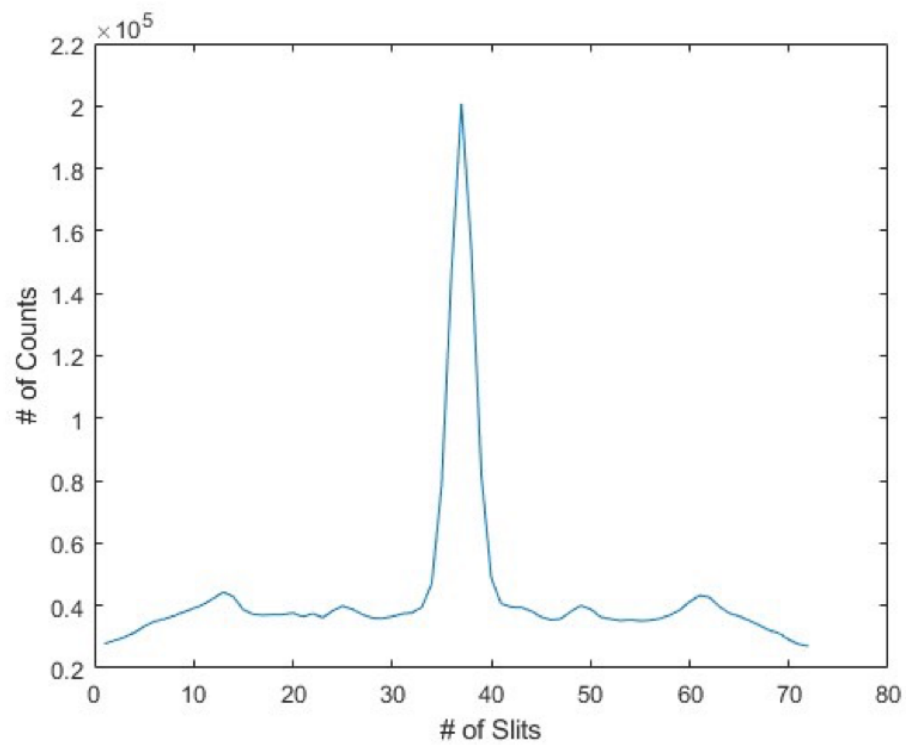


Figure 63. Number of counts per source position for a source located at the center of the FOV.

This could not be fixed afterwards; rather, it was better to carry this change through the analytic code. The new effective thickness was applied to each part of the collimator to get the analytic response to match the as-built model.

7.2.4 Angular Code Changes

One required change to the code was the addition of angular dependences. The slit scattering portion of the code needed to include a directional component indicating the angle between the collimator slit and the line defined by the detector and the center of the volume. The directional factor was added in the form of a cosine of the angle defined by the dot product of the unit vector from the detector to the center of the FOV and unit vector from the detector to the slit opening. Geometrically, adding this factor accounts for the angle and direction of the slit, which is different from geometry dealing with the slit opening to the center alone. This factor of slit length and direction accounts for how neutrons are detected in the system, rather than simply how neutrons enter the slit of interest.

7.3 Representative Reconstructions

Following changes to the code, a series of characteristic reconstructions were made using the two representative points and the five points previously reconstructed in Section 6.2.2. Reconstruction images were made using 357 pixels by 357 pixels to create the image. Each pixel represents 1 square millimeter in the inspection volume. Given that the gaps between fuel pins are 3.28 mm, a pixel resolution of 1 mm was chosen to ensure that the pixel resolution is small enough to image the fuel pin gap.

7.3.1 Inspection Volume Sampling

The number of lines of response which intersect in a particular point as the collimator rotates define how well each position is sampled by the system. Figure 64 shows where the centerline of each line of response crosses the inspection volume for one collimator position. As the analytic model rotates and changes collimator position, different locations within the inspection volume are sampled by the collimator's lines of response. However, the analytic

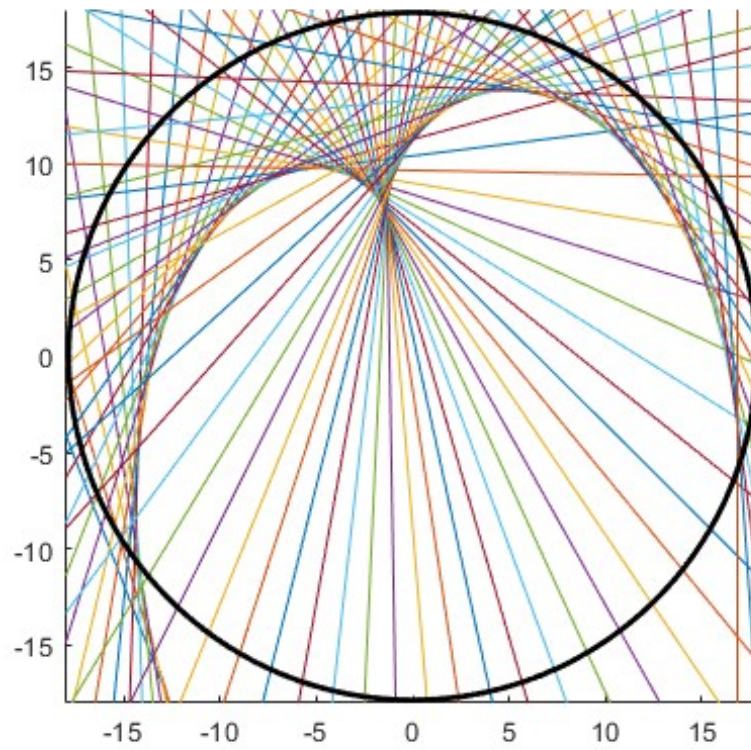


Figure 64. Lines of response centerlines in across the field of view

model does not sample all locations in the FOV equally.

In order to understand the extent to which the sampling differs radially, a calculation based on the direct and edge transmission portion of the analytic neutron response model was performed to determine how often each pixel within the inspection volume was sampled for 288 projections. Section 2.7.2 details Equation 12 describing how the direct and edge transmission portions of the analytic neutron response are calculated. Equation 12 uses two factors, labelled f_1 and f_2 , to determine if the location of interest is within the direct path of transmission for each collimator slit or within the slit edge region where the neutron will still manage to traverse the collimator slit and reach the neutron detectors. If f_1 and f_2 are both equal to 1 then the location of interest within the inspection volume is completely within the direct path of transmission. If one factor is equal to 1 and another is less than 1 but greater than zero, the location of interest is within the edge region and a fraction of neutrons emanating from this position will still traverse the slit. If both f_1 and f_2 are equal to zero, then the location of interest is fully occluded and neutrons emanating from the location will not travel down the slit as defined by direct or edge transmission.

For the sampling calculation, two 357x357 variables were created to hold counting information about the number of times each location of interest, namely each 1 mm by 1 mm pixel, was sampled. The first variable counted how many times each pixel had factors f_1 and f_2 equal to 1 for each collimator slit for 288 collimator positions. This is the number of times that any given pixel was fully within the direct transmission path of any collimator slit. This is also the number of times that each pixel is directly sampled by the collimator slits for 288 projections. Figure 65 shows the values for direct sampling frequency across all pixels. The center of the inspection volume, the pixel located at (179, 179), is sampled the most frequently with 288 samples of that pixel across the 288 projections. The amount of sampling immediately drops off radially from the center to values which fluctuate from 35 to 70 direct samplings per pixel. At a radial distance of 6.4 cm from the center of the inspection volume, the amount of direct sampling begins to increase and continues to do so until a radius of 16.7 cm from the center of the inspection volume. The sampling then drops off again over the last 1.2 cm from the 16.7 cm peak to the edge of the inspection volume. Figure 66 shows a plot of the sampling happening in a cross section of the center of the FOV by holding the y-value of the pixel constant at 179 and

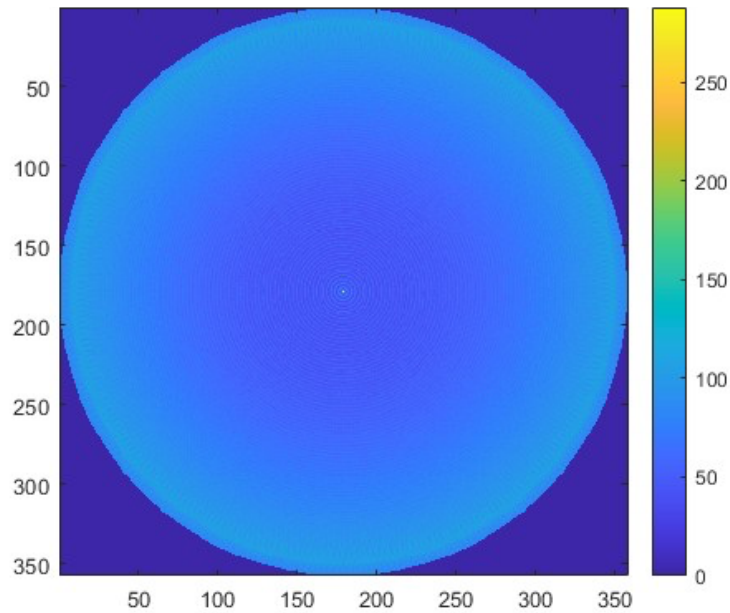


Figure 65. Image showing the number of times each pixel within the inspection volume is sampled directly for a complete analytic model with 288 projections.

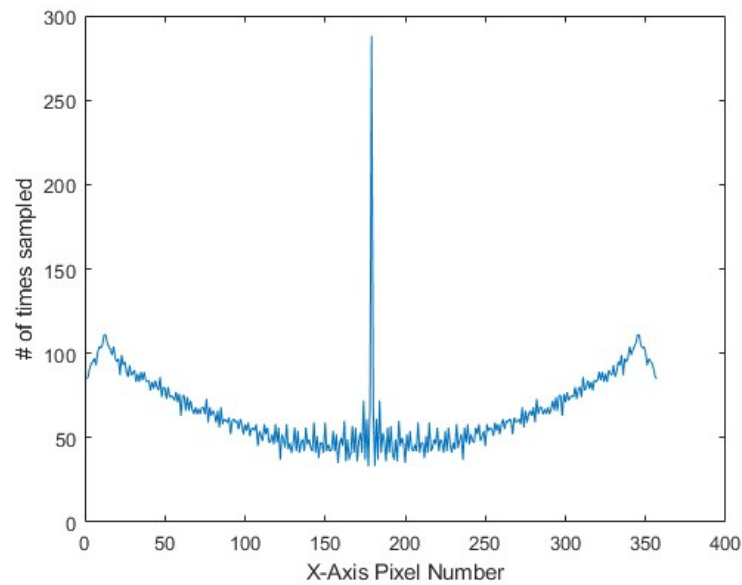


Figure 66. Plot of the cross section of Figure 65 depicting the changes in direct sampling frequency along the line where the y-value of the pixel is 179 and the x-value changes from 1 to 357.

changing the x-values for the pixels from 1 to 357. The cross section emphasizes the difference in the radial sampling from the center at pixel 179 to the edges of the FOV at pixels 1 and 357.

The second variable counted how often each pixel was located within the edge region of a slit for all 72 slits for 288 collimator positions. These would be places where either factors f_1 or f_2 were greater than zero but less than one. This determined how often each pixel was edge sampled. Figure 67(left) shows the edge sampling frequency for all pixels. In this image, the pixel (179, 179) is sampled 576 times, but unlike for the direct sampling, this is not the most frequently sampled edge location. There is instead a ring around pixel (179, 179) where the most sampled locations occur. Figure 67(right) zooms in on this ring which extends from a radius of 2 to 5 mm from the center pixel and includes 6 locations where the pixels are sampled a maximum number of 766 times. These locations are at pixels (174, 177), (176, 179), (177, 184), (181, 174), (182, 179), and (184, 181) and are marked with black outlines around the pixel. After this 5 mm radius, the sampling of each position decreases radially to the edge of the FOV. Figure 68 shows the cross section where the y-value pixel is 179 and the x-value changes from 1 to 357. The cross section shows a steep drop off in edge sampling over the 2.5 cm distance between a radial distance of 15.9 cm and the edge of the FOV.

The sum of the edge and direct sampling per pixel is shown in Figure 69. When both the direct and edge sampling are considered, the origin at pixel (179, 179) has the highest sampling frequency across all pixels in the FOV. The frequent edge sampling of the ring of pixel locations surrounding the origin makes the pixels located within 5 mm of the origin the highest sampled pixels in the system. The lowest sampling frequencies are around the outer edge of the FOV and there is a steep decrease in sampling rate from a radial distance of 16.6 cm to the edge of the FOV at 17.8 cm. This can be better seen in Figure 70 where the cross section of the sampling frequency is shown across the center of the FOV. Again, this figure is created by plotting the sampling frequency at the pixels with an x-value from 1 to 179 and a constant y-value of 179. With less sampling of a particular location comes higher uncertainties and lower spatial resolution since there is less data available. Places where the sampling frequency is the highest would have the best spatial resolution whereas places where the sampling is the lowest would have more uncertainty.

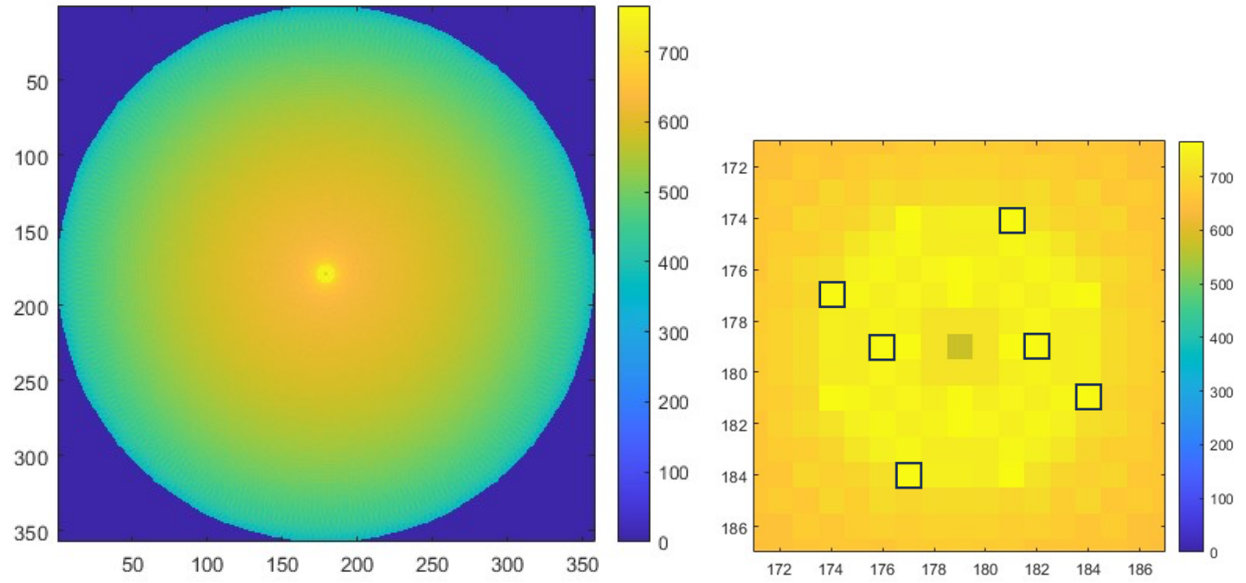


Figure 67. Image showing the number of times each pixel was within slit edge region of a collimator slit for 288 projections (left) and a zoomed in version of the image focusing on the center point (179,179) and the surrounding ring of increased sampling with the highest sampled pixels outlined in black (right).

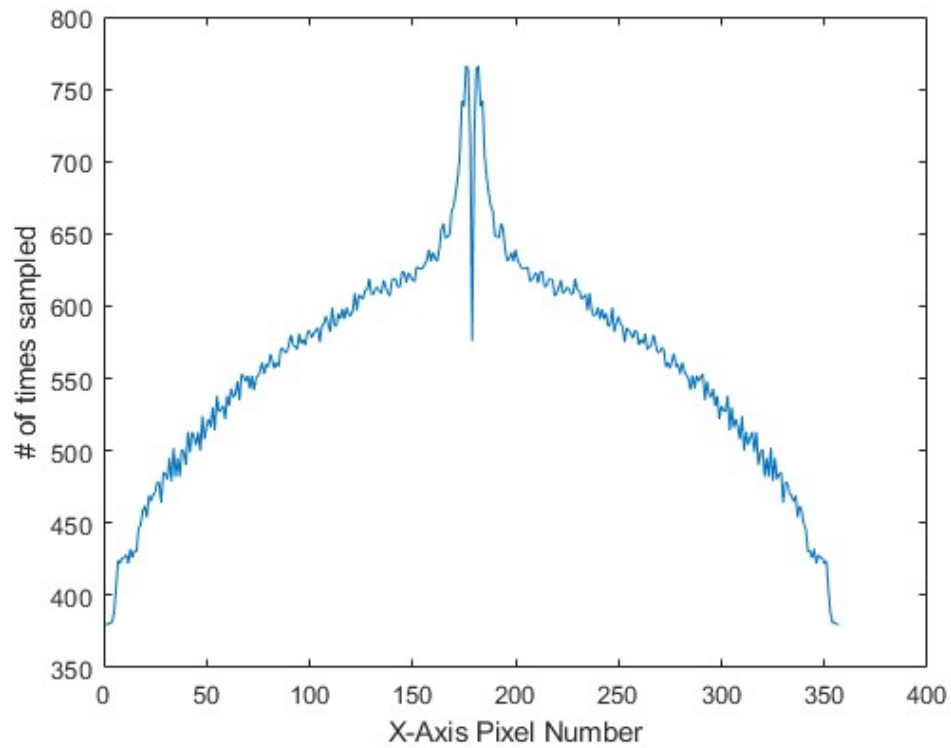


Figure 68. Plot of the cross section of Figure 67 depicting the changes in edge sampling frequency along the line where the y-value of the pixel is 179 and the x-value changes from 1 to 357.

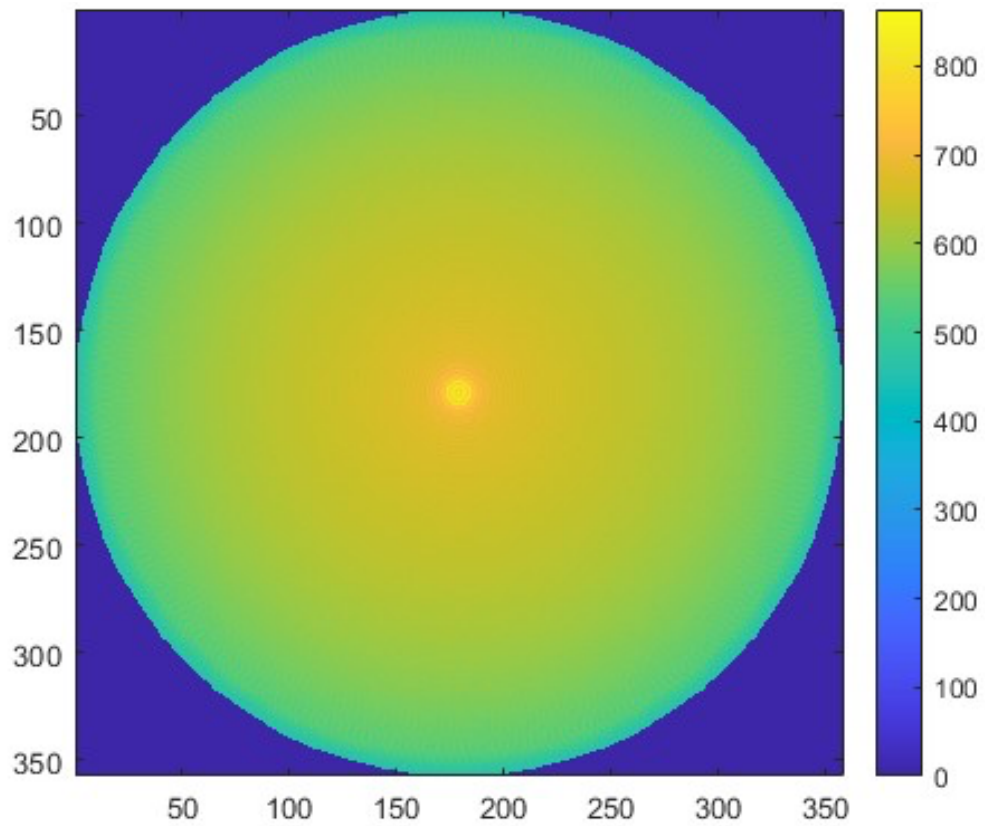


Figure 69. Image showing how many times each pixel was included within a direct or edge scattering path for any collimator slit summed over 288 projections.

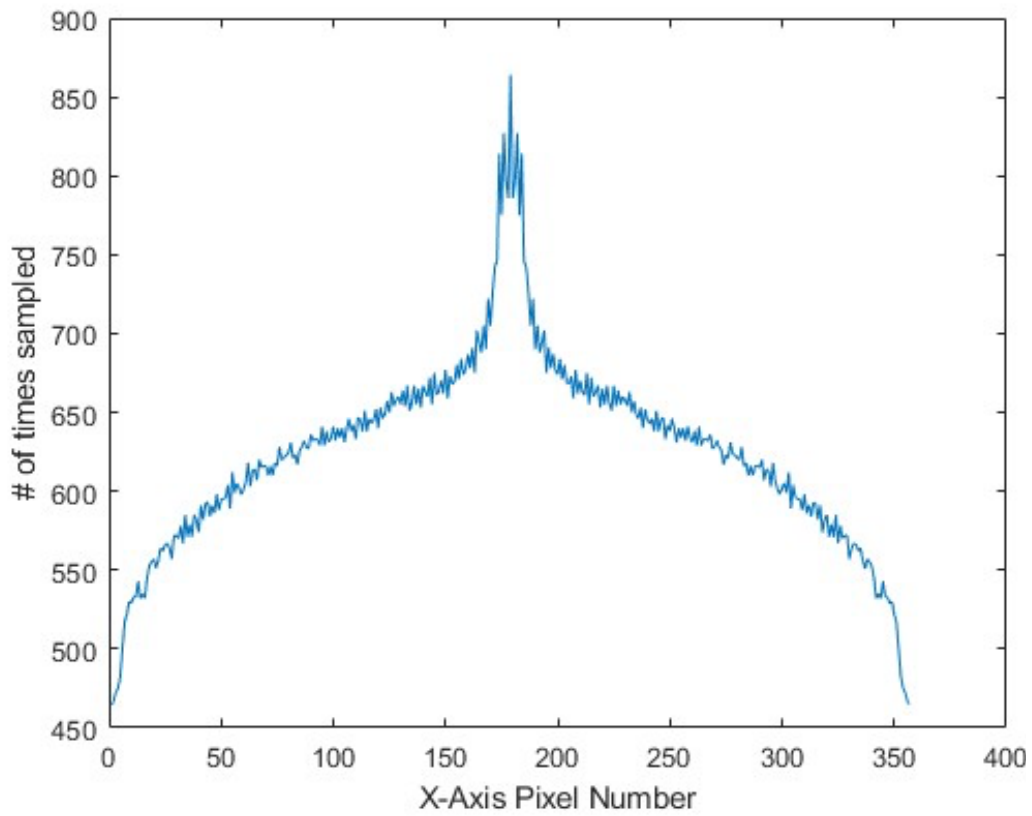


Figure 70. Plot of the cross section of Figure 69 depicting the changes in total sampling frequency along the line where the y-value of the pixel is 179 and the x-value changes from 1 to 357.

7.3.1 *Representative Source Points Reconstruction*

To determine how well the analytic model was matching the measured data, two representative points were chosen for analysis. The origin point, located at the center of the inspection volume, and a corner point, located at the furthest extent from the origin in the corner of the PWR 17x17 assembly, were analyzed. Figure 71 shows the locations of these points within a fuel assembly grid. These points represent the “best case” and “worst case” sampling scenarios for fuel rod locations when there is no attenuation of neutrons occurring from interactions in fuel. With both a best location and a worst location, the neutron response model can be evaluated at both extremes.

The origin measurement is considered a best-case source location because the lines of response frequently cross and the origin point is the most frequently sampled location in the FOV at 864 times. The corner measurement is considered the worst fuel pin location for imaging. The center of the fuel pin is located at a radial distance of 15.3 cm from the center of the FOV. Given that the sampling frequency decreases toward the edge of the FOV, that results in less measured data at the corner fuel pin position. The center pixel of the evaluated fuel pin located in the corner of the assembly grid is (280, 73), which is sampled 580 times. The corner fuel pin is sampled 32.8% less often than the center pin.

The origin measurement used as the representative point was previously described in Section 7.2.3. The measurement data collected was binned into 72 positions to match the averaged 72 positions used in the analytic neutron response code. Figure 72 (left) shows the sinogram of the measured data at the origin where the neutron counts for each detector at each position are reported by the color bar. Figure 72(right) shows the sinogram created using the analytic neutron response code to generate a response for a source located at position (179, 179). In order to compare the neutron counts in each detector for the measured sinogram and the analytic sinogram, the analytic sinogram shown in Figure 72(right) was multiplied by the total number of neutrons emitted from the source used in the measured data. This is because the analytic code reports neutron counts in terms of the fraction of neutron counts detected per source neutron emitted. After the correction the two sinograms can be compared directly. Both sinograms show two bright lines representing the origin source counts for each detector and position. There are places where there is a slight difference in the source intensity. These

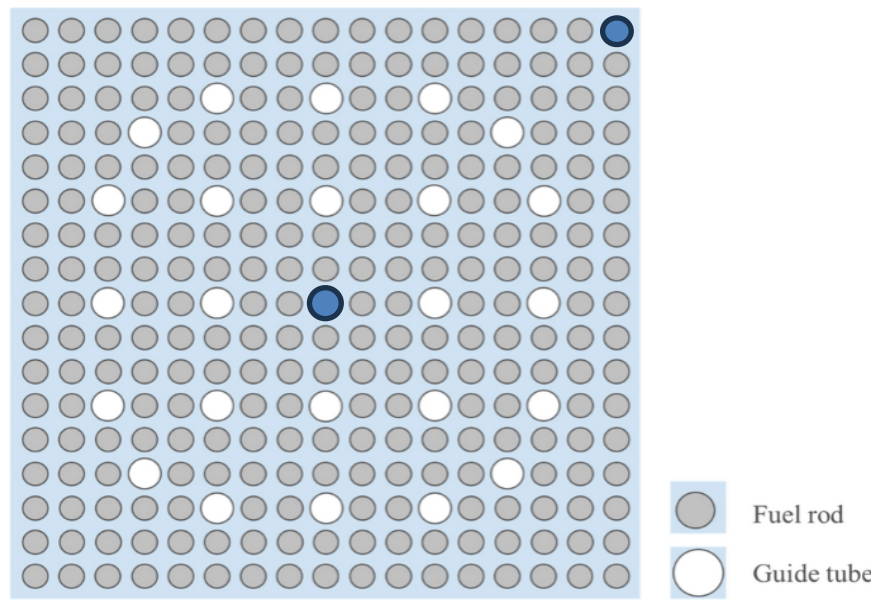


Figure 71. PWR assembly grid with (0, 0) and (8, 8) labelled with blue dots.

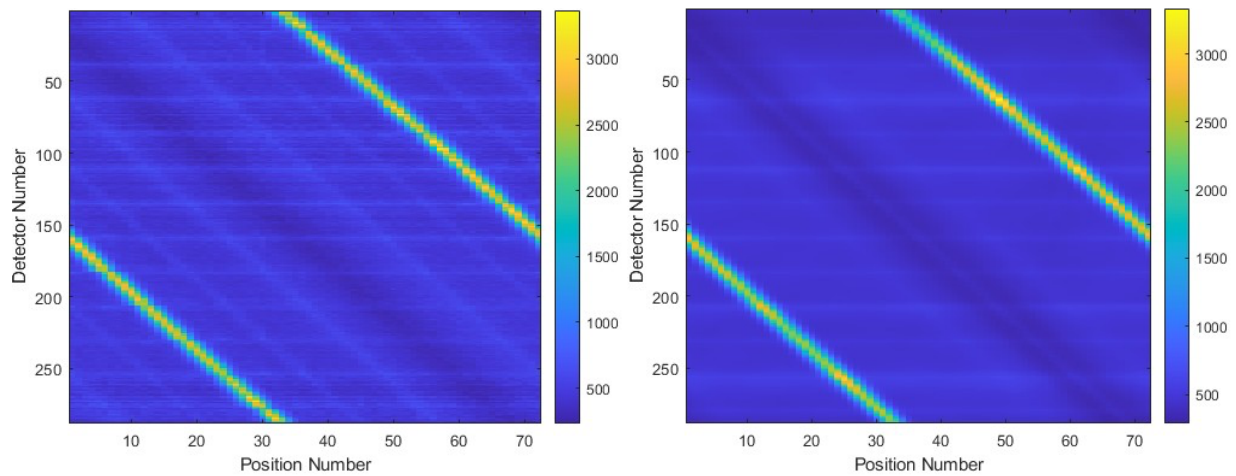


Figure 72. Measured sinogram (left) and analytic code generated sinogram for (0, 0).

differences may be caused by the room return in the system or by differences where the factors added to the analytic code to account for the variations in collimator thickness are either too low or too high for some detectors at certain positions.

Following examination of the sinograms, a reconstruction was performed on the source located at the origin. Figure 73(left) shows the reconstruction for the full FOV with a source located at (0, 0). Visually, the reconstruction does show source intensity emanating from a point in the center of the inspection volume. Figure 73(right) shows a zoomed in version of the source point which better shows the pixels where source is primarily being attributed. The reconstruction pixel with the highest source intensity value is (178, 179) which has a source intensity of 2.33×10^6 . The exact center of the inspection volume is pixel (179, 179). The difference between the pixel with the highest source intensity and the exact center pixel of the inspection volume is 1 mm and this difference can be attributed to the placement of the mock fuel assembly grid being slightly unaligned with the exact center of the FOV. This difference in grid placement is taken into account for analysis of other measurements and reconstructions.

When determining how well a source is reconstructed, there are various characteristics of a reconstruction which can be examined. In addition to looking at where the source point was reconstructed versus where it is expected to be reconstructed and the source intensity of that pixel, factors such as the reconstruction inferred source strength and the true source strength can be examined. The true source strength is the number of counts in the sinogram where a detector was directly sampling the neutron source. For source pin (0, 0), the source strength was 2.49×10^6 counts. This can be compared to the total number of counts in the sinogram, in this case 1.14×10^7 counts. The neutron source represents 21.8% of the total source counts. In comparison, when the fuel pin is reconstructed the intensity of source attributed to the fuel pin in the reconstructed image is 4.17×10^7 . The total intensity in the reconstructed image is 8.65×10^7 which makes the fuel pin intensity 48.2% of the total intensity in the reconstruction.

Each fuel pin has a radius of 0.4812 cm and the gap between fuel pins is 0.3280 cm. If the pixel with the highest source intensity is treated as the center of the fuel pin, the total amount of source attributed to the fuel pin can be calculated and compared to the total amount of source attributed to the gap around the fuel pin. Figure 74(left) shows the 69 mm² area attributed to the

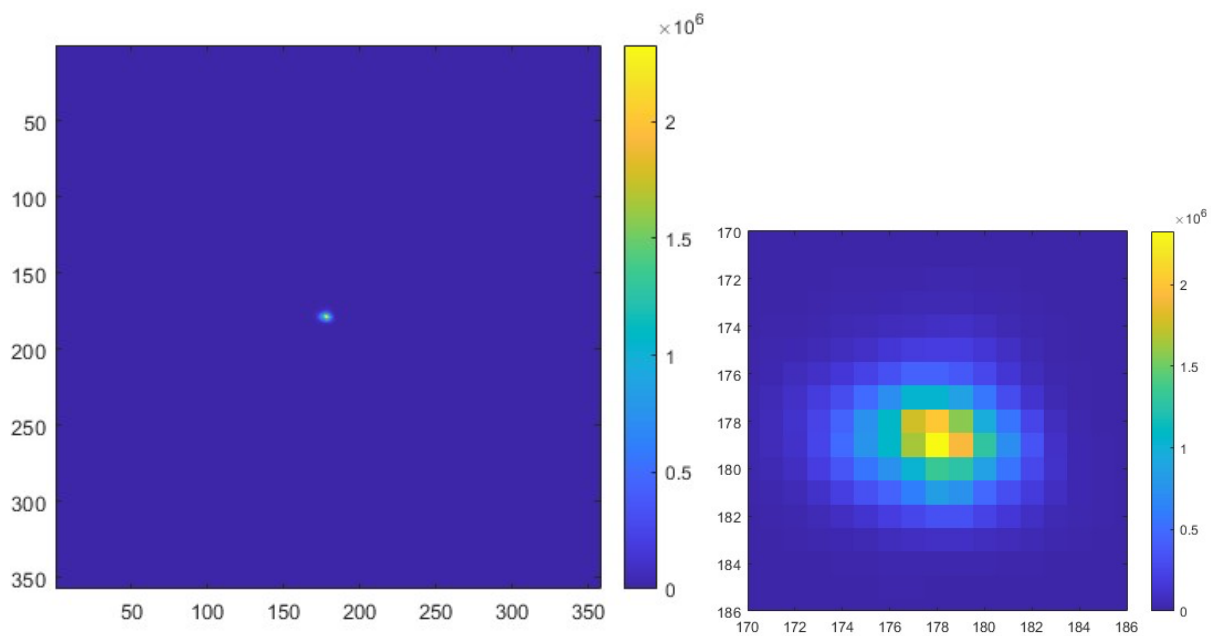


Figure 73. Reconstruction of the neutron source at the point (0, 0), the center of the inspection volume within the entire FOV (left) and zoomed in on the center point (right).

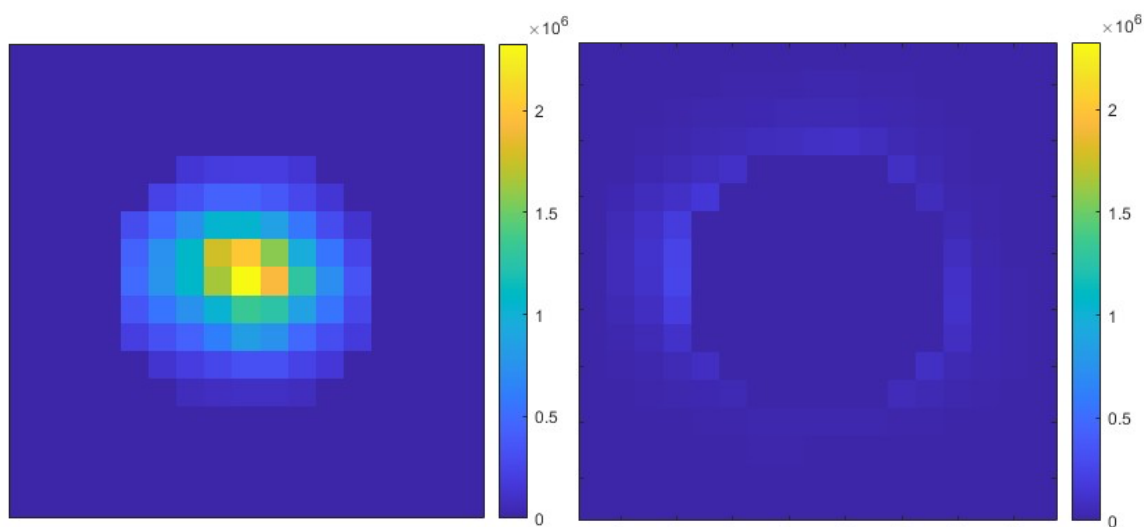


Figure 74. Source within the fuel pin radius (left) and source within the fuel pin gap (right).

origin fuel pin and the amount of source intensity attributed to each part of the fuel pin. Figure 74(right) shows the 124 mm² of area attributed to the fuel pin gap and the source intensity within each portion of the fuel pin gap for a fuel pin located at (0, 0). A high ratio of the fuel pin intensity to the gap intensity is desirable and demonstrates how well a source pin is reconstructed. For (0, 0), the ratio of the pin intensity to the halo intensity is 8.24 showing that the origin point is very well reconstructed as a single point. The full width half maximum (FWHM) area around the pixel with the highest source intensity also demonstrates how well the reconstruction attributes source. The FWHM area for (0, 0) is 9 mm² for this best-case scenario reconstruction. These same attributes can be used to compare single reconstruction of other pins as well.

After examining the best-case scenario single pin reconstruction, looking at the worst-case single pin measurement was the next step. The corner measurement was taken by placing the neutron source at a position (8, 8) and rotating the collimator to take data. Data was taken for one hour in detector position 1 and one hour in detector position 2. This yields values for sections of the FOV which are otherwise difficult to model using the neutron response code. Figure 75(left) shows the measured data sinogram of a source at (8, 8), while Figure 75(right) shows the analytic generated sinogram for the same fuel rod position. Again, there are differences between the two sinograms; however, the differences this time are larger. Figure 76 shows the ratio of the analytic sinogram to the measured sinogram which better displays some of the differences between the two sinograms. In the ratio plot there almost appears to be an echo of the source intensity. This echo exists because the measured sinogram is spreading the source intensity across more positions than the analytic code. As discussed in Section 7.1.1, the constant collimator rotation means that the measurement bins are wider than the analytic model bins for the projections. Despite attempts to correct for this in the analytic model by expanding the projections, without more accurate data binning, the data is being spread over multiple projections. The echo of the source intensity in the sinogram shows the places where the measured data is spreading the source intensity beyond where the analytic sinogram does.

Figure 77(left) shows the reconstruction at (8, 8). After correcting for the mock grid's misalignment, which was discovered during the reconstruction of the fuel pin at the origin, the center of the fuel pin located at grid position (8, 8) corresponds to pixel (279, 73). The source

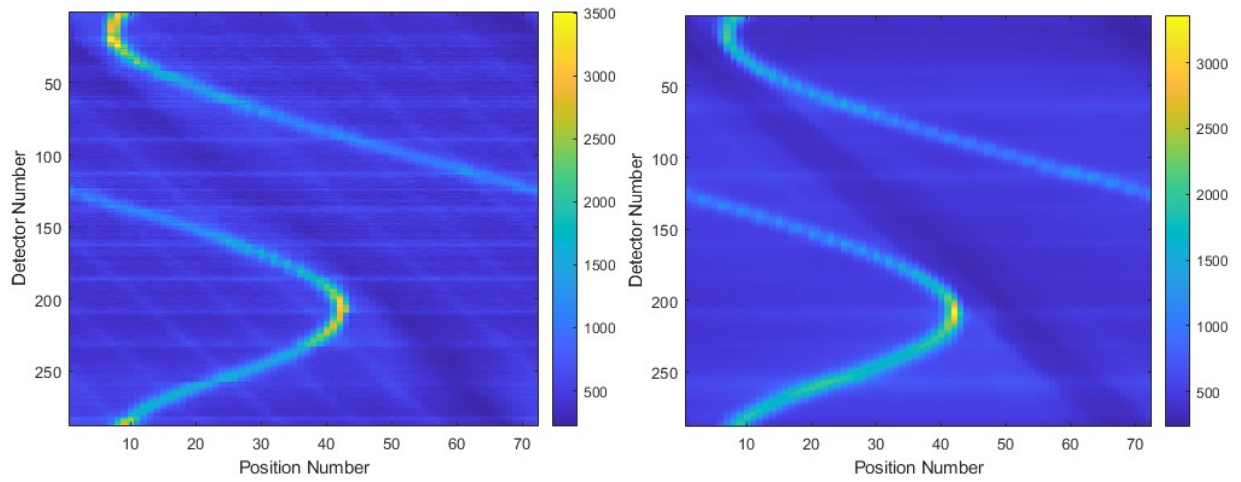


Figure 75. Sinogram created from the measured data with a source at fuel rod position (8, 8) (left) and the analytic generated sinogram for a source at (8, 8) (right).

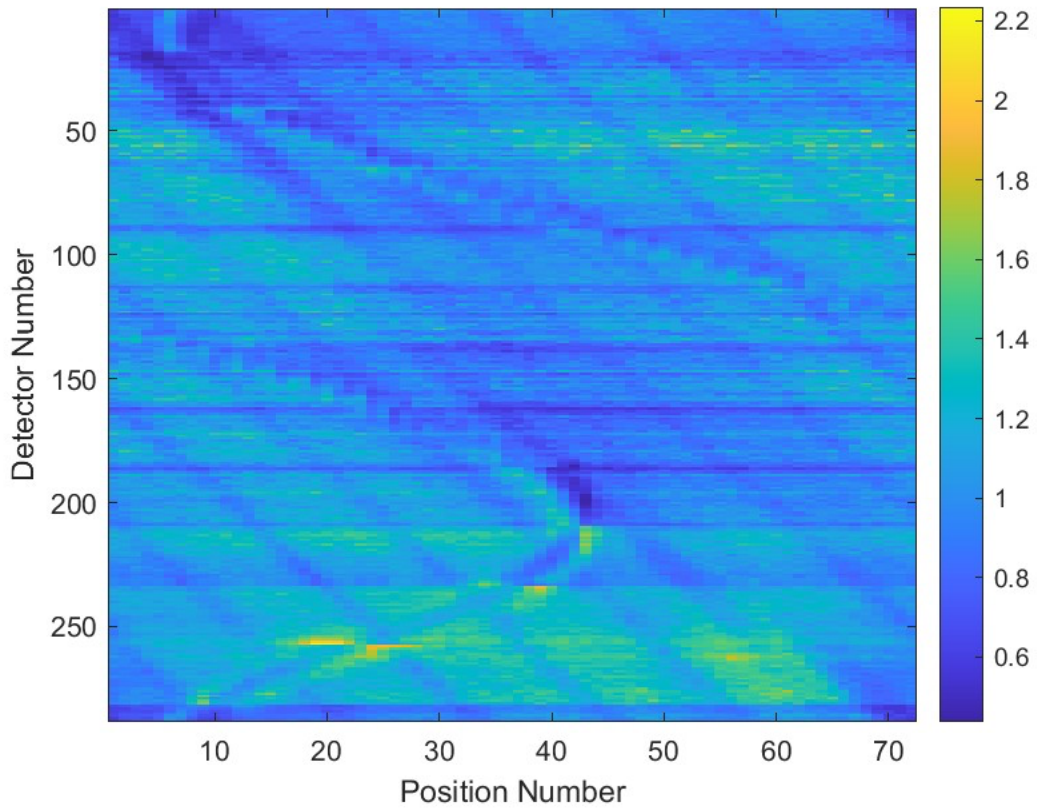


Figure 76. Ratio between the measured and analytic generated sinograms.

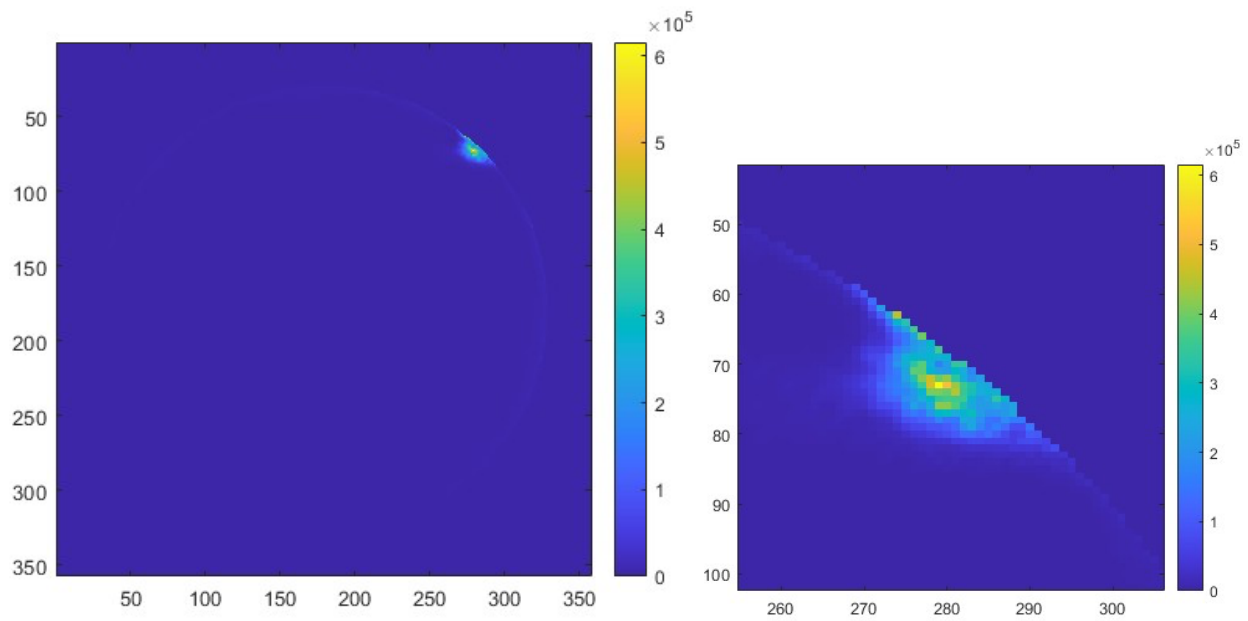


Figure 77. Reconstruction of the neutron source at point (8, 8) within the whole FOV (left), and a closer version of the point (8, 8).

located at (8, 8) has a larger “smear” of green surrounding the point and lining the edge of the field of view. The extent of the smear around the point can be seen more clearly in the enlarged image of the corner position, shown in Figure 77(right). The smear is the result of the difference in the source intensity position that was seen in the ratio sinogram in Figure 76. The source intensity in this area is 6.16×10^5 or 0.66% of the total source intensity. If a fuel pin is centered at (279, 73) the fuel pin intensity to halo intensity is 1.34 and the FWM area for the fuel pin (8, 8) is 24 mm^2 . This pin to gap ratio for the smeared corner measurement is much lower than the 8.24 value for (0, 0). This gives a point of comparison for best-case pin to gap ratio and worst-case pin to gap ratio. The FWHM value for the corner measurement is slightly misleading because of how close the fuel pin is to the edge of the FOV. Beyond the edge of the FOV there are no source counts, and these zero values truncate the FWHM.

In order to determine how much the data is smeared out, the angle with respect to the center of the inspection volume spanning the full extent of the smeared source was calculated to be approximately 12.7 degrees. While the full extent of the smear is 12.7 degrees, the data spread is due to much of the source being attributed to the edge of the inspection volume. Disregarding source intensity collecting along the edge of the field of view and focusing on the smear surrounding the fuel pin position, the source extent is 5.8 degrees with a relative intensity that is 0.057% of the total source intensity. The angular difference between each of the 72 projection locations is 5 degrees. This would imply that a smear extent of 5.8 degrees represents a smear across 2 different projection positions, which fits with the differences which can be seen between the measured and analytic sinograms. The green smear of the source intensity results from trying to reconstruct a point which is both sampled less often than other points and spreads source intensity over multiple positions due to poor data binning. The pixel within the smear with the highest source intensity is (279, 73) which means the source is placing the intensity in the correct spot, even if it does not resolve into a circular pin-like shape.

The next step was to see if both the corner and the origin could be reconstructed simultaneously. In order to create a measured sinogram for the two points, the measured sinogram created for (0, 0) and the measured sinogram for (8, 8) were each scaled to a mean number of total counts in each measurement and added. This was to correct for any slight variation in data collection time. The same procedure was repeated for the analytic code to

generate an analytic response sinogram. Figure 78 shows the sinogram created from the sum of the measured data. This sinogram shows the different shapes of each fuel pin overlapping one another. Figure 59 shows the reconstruction of the two points created from the summed sinogram. The corner point still did not resolve into a circular shape due to the previously mentioned issues with the corner measurement. The pixel with the greatest source intensity is still located correctly at (280, 76) for the corner smear and at (179, 179) for the origin point. When the source intensity at the origin point and the source intensity for the corner point smear are summed, the origin point is determined to have greater source intensity. The ratio of the corner measurement sinogram total counts to the origin measurement sinogram total counts is 0.983, whereas the same ratio for source intensities in the reconstruction is 0.931. Some corner measurement source is being lost in the smear and attributed elsewhere in the reconstruction, potentially either to the center of the reconstruction FOV due to increased sampling or along the edges of the FOV due to poor position binning.

One of the reasons why there is extra source intensity attributed to the center of the FOV has to do with the sampling frequency at that location. Given how much more often the center of the volume is sampled compared to the edge, more weight is given to the pixels in the center of the FOV when reconstructing than the edges. MLEM reconstruction algorithms use a statistical weighting when reconstructing images to try to increase convergence speed. The additional sampling of the center of the FOV versus the edges increases the weighting of these pixels. With slightly higher weights, more source is attributed to these areas.

The reconstruction created from summing the two sinograms can be compared to a reconstruction image created by reconstructing each sinogram separately and then scaling and summing the reconstruction images. Table 13 shows the details of each of these reconstruction methods with the details of the reconstruction created from the sum of two sinograms being labelled “Summed Sinogram” and the reconstruction details coming from the sum of each pin reconstructed individually and then summed labelled as “Summed Reconstruction”. The most accurate reconstructions that can be created with this code are cases where just a single fuel pin is reconstructed because there is no extra noise added to the system unlike when multiple sinograms are reconstructed simultaneously. By examining the summed reconstruction image,

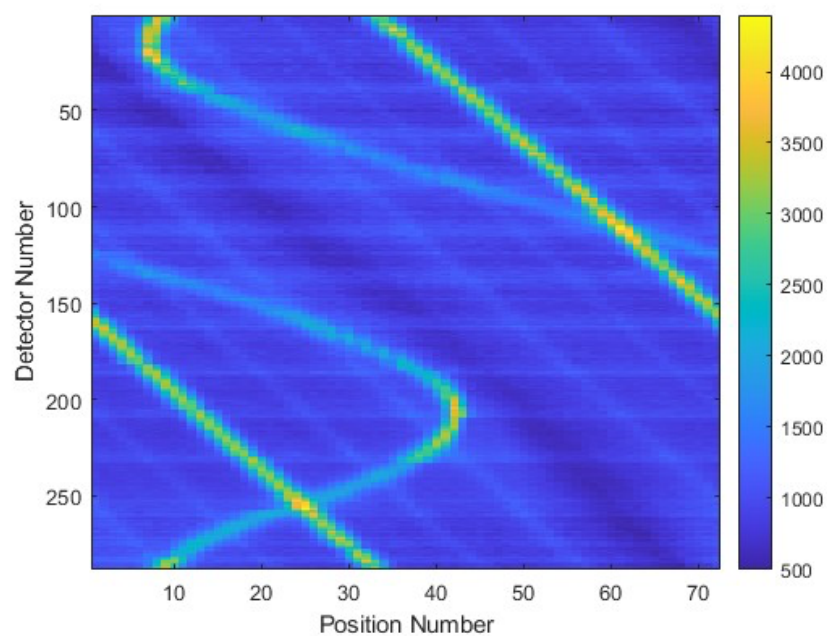


Figure 78. Sum of the two measured sinograms at $(0, 0)$ and $(8, 8)$.

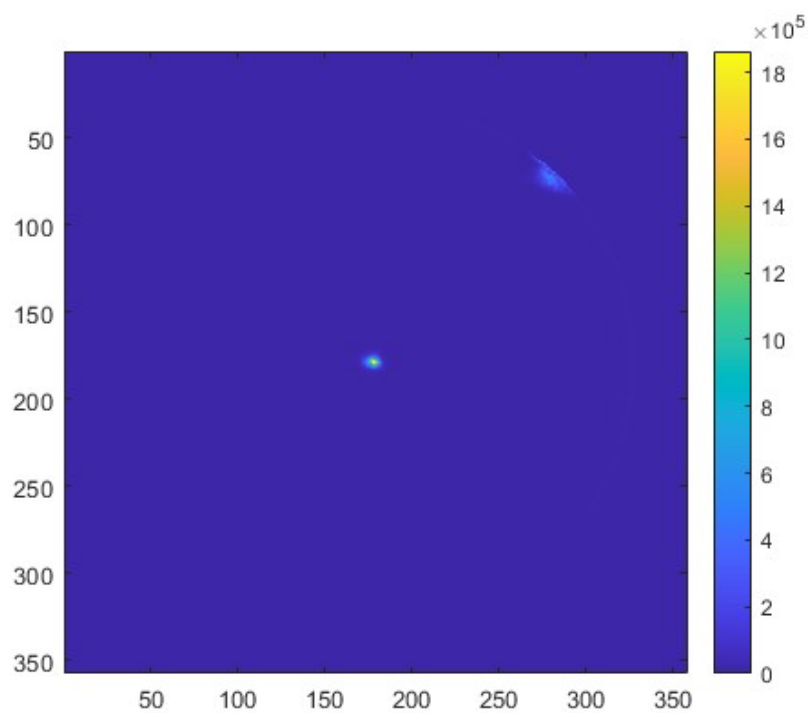


Figure 79. Reconstruction of source points at both $(0, 0)$ and $(8, 8)$.

Table 13. Summary of the results of (0, 0) and (8, 8) reconstructed from the summed sinogram, and the reconstruction image created when the reconstructions of each individual pin were reconstructed then those images summed.

Fuel Pin	Expected Pixel Location	Actual Pixel Location	Δ (mm)	Source Intensity at Pixel	% Total Reconstructed Area	FWHM Area (mm ²)	Ratio Of Pin to Halo
Summed Sinogram Fuel Pins							
(0, 0)	(179, 179)	(178, 179)	1	2.18×10^5	23.2%	12	9.13
(8, 8)	(280, 73)	(279, 73)	1	6.75×10^5	11.9%	27	1.41
Summed Reconstruction Fuel Pins							
(0, 0)	(179, 179)	(178, 179)	1	2.43×10^6	24.1%	9	8.23
(8, 8)	(280, 73)	(279, 73)	1	6.91×10^5	11.2%	18	1.34

a baseline for how well the pins could reconstruct is seen. This best-case reconstruction image can then be compared to the summed sinogram image to see how differently the code reconstructs data where the uncertainty and noise in each sinogram add together.

The pin to halo ratios for each of these fuel pins for the two different reconstruction methods show differences which can be attributed partially to the counts from the corner measurement being attributed more to the center of the FOV. Despite having a lower source intensity at the highest pixel in the summed sinogram than in the summed reconstruction version, the ratio of the pin to halo is higher for both in the summed sinogram. More of the smear of source from the corner being attributed to the center of the FOV would raise the pin to halo ratio. Removing some of the excess smear from the corner reconstruction would also increase the pin to halo ratio of the corner measurement. The FWHM values in the summed sinogram reconstruction are generally larger. The source intensities at the highest pixel for each fuel pin for the summed sinogram are lower than the intensities for the summed reconstruction. This, combined with the pin to gap ratio values, imply that the source in the summed sinogram reconstruction is being spread out slightly more than in the summed reconstruction version. With decreased count rates at the peak pixel location for each fuel pin, the “half maximum” of the FWHM is a lower value as well. If the source is spread across more pixel, but those pixels are still within the fuel pin equivalent area, that would result in the higher FWHM and a higher pin to halo ratio seen in the summed sinogram reconstruction. This helps to demonstrate how the analytic code reconstructs multiple points at once. The source is still well attributed to the source pin location but spread out among more pixels than if each fuel pin was reconstructed individually.

7.3.2 Five Points Reconstruction

In order to analyze the updated analytic code, points (0,0), (2, 0), (2, 1), (1, 3), and (0, 4) were re-measured and reconstructed. Each position was recorded with the collimator rotating at 1 rev/min for 10 min with the detectors in position 1 and repeated for 10 min with the detectors in position 2. The results were summed to get a full measurement of each location. Due to the symmetry of the system, the source positions which were measured to get this data corresponded to mock fuel rod grid positions (0, 0), (0, 2), (1, 2), (3, 1), and (4, 0). The sinogram used for the

analysis and reconstruction of these 5 points is the sum of the individual sinograms for each point. Figure 80 shows the individual sinograms created for each of these pins and the summed sinogram to better illustrate how the distinct sinogram shapes that each fuel pin location creates are added together to get one total sinogram. Places where these sinogram shapes overlap add more source counts to those detectors at those positions.

Each fuel pin location was reconstructed independently first using the sinograms shown in Figure 80 to have a point of comparison for the summed sinogram reconstruction. For fuel pin (0, 0), the center of the fuel pin would be located at pixel (179, 179). In the reconstruction of the individual sinogram at (0, 0), the pixel with the highest source intensity for that fuel pin located at (0, 0) was (177, 178). The difference in these locations is 2.24 mm and is likely the result of the mock fuel assembly grid being placed within the imager system slightly off center. For comparison of all points going forward, (177, 178) is treated as the center of the fuel assembly grid, the location of fuel pin (0,0). The expected center pixels of all other fuel pin positions are calculated relative to (177, 178).

Table 14 sums up some key features of each of the individually reconstructed points. This summation includes information about the where the center of each fuel pin is expected to reconstruct, where it actually constructed, and the distance in millimeters between those two pixels. For all 5 positions the x value of each pixel reconstructed correctly; however, the y values of (0, 2), (1, 2), and (3, 1) are different from where they were expected to be. Fuel Pin (0, 2) reconstructed the farthest away from its expected location position at 6 mm. The FWHM and pin to halo ratios for each of these fuel pins are larger than the ideal (0, 0) case where one hour of measurement time resulted in a FWHM of 9 mm² and a pin to halo ratio of 8.24. In addition to having the furthest reconstruction point from its expected point, fuel pin (0, 2) also has the worst pin to halo ratio and the largest FWHM. This indicates that the reconstruction of (0, 2) spreads the source intensity across a larger area than the other fuel pin reconstructions do. The column in Table 14 labelled “% Fuel Pin Counts” represents the percentage of the total counts recorded during the measurement that come from the direct sampling of the fuel pin. The column labelled “% Fuel Pin Intensity” represents the percent of total source intensity in the FOV which are attributed to a fuel pin with a center located at the actual pixel location. For pin (0, 2), despite the percentage of neutron counts coming directly from the source location being mostly in

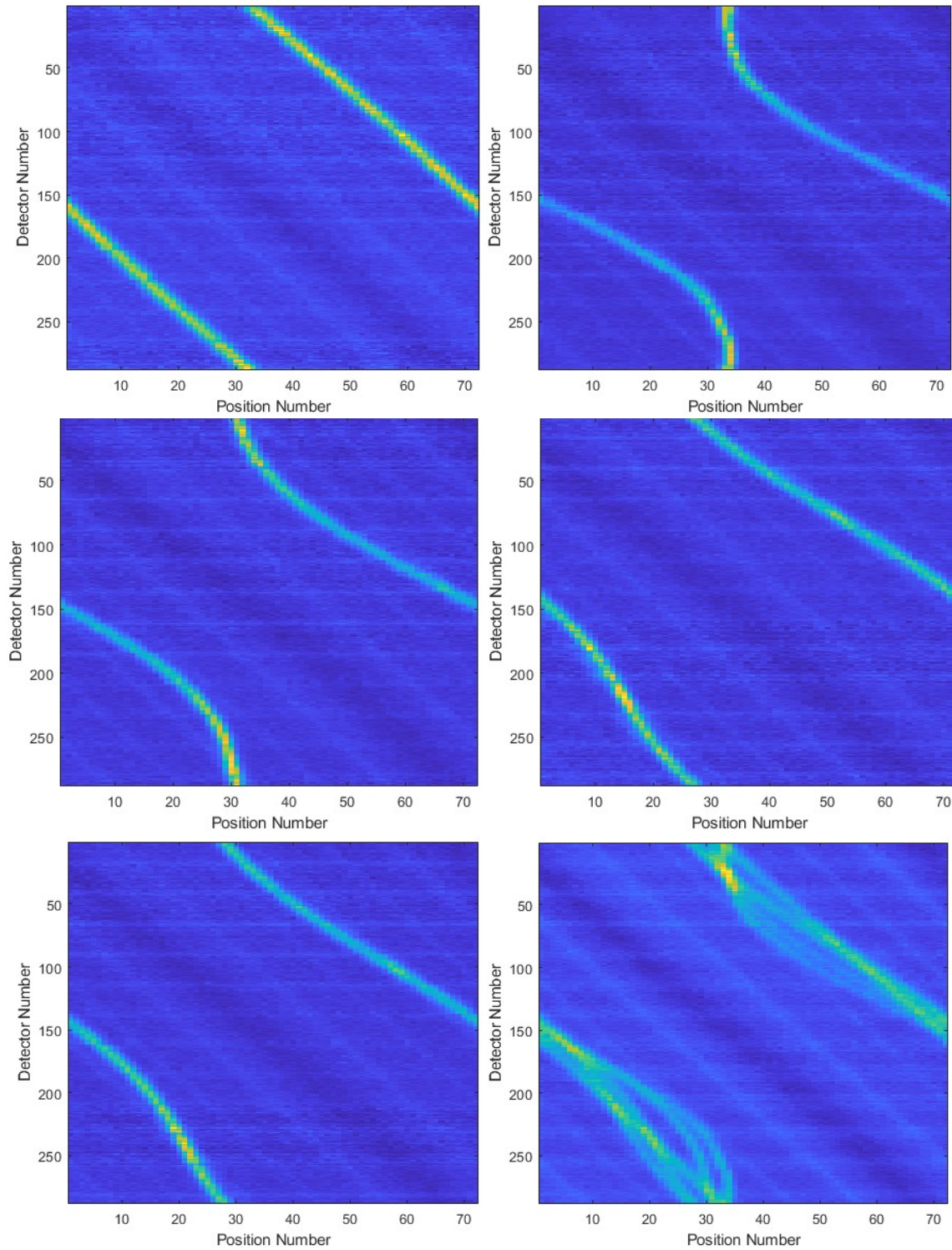


Figure 80. Sinograms for fuel pins (0, 0) (top left), (4, 0) (top right), (3, 1) (middle left), (2, 0) (middle right), (2, 1) (bottom left) and the sinogram created when all 5 sinograms are scaled and summed (bottom right).

Table 14. Data comparison for 5 fuel pins reconstructed individually.

Fuel Pin	Expected Pixel Location	Actual Pixel Location	Δ (mm)	Count Rate at Pixel	% Fuel pin Counts	% Fuel Pin Intensity	FWHM (mm ²)	Ratio of Pin to Halo
(0, 0)	(177, 178)	(177, 178)	0	2.19×10^5	21.1	40.2	17	3.77
(0, 2)	(177, 152)	(177, 158)	6	1.16×10^5	18.8	27.7	26	2.21
(1, 2)	(190, 152)	(190, 157)	5	1.53×10^5	18.7	35.7	25	3.35
(3, 1)	(216, 165)	(216, 169)	4	1.53×10^5	18.4	35.5	20	3.46
(4, 0)	(229, 178)	(229, 178)	0	1.95×10^5	18.4	40.7	18	4.86

agreement with the other pin locations, the percentage of the source intensity attributed to the fuel pin is lower than for the other pins. Based on these individual reconstructions, it would be expected that (0, 0) and (4, 0) would reconstruct the most accurately. Their FWHM values are the smallest, their pin to halo ratio values are both the largest values of the individual reconstructions, and they reconstruct at the expected location. Fuel pins (1, 2) and (3, 1) would be expected to reconstruct slightly less accurately since their distances from the expected locations are 5 mm and 4 mm respectively, and their pin to halo ratios are similarly less than those of (0, 0) and (4, 0). Fuel Pin (0, 2) would be expected to have the worst accuracy in reconstruction given the issues discussed.

Following the analysis of the individual fuel pin reconstructions and sinograms, the summed sinogram was reconstructed. Figure 81(left) shows the total FOV reconstruction created from the summed sinogram of the 5 points, while Figure 81(right) shows an enlarged version of the reconstructed 5 points. Visually in the reconstruction, 5 distinct source rod positions can be seen. To better examine the quality of the reconstruction and spatial resolution of the imager, an additional reconstruction was created where each of the individual reconstructed images for each of the five fuel pins were scaled and summed together to get a summed reconstruction. Like in the (0, 0)/(8, 8) fuel pin reconstruction analysis these two different reconstructions are referred to as the “Summed Sinogram” image and “Summed Reconstruction” image. Figure 82 shows the resulting image for the summed reconstructions and Table 15 details the differences between the reconstructions of each fuel pin for the two different images.

In general, the summed sinogram results in larger FWHM values and smaller pin to halo ratios. Much more uncertainty in fuel pin location is introduced to the system as more fuel pins are being reconstructed simultaneously. More uncertainty in fuel pin location leads to spreading of source intensity around the area of interest and less clearly defined fuel pins. While each fuel pin can be resolved in both reconstructed images, the resolution in the summed sinogram image is worse.

For fuel pin (0, 0), the reconstruction results are the most similar in the two images than for any of the other fuel pins. Similarly to the (0, 0)/(8, 8) case, the FWHM of the summed sinogram pin reconstruction increased as well as the pin to halo ratio value compared to the summed reconstruction image. Again, this is likely due to other fuel pins attributing some

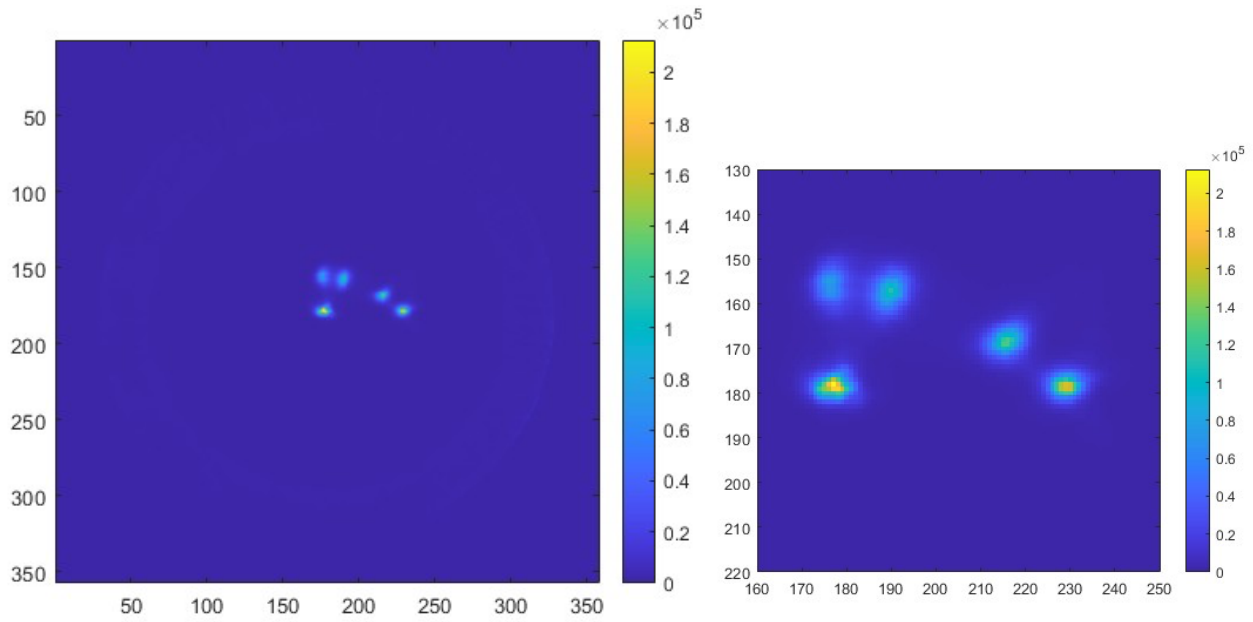


Figure 81. Reconstruction of 5 points at (0,0), (2,0), (2,1), (1,3), and (0,4) created from reconstructing the summed sinograms of all 5 points with full image displayed (right), and a zoomed in version of the reconstruction focusing on the points of interest (left).

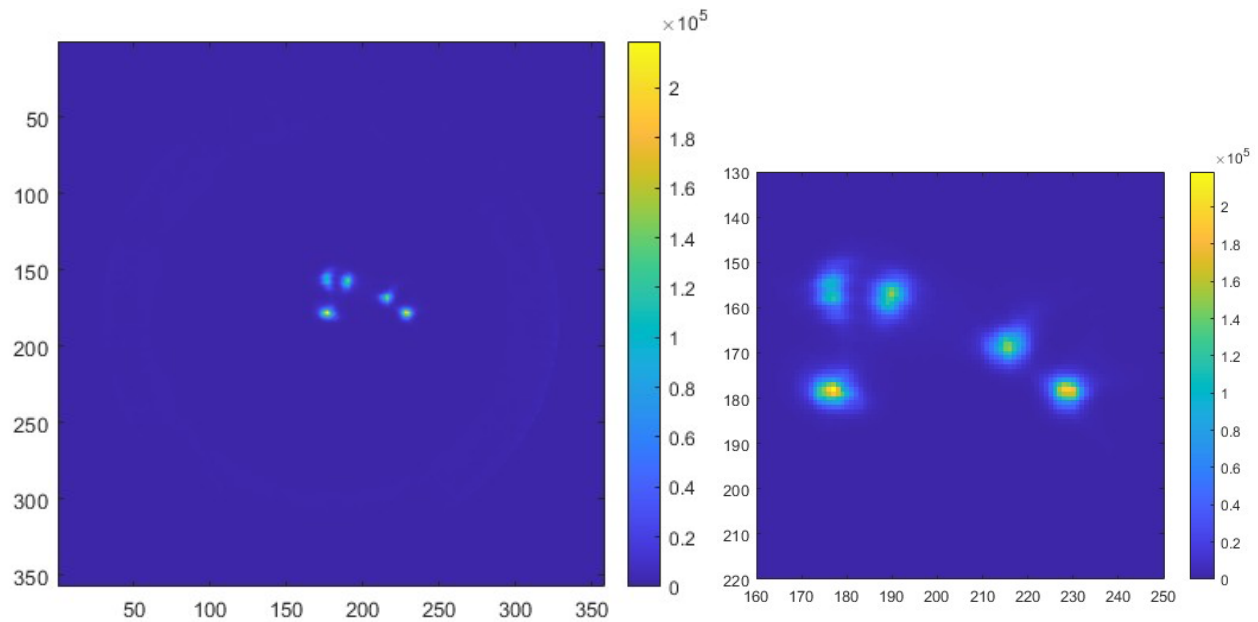


Figure 82. Reconstruction of 5 points in the whole FOV when each pin is reconstructed separately, and the reconstructions are scaled and summed together (left) and an enlarged version focusing on the points of interest (right).

amount of source towards the center of the FOV when multiple pins are reconstructed simultaneously. For fuel pin (0, 2), the summed sinogram reconstruction is more accurate to the expected location than the individually reconstructed sinogram at only 3.16 mm from the expected location instead of 6 mm. Fuel pin (2, 1) still reconstructs 5 mm from its expected location, just as in the summed sinogram image. Pins (1, 3) and (0, 4) reconstruct 3 mm from their expected position and at their expected position respectively.

Reconstructing all 5 points simultaneously results in a larger FWHM and smaller pin to halo ratio for (0, 2) as well as a decreased source intensity. The source intensity attributed to fuel pin (0, 2) in the summed sinogram reconstruction is much less than the intensity attributed to the other fuel pins in the same reconstruction and less than the source intensity in the summed reconstruction image. The intensity of the fuel pin radius accounts for only 4.71% of the total source intensity, whereas for pin (1, 2), 6.62% of the total source intensity is attributed to that position. Given that all the measurements were made with an identical neutron source, the fuel pin intensities for each of these positions should be identical. The fact that they are not identical points to an issue with the reconstruction. For all points in the summed sinogram except for (0, 0), the percentage of counts attributed to each fuel pin is less than the percentage attributed in the summed reconstruction image.

Pins (0, 0) and (4, 0) in the summed sinogram image reconstruct the most accurately and the most similarly. The pin to halo ratio for point (4, 0) is slightly higher than for (0, 0), but this is explained by the excess source attributed to the area around the origin pin decreasing the ratio for the pin at (0, 0). Pin (3, 1) reconstructs as anticipated, but pin (1, 2) does not reconstruct as well as expected when examining the individual reconstructions. This is due to how close together fuel pins (0, 2) and (1, 2) are located. These pins are located at the closest distance to one another, only separated by 0.328 cm. Given how closely these two pins are located, any spreading of the source intensity around one of the pins impacts the one beside it. The two source positions share parts of their fuel pin halo. Uncertainties in the source location add to the amount of source attributed to the shared portion of their halo. This creates more noise between the two positions and makes it more difficult to distinguish the fuel pins. The compounding location uncertainties between these two pins may be part of the reason why the expected pin location and the actual pin location are different. In addition to the compounding uncertainties, one potential

Table 15. Comparison of 5 fuel pins with summed reconstruction and reconstructed from the summed 5 fuel pin sinogram.

Fuel Pin	Expected Pixel Location	Actual Pixel Location	Δ (mm)	Count Rate at Pixel	% of Total Reconstructed Counts	FWHM (mm ²)	Ratio of Pin to Halo
Summed Reconstruction Fuel Pins							
(0, 0)	(177, 178)	(177, 178)	0	2.19×10^5	9.04	17	3.41
(0, 2)	(177, 152)	(177, 158)	6	1.16×10^5	5.57	26	2.05
(1, 2)	(190, 152)	(190, 157)	5	1.53×10^5	7.48	25	2.83
(3, 1)	(216, 165)	(216, 169)	4	1.53×10^5	7.13	20	3.32
(4, 0)	(229, 178)	(229, 178)	0	1.95×10^5	8.14	18	4.79
Summed Sinogram Fuel Pins							
(0, 0)	(177, 178)	(177, 178)	0	2.13×10^5	9.04	20	3.54
(0, 2)	(177, 152)	(176, 155)	3.16	7.71×10^4	4.71	38	1.79
(1, 2)	(190, 152)	(190, 157)	5	1.13×10^5	6.62	31	1.90
(3, 1)	(216, 165)	(216, 168)	3	1.28×10^5	6.82	27	2.70
(4, 0)	(229, 178)	(229, 178)	0	1.66×10^5	7.48	20	3.79

reason why the source rod may not be attributing the location to exactly the correct spot is due to the room return inherent in the measurement which has an impact on the signal to noise ratio since room return is effectively adding noise to the system. The rotation of the collimator also likely plays into the difficulties with matching a source to a location. The position binning used in creating the measured sinogram spreads source data across multiple positions, which makes reconstruction more difficult.

Chapter 8. Full Fuel Assembly Measurements and Reconstructions

8.1 *Experimental Procedure*

Following these characterizations, multiple measurements were made to represent an entire fuel assembly. The neutron source used in these measurements is the same ^{252}Cf neutron source described in Section 7.2.1.

Only one quarter of the positions in the fuel assembly grid need to be measured within the imager since both the imager and PWR assemblies are symmetric. This means that measuring a fuel rod in position (8, 8) would give a translated version of the same neutron response as a fuel rod in position (-8, 8), (8, -8) or (-8, -8). By translating the sinogram position rows by $\frac{1}{4}$ of the positions the results can be effectively rotated about the origin. Figure 83 shows a full fuel assembly indicating where rod positions would be. The orange box shows the fuel positions measured during this experiment. All the data will be summed to yield the representation of the full assembly and the image of all the source points reconstructed.

For the unattenuated source measurements, the source rod was positioned in one location and the collimator rotated at a constant rate at 1 rev/min for 10 minutes while taking data at a rate of 5 readouts per second. This measurement was performed once with the detector modules held in position 1 and then a second time with the detector modules in position 2. This is the same process as described in Section 5 for taking measurements with the exception that the collimator speed, measurement time and neutron source were updated for this campaign. The measurement process was repeated, moving the source rod to a different location until enough data was taken to represent an entire spent fuel assembly.

The attenuated measurements used stainless-steel rods with the same outer diameter as the source rod to represent attenuation which would happen in the fuel assembly due to additional fuel rods being in place. The stainless-steel rods are placed into the fuel assembly grid in all positions where a fuel rod would be located except for the location at which the measurement is occurring. In that position, the aluminum source rod continues to be used. Figure 83 shows diagram of where fuel rods would be in an assembly as well as which spaces in an assembly are devoid of fuel rods and used for control rods or other reactor instrumentation.

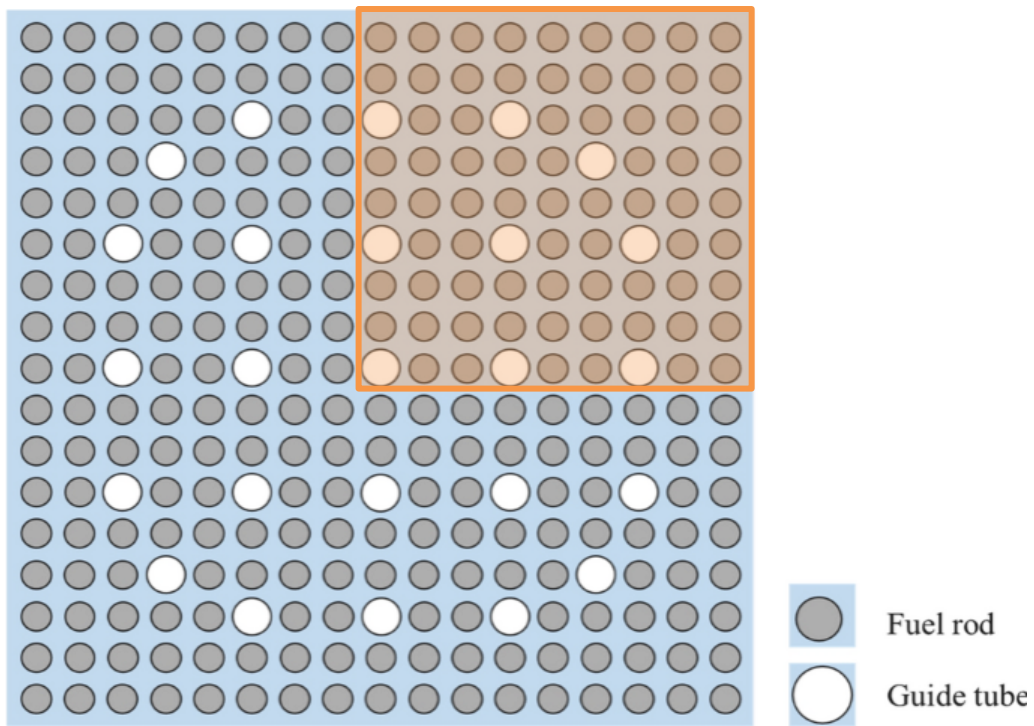


Figure 83. Diagram of measured locations for full fuel assembly equivalent.

Figure 84 shows the stainless-steel attenuation rods loaded into the mock fuel assembly grid within the imager's FOV matching the fuel rod pattern shown in Figure 83. Some of the fast neutrons from the ^{252}Cf source are attenuated by the stainless steel in the assembly, similarly to how other fuel rods would attenuate some of the fast neutrons emitted from a single fuel rod. The source was moved from position to position by removing the stainless-steel rod from the position of interest, placing the aluminum source rod in that position and then using the stainless-steel rod to fill in the previous measurement position. Once again, each measurement was performed twice, once with detectors in position 1 and then again in position 2. Data was recorded for 10 min in each of these positions and the collimator was rotated at a constant speed of 1 rev/min with a data readout rate of 5 readouts per second.

8.2 *Reconstruction and Discussions*

8.2.1 *4 Points Unattenuated*

Given the increased sampling of the lines of response closer to the center of the inspection volume, reconstructions were begun with a few sources close to the center of the inspection volume and work out to the edges, where the sampling is worse. The first few reconstructed sources include positions (0, 0), (0, 1), (1, 0), and (1, 1). Figure 85(left) shows the summed measured sinogram and Figure 85(right) shows the analytically generated sinogram.

Similar to how the 5 fuel pins were analyzed in 7.3.2, individual sinograms and reconstructions for each fuel pin were examined as well as the summed sinogram and reconstructed image. Figure 86 shows the individual sinograms for each of the fuel pins and Table 16 details the individual reconstructions. All 4 sinograms have two lines across the sinogram showing at which positions the detectors sampled the neutron source directly. Given how close together each of the sources was, the sinograms of each fuel pin are similar. The differences between them come from small bends in the direct sampling lines. These bends contain the location information about each of the fuel pin positions. When added together it is difficult to differentiate each of those different bended sinogram lines. Neither the measured sinogram or the analytic sinogram in Figure 85 show any distinct bends. The location information of each source position is now also partially dependent on the locations in the

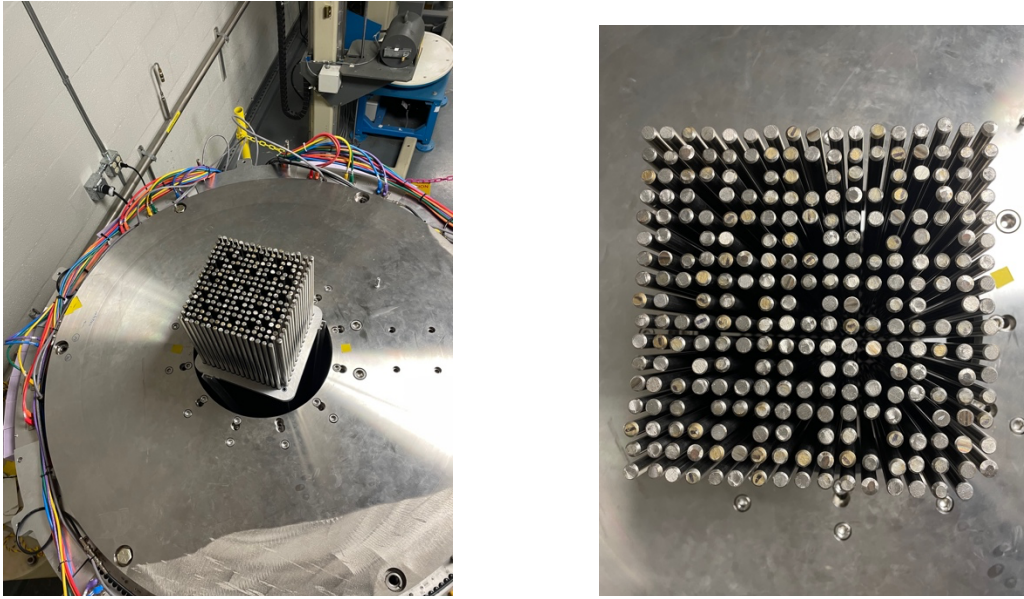


Figure 84. Photos of the mock assembly grid loaded with stainless steel rods within the imager (left) and a closer view of the stainless-steel rods within the assembly grid (right)

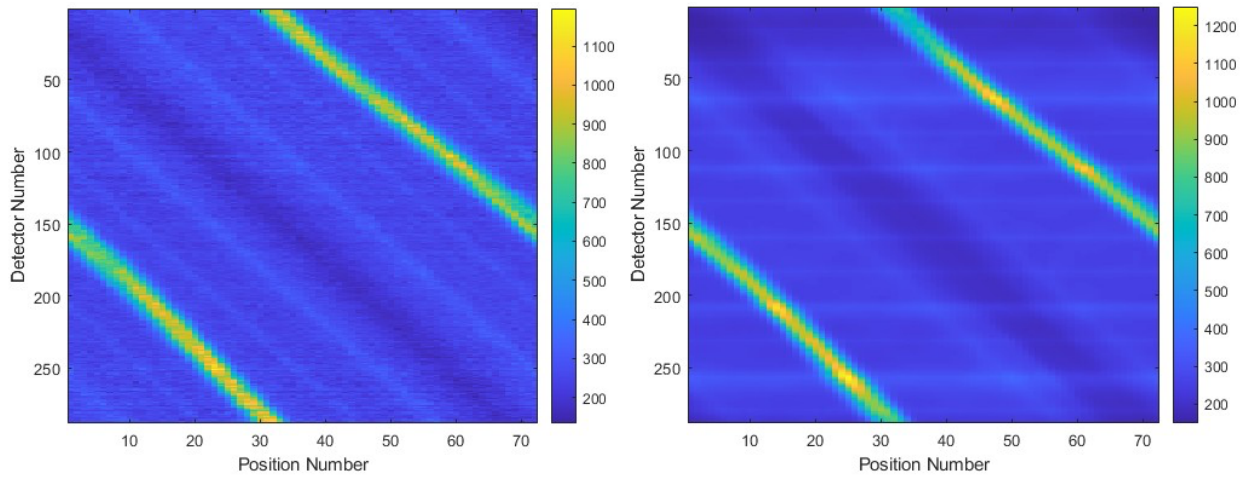


Figure 85. Summed sinogram of the measured data at 4 points: $(0, 0)$, $(1, 0)$, $(0, 1)$, $(1, 1)$ (left) and the analytic generated sinogram of the 4 points (right).

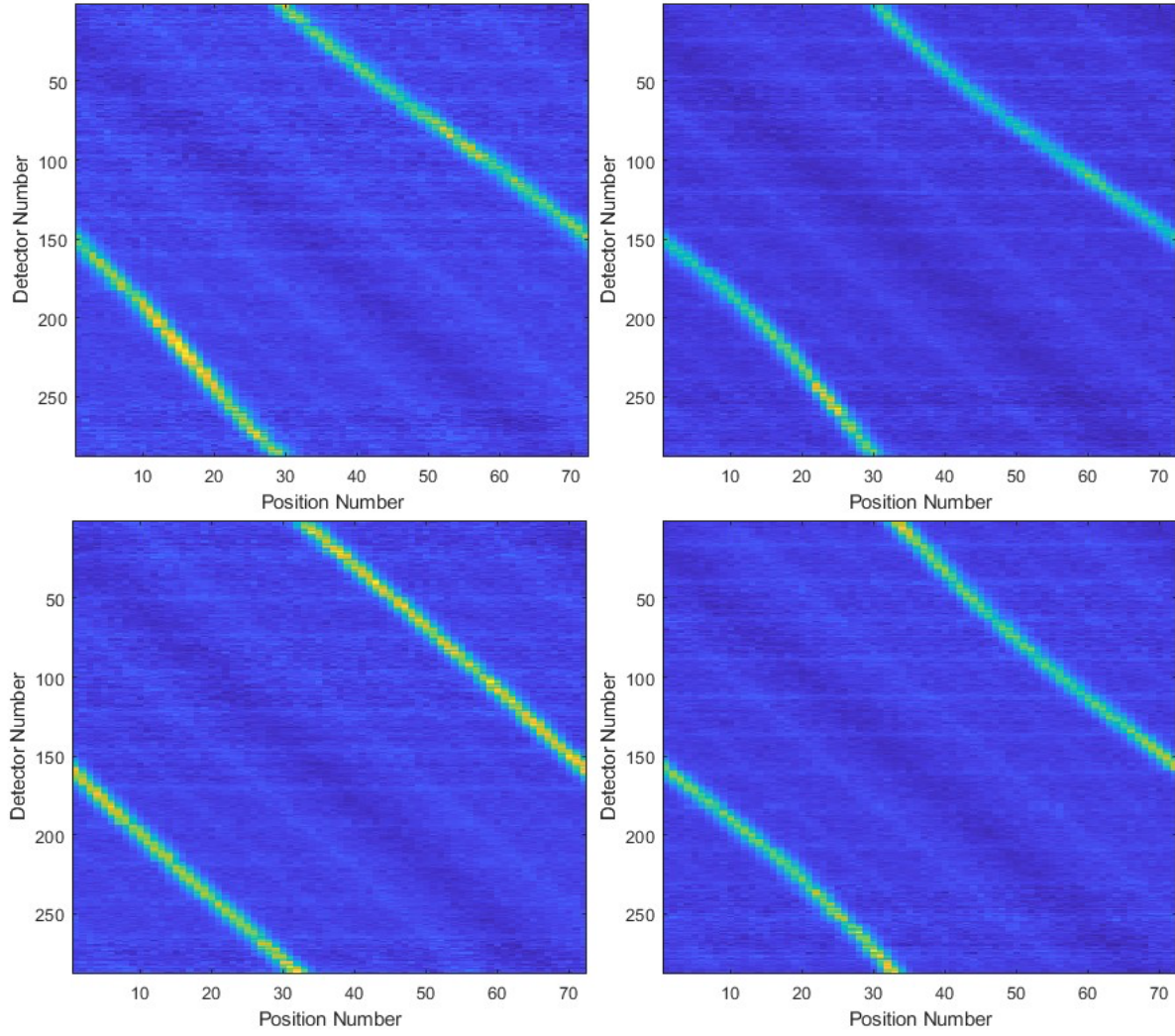


Figure 86. The measured sinograms of $(0, 1)$ (top left), $(1, 1)$ (top right), $(0, 0)$ (bottom left), and $(1, 0)$ (bottom right).

Table 16. Summary of individual reconstructions of fuel pins (0, 0), (0, 1), (1, 0), and (1, 1).

Fuel Pin	Expected Pixel Location	Actual Pixel Location	Δ (mm)	Count Rate at Pixel	% Fuel pin Counts	% Fuel Pin Intensity	FWHM (mm ²)	Ratio of Pin to Halo
(0, 0)	(177, 178)	(177, 178)	0	2.19×10^5	21.1	40.2	17	3.77
(0, 1)	(177, 165)	(179, 166)	2.24	9.78×10^4	19.2	32.7	34	2.36
(1, 0)	(190, 178)	(187, 179)	3.16	1.87×10^5	19.8	38.7	16	3.53
(1, 1)	(190, 165)	(188, 168)	3.61	1.54×10^5	19.3	34.9	21	2.83

summed sinogram where each individual pin sinogram overlaps the others and results in a higher number of counts. In some places the overlaps match very well between the measured sinogram and the analytically generated sinogram. However, the scales on these two sinograms are different. There are fewer counts in the measured sinogram than in the analytically generated sinogram. This means that the places where the overlaps occur have a lower source intensity than the analytic model anticipates. There also appear to be more places where the overlaps pile up in the measured sinogram compared to the analytic model generated sinogram. This is related to a previously discussed issue. The binning of the collimator positions when creating the measured data sinogram is not sufficiently precise to match the analytic code sinogram and as a result the direct source counts are being spread over 2, 3 or 4 positions whereas the analytic code only spreads them over 1-2 positions. When the source locations are so close together, this problem becomes particularly apparent, as evidenced by the reconstruction of the summed sinogram, pictured in Figure 87.

The reconstruction of these 4 points did not result in an image with 4 distinct source locations. The reconstruction image, shown in Figure 87, shows a center source point and then 3 blurred sources around that origin where the other points did not fully resolve. The unresolved source points occur because the rotation of the collimator blurs the source detection enough that the measured sinogram is not sufficiently precise. For example, Figure 88 includes the analytic and measured sinograms of point $(-1,-1)$. The source sinograms have roughly the same shape and the highest source intensities are in the same places in each sinogram, but the measured data is blurred over 2-3 positions because of the way that the data is binned. This can be seen better in Figure 89 which is the sinogram created by taking the ratio of the number of counts in each position for the analytic code generated data over that of the measured data. The line which defines the positions where the detectors see the highest source intensity is much sharper on the analytic sinogram. There is also a dark blue echo of the directly detected source points that can be seen in the ratio plot. That echo shows where the analytic sinogram has fewer counts than the measured sinogram. Just to the left of that echo is a place where the two sinograms have similar source counts. This shows that the measured sinogram has spread the direct source out over more positions than the analytic sinogram. The excess spread is represented by the dark blue echo. When the source intensity is spread over 2 positions, as in the measured data, some of the more

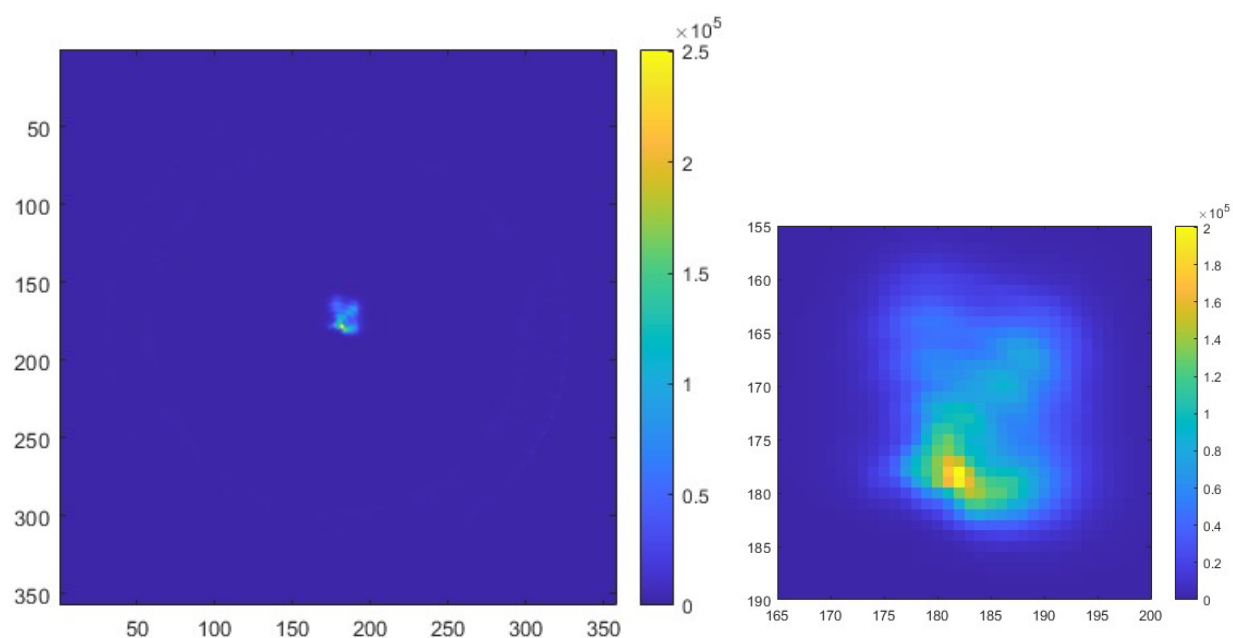


Figure 87. Reconstruction of fuel pins (0, 0), (0, 1), (1, 0), and (1, 1) in the whole FOV (left) and a zoomed in version (right).

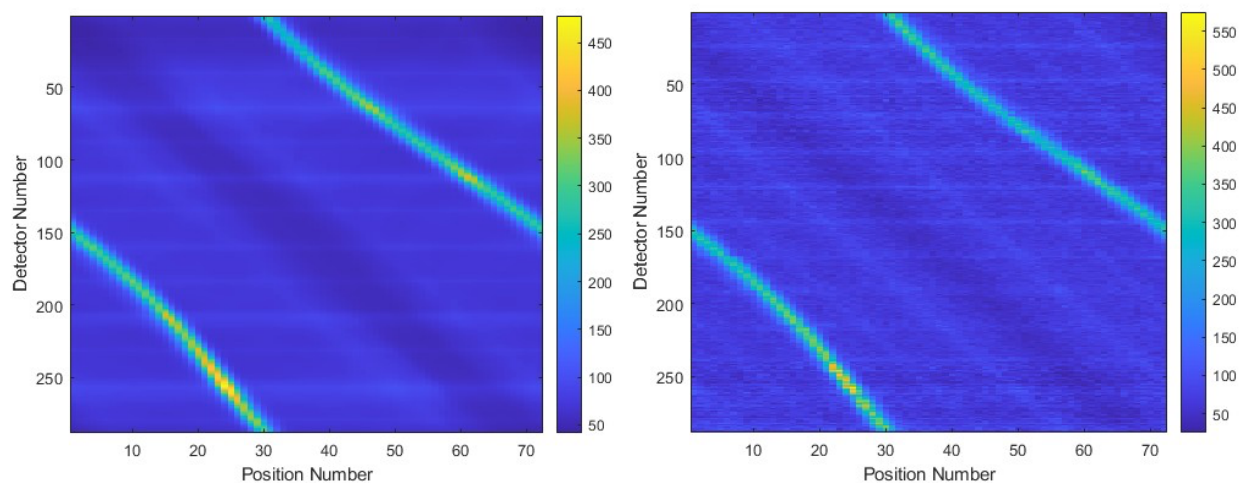


Figure 88. Comparison between the analytic sinogram for point (1, 1) (left) and the measured sinogram (right).

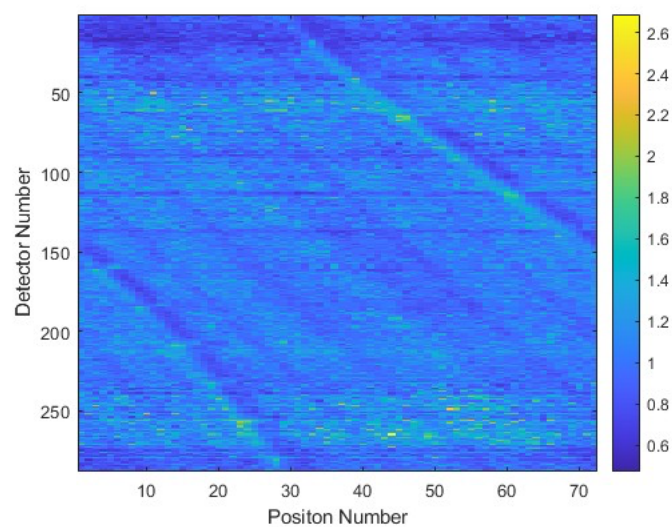


Figure 89. Sinogram created by taking the ratio of the measured sinogram for (1, 1) to the analytic sinogram for (1, 1).

defining features of the sinogram are blurred. Expanding the analytic code to match the blurring from the measured data loses some location data. When all these smeared source points are combined into one sinogram to try to reconstruct all points, there are not enough distinct differences in the sinogram to localize the points. The measured data for each source position does not preserve the shape of the location well enough to distinguish individual pins. The spatial resolution is defined by how well the measured data sinograms can preserve the individual shape of each of the source positions. In order to preserve the shape perfectly the spatial resolution would have to be sufficient to differentiate whether a source is present for each millimeter across the field of view. The current spatial resolution does not do so and cannot tell the difference between fuel pins with a gap of 3.28 mm.

The issues with the spatial resolution come from how the measured data is binned into various positions. The constant rotation of the collimator results in a constant stream of data coming from slightly different collimator positions that are binned into one single position. Previously, this issue was mentioned in Section 7.1.1 and the analytic code was expanded to try to account for the collimator rotation. However, without location data corresponding to the collimator location for each read out, it is difficult to bin the data such that each position is even as distinct as the 4 positions averaged together in the analytic model. The rotation of the collimator means that the locations for each source peak are smeared across two collimator positions where they should be in only one. If more exact location data were available correlating the readout data with the location of the collimator, the data would be able to be binned in a more precise way. Instead of the bins accounting for a 5-degree spread around the collimator position location, more precise collimator location data could reduce that 5-degree spread to potentially 2 or 3 degrees. While this would result in the elimination of some collected data to create more precise bins, increasing the collection time and number of rotations would allow the system to compensate for the decreased counts used in the detector bins.

A reconstruction image was created by separately reconstructing each of the fuel pins and then summing the resulting reconstructions into one “Summed Reconstruction” image. Figure 90 shows the results of this process. When the pins are reconstructed individually, 4 distinct fuel pins can be seen. Table 17 details the differences between the two sinograms. Since there are no distinct fuel pins in the summed sinogram reconstruction, the values calculated in the table for

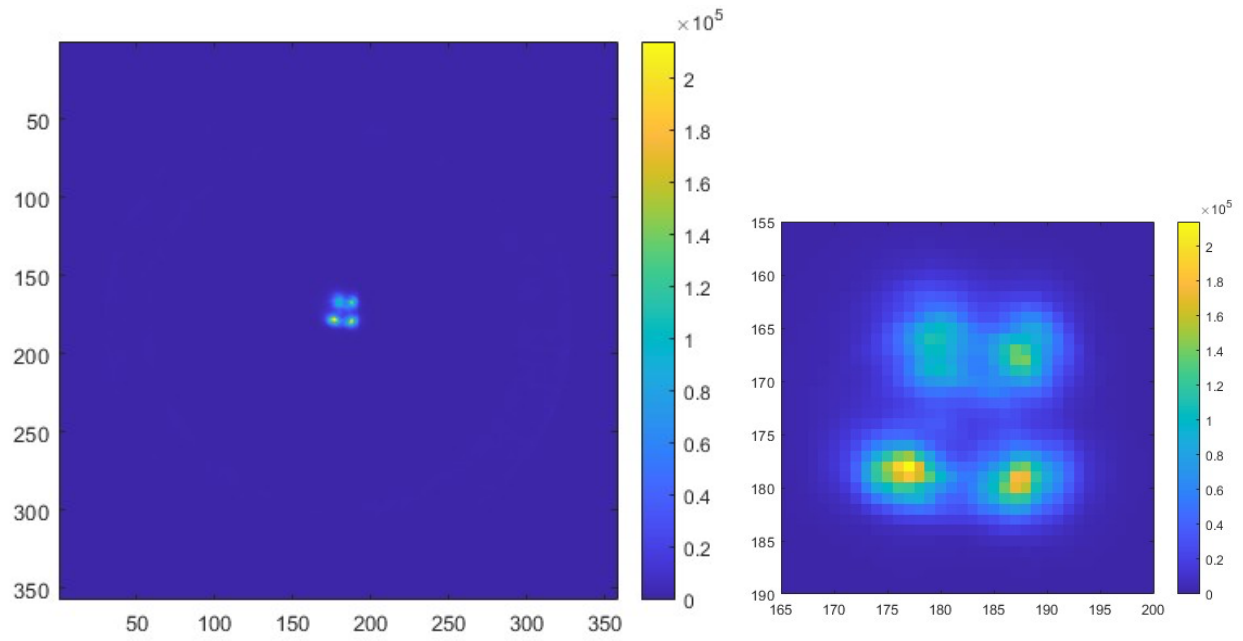


Figure 90. Reconstruction created by reconstructing fuel pins (0, 0), (0, 1), (1, 0), and (1, 1) individually and then summing the reconstruction results in the whole FOV (left) and enlarged to focus on the pins (right).

Table 17. Comparison of fuel pins (0, 0), (0, 1), (1, 0), and (1, 1) when reconstructed separately and then added together and when reconstructed simultaneously from one sinogram.

Fuel Pin	Expected Pixel Location	Actual Pixel Location	Δ (mm)	Count Rate at Pixel	% of Total Reconstructed Counts	FWHM (mm ²)	Ratio of Pin to Halo
Summed Reconstruction Fuel Pins							
(0, 0)	(177, 178)	(177, 178)	0	2.14×10^5	11.7	18	3.41
(0, 1)	(177, 165)	(179, 166)	2.24	1.10×10^5	9.03	64	2.05
(1, 0)	(190, 178)	(187, 179)	3.16	1.78×10^5	10.4	20	2.83
(1, 1)	(190, 165)	(188, 168)	3.61	1.41×10^5	9.35	40	3.32
Summed Sinogram Fuel Pins							
(0, 0)	(177, 178)	(177, 178)	-	7.00×10^4	0.00731	106	0.702
(0, 1)	(177, 165)	(179, 166)	-	4.60×10^4	0.00609	148	0.686
(1, 0)	(190, 178)	(187, 179)	-	1.11×10^5	0.0114	119	0.904
(1, 1)	(190, 165)	(188, 168)	-	7.59×10^4	0.00863	150	0.850
	-	(182, 179)	-	1.59×10^4	0.0141	46	1.23

the summed sinogram are using the actual locations in the summed reconstruction for the center of the fuel pins. As a result, no difference between the expected and actual locations is reported for the summed sinogram. There is a fifth row at the bottom of the table under summed sinogram. The same calculations for each fuel pin were performed on the spot of highest source intensity in the summed sinogram image. This pixel is (182, 179) which is located along the x axis halfway between the actual fuel pin locations for (0, 0) and (1, 0). Given that the individual reconstructions yield similar intensity results for (0, 0) and (1, 0), the reconstruction code appears to just combine the two points into one higher source intensity location halfway between the two.

Since the reconstruction code could differentiate between two sources located at the smallest distance apart from one another (0.328 cm) in 7.3.2, it made sense to examine the summed reconstructions of just 2 of the 4 pins. First, pins (0, 0) and (0, 1) were reconstructed together. Figure 91 shows the results of these two. The reconstruction does not resolve into two distinct fuel pins. Neither does the reconstruction of fuel pins (0, 0) and (1, 0), as shown in Figure 92. In both images the source intensity is smeared between the two fuel pin locations. Fuel pins (0, 0) and (1, 1) are located farther away from one another at 0.464 cm after determining that the origin position cannot be resolved with the closest fuel pins, the next smallest fuel pin gap was examined. The reconstruction of these two pins is shown in Figure 93. When these two points are reconstructed simultaneously two distinct source pins can be differentiated; however, one of the source pins is not located where expected. Fuel pin (0, 0) is no longer located at the center of the inspection volume, but instead was reconstructed at pixel (187, 179). This is the same location where fuel pin (1, 0) is expected to reconstruct. In comparison, fuel pin (1, 1) reconstructs to the location of which matches the reconstruction of the individually reconstructed sinogram, pixel (188, 168).

It has been established that the reconstruction code attributes more source to the center of the FOV due to increased sampling in the center of the inspection volume. As a result, certain combinations of fuel pins were examined without the fuel pin (0,0) present to see how that impacted the reconstruction. Fuel pins (1, 0) and (1, 1) were reconstructed first and the resulting image is shown in Figure 94. These are two separate fuel pins with their centers in the same location as in the individually reconstructed images. Between the code's ability to resolve fuel

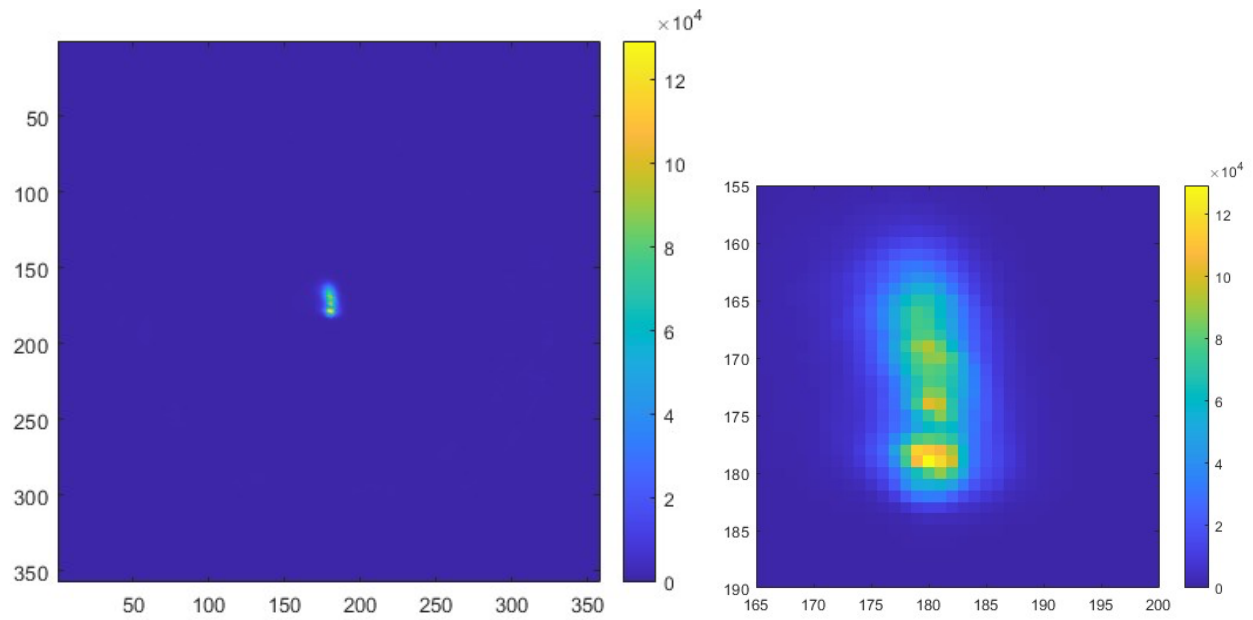


Figure 91. Simultaneous reconstruction of fuel pins(0, 0) and (0, 1) in the full FOV (left) and a close up (right).

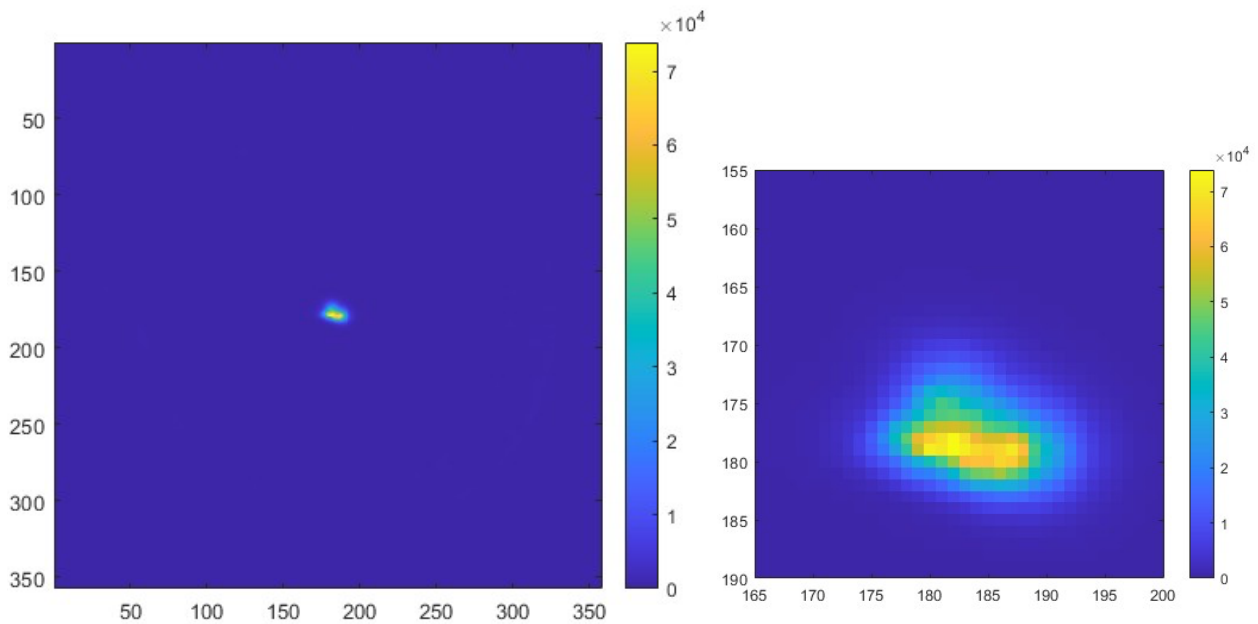


Figure 92. Simultaneous reconstruction of fuel pins (0, 0) and (1, 0) in the full FOV (left) and a close up (right).

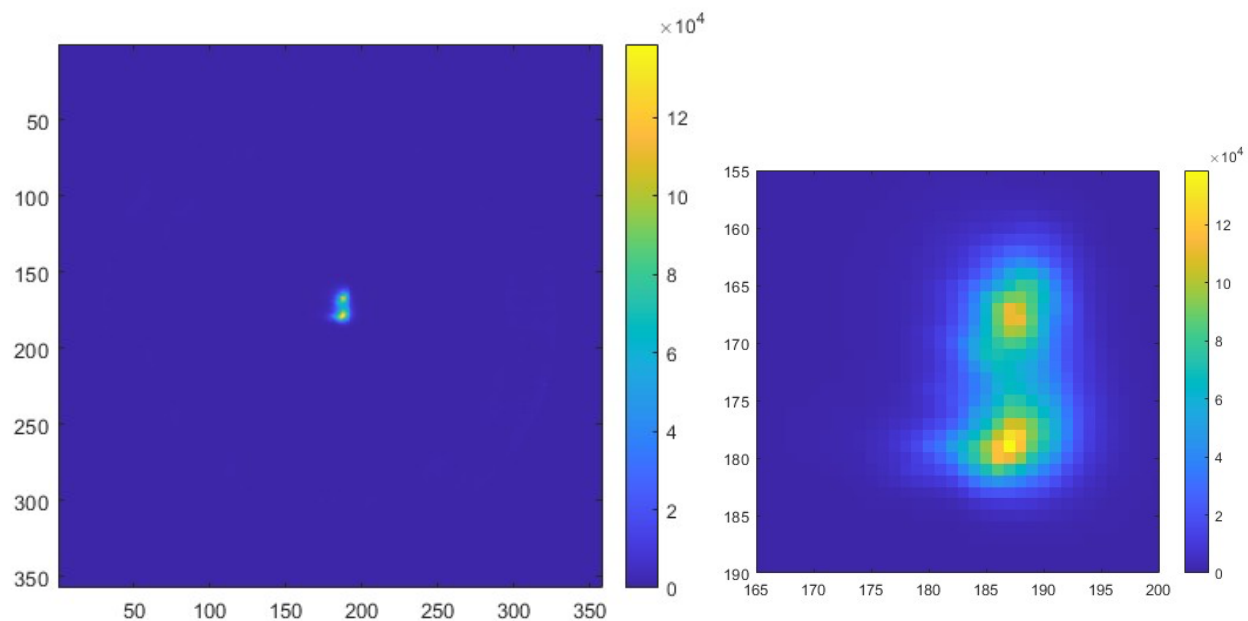


Figure 93. Simultaneous reconstruction of fuel pins (0, 0) and (1, 1) in the full FOV (left) and a close up (right).

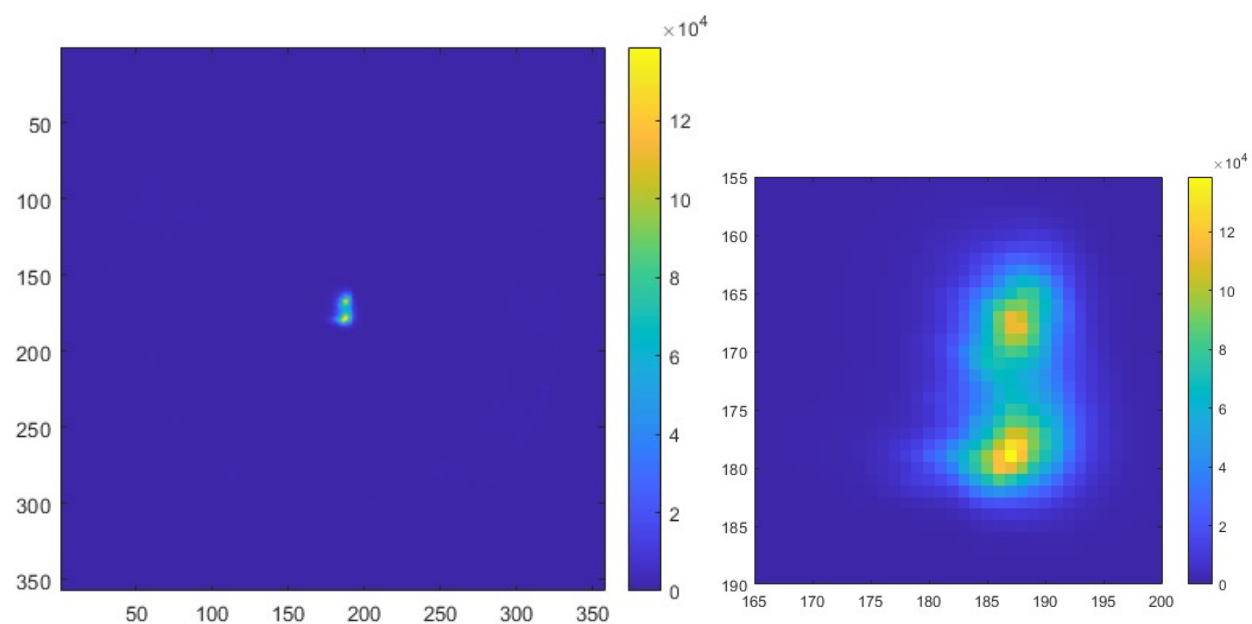


Figure 94. Simultaneous reconstruction of fuel pins (1, 0) and (1, 1) in the full FOV (left) and a close up (right).

pins at positions (0, 2) and (1, 2) and the ability to reconstruct fuel pins at positions (1, 0) and (1, 1), it appears the reconstruction code can differentiate fuel pins which are 0.328 cm apart if the fuel pin at (0, 0) is not one of the pins. The extra source attributed to the center of the FOV combined with poor position binning makes it such that there is too much extra noise and uncertainty in the reconstructions when (0, 0) is reconstructed next to another pin.

Following this realization fuel pins (1, 0), (0, 1) and (1, 1) were all constructed simultaneously to determine if the imager could resolve 3 pins that close together. Figure 95 shows the resulting image. Again, the result is a blur of source intensity between the 3 expected fuel pin positions. The imager is not capable of resolving 3 measured fuel pins at close distances. Fuel pin (1, 0) can be seen as a distinct pin, but pins (0, 1) and (1, 1) are blurred together. The spatial resolution of the imager is not sufficient to resolve 3 measured pins in close proximity to one another, let alone a full measured fuel assembly.

8.2.2 *Simulated Reconstructions*

After determining that the measured data could not reconstruct three pins simultaneously, it was decided to use simulated data to see if that would be reconstructed. Using the analytic neutron response code, a simulated sinogram was created for points (0, 0), (1, 0), (0, 1), and (1, 1). Reconstructing this simulated sinogram proved possible as shown in Figure 96. All 4 pins show up as distinctly different locations with well-defined gaps between them. After reconstructing 4 simulated pins successfully, it was relevant to see how many fuel pins could be added to the simulation and still reconstruct. Fuel pins were added to the system in layers, shown in Figure 97. Beginning with the 4 pins in the black square, pins were further added where a fuel pin would be located. Guide tube locations were left empty. Following the 4 pins measurement, the 8 pins outlines with an orange square were reconstructed. Reconstructions were run for each colored square successively, ending with the pink square containing 72 pins which make up one quadrant of the assembly.

The sinograms of the 8, 21, 31, 40, 54, and 72 fuel pin configurations are shown in Figure 98. As progressively more pins are added the sinograms become congested as the response from each pin begins to overlap with other pins. When adding the sinograms of each individual fuel pin response together the direct transmission of neutrons is accounted for, but so is all of the

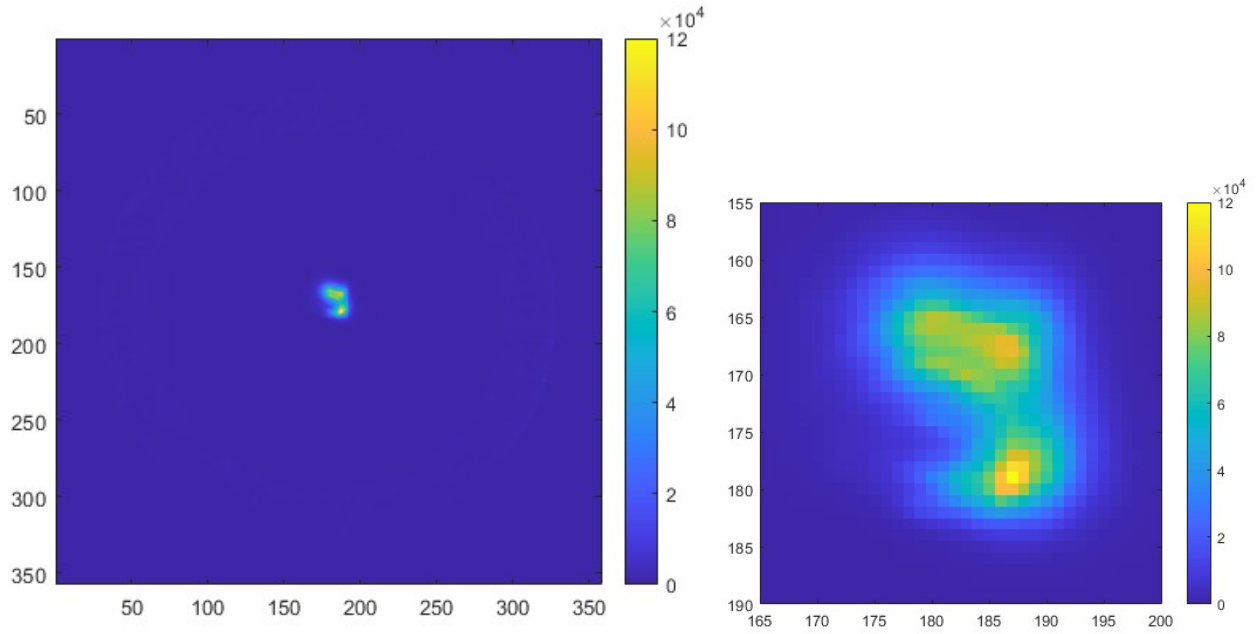


Figure 95. Simultaneous reconstruction of fuel pins (0, 1), (1, 0), and (1, 1) in the full FOV (left) and a close up (right).

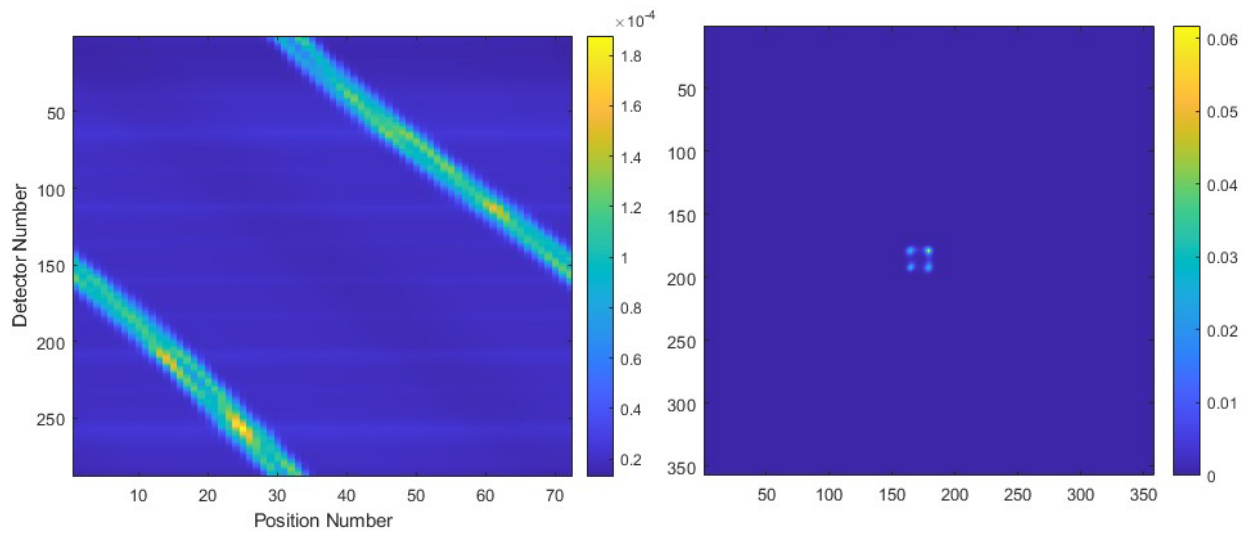


Figure 96. Simulated sinogram for 4 fuel pins (left) and the reconstruction of the pins (right).

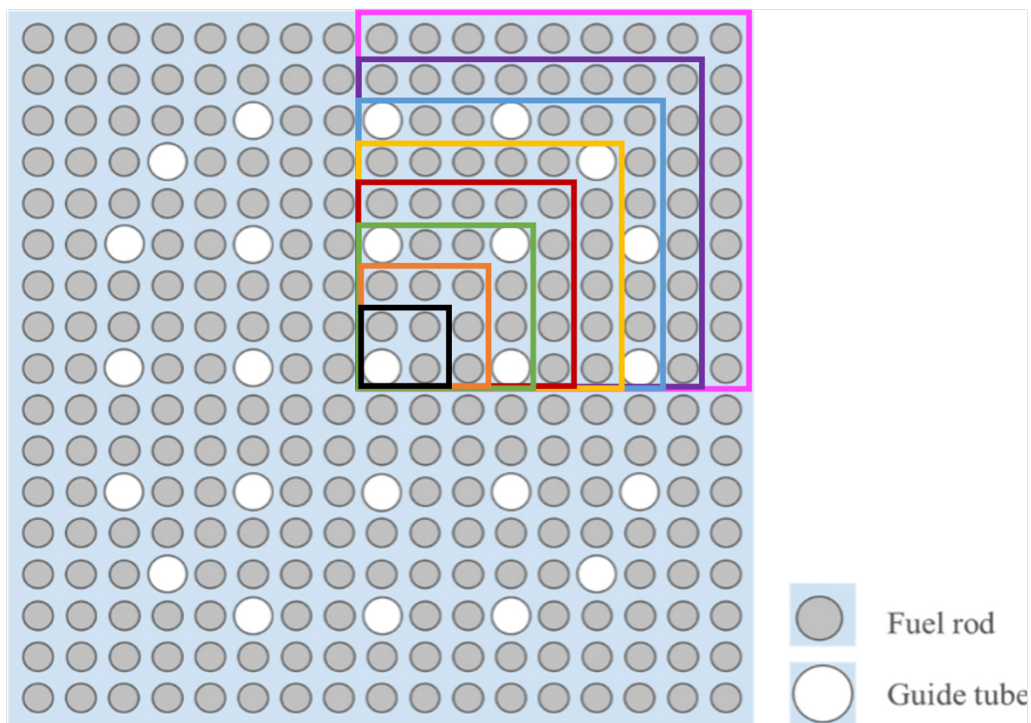


Figure 97. Fuel pin configurations defined by each square on the full fuel assembly.

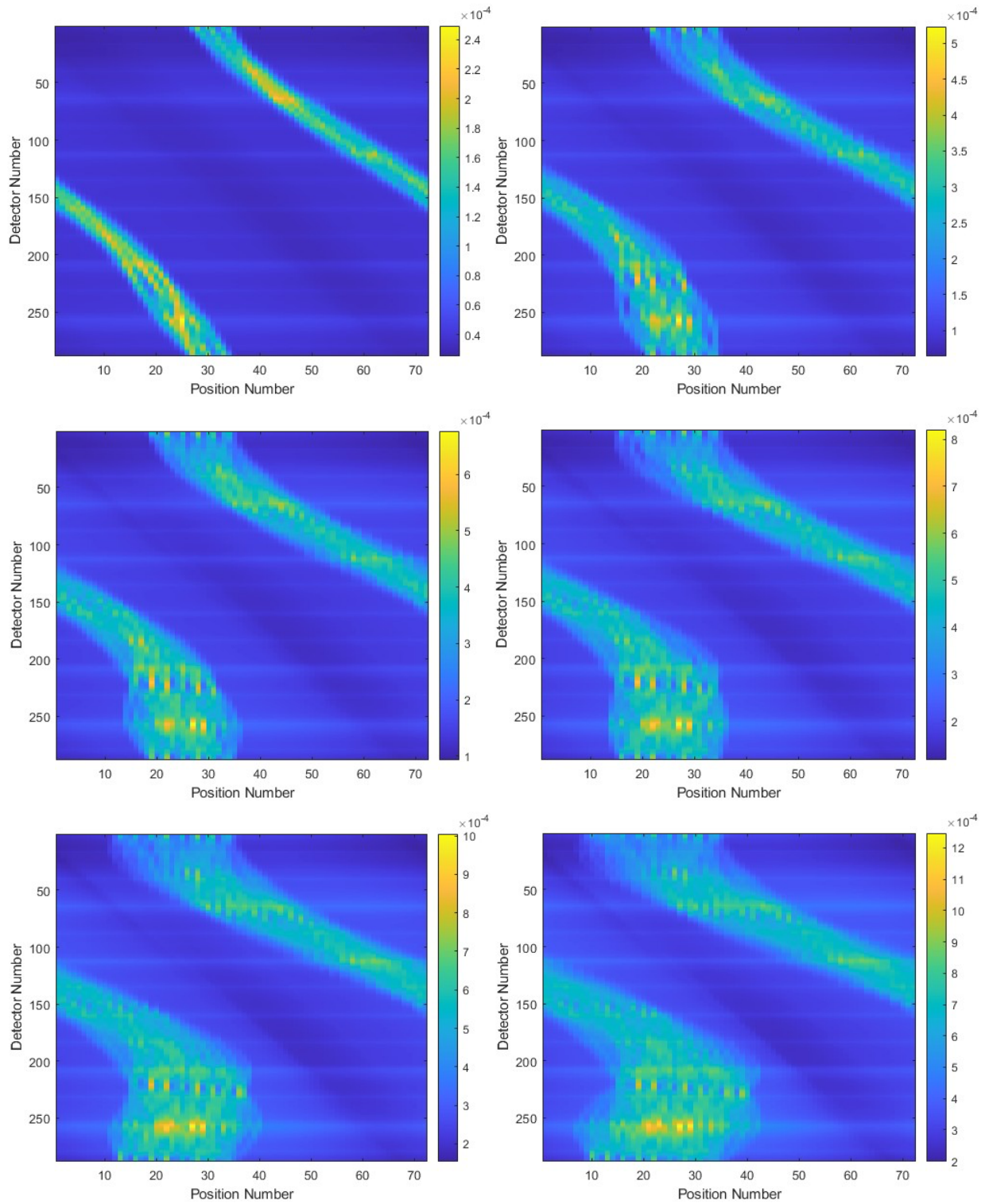


Figure 98. Select simulated sinograms for 8 pins (top left), 21 pins (top right), 31 pins (middle left), 40 pins (middle right), 56 pins (bottom left) and 72 pins (bottom right).

neutron scattering happening in each simulated fuel pin. The background scattering neutrons recorded from the summed positions increases. With the higher background scattering as well as more direct transmission from all pins summed together, the reconstructions at certain pins begin to worsen. Figure 99 shows the reconstructions for 8, 21, 31, 40, 54, and 72 fuel pin configurations. The reconstruction code does well with identifying 8 pins simultaneously, but as the number of pins is increased, the fuel pins located toward the center of the inspection volume become harder to distinguish. Fuel pins (0, 1) and (0, 2) as well as (1, 0) and (2, 0) become smeared together enough that it is not possible to identify them as separate pins for pin configurations larger than 21 pins. Additionally, for fuel pin configurations 54 and 72 the fuel pins (2, 1), (1, 2), and (2, 2) are very faded and pin (1, 1) is nearly unidentifiable as a fuel pin.

The reconstruction code is spreading the source intensity from (1, 1) to the area between (2, 1), (1, 2), and (2, 2). The guide tube positions where there is no source pin present are easily identifiable as well. While the reconstruction code can accurately reconstruct many simulated source pins, beyond 31 fuel pins the code is not able to continue accurately reconstructing the center fuel pins. Comparing the 8 pin sinogram and the 72 pin sinogram shown in Figure 98, the fuel pins located at the center of the volume have nearly the same number of counts as the background scattering neutrons. The source pin counts are approximately 6 times larger than the background scatter for the 8 pin case. However, for the 72 pin configuration, the source pin counts are approximately 1.6 times the background scatter neutrons. For fuel pins located towards the center of the volume, the source neutrons are only 1.2 times the background scatter. The increased background from adding more fuel pins makes it difficult to tell the center fuel pins apart from the background scatter. As a result, the source intensity is smeared around the region of those pins obscuring the location of the pins. More work would have to be performed to improve the analytic neutron response model for the fuel pins close to the origin pin.

8.2.3 *Attenuated Reconstructions*

Despite the system being unable to reconstruct a measured full fuel assembly with or without attenuation, some of the previously examined source measurements can be re-examined with attenuation rods present. The addition of the attenuating rods serves to decrease the signal to noise ratio by decreasing the amount of signal detected.

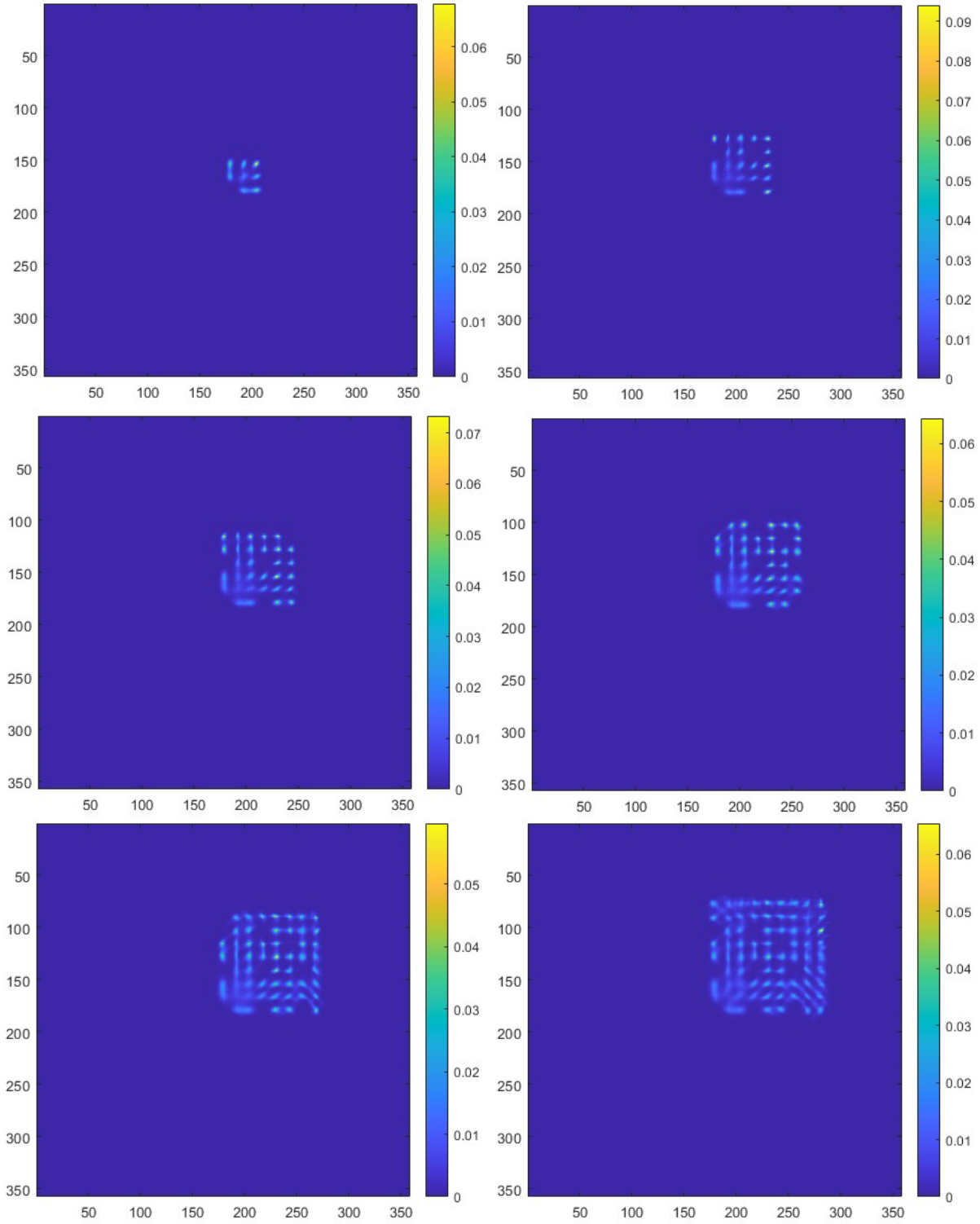


Figure 99. Select simulated reconstructions for 8 pins (top left), 21 pins (top right), 31 pins (middle left), 40 pins (middle right), 56 pins (bottom left) and 72 pins (bottom right).

The origin measurement is surrounded by the attenuating rods, meaning that in all directions fast neutrons from the center pin have a high likelihood of interaction. This adds a lot of scattering to the system which increases the noise in the sinogram. Figure 100 shows the measured sinogram for fuel pin (0, 0) with attenuation rods (left) compared to the measured sinogram without attenuation rods in order to demonstrate how much more noise is added to the system. The background in the attenuated sinogram is much higher than in the unattenuated and the counts coming from fuel pins which directly sample the source pin are much lower. In the unattenuated sinogram, 21.1% of the total counts detected during the measurement are from direct sampling of the fuel pin. In comparison, only 12.0% of the total counts in the attenuated measurement are from direct sampling of the fuel pin. The total number of counts in the attenuated sinogram is 1.3067×10^6 , whereas the total number of counts in the unattenuated sinogram is 1.4148×10^6 . So, the total number of counts detected in the attenuated measurement is only 92.36 % of the unattenuated counts, and of the counts that are detected fewer of them come from direct transmission from the fuel rod. The other 88% of the counts in the attenuated measurement come from scattering.

The reconstruction of fuel pin (0, 0) with attenuation is shown in Figure 101. The difference between the unattenuated and attenuated sinograms lead to differences in the respective reconstructions. In comparison to the unattenuated reconstruction which has a FWHM of 17 mm² and a pin to halo ratio of 3.77, the attenuated measurement has a FWHM of 19 mm² and a pin to halo ratio of 2.53. The addition of noise to the system coming from scattering off the attenuated rods led to spreading of the source intensity across more pixels around the center of the FOV. The percentage of source intensity attributed to a fuel pin located at (0, 0) with no attenuation is 40.2% of the source intensity, whereas that value has decreased to 14.5% in the attenuated measurement. Much more of the source intensity in the attenuated measurement is attributed to other locations outside of the fuel pin. Some of this source is being attributed to faint artifacts that can be seen in the reconstruction in Figure 101. There are two faint rings which can be seen surrounding the origin fuel pin. With the addition of noise to the system, the excess source counts are being attributed to these rings. Despite having accounted for the collimator tie rods previously in Section 7.2.3, the addition of more scattering in the system makes the excess counts detected around the tie rods higher than for the unattenuated measurement. The ring

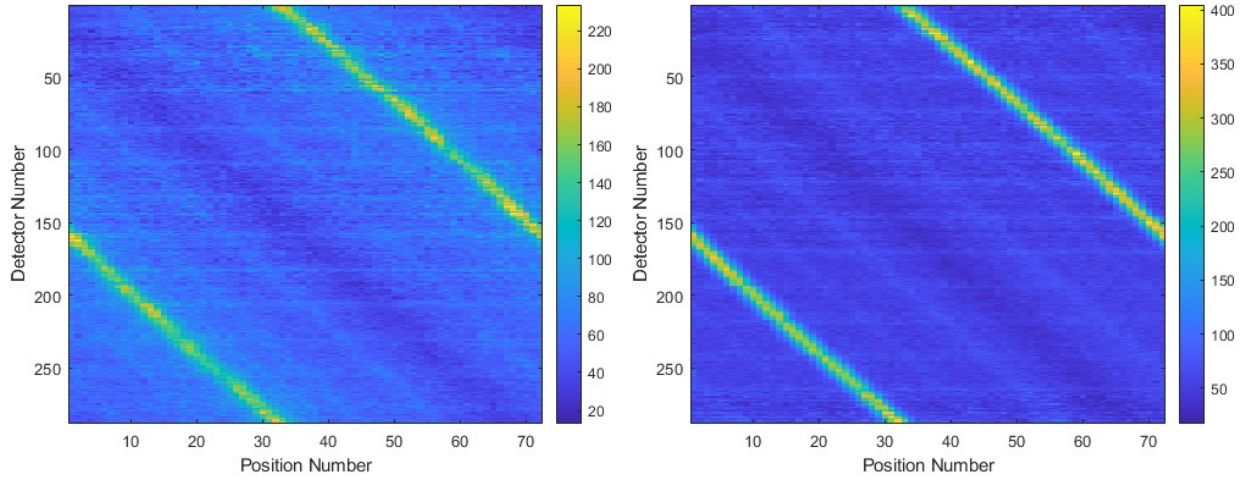


Figure 100. Measured sinogram for (0, 0) with attenuation rods in place (left) and measured sinogram for (0, 0) without attenuation rods (right).

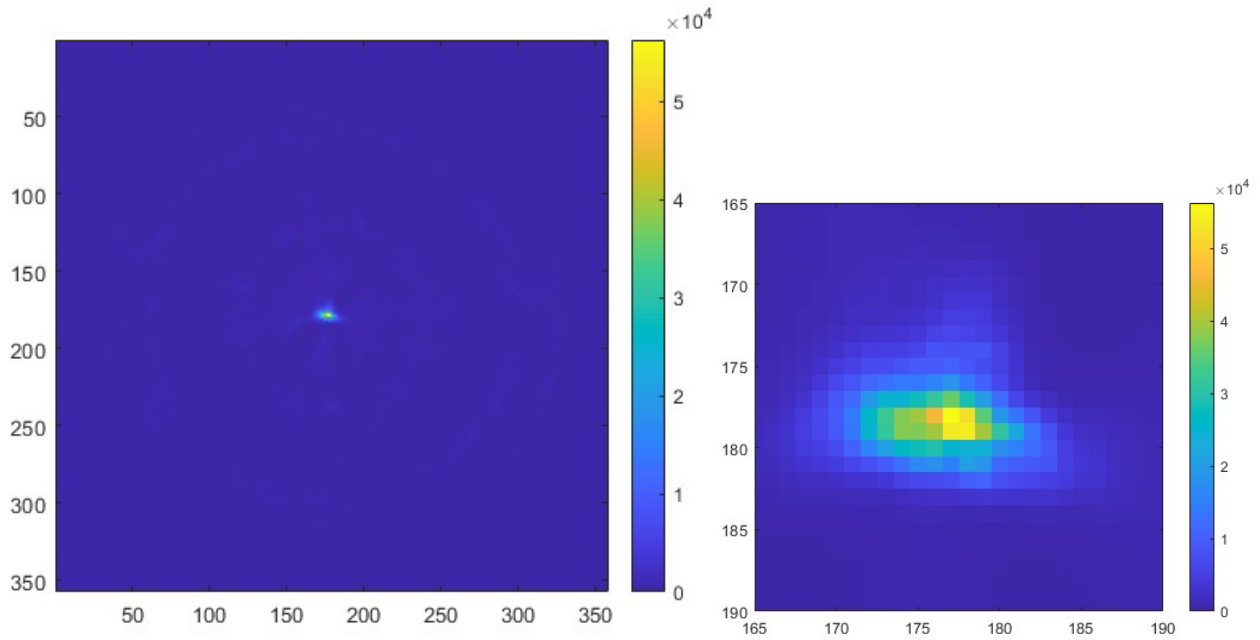


Figure 101. Reconstruction of (0, 0) with attenuation rods in place within the full FOV (left) and zoomed into the center (right).

artifacts are a result of extra source counts near those tie rods being reconstructed in the system. Better accounting for these tie rods with more scattering occurring would likely remove the ring artifacts from the reconstruction.

The corner measurement has the same challenges as mentioned in Section 7 due to the lower sampling rate along the edges of the inspection volume, but the reconstruction becomes more challenging when the attenuating rods are added. Figure 102 shows the attenuated sinogram compared to the unattenuated sinogram for fuel pin (8, 8). The attenuation rods decrease the signal going to part of the detector, while some of the signal remains the same because of the location of the source rod. The attenuating rods do not surround the corner measurement and so they only decrease the neutron intensity in certain directions. This can be seen in the sinogram where the parts with the greatest source intensity are those where there are no rods between the detector and the source. In the places where there are many rods between the source and the detector, the source is so attenuated that the source nearly blends in with the increased scattering background from neutron scattering off the stainless-steel rods. The total number of counts in the attenuated sinogram is 1.1688×10^6 and of that 15.7% of the counts are from directly sampling the source. In the unattenuated sinogram, the total number of neutron counts is 1.4114×10^6 and 17.3% of those come directly from the fuel pin. The attenuation rods decrease the total number of counts by 17.19% between the two measurements.

The attenuated reconstruction generated for (8, 8) is shown in Figure 103. Much more of the source is attributed to the edges of the inspection volume in the attenuated measurement than in the unattenuated. This includes the highest source intensity pixel, (274, 63), which is 14.3 mm from the source location and along the edge of the inspection volume. In fact, very little source is attributed to the location (279, 73) at all. The pixel with the second highest source intensity which is not located along the edge of the inspection volume is (284, 85). This is the highest pixel within the source blur, and it is located 9.85 mm from the corner source rod. The amount of source attributed to the corner location of (279, 73) where the center of the source rod is located is only 8.9% of the source at pixel (284, 85).

Finally, the 5 points were again examined with attenuation rods in place. Figure 104 shows the comparison of the measured attenuated sinogram and the measured unattenuated sinogram. The unattenuated sinogram has a total count of 7.8159×10^6 neutrons, while the attenuated

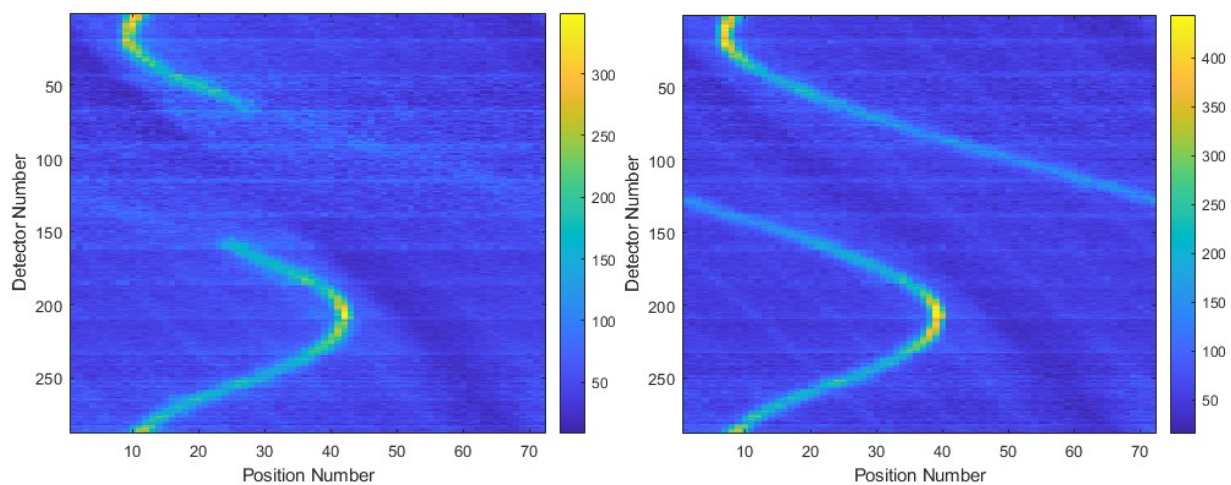


Figure 102. Measured sinogram for (8, 8) with attenuation rods in place (left) and sinogram for (8, 8) without attenuation rods (right).

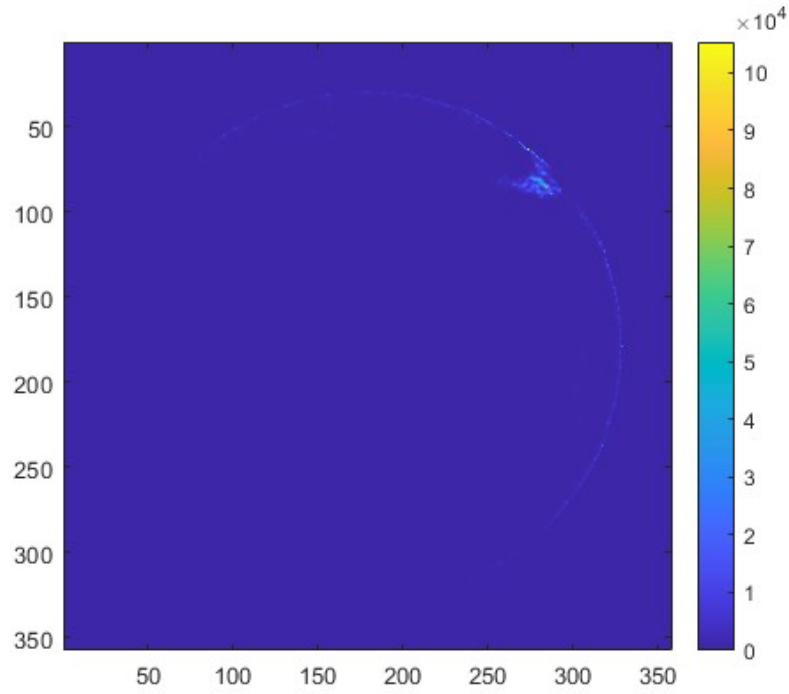


Figure 103. Reconstruction of (8, 8) with attenuating rods in place.

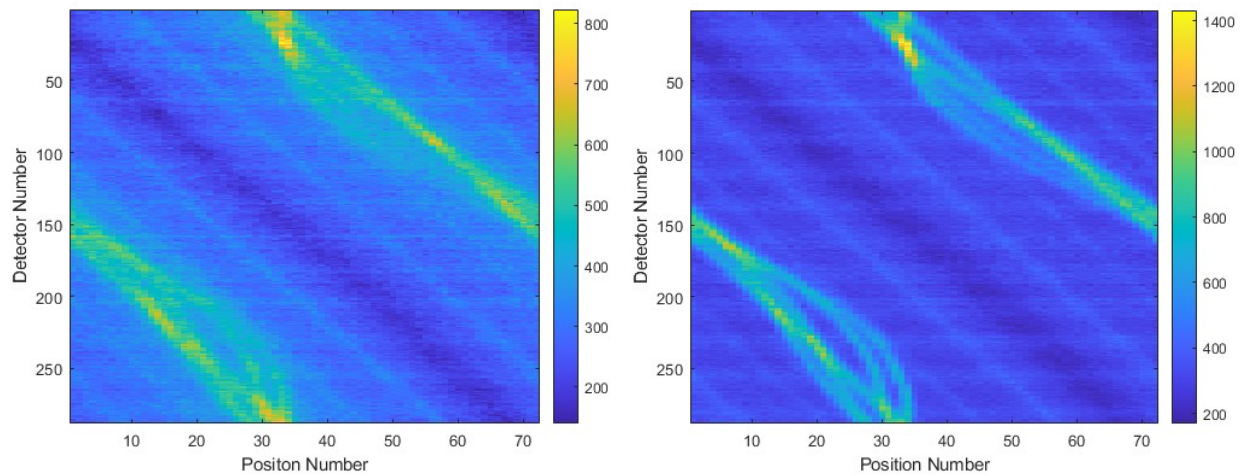


Figure 104. Measured sinogram for fuel pins (0, 0), (0, 2), (1, 2), (3, 1), and (4, 0) with attenuation rods in place (left) and sinogram for the 5 fuel pins without attenuation rods (right).

measurement detected a total of 6.6725×10^6 neutrons. This is only 85.37% of the total counts for the unattenuated measurement. Additionally, fewer counts come from the direct detection of fuel pins in the attenuated measurement.

Figure 105 shows the reconstruction image created by summing the sinograms of each of these 5 points. While it is still possible to visually tell that 5 separate sources are in the image, many more artifacts are introduced. There appear to be ring artifacts surrounding the origin point at different radii, which as previously discussed, likely come from the increased scattering counts around the collimator tie rods. Each of the fuel pins appears to have an oblong shape as well. For comparison, a summed reconstruction sinogram was created from reconstructing each attenuated fuel rod separately and then scaling and summing their results into one image. Figure 106 shows that summed reconstruction. The summary of the differences between the two reconstruction images is reported in Table 18. Of note is the difference between the summed sinogram FWHM for (2, 0) of 51 mm^2 and the individually reconstructed FWHM for pin (2, 0) which is 140 mm^2 . A fuel rod in the system takes up an area which would be 69 mm^2 the source intensity around this point is so diffuse that the FWHM goes far beyond the fuel pin area and into the 0.328 cm gap between pins. The amount of source attributed to this location is so spread out that there is no visible difference between where the fuel pin is and where the gap would be. This is another instance where the imager can resolve 2 fuel pins which are 1 fuel pin gap apart but adding a third fuel pin close to the other two would not be resolvable.

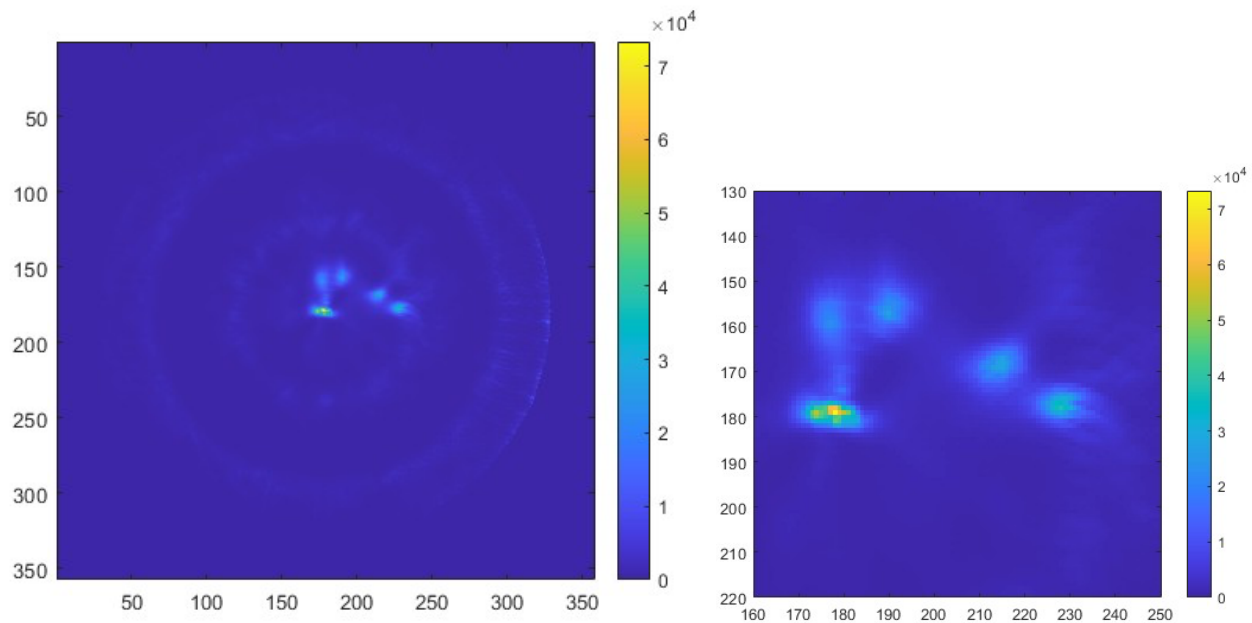


Figure 105. Reconstruction of fuel pins (0, 0), (0, 2), (1, 2), (3, 1), and (4, 0) with attenuation rods present.

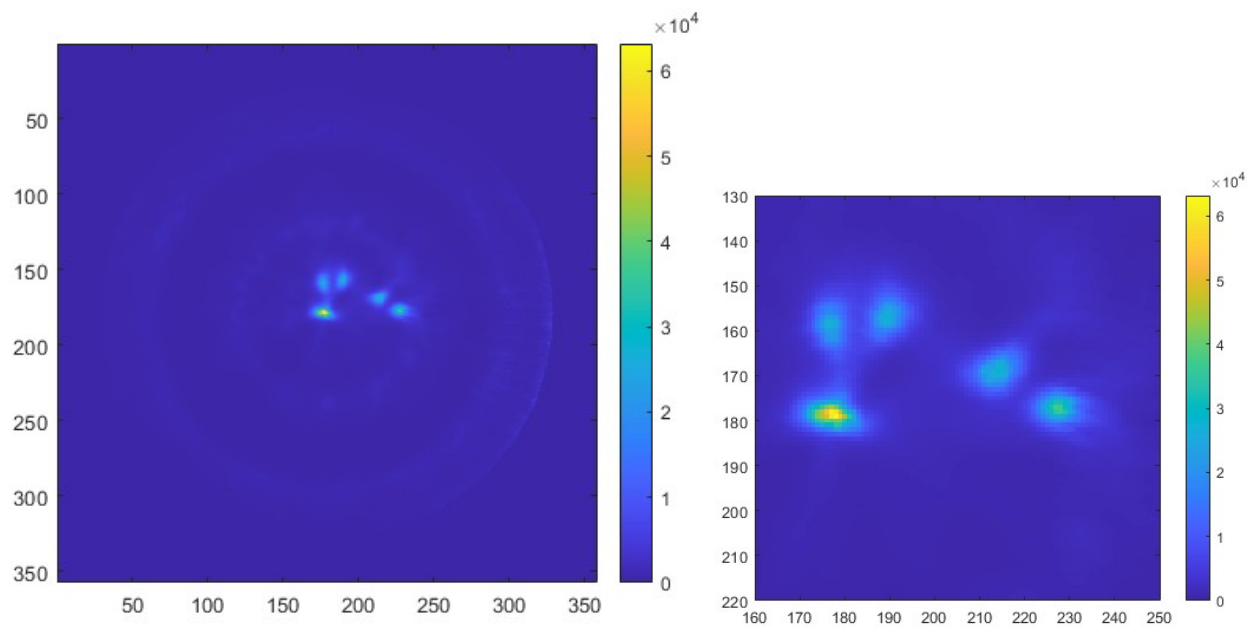


Figure 106. Summed reconstruction of fuel pins (0, 0), (0, 2), (1, 2), (3, 1), and (4, 0) with attenuation rods present.

Table 18. Comparison of fuel pins (0, 0), (0, 2), (1, 2), (3, 1), and (4, 0) with attenuation rods present when reconstructed separately and then added together and when reconstructed simultaneously from one sinogram.

Fuel Pin	Expected Pixel Location	Actual Pixel Location	Δ (mm)	Count Rate at Pixel	% of Total Reconstructed Counts	FWHM (mm ²)	Ratio of Pin to Halo
Summed Reconstruction Fuel Pins							
(0, 0)	(177, 178)	(177, 178)	0	6.14×10^4	4.06	31	1.76
(0, 2)	(177, 152)	(177, 158)	6	2.73×10^4	2.20	51	1.27
(1, 2)	(190, 152)	(189, 157)	5.01	2.81×10^4	2.30	51	1.27
(3, 1)	(216, 165)	(214, 169)	4.42	2.89×10^4	2.46	51	1.48
(4, 0)	(229, 178)	(227, 177)	2.24	3.76×10^4	2.72	38	1.71
Summed Sinogram Fuel Pins							
(0, 0)	(177, 178)	(178, 179)	1	5.91×10^4	3.84	35	1.61
(0, 2)	(177, 152)	(177, 159)	7	1.79×10^4	1.94	140	0.901
(1, 2)	(190, 152)	(190, 157)	5	2.07×10^4	1.95	98	1.00
(3, 1)	(216, 165)	(215, 168)	3.16	2.35×10^4	2.16	71	1.71
(4, 0)	(229, 178)	(228, 178)	1	3.09×10^4	2.51	48	1.41

Chapter 9. Conclusion

9.1 Conclusions

In this dissertation a fast neutron emission tomography system was described, constructed, characterized, and tested. Picking up from Dr. Iyengar's NET Imager design, modifications had to be made when constructing the imager. These modifications included the reducing of the collimator slits from 96 to 72 and a change in the spacing between the collimator slits. An analytic neutron response model as well as a radiation transport model were also updated to reflect this change. Measurements were taken to verify the absolute efficiency of the detector modules as well as determine the bias voltage and trigger threshold which yield the best efficiency for the modules. The modified imager was constructed onsite at ORNL. An initial measurement campaign was performed using the imager which demonstrated its ability to collect and reconstruct measured data using the analytic neutron response model. Following the initial observations, further characterization of the imager took place. First, the relative efficiency of each detector in the imager was determined through measurement and incorporated into the analytic response code. Following that, representative source locations at (0, 0) and (8, 8) were used to determine changes needed for the analytic model. This included the addition of an angular factor to the slit scattering calculation as well as changes to the collimator's effective thickness values. Finally, a measurement campaign took place to find the imager response to fuel rod positions with and without attenuation rods in place and to determine if the spatial resolution of the system would be less than 3 mm.

The NET system as currently built is not able to be used to image whole spent nuclear fuel assemblies. The spatial resolution is not sufficient to be able to distinguish multiple fuel pins at gaps of 3.28 mm. The imager can distinguish many cases where two fuel pins are 3.28 mm apart, but the addition of another pin at 3.28 mm would not be able to be reconstructed currently. The room return of the system accounts for as much as 12% of the total source counts in a measurement. Increased background counts coming from room return, as well as limited data about the collimator location for each data read out, render the measurements too imprecise to image multiple fuel pins. Without the addition of more borated polyethylene to the system or the addition of the missing 6 detector modules, the room return of the system cannot be decreased.

Fast neutrons from ^{244}Cm have an average energy of 2 MeV when produced by spontaneous fission. These neutrons have a mean free path of 6.75 cm in borated polyethylene. Adding an additional 10 cm of borated polyethylene to the exposed sides of each detector module would significantly decrease the room return of the system by attenuating these neutrons. Alternatively, having all 12 detector modules present with 10 cm of borated polyethylene behind them would also reduce the room return. Reducing the system noise by 10% would allow for better signal to noise ratios and better fuel pin reconstructions. Additionally, without being able to determine the collimator rotation angle with 0.8 degree precision, the data cannot be more precisely binned and evaluated.

9.2 *Future Work*

Going forward, there is some additional work which could be done with the imager. The first thing that would be necessary would be to ensure that collimator rotation position was accurately and frequently reported in output data. The best way to do this without making significant changes to the imager would be to integrate the motion of the motor into the DAQ scalar output. Currently, the analytic code uses 288 collimator positions to generate lines of response. This corresponds to a collimator position at every 0.8 degrees. In order to match this number of positions, a scalar value for collimator position should be output by the DAQ which can track the collimator location with the same angular accuracy or better through all collimator rotations. It would be beneficial to start collecting collimator data with the collimator starting in an established “home” position from where all measurements begin. Synchronizing the start of collimator movement with the start of data collection would allow the user to be able to read out exactly where the data collection starts for each rotation. This would be sufficient to match the analytic model’s positioning. For the present work, the readout rate of the DAQ system for the imager was set to 5 readouts per second. At 1 revolution per minute, this works out that the DAQ is currently collecting data at rate of 1.2 degrees per readout. In order to match the analytic model and collect data at a rate of 0.8 degrees per readout, the motor would have to rotate at a speed of 0.67 revolutions per minute, instead of the current 1 revolution per min. That means that each measurement would take 1.5 times as long as it currently does to match the analytic code.

Each measurement is currently taken for 10 min with the detector modules in one position and then taken for another 10 min with the detectors in the second position. If the 6 additional detectors were added to the system, the measurement would not need to be repeated. Despite rotating the collimator at a slower rate, making a single 15 min measurement with all 12 detectors would take less time while increasing the measurement precision since the data would be able to better match the analytic model. Additionally, the addition of the remaining 6 detector modules to the system would improve the quality of the data as it would remove the issues with room return increasing the neutron background. If finer collimator rotation outputs are desired, the scalar DAQ could output the location of the imager every 0.1 degrees as well as the counts in each detector every 0.1 degrees. This would require much longer measurements times if the DAQ readout rate is left unchanged. To output at a 5 readouts per second rate with outputs every 0.1 degrees, the collimator would need to rotate at a speed 0.0833 revolutions per minute or 1 revolution every 12 minutes. To complete 10 rotations would take 2 hours. At that point the readout rate for the DAQ should be changed to output at a faster speed. A rate of 20 readouts per second would allow a full rotation to occur every 3 minutes, decreasing the measuring time for 10 rotations down to 30 min per measurement.

Following these changes to the detector and the DAQ system another set of measurements should be performed as described in Section 8. The source rod data should be compiled until a whole spent nuclear fuel assembly equivalent is being reconstructed both with and without attenuation rods in place. This would demonstrate the ability of the imager to reconstruct whole spent fuel assemblies. Source points of interest should be removed from the whole assembly equivalent to determine if the imager sufficiently reconstructs missing fuel pins when there are so many sources present in the inspection volume. Measuring and reconstructing sources with different source strength would also be helpful to mock up spent fuel verification of fuel rods with different burnups to determine if the system can be used to detect rod replacements which have lower burnup than the surrounding rods.

Most PGET systems have been used to image other types of fuel aside from PWR 17 x 17 fuel assemblies, including VVER-1000 fuel. In order to be able to compare the image quality with the images from PGET, the NET system should be used to try to measure and reconstruct these smaller fuel assemblies and see how sensitive it is to missing rods. Both simulations and

measurements of other fuel assemblies should be done with the NET system to compare to the PGET systems currently in place.

In summary, a proof-of-concept fast neutron emission tomography imager has been constructed, operated, and characterized with spent nuclear fuel verification for international safeguards applications in mind. While the as built imager does not have sufficient spatial resolution to image spent fuel pins, the imager spatial resolution should be reevaluated with the suggested changes to the imager made.

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Vita

Originally from the northeast, Mairead E. Montague grew up in Bridgewater, NJ. After high school, she attended the University of California, Berkeley and received a Bachelor of Science degree in Nuclear Engineering in 2018. Before graduating with her undergraduate degree, she knew she wanted to attend graduate school. She chose to attend the University of Tennessee, Knoxville beginning in Fall 2018 to pursue a Doctor of Philosophy degree in Nuclear Engineering while pursuing a concurrent Master of Science degree in Nuclear Engineering. During this time Mairead was recognized with the Tennessee Fellowship for Graduate Excellence. Mairead completed her MS in 2019 and continued with PhD research at Oak Ridge National Laboratory. Her research interests include the development of novel detector systems for nuclear safeguards and security. Mairead has worked with neutron detection as well as scintillator testing.

After graduation, she will begin a post-doctoral appointment with Dr. Sara Pozzi at the University of Michigan as a Distinguished Postdoctoral Research Fellow with the Consortium for Monitoring, Verification, and Technology. She is incredibly grateful for all the support from her husband, family, and friends as she begins her new career.