

Characterization of the energy and angular dependence of optically stimulated dosimeter
responses in broad beam neutron radiation fields

by

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Abstract

Personal dosimeters are used in facilities where occupational radiation doses may exceed 10% of the annual radiation worker dose limit (10 CFR § 835.402, 10 CFR § 20.1502). In years past, the Panasonic UD-809 dosimeter was the preferred choice for neutron albedo dosimetry within Department of Energy (DOE) facilities [1]. However, the advent and increasing operational acceptance of Optically Stimulated Luminescence (OSL) dosimeters have shifted preferences toward Landauer's OSL-N model, particularly in DOE and Nuclear Regulatory Commission (NRC) licensed facilities [2]. Studies in the early 2000s examined the angular response of the UD-809 dosimeters from select mono-energetic neutron sources [3]. This study builds on that earlier work by modeling the expected energy and angular responses from Landauer's OSL-N from multiple mono-energetic neutron fields using the Monte-Carlo N-Particle (MCNP) transport code [4]. The objective is to refine the existing assessment framework by integrating modern Monte-Carlo methodologies and current dose quantification techniques. The aim is to enhance the accuracy of personal dosimeters and facility area monitoring systems, especially with the introduction of OSL dosimetry. This study compares the effect of angular responses with the operational dosimetry quantities ambient dose equivalent, $H^*(10)$, used in Facility Area Monitoring (FAM), and the personal dose equivalent, $H_p(10)$, used in individual monitoring. Results show that at increasing incident angle, the theoretical dosimeter response is significantly depressed compared to the dose that would be imparted to a real-world individual or real-world FAM phantom. This effect is exaggerated at lower neutron energies (near thermal), where the dosimeter response may poorly quantify the dose to personnel. At higher neutron energies above 1 MeV, a resonance effect is observed where dosimeter response is relatively constant within certain angular domains.

These insights underscore the need to consider the primary incident angle of neutron radiation when processing dosimeter results, to properly quantify and assign dose to workers or members of the public. Further, it is shown that at angles coinciding with radiation fluence from the posterior to the anterior, use of facility area monitors with phantoms may provide results that underestimate the dose imparted significantly if corrective measures are not implemented to account for the dominant fluence vector.

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Dedication

This paper is dedicated with love and gratitude to my wife, Suttan, and my son, Sonny. To Suttan, for the invaluable gift of time and space that allowed me to complete this endeavor, for instilling in me the drive to achieve beyond my aspirations, and for bringing our incredible son into our lives. And to Sonny, who might not yet grasp the significance of this work, but whose patience has been a heartwarming reminder of what truly matters—his simple desire to be near his dad, even as I navigated the demands of my studies.

Chapter 1 - Introduction

1.1 Dosimeter Concepts and Applications

Neutron albedo dosimeters are widely used in quantifying neutron radiation by interpreting the scattered and thermalized neutron responses arising from the radiation field encountered by the scattering object [5]. These devices are indispensable in tracking radiation doses in settings where neutron radiation is present. Examples include nuclear power plants, where neutron flux monitoring is critical for safety and regulatory compliance [6], medical facilities specializing in radiotherapy and radiopharmaceutical production, where precise dose measurement ensures therapeutic efficacy and safety [7], and a wide range of industrial and academic research laboratories engaged in material science and nuclear physics experiments [8]. Each of these environments presents unique challenges in radiation monitoring, underscoring the importance of sophisticated dosimetry tools like neutron albedo dosimeters for maintaining operational safety, regulatory compliance, and ensuring the health protection of workers and the public. In addition, relating the thermalized and scattered neutron response of the dosimeter to specific dose quantities towards direct or indirect human exposure can become involved. Without proper consideration of known factors leading to the specific dosimeter response, dose estimates based on the measured response can vary widely [6].

Figure 1.1 displays a rendering of a typical configuration and function of an albedo dosimeter. The dosimeter is worn on the outside of the person, in the region of the body where the measurement is meant to quantify dose received. Whole body dosimeters are often worn on the upper torso as displayed in the rendering. As radiation penetrates the body, some of the radiation is scattered into the dosimeter and deposits its energy on a film or element.

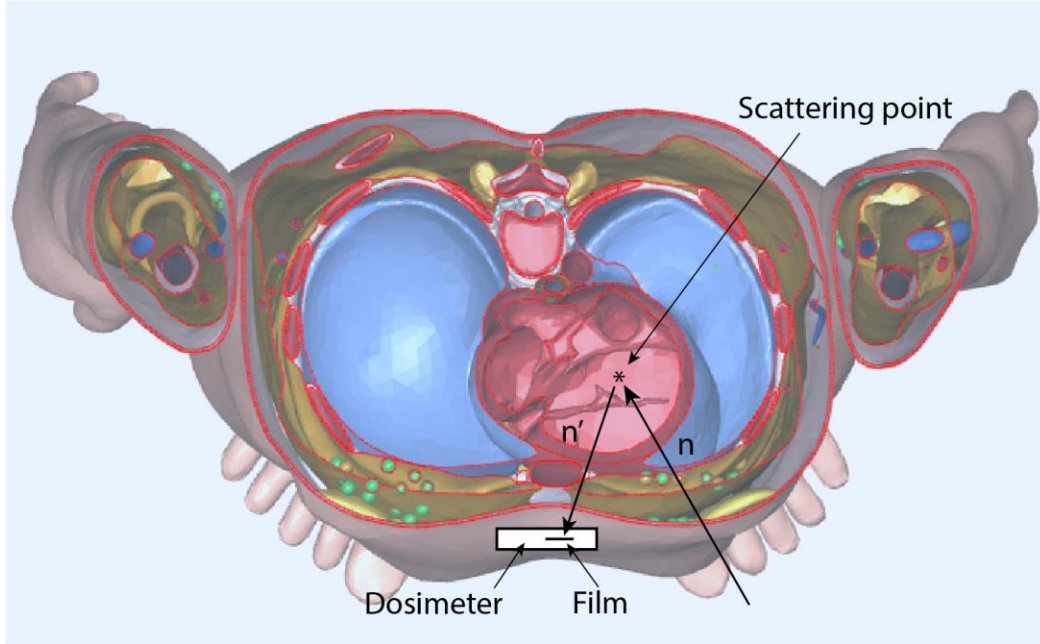


Figure 1.1. Representation of an albedo dosimeter on a Mesh-type Reference Computational Phantom [9]. Some portions of the incident neutrons (n) are scattered by the body into the dosimeter film. The film can be processed to estimate the neutron field and dose received.

Generally, when worn by workers to estimate the effective personal dose, neutron albedo dosimeters are used to measure the personal dose equivalent, $H_p(10)$. When used for Facility Area Monitoring (FAM) they are used to measure the ambient dose equivalent, $H^*(10)$. The personal dose equivalent is the dose equivalent in soft tissue at an appropriate depth, d , below a specified point on the human body, and is measured in units of $\text{J} \cdot \text{kg}^{-1}$ where one Sievert (Sv) is equal to one $\text{J} \cdot \text{kg}^{-1}$ at a depth of 10 mm in the International Council of Radiation Units and Measurements (ICRU) 4-element soft tissue [7]. The specified 10 mm depth point is usually given by the position where the individual dosimeter is worn. The ambient dose equivalent, $H^*(10)$, is defined as the dose equivalent at a point in a radiation field that would be produced by the corresponding expanded and aligned field in the ICRU 4-component tissue equivalent sphere. The point is at a depth of 10 mm on the radius vector opposing the direction of the

aligned field and is measured in the same units and the personal dose equivalent [10]. An expanded and aligned field is one where the fluence and its energy distribution are the same and the fluence is unidirectional [11]. Figure 1.2 visualizes the differences between the ambient dose equivalent and the personal dose equivalent based on an expanded and aligned field incident onto a spherical phantom.

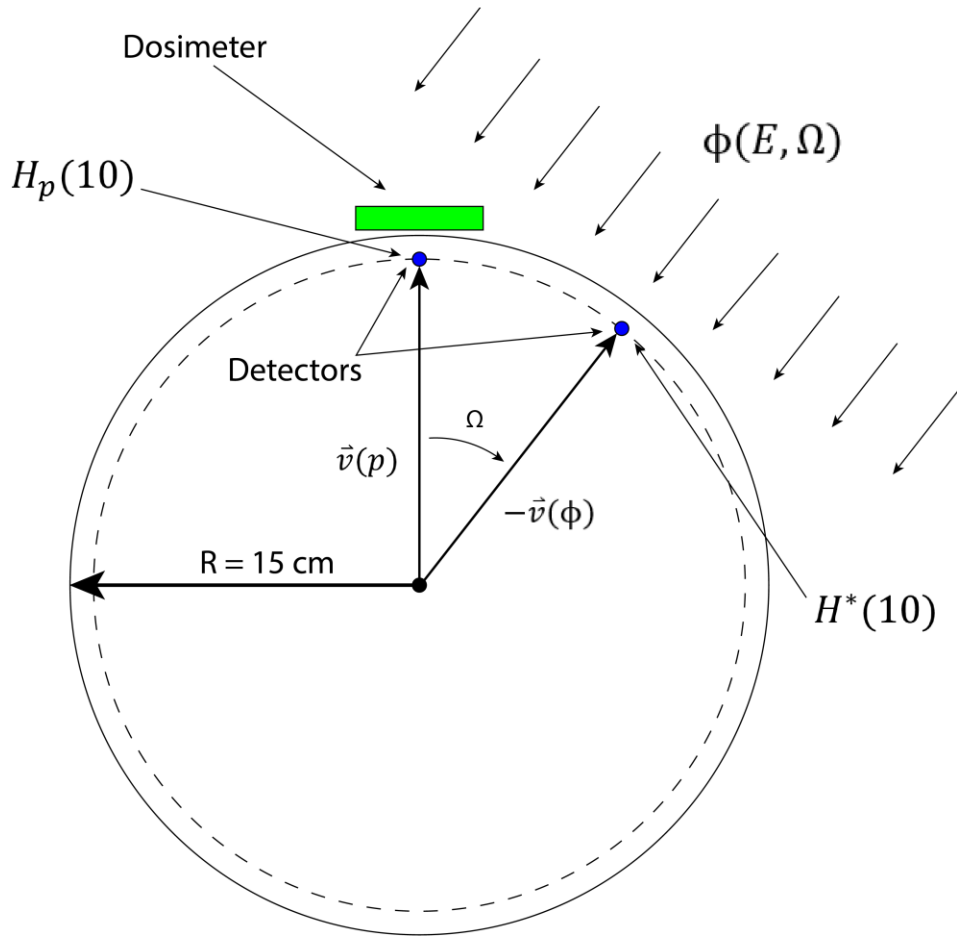


Figure 1.2. $H_p(10)$ and $H^*(10)$ theoretical dose points in a reference ICRU sphere. $H^*(10)$ is dependent on the incident expanded and aligned field on the ICRU sphere, whereas $H_p(10)$ is generally dependent on the dosimeter location and the ICRU tissue at the dose point.

Neutron albedo dosimeters typically use materials like polycarbonate, which is etched by secondary neutron interaction products as they deposit their energy along tracks within the

dosimeter substrate. The number of tracks correspond to, by way of the interaction coefficients, the number of neutrons encountered over the time of irradiation. This is then correlated to a dose via calibration to specific irradiation field characteristics. Some advanced dosimeters use enriched boron under special cadmium cover arrangements to improve sensitivity, particularly for detecting epithermal neutrons in addition to fast neutrons [6]. Ultimately though, the design of neutron albedo dosimeters often involves trade-offs in terms of sensitivity and energy response. For example, a comparative analysis between a boron carbide shielded and polyethylene moderated $^6\text{LiF:Mg,Ti}$ – $^7\text{LiF:Mg,Ti}$ albedo dosimetry system made by Instituto de Radioproteção e Dosimetria (IRD) and a boron loaded plastic with two boron-free windows $^6\text{LiF:Mg,Ti}$ – $^7\text{LiF:Mg,Ti}$ albedo thermo-luminescent dosimeter (TLD) system, manufactured by Alnor, revealed distinct differences in sensitivity and applicability, particularly in environments dominated by radionuclide sources lacking moderation. Specifically, the IRD system demonstrated a higher sensitivity—approximately five times greater than the Alnor system. This enhanced sensitivity renders the IRD system more adept at measuring low doses in occupational settings, especially in the context of bare ^{252}Cf and $^{241}\text{Am-Be}$ neutron fields where albedo dosimeters typically struggle with energy response. Consequently, the IRD TLD albedo system proved capable of accurately measuring doses as low as the Brazilian monthly recording level of 0.2 mSv, a task the Alnor system could only achieve with substantial moderation of these neutron sources [7]. This research, and other similar studies, underscore the necessity of selecting a dosimeter that aligns with the unique energy characteristics and sensitivity requirements of the neutron field in question for precise dose measurement.

Unfortunately, the practice of nuclear and radiological facilities tailoring their dosimeter selections to match the unique characteristics of each neutron field, produced by diverse

methods, is not widely adopted. Similarly, calibrating neutron dosimeters for specific individual fields remains uncommon. The practicality of implementing such dosimeter programs generally falls far out of reach in terms of the specialized expertise required, implementation time, and costs of an optimized radiation protection program [8]. More often, a single dosimeter, or category of dosimeter (generally selected based on a chosen single manufacturer) is implemented to quantify a wide range of neutron fields present at any one radiological facility. As the manufacturer and/or dosimeter processing laboratory may be blind to the fields involved in producing the irradiation of their dosimeters, these processing laboratories can only implement certain energy corrections based on characteristic dosimeter aspects like comparison of the response of different elements within multi-element dosimeters, with each element shielded by a different material providing a very coarse measurement of the neutron energies encountered [9].

Neutron albedo dosimeters contribute significantly more than just measuring radiation levels; they are pivotal in safeguarding the health and safety of individuals working in radiation-prone environments. By assisting in keeping radiation exposure within the safe thresholds recommended, these devices are essential tools in the operational health and safety arsenal. The International Commission on Radiological Protection (ICRP) is at the forefront of setting these safety guidelines. The enactment of ICRP Publication 103, for instance, has notably impacted neutron dosimetry practices. This publication has led to the reporting of lower effective doses from neutron dosimeters, as a direct result of updated neutron weighting factors [10]. This adjustment emphasizes the evolving understanding of radiation effects and the continuous effort to refine protective measures based on the latest scientific evidence

1.2 Operational Dose Quantification

The ambient dose equivalent and the personal dose equivalent both apply to strongly penetrating radiation, quantifying dose in a medium designed as a theoretical model of soft tissue. Their distinctions lie in the location of the dose measurement point, the specific medium surrounding this point, and the angle at which radiation is measured. The ambient dose equivalent is specifically associated with the ICRU sphere and presumes a consistent positional relationship to the radiation source, adhering to a standard field orientation and expansion [10]. Conversely, the personal dose equivalent does not adhere to a fixed anthropomorphic model or require a predefined field alignment, focusing solely on the medium at the measurement point [11]. Therefore, while the ambient dose equivalent calculation solely depends on the radiation field's fluence and energy characteristics, the personal dose equivalent also considers the angle of incidence of the radiation. In practice, when measuring for the ambient dose equivalent via facility area monitoring, the incident angle of radiation intuitively will also affect the measurement as there is no physical way to expand and align the field incident on a real-world dosimeter. All that can be practically done is to face the dosimeter in the direction of the neutron source and attempt to correct the response to the ICRU-sphere after processing. However, this may become impractical, and the general practice of facility area monitoring is to mount dosimeters adjacent to the field onto a wide variety of backing materials such as a wall or stand. More prudent facilities, without the expertise and/or man-hours to post process FAMs, may use physical phantoms such as polymer-based structures, or even simply containers of water, as a backing for the dosimeter. To accommodate the unpredictable outcomes of operational decisions, facilities typically employ what are believed to be conservative assumptions, or at the very least, approaches considered to be on the side of caution. These strategies aim to produce observed

responses that are easier to compare with theoretical values. Consequently, each facility may adopt a diverse array of practices to ensure these comparisons are as accurate as possible.

With the wide variety of possible implementation strategies, it becomes clear that the dosimeter readings for any given defined field can vary significantly depending on how monitoring is applied. A dosimeter worn by a worker, aimed at measuring personal dose equivalent, naturally includes the effect of angular dependence since the worker moves, causing the dosimeter to experience radiation from different angles. On the other hand, a dosimeter used for Facility Area Monitoring (FAM) measures the ambient dose equivalent. This setup, being stationary, primarily accounts for energy dependence due to its fixed position and orientation relative to the radiation source. This distinction facilitates understanding of how worker movements can lead to variations in dosimeter readings. The importance of this being consideration of both the angular and energy dependencies in dosimetry practices. For example, a FAM may be used to quantify the ambient dose equivalent in a hallway adjacent to a room where a source is located. A practical approach for positioning a dosimeter is to affix it to the wall nearest to the source room, ensuring that the dosimeter and any phantom used for simulations are aligned perpendicularly to the direction of the radiation source. Personnel may most often be aligned perpendicular to the FAM orientation as they traverse the hallway, so the position of workers or members of the public within the hallway may significantly differ from the position of the FAM. Currently, there is no straightforward or concise method to reconcile the differences in measured doses versus those that would have been recorded by a physical dosimeter worn by an individual. It is largely up to the licensee's radiation protection professionals to perform independent studies to determine any correction factors required—if the effort is made at all.

1.3 Optically Stimulated Luminescence Neutron Dosimeters

Neutron detection relies on interactions of neutrons within a material to produce secondary charged particles, which can then deposit energy in a detection material and produce a measurable signal. When luminescence dosimeters are used as the detection and signal producing material, these secondary particles and photons ionize the detector substance, producing electron-hole pairs. Generally, these carriers collide to produce secondary electrons and eventually recombine to lose their excess energy. When the substrate contains capture centers, or electron traps, these free electrons and holes relax into these traps, where they are left in a metastable state. Stimulation can remove the electrons and holes from these traps where they can recombine to a stable state by emitting excess energy in the form of photons [14]. The process of stimulating trapped electron-hole pairs via certain wavelengths of light is known as optically stimulated luminescence (OSL) and is illustrated in Figure 1.3. The emitted light photons from the return to ground-state process can be quantified via optical detection, and the number of light photons detected can be correlated to dose in the material and further to dose received by an individual [12].

Albedo neutron dosimeters rely on neutrons scattered by the user's body in the albedo configuration. Neutrons interact with a body and a fraction of those neutrons are scattered in the direction of the neutron dosimeter. The neutron dosimeter is generally primed to detect lower energy, scattered neutrons. After determining the total energy absorbed by the detection element, which is based on the proportion of neutrons likely to be deflected into it, this energy can be inversely calculated to estimate the total neutron flux the body was exposed to, and therefore, the overall dose received during the exposure period [13].

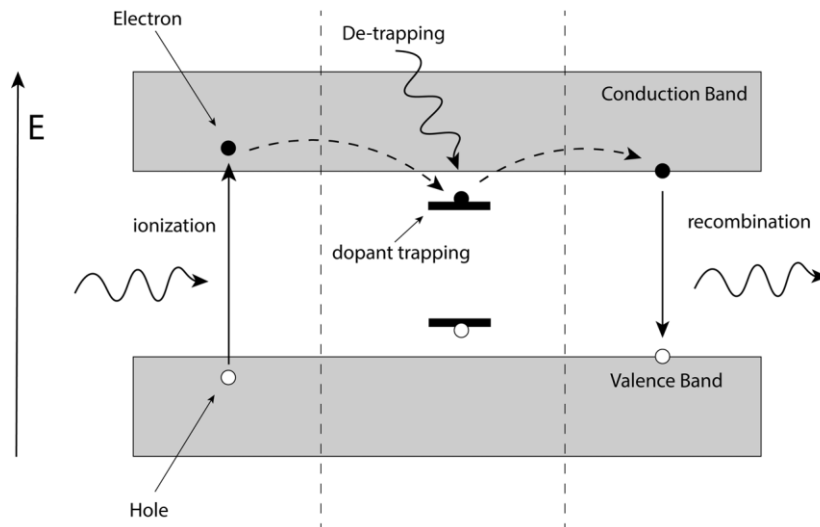


Figure 1.3. A basic diagram of the OSL trapping and de-trapping mechanism. Left to Right: ionizations move electrons into the conduction band, the electrons fall into traps created by the dopant, de-trapping excites electrons back into the conduction band, then recombine back into the valence band by emitting their excess energy as a photon.

Calibration of albedo neutron dosimeter response is dependent on the characteristics of the neutron field encountered i.e. the energy profile. If one were to model the characteristics of the neutron field via a center-of-mass system, the angular dependence of the incident field would not be a dependent factor, as neutron scattering is considered to exhibit 4π directionality in that model. Therefore, only the interaction volume and element orientation would need to be further considered. However, in a laboratory coordinate system, the energy of the scattered neutron is correlated with the angle of scatter and because the laboratory coordinate system generally represents a more macroscopic real-world, observable scattering mechanism, the angular dependence of incident neutrons should be considered [14]. The ICRP has established fluence-to-dose coefficients to facilitate the estimation of dose received by an individual, considering their spatial orientation relative to a radiation source and previous research has highlighted how the orientation of an individual's body can significantly impact the dose received [17], [15], [20]. Despite this, there has been relatively limited progress in studying how operational dosimeters'

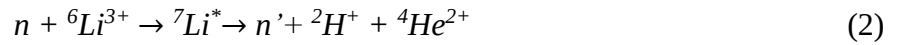
readings are affected by the angle of incidence and how correlating these readings with the variations in body orientation are observed and calculated.

Optically stimulated luminescence (OSL) dosimeters have become increasingly popular and are now the dosimeter of choice for Department of Energy facilities, which make up a sizable portion of neutron source research and development [2]. While sensitive to photons and charged particles, the most common OSL materials used commercially, $\text{Al}_2\text{O}_3\text{:C}$ and BeO , are not inherently sensitive to neutrons. The cross sections for neutron interactions in these materials are low and a direct mechanism for ionization of electrons via neutron irradiation is not probable enough to be in practical use for operational use-cases [16]. For this reason, neutron converters are used, usually containing ^6Li or ^{10}B , to interact with the incident neutrons and produce secondary charged particles, which can then create the required ionizations in the primary OSL substrate. $\text{Al}_2\text{O}_3\text{:C}$ combined with a $^6\text{Li}_2\text{CO}_3$ neutron converter has emerged as a practical, commercially available coupling [17]. The Landauer OSL-N relies on this pair by coating micro-pellets of the $\text{Al}_2\text{O}_3\text{:C}$ with a thin layer of $^6\text{Li}_2\text{CO}_3$ thus providing a neutron interaction secondary material with a high likelihood of deposition of secondary charged particles into the primary substrate [18], [21]. The thin layer has not been shown to significantly decrease the luminescence properties of the primary substrate and its usefulness in the field of health physics has been adequately demonstrated for the Landauer OSLN to take a dominant portion of the market in terms of OSL dosimeters used today [19].

The primary reaction mechanism used in the Landauer OSLN is the $^6\text{Li}(n, t) ^4\text{He}$ reaction as shown in equation (1).



This mechanism has an exceedingly high cross-section for thermal and epi-thermal energies, but the cross section decreases with increasing energy. At energies approaching 1 MeV, other reactions reach their threshold and start to approach the same probability [20]. As energy increases beyond 1 MeV, some literature notes that a new reaction begins to dominate, ${}^6\text{Li}(n, n' + d){}^4\text{He}$, as shown in (2) and displayed in Figure 1.3 [27].



In some research this reaction is attributed to an excited state of ${}^7\text{Li}$ following capture, but most databases do not explicitly list the meta-stable state [21]. In fact, the only cross section data listed in the in the Nuclear Energy Agency's Java-based Nuclear Information Software (JANIS) library for this reaction mechanism is from the Chinese Evaluated Nuclear Data Library (CENDL) and Joint Evaluated Fission and Fusion (JEFF) nuclear databases [25], [26], [27]. The National Nuclear Data Centers Evaluated Nuclear Data File (ENDF) library currently does not list cross sections for the reaction, though it does list an inelastic scattering (n, n') cross section for ${}^6\text{Li}$ [31]. Unfortunately, the resultant of simply the inelastic cross-section would not necessarily be the creation of secondary charged particles that can deposit energy into the $\text{Al}_2\text{O}_3:\text{C}$ dosimeter element, which is the mechanism of interest.

Figure 1.4 displays the cross sections for the ${}^6\text{Li}(n, t){}^4\text{He}$ and ${}^6\text{Li}(n, n' + d){}^4\text{He}$ reactions according the ENDF, JEFF, and CENDL databases. As can be observed, as neutron energy increases the cross section of the ${}^6\text{Li}(n, n' + d){}^4\text{He}$ reaction reaches its threshold and quickly approaches, and then exceeds, the cross section of the ${}^6\text{Li}(n, t){}^4\text{He}$ reaction. While the ENDF and JEFF databases agree with respect to the ${}^6\text{Li}(n, t){}^4\text{He}$ reaction, there are some differences observed between the CENDL and JEFF libraries for the ${}^6\text{Li}(n, n' + d){}^4\text{He}$. This may indicate a need for further refinement of this reaction mechanism.

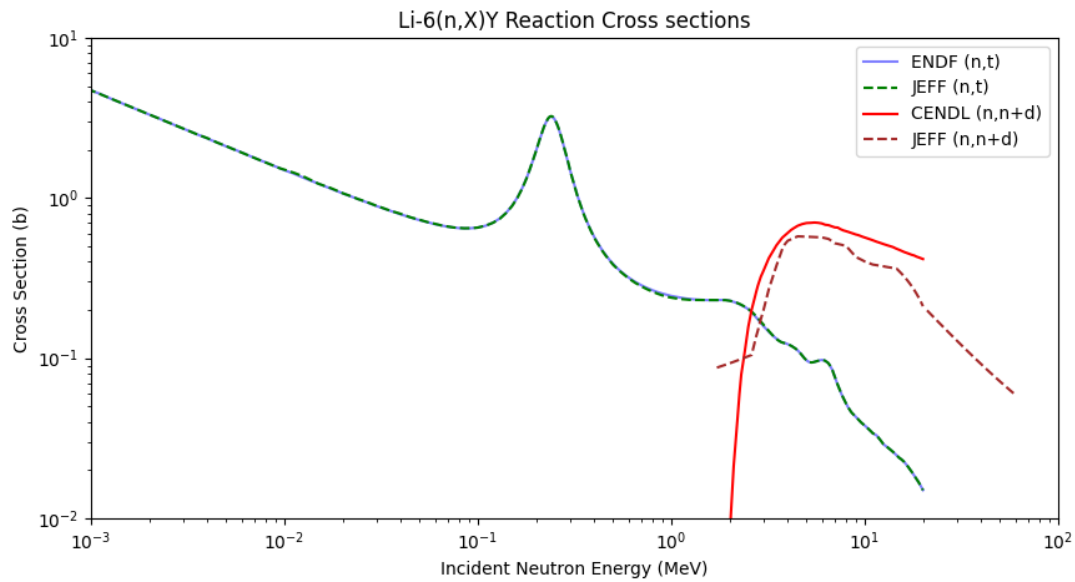


Figure 1.4. Li-6 neutron induced reaction cross sections for (n, t) and (n, n + d) (JANIS). Above 1 MeV (n, n + d) becomes more probable, according to CENDL and JEFF libraries.

1.4 Objective

Applying personal and facility area monitoring dosimeters to accurately quantify doses in specific environments has consistently presented challenges for user facilities. Manufacturers are tasked with creating dosimeters sensitive to neutron radiation, while processing facilities, often associated with the manufacturers themselves, process these dosimeters and perform basic dose assessments. These assessments rely on algorithms, created either by the manufacturers or third parties, to link dosimeter readings with radiation doses. For dosimeters comprising multiple elements, varying shielding configurations are applied to each element to better interpret the field's energy spectrum and refine dose estimates [10]. However, a significant gap in this process is the accounting for the angle of incidence, a factor typically overlooked by processors but critical for accurate dose assignment by the user facilities. These facilities must then adjust for unaccounted variables, including the angle of incidence, to ensure regulatory compliance; a task emphasized by the International Organization for Standardization (ISO) publication 21909

(generally adhered to for non-U.S. based organizations), DOE's Standard 1095-2018 (for DOE facilities), and the American National Standards Institute (ANSI) N13.32-2018 for other U.S. based facilities [22], [23]. Notably though, ANSI N13.32-2018 does not *require* an angular dependence study for standard adherence, which lack is reflected in the National Voluntary Laboratory Accreditation program (NVLAP), while ISO and DOE based accreditation programs implement it as a requirement. This leaves a potential discrepancy for NVLAP accredited laboratories, and even so, for ISO and DOE accredited laboratories the study implemented is generally limited because addressing these complexities often necessitates advanced methodologies, such as Monte-Carlo simulations, especially in structurally intricate settings. Therefore, the aim of this study is to (1) demonstrate the importance of considering incident angle of radiation when correcting dosimeter response, (2) correlate energy and angular response to provide an updated framework for post-processing corrections, and (3) provide directly applicable computational data to user facilities in order to help simplify their post-processing efforts.

Chapter 2 - Methodology

2.1 Computational Tools

Monte-Carlo N-Particle Transport Code (MCNP) version 6.3 was utilized to simulate the radiation transport incident on the ICRU sphere [4]. This version was chosen as it is currently the latest release with the most up to date ACE databases. The ICRU sphere is a sphere with a radius of 15 cm composed of four elements, namely oxygen (wt.%: 76.2%), carbon (11.1%), hydrogen (10.1%), and nitrogen (2.6%). It is meant to provide a standardized tissue equivalent material for soft tissue dose estimation [31]. The definition of the ambient dose equivalent relies directly on the dose quantified in the ICRU sphere [11]. The personal dose equivalent relies on the dose quantified in the same material as used in the sphere. Also, of note is the ISO-slab which is a 30 cm × 30 cm × 15 cm rectangular parallelepiped composed of the same material as the ICRU sphere [24]. Both geometries contain about 14 liters of soft tissue equivalent material. Generally, in real-world dosimeter calibrations, a phantom approximating the ISO-slab is used due to the ease of manufacturing a rectangular block [24]. When applying computational tools, the ICRU-sphere is generally used because it is in direct alignment with operational quantity definitions. As a computational study, this work relies on the ICRU-sphere as the primary method of determination of the personal dose equivalent and as a computational scattering phantom object for dosimeter response.

Recently, ICRP published and made free to download ICRP Publication 145, Adult Mesh-type Reference Computational Phantoms (MRCPs) [25]. The MRCPs include all source and target regions needed for estimating effective dose and can be implemented directly into Monte Carlo particle transport codes for dose calculations without voxelization, fully maintaining the advantages of the mesh geometry [36]. This provides a highly accurate to real-

world dose quantification and scattering object for determination of dosimeter response. Memory and computing power demands for the increased detail are very high and for an equal computational system, results in a drastic increase in required run-time as the particles and secondary particles are transported intricately through the unstructured mesh. This increased computational cost coupled with the direct definition of the ambient dose equivalent being in the ICRU-sphere, the MRCP was not utilized for this study.

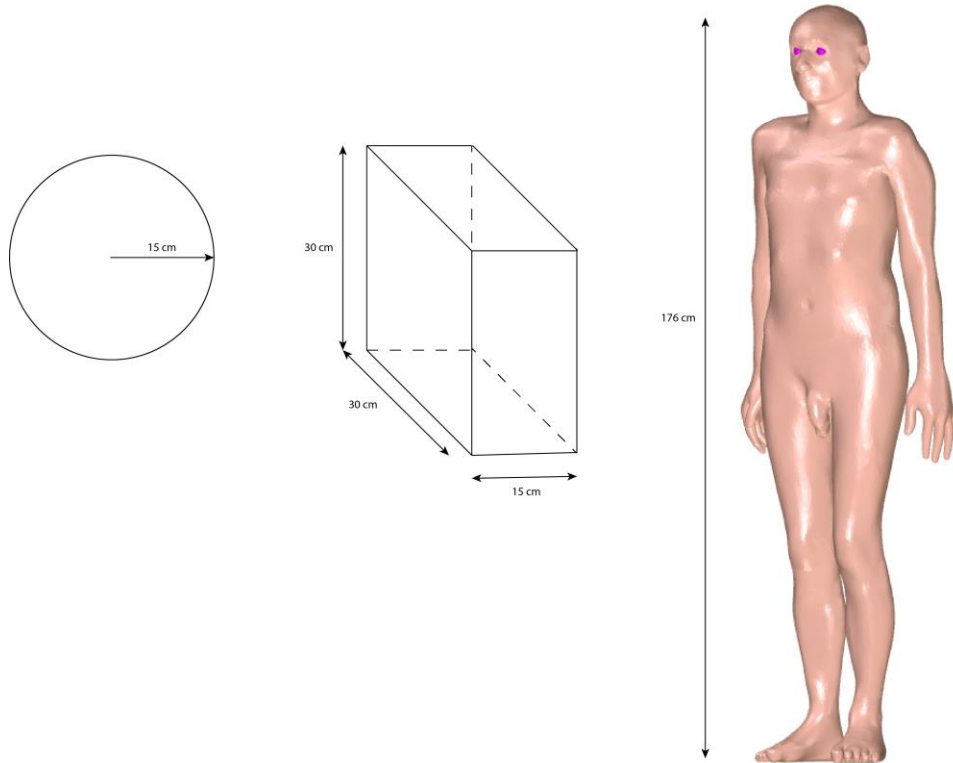


Figure 2.1. Visual comparison of computational phantoms used in dosimetry. Left to right; ICRU sphere, ISO slab, and ICRP Adult Male MCRP [9]. Images are not to scale.

2.2 Dosimeter Model

The Landauer OSL-N dosimeter was modeled utilizing openly available published technical specifications and real-world physical measurements. The four-element dosimeter is housed in the same configuration as the Panasonic UD-809 dosimeter, using an acrylonitrile

butadiene styrene (ABS) slide inserted into an ABS casing, which holds the filters [26]. For use at Department of Energy facilities, the OSL-N slide and casing are often housed in a protective clear polyethylene terephthalate (PET) housing that allows the dosimeter to be attached by clip and worn by personnel [1]. Figure 2.2 shows an image of the OSLN elements, casing, and holder.



Figure 2.2. Landauer OSL-N albedo dosimeter. The ABS element casing and PET badge holder are identical to that commonly used with the Panasonic UD-802 [37].

For the purposes of computational modeling, the ABS slide, and casing were considered one unit carrying the four elements. The PET housing was modeled separately and air gaps between the elements and their casing, as well as the casing and the housing were maintained. The housing itself was modeled as a 3 mm thick rectangular parallelepiped (RPP) with outer dimensions of $5.7 \text{ cm} \times 3.2 \text{ cm} \times 1 \text{ cm}$, with a density of $1.3 \text{ g} \cdot \text{cm}^{-3}$. The sizing of the ABS enclosure necessitated its division into two parts: the top section, which accommodates the four elements, and the bottom piece that serves to fasten the enclosure securely its holder. Together, the main portion and lower portions were centered inside of the housing with the main casing having an outside dimension of $4.9 \text{ cm} \times 1.5 \text{ cm} \times 0.55 \text{ cm}$, and the lower portion having dimensions of $4.9 \text{ cm} \times 0.8 \text{ cm} \times 0.3 \text{ cm}$. The ABS material density was estimated to be $1.26 \text{ g} \cdot \text{cm}^{-3}$. RPPs were also used to model the Copper and Aluminum filters contained within the

casings, with dimensions of $0.7\text{ cm} \times 0.1\text{ cm} \times 0.4\text{ cm}$ and $0.7\text{ cm} \times 1\text{ cm} \times 0.7\text{ cm}$, respectively. Copper was modeled with a density of $8.96\text{ g} \cdot \text{cm}^{-3}$ and aluminum was considered to have a density of $2.69\text{ g} \cdot \text{cm}^{-3}$ [27].

Within the casing and the housing right circular cylinders (RCCs) were used to model ports, model the element themselves, and the polyimide and polycarbonate element protective coatings. Each element had a thickness of 0.03 cm and a radius of 0.35 cm centered about the casing port. In addition, a polycarbonate coating with a thickness of about 0.02 cm was modeled on the front and back outer surfaces of the elements. The elements are numbered from right to left as in Figure 2.3. Element one, which is sensitive to beta radiation, was simulated with a 0.15 cm layer of polyimide film on both its front and back, covering the polycarbonate layer. Additionally, the model included an extension of element one's port beyond the casing, simulating an actual opening in the dosimeter's housing. This meticulous modeling approach was adopted to enhance the accuracy of the calculations by considering the scattering and deposition of charged particles resulting from interactions within the dosimeter itself.

For the neutron sensitive element, modeling the randomization of the lithium coated pellets is difficult and using lattice geometries can add significant run-time for each individual input file. Due to the high number of runs required an approximation was adopted to model the element substrate and $^6\text{Li}_2\text{CO}_3$ converter as matrixed together as a single unit based on the published mass fractions of the converter and the substrate. Together the mixture had a density of $3.03\text{ g} \cdot \text{cm}^{-3}$ with mass fractions displayed in Table 2.1. This assumption may be problematic if real-world particle trajectories significantly affect charged particle deposition in the $\text{Al}_2\text{O}_3\text{:C}$ portion of the neutron element [39]. Due to the high number of particles involved though, and the albedo effect utilizing non-primary field neutrons but the thermalized and scattered neutrons

from all incident angles, it was accepted as a sufficient methodology for the scope of this study. Follow-on research may be required to determine the distinguishable effects, if any, that particle trajectories may have on randomized lattice vs matrixed element construction for dosimeter use.

Table 2.1. Dosimeter Element Mass Fractions [35]

Element	Mass Fraction
^6Li	0.078670
^7Li	0.004829
C	0.083365
O	0.568507
Al	0.264628

The assembled dosimeter was then aligned with the ICRU-sphere to be in the center cross-section at 0.4 cm outside of the outer surface of the sphere. This was the distance recommended by Landauer to account for the users clothing at the upper breast, which is generally where individuals wear chest dosimeters either attached directly to their clothing via the dosimeter clip or hung on a lanyard [37].

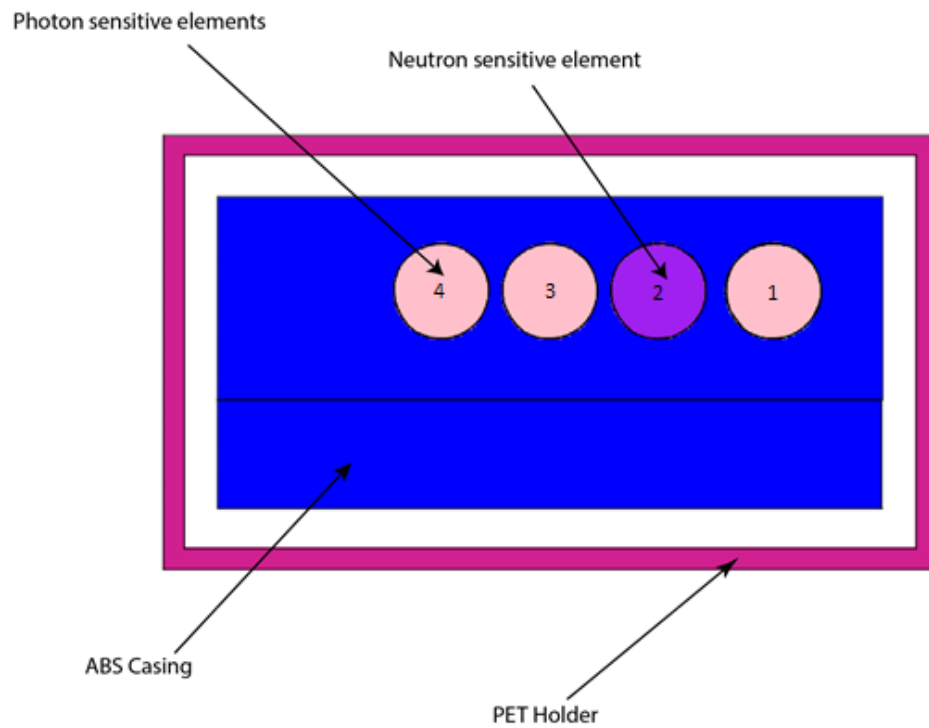


Figure 2.3. MCNP Model of OSLN Dosimeter, Front View

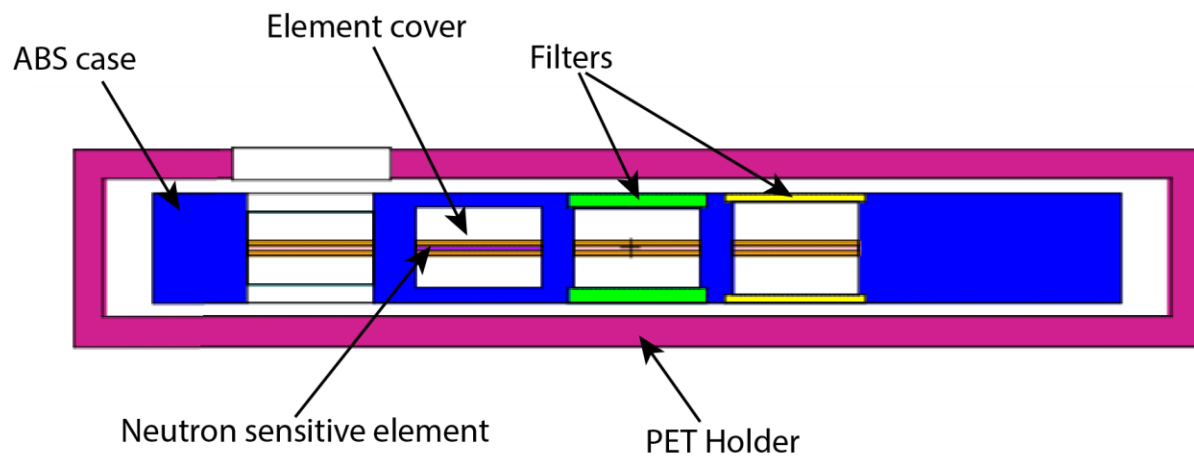


Figure 2.4. Dosimeter Elements in ABS holder, Top View

2.3 Dose Estimation

Since the ambient dose equivalent is defined using the ICRU sphere, theoretically, no matter the angle of the expanded and aligned field the dose will be the same. When modeling the result using Monte-Carlo, there should only be slight differences within a statistical uncertainty due to the stochastic calculation method and nature of radiation transport. The personal dose equivalent, on the other hand, is defined at a fixed point—1 cm depth where the dosimeter is worn. Due to this key difference, the personal dose equivalent and ambient dose equivalent are compared in relative terms, based on the personal dose equivalent. To estimate these values, a sphere of volume of 1 cm³ was constructed within the ICRU sphere, centered at a depth of 1 cm behind the dosimeter for $H_p(10)$, while another sphere was located at a depth of 1 cm in the opposite vector of the expanded and aligned field for each angle for $H^*(10)$. Figure 2.5 provides a visual representation of the personal dose equivalent (left) with an incident radiation field normal to the dosimeter. On the right is a comparison of the ambient dose equivalent and the personal dose equivalent points are displayed, with the ambient dose equivalent moving along the 1 cm depth track, opposing the direction of the expanded and aligned field.

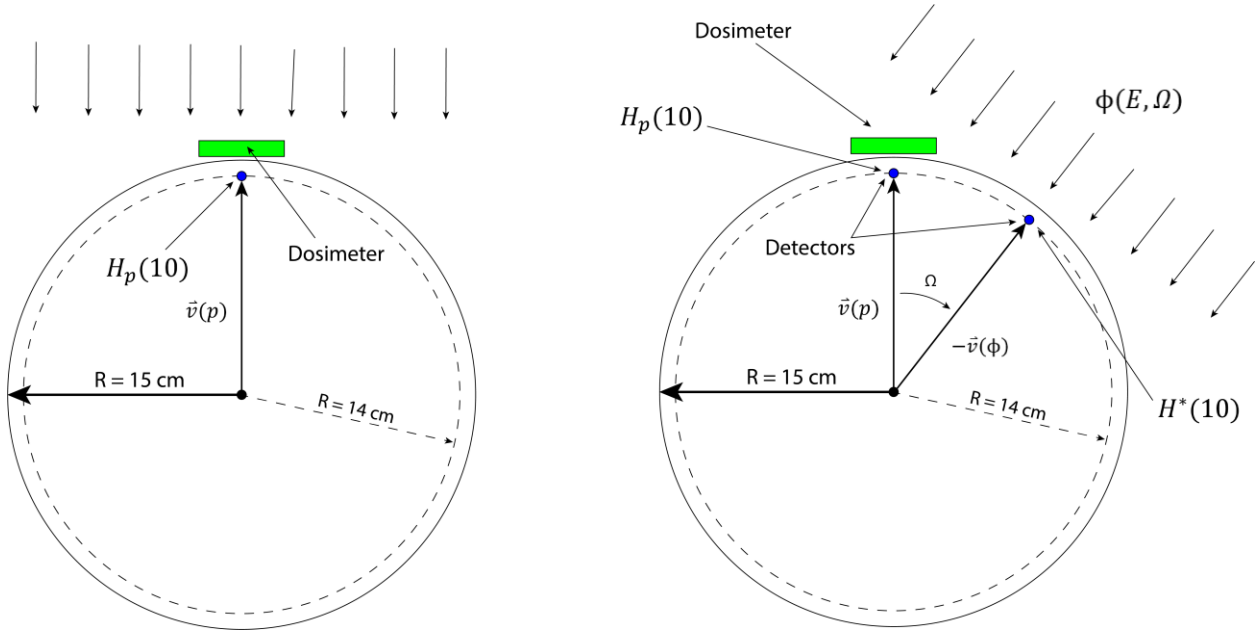


Figure 2.5. MCNP Model Dose Point Layout. Personal dose equivalent (left) and the ambient dose equivalent (right) in comparison to the personal dose equivalent.

2.4 Radiation Simulation

To maintain the expanded and aligned field on the phantoms, a vacuum was required to transport the particles to the phantom, otherwise neutrons could scatter due to modeled air and the incident neutrons would not represent a true expanded and aligned field [28]. While this would likely not affect the results of higher energy particles, when modeling lower energies, it could significantly alter the results. Fields with energies ranging from thermal (2.5×10^{-8} MeV) to 20 MeV were sampled on a logarithmic scale until an approximate equal logarithmic distance between energies less than 1 MeV was achieved. For energies between 1 and 20 MeV, common emission energies were modeled and, as resources permitted, linear half-distances were used to sample more energies. In total, more than 50 different energy fields were modeled, with an expanded and aligned field with angles from 0 to 180 degrees normal to the surface of the

dosimeter, at increments of 10 degrees. In all, over 1000 unique individual MCNP runs were tallied.

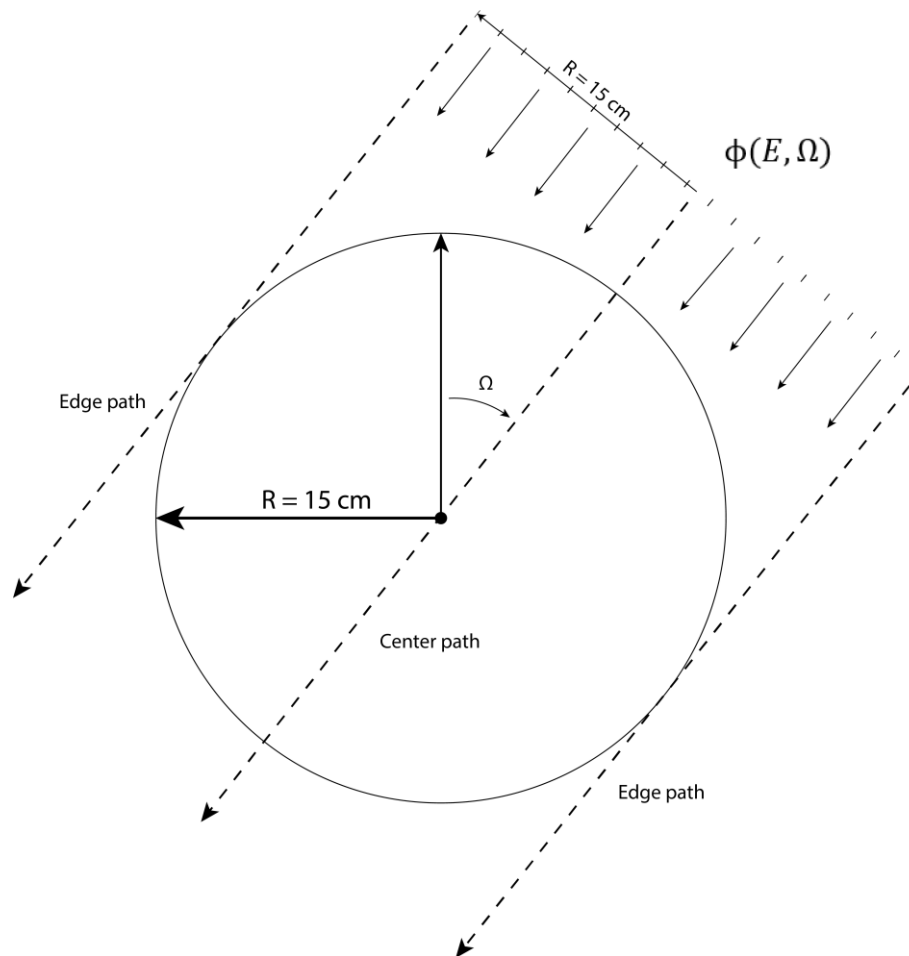


Figure 2.6. Expanded and Aligned Field Description. Outer boundaries of the source circle are tangential to the ICRU sphere

One consideration to address when modeling tissue and neutron interactions is incoherent inelastic scattering which may be impacted by the molecular structure of soft tissues [40]. The application of the $S(\alpha, \beta)$ scattering function though is not directly relevant when using an ICRU sphere or ICRU to model soft-tissue [42]. The primary purpose of the ICRU sphere is to provide a standardized model for estimating dose and dose equivalent in human tissues from various types of radiation, including neutrons. The ICRU sphere is used for *simplified* modeling of radiation interactions with human tissues, focusing on macroscopic dosimetry quantities [41]. It generally does not require the detailed microscopic information about atomic and molecular motions that $S(\alpha, \beta)$ provides because it assumes a homogeneous medium, which generally suffices for macro-dosimetry calculations but does not account for the detailed atomic-level interactions and dynamic behaviors captured by $S(\alpha, \beta)$ scattering [31]. The main goal in using an ICRU sphere is to calculate radiation doses involving averaging over many scattering events and considering the overall energy deposition in the tissue-equivalent material, rather than the detailed scattering dynamics.

For comparison, both the use of free-gas model and the use of the $S(\alpha, \beta)$ scattering were considered for this study. Main body results report the use of the free-gas scattering assumption with the ICRU sphere. **Error! Reference source not found.** provides a comparison for the injection of $S(\alpha, \beta)$ scattering function for hydrogen in light-water, as body composition by weight is generally 60 to 70% water [43]. Injecting this assumption though may result in over-reliance of the light water scattering function as molecular structure cannot be specified, so all hydrogen scattering is treated as hydrogen in light-water, while soft-tissue is comprised of various organic molecules with hydrogen bonds. The total effective cross-sections between both

methods are not overly dissimilar though, so the difference between both methods is not expected to be significant [45].

2.5 Derivation of the Heavy Charged Particle Deposition

When attempting to computationally estimate the expected response of a dosimeter, one of the most crucial factors is ensuring that the tallied parameter corresponds to the real-world physical phenomenon. As neutrons are uncharged particles, their measurement is dependent on the interactions of the neutrons with the detector materials to produce secondary effects that can be directly measured [16]. Interactions of the ^6Li with the neutrons produce heavy secondary charged particles which deposit their energy locally in the $\text{Al}_2\text{O}_3:\text{C}$. This deposition of energy causes excitation of the electrons into electron traps. As the electron traps are optically stimulated, they give off excess energy as light and thus fall back down to their relaxed state on valence band. Thus, the energy deposited by the neutrons via creation of secondary charged particles is proportional to the light emitted by dosimeter element when it is read [19], [29], [45].

To determine dosimeter response, an idealized scenario would be to model the creation and transport of these secondary charged particles to determine the relative energy deposition of those particles in the primary substrate. Unfortunately, modeling charged particles computationally in MCNP increases run-time drastically. Due to the large number of individual runs and source particles required in each run to maintain output statistics within a satisfactory error, this approach was impractical for this study. In lieu of direct tallying, techniques are used to estimate the charged particle energy deposition in the neutron sensitive element, without specifying transport of heavy ions explicitly. The F6 tally provides the energy deposition for particle types in units of MeV/g per source particle by using a hybrid of track-length and collision effects. Track-length tallies are performed using heating numbers. These heating

numbers include the kinetic energy for all secondary particles except photons. If available, the partial heating numbers of particles on the MODE card are removed to ensure consistency. If not, the energy of the secondary particle is subtracted out from only the +F6 tally at the point of collision. The energy from non-transported secondary charged particles is deposited along the track (for projectiles with heating numbers) or at the point of collision (for everything else) [4]. Therefore, it is possible to observe the contribution from specific secondary particles by using combinations of tallies for standard particles.

To provide a good estimate of the heavy charged particle energy deposition created by neutron interactions with the ${}^6\text{Li}$, it is possible to use F6 neutron and photon tallies and calculate the difference [46]. This will remove all energy deposition in the $\text{Al}_2\text{O}_3\text{:C}$ film from photons and electrons (as the F6 photon tally couples both photon and electron deposition), leaving approximately only the deposition of the heavy charged particles from the neutron interactions. The direct energy deposition of neutrons would be insignificant compared to the deposition from the charged particles due to the low capture cross-section of the dosimeter material. When compared across different source energies and incident angles, this should provide a reasonable relative estimate of the neutron response of the dosimeters.

Of course, when combining tally results, error needs to be propagated through the solution. MCNP provides a mean value per source particle $\bar{x}_i$ and estimated relative error R . The relative error is simply error s_i divided by the mean value. To propagate the error of the combined tallies, a standard error propagation formula was used where the propagated variance s_i^2 is a function of the combined variances and covariance, and the contribution of each is determined by the partial derivative for each variable of the function $f(x, y)$.

$$s_f^2 = \left| \frac{\partial f(x, y)}{\partial x} \right|^2 s_x^2 + \left| \frac{\partial f(x, y)}{\partial y} \right|^2 s_y^2 + 2 \frac{\partial f(x, y)}{\partial x} \frac{\partial f(x, y)}{\partial y} s_{xy} \quad (3)$$

$$f(x, y) = X - Y \quad (4)$$

$$\frac{\partial f(x, y)}{\partial x} = 1 \quad \frac{\partial f(x, y)}{\partial y} = -1 \quad (5)$$

$$s_f^2 = s_x^2 + s_y^2 - 2s_{xy} \quad (6)$$

$$s_x^2 = (R_x \cdot \bar{x})^2 \quad s_y^2 = (R_y \cdot \bar{y})^2 \quad (7)$$

Since the relative error for tally x and tally y are given in the result, calculation of the variance for each tally is straightforward, where the result is the square of the relative error multiplied by the mean. More complicated is the treatment of the covariance s_{xy} since the covariance is calculated from the series of data points about the tally output mean.

2.5.1 Determination of the Covariance

MCNP produces tallies as a mean value over n total trials with an estimated relative error $(\bar{x}_n \pm R_n)$. When combining tally results, this leaves only the single data point for each tally. When only single data points are available, insufficient data exists to calculate the covariance in the traditional sense because the covariance is a measure of how two random variables vary together and is calculated from a series of data points, i.e. $(x_i, x_{i+1}, x_{i+2}, \dots)$ and $(y_i, y_{i+1}, y_{i+2}, \dots)$, about their respective mean values [30]. If no other data exists, one could hypothesize about the relationship between $\bar{x}$ and $\bar{y}$ based on domain knowledge or previous experience.

However, if $\bar{x}$ and $\bar{y}$ are two observations from larger populations that are available, or larger populations can be determined from the given output information, covariance can be calculated. A common method for obtaining more data sets is to run multiple MCNP files corresponding to different random number seeds. This provides multiple data points

corresponding to the same tally that can be considered randomized about a normal distribution [31]. Due to the extensive time that each MCNP file requires to run, when real time is important, this strategy cannot be reasonably completed. Another method is to use ptrac file information which can provide tally results for each individual score [32]. Ptrac files creation also increases the runtime for each individual run and can become unreasonable when many runs are required. Finally, if batch statistics are available for the total results ($\bar{x}_n \pm R_n$), one can calculate data points based on the tally dumps over n histories. All that is required for this is implementation of the standard output tally results or summarized results printed to a separate output file [33], [34].

The MCNP runs were specified to dump every one-million source histories. If the previous dump represents the mean value for n histories ($\bar{x}_n$), the tally output for the following dump represents the mean value over $n + m$ histories, or $\bar{x}_{n+m}$. Therefore, it is possible to extract what the mean over m histories must have been to produce the $\bar{x}_{n+m}$ result. These $\bar{x}_m$ values can be considered “pseudo” $\bar{x}_i$ data points. The process can be repeated for each dump to extract pseudo data sets that can be used to estimate the correlation coefficient required to calculate the covariance, and thus more accurately compute the propagated error. When batch statistics also are unavailable, the assumption of the tallies would be that they are independent which may offer a conservative approach if there is sufficient reason to suspect the results are not negatively correlated. This tends to be a safe assumption for tally results that are estimating contributions from different particles, if contributions from the particles of one tally do not contribute to the result of the other.

To determine the mean $\bar{x}_m$ and estimate the relative error R_m of m histories based on the information from $\bar{x}_n$ and $\bar{x}_{n+m}$, we first calculate the total sums for each dump, n and $n + m$

(8). From there, we can derive the $\bar{x}_m$ value as the sum of the difference divided by m source particle histories (11).

$$\bar{x}_n = \frac{1}{n} \sum_{i=0}^n x_i \rightarrow \sum_{i=0}^n x_i = n \cdot \bar{x}_n \quad (8)$$

$$\bar{x}_{n+m} = \frac{1}{n+m} \sum_{i=0}^{n+m} x_i = \frac{1}{n+m} \left[\sum_{i=0}^n x_i + \sum_{i=0}^m x_i \right] \quad (9)$$

$$\sum_{i=0}^m x_i = (\bar{x}_{n+m})(n+m) - \sum_{i=0}^n x_i = (\bar{x}_{n+m})(n+m) - (\bar{x}_n)(n) \quad (10)$$

$$\bar{x}_m = \frac{1}{m} \sum_{i=0}^m x_i \quad (11)$$

To calculate the variance for the m histories, we can follow a similar process to extract that information from the dumped data set, following the determination of $\sum_{i=0}^m x_i$ over m data points, as specified in equations (12) through (15).

$$s_n^2 = \frac{1}{n} \sum_{i=0}^n (x_i - \bar{x}_n)^2 \rightarrow \sum_{i=0}^n (x_i - \bar{x}_n)^2 = s_n^2 \cdot n \quad (12)$$

$$s_{n+m}^2 = \frac{1}{n+m} \sum_{i=0}^{n+m} (x_i - \bar{x}_{n+m})^2 = \frac{1}{n+m} \left[\sum_{i=0}^n (x_i - \bar{x}_n)^2 + \sum_{i=0}^m (x_i - \bar{x}_m)^2 \right] \quad (13)$$

$$\sum_{i=0}^m (x_i - \bar{x}_m)^2 = (s_{n+m}^2)(n+m) - (s_n^2)(n) \quad (14)$$

$$s_m^2 = \frac{1}{m} \sum_{i=0}^m (x_i - \bar{x}_m)^2 \quad (15)$$

The estimated sample error of each m histories is thus simply the square of the estimated variance over m histories.

$$s_m = \sqrt{s_m^2} \quad (16)$$

To use this data to estimate the covariance between the two data sets, the covariance $Cov(X, Y)$ needs to be defined in terms of the Pearson correlation coefficient ρ and the total error from the X and Y datasets as in equation (17).

$$Cov(X, Y) = \rho \cdot s_X \cdot s_Y \quad (17)$$

It is then the Pearson correlation coefficient that can be estimated by the pseudo extracted data sets for each tally by using the sum of squares method as in equation (18).

$$\rho = \frac{\sum(\bar{x}_m - \bar{x})(\bar{y}_m - \bar{y})}{\sqrt{\sum(\bar{x}_m - \bar{x})^2 \sum(\bar{y}_m - \bar{y})^2}} \quad (18)$$

From here the estimation of the propagated uncertainty of combining the two tallies can be found using the total errors from the X and Y datasets, and the estimated correlation coefficient as determined from the pseudo data set (19).

$$s_{X-Y}^2 = s_X^2 + s_Y^2 - 2Cov(X, Y) = s_X^2 + s_Y^2 - 2(\rho \cdot s_X \cdot s_Y) \quad (19)$$

2.5.2 Charactering Dosimeter Response

For this study, the interest was in the *neutron* response of the dosimeter. As elements one, three, and four are relatively insensitive to neutrons, the energy deposition in those elements were not quantified. Still, to account for their potential influence on element two via scattering, they were incorporated into the model. For the output tally results of MeV/g, discussed in a previous section, post processing was applied to convert the energy deposition in the dosimeter to an estimated effective dose and the dose was compared across all included energies. The post processing was calibrated at a 0-degree incident angle, using the personal dose equivalent response of the phantom. At a 0-degree orientation, both the personal and ambient dose equivalents originate from the same measurement point. At this orientation, the energy

deposition in element two was compared to the phantom personal and ambient dose point as a ratio where a calibration factor, cal , is equal to the ratio of the response at 0 degrees to the response at the dose point (20). This ratio was then utilized across all measurement points to derive a relative value representing what a dosimetry lab might report and normalize the responses to by multiplication of the calibration factor to the observed response (21).

$$Cal = \frac{R_D(0^\circ)}{H_p(10)} \quad (20)$$

$$H_p(10, D^\circ) = R_D(D^\circ) \times Cal \quad (21)$$

Each response at the respective incident angle was compared with the response at the 0-degree orientation, and the response at each energy level was also compared for every angle. Combination of the two viewpoints (Response v Angle, and Response v Energy) results in a 3-dimensional data product with the x-axis corresponding to the energy of the expanded and aligned field (E), the y-axis with the cosine of the angle of incidence for the expanded and aligned field (simply θ here), and the z-axis with the normalized combined energy and angular response. Formulation of the correction factor is represented in equation (22 where M is the measured dose, $R(E)$ is the normalized response dependent on incident energy and $C(\theta)$ is the normalized response dependent on incident angle. The normalization leads to unparametrized values ranging from 0 to 1 for both variables. The correction factor F would be the inverse of the multiple of the two factors.

$$D = M \cdot F = M \cdot \left[\iint R(E)C(\theta)dEd\theta \right]^{-1} \quad (22)$$

For the multiple regression we can express the energy dependence and angular dependence as two different variables and combine each polynomial. The regression can then be used if the field characteristics are known. Algebraically the multiple regression $\hat{y}_{E,\theta}$ takes the

form of the combination of the regression over the energy $\hat{y}_E$ (23) and the regression over the angle $\hat{y}_\theta$ (24). A linear combination is displayed in equation (25). Here $\hat{\beta}_i$ and $\hat{\alpha}_i$ coefficients are for the variable terms x_i and x_i^2 , respectively. For the multiple regression $\hat{\delta}$ is introduced as the coefficient for the combined term. In practice, the multiple regression was non-linear and took more of a sigmoidal form, which is introduced in the next chapter.

$$\hat{y}_E = \hat{\gamma}_E + \hat{\beta}_E x_E + \varepsilon_E \quad (23)$$

$$\hat{y}_\theta = \hat{\gamma}_\theta + \hat{\beta}_\theta x_\theta + \varepsilon_\theta \quad (24)$$

$$\hat{y}_{E,\theta} = \hat{\gamma}_\theta + \hat{\beta}_\theta x_\theta + \hat{\gamma}_E + \hat{\beta}_E x_E + \hat{\delta} x_E x_\theta + \varepsilon \quad (25)$$

For this study, finding the goodness of fit is the same as finding the fit for a linear regression model over one variable. The R^2 value is just the ratio of the sum of square errors to the total sum of squares and representative of the perfect of data points which can be explained by the variables in the regression model.

$$R^2 = 1 - \frac{\sum_{i=1}^n (y_{E,\theta,i} - \hat{y}_{E,\theta,i})^2}{\sum_{i=1}^n (\hat{y}_{E,\theta,i} - \bar{y}_{E,\theta})^2} \quad (26)$$

Chapter 3 - Results and Discussion

3.1 Model Validation

Ensuring accuracy of computational models is a key factor before interpretation of results. For this model, the use of the ICRU-sphere and ICRP dose quantities makes the process straightforward as the energy-dependent conversion factors are derived from computational models. For this study, the model was validated using ICRP publication 74, ICRP publication 119, and ICRP publication 116 values. Equation (27) describes derivation of the coefficients where $F(E)$ is the neutron fluence-to-effective dose conversion coefficient, $H_p(10)$ is the observed dose in $\text{MeV} \cdot \text{g}^{-1}$ at the personal dose equivalent point, $W_R(E)$ is the energy dependent neutron absorbed dose to effective dose weighting factor for neutrons, and $\phi(E)$ is the fluence of the field in $\text{source particle} \cdot \text{cm}^{-2} \cdot \text{MeV}^{-1}$ [43], [7], [17]. The $\phi(E)$ is a function of the total number of source particles samples (n), and the radius of the source geometry (r) as shown in Eq. (28). The energy (E) is of the expanded and aligned field. Volumetric conversion of the observed $H_p(10)$ was not required, as the quantified dose point had a volume of 1 cm^3 and the density of the ICRU 4-component soft tissue is $1 \text{ g} \cdot \text{cm}^{-3}$.

$$F(E) = (1.602 \times 10^{-13} \text{ J} \cdot \text{MeV}^{-1}) \times H_p(10) \times W_F(E) \times \phi(E) \quad (27)$$

$$\phi(E) = n/\pi r^2 \quad (28)$$

The model was evaluated over three different energies: thermal ($2.5 \times 10^{-8} \text{ MeV}$), intermediate (1.0 MeV), and fast (10 MeV). These energy values were chosen due to the existence of published values at those energies, eliminating any need for interpolation. The radiation weighting factors $W_R(E)$ were determined by a piece-wise formula, as specified in

ICRP publication 116 and described in equations (29) and (30) [7]. ICRP 74 and 119 utilize averaged radiation weighting factors, but these weighting factors are derived from the same equations, as specified in ICRP publication 60 [35]. Therefore, the averaged factors were not considered and only directly calculated values from equations (29) and (30) were used.

$$W_R(E < 1 \text{ MeV}) = 2.5 + 18.2e^{-\ln(E)^2/6} \quad (29)$$

$$W_R(1 \text{ MeV} \leq E \leq 50 \text{ MeV}) = 5.0 + 17.0e^{-\ln(2E)^2/6} \quad (30)$$

Table 3.1. Comparison of ICRP values and dose results

Energy (MeV)	Published Standard Values		Model Derived Values		Relative Error	
	ICRP 74/119 $F(E)$ (pSv · cm ²)	ICRP 116 $F(E)$ (pSv · cm ²)	$H_p \pm 1s$ (eV · g ⁻¹)	$F(E) \pm 1s$ (pSv · cm ²)	ICRP 74/119	ICRP 116
2.5×10^{-8}	7.60	4.0	24.1 ± 6.72	6.81 ± 1.90	10 %	70 %
1.0	282	301	104 ± 4.42	244 ± 10.3	14 %	19 %
10	499	500	566 ± 14.7	564 ± 14.7	13 %	13 %

After determining $W_R(E)$ at each energy, fluence-to-dose conversation factors $F(E)$ were calculated based on the results of the MCNP outputs. Percent error was then determined in comparison to the values in ICRP publication 74 and ICRP publication 119, as well as ICRP publication 116. Table 3.1 shows the MCNP results, derived $F(E)$, and the error compared to all three publications. The observed fluence-to-dose derivations from the model were roughly in agreement with the published values ICRP publication 74 and ICRP publication 119 values [44], [17]. All values for these publications were less than 15% error. When compared to ICRP publication 116 values, the error for the two lower energy values was increased to 70 % at 2.5×10^{-8} MeV and 19% at 1.0 MeV [7]. This difference may be attributable to the reliance of

ICRP publication 116 on a voxel-type anthropomorphic phantom as introduced by ICRP publication 110 [45]. ICRP publication 74 and ICRP publication 119 values rely on simpler anthropomorphic models composed of the specifications in ICRP publication 23 and ICRP publication 89, respectively. Based on the results of this study, these simpler anthropomorphic models seem to respond to neutrons more similarly to the ICRU sphere, especially at lower neutron energies [44], [45]. With this information, we can reasonably conclude that our model largely agrees with the expected values, and interpret new values based on this model as having some validity to what would be expected to be observed. Still though, comparison of the ICRP Publication 74/119 and 116 results, along with the results of this study, may provide more justification for implementing more advanced human models in follow-on research.

3.2 Correlation Method Validation

The proposed method of estimating the correlation coefficient is a novel method inspired by recent work awaiting peer-review as of the completion of this study [34]. As such, validation is required to ensure that the method does not introduce significant biasing in the result. For MCNP models with a sufficiently high number of source particles, the relative error should be a minor (< 0.05) component of the overall determined value. Therefore, if the pseudo and traditional method agree within a few percent, there should be no significant effect on the overall outcome. A computational battery of tests was developed to probe the two methods, the traditional (Pearson) method and novel pseudo method. Correlated values were randomly sampled from normal distributions about a randomly selected mean value. These mean values represent the “true” mean values of the X and Y datasets. In total, each dataset consisted of 100,000 data points. From the X and Y datasets, the Pearson correlation coefficient was determined in the traditional way, as shown in equation (18). Every 100 data points was then

grouped into a subset. The means of these subsets were determined and represented the pseudo-X and Y values. From the pseudo-X and Y values, and the sample mean of the full data set, the pseudo correlation coefficient was determined. The test was run 100 times for each specified correlation value. The difference between the traditional and pseudo values of each run was calculated and no single run, for any specified correlation, was determined to be more than 0.06 away from the Pearson coefficient, with an average difference of 0.02. Figure 3.1 displays the mean Pearson and pseudo coefficients for each specified correlation. It also displays the 1-sigma error from the repeated runs for each point. As can be seen from the graph, not only are the mean values nearly identical, but so are the distributions.

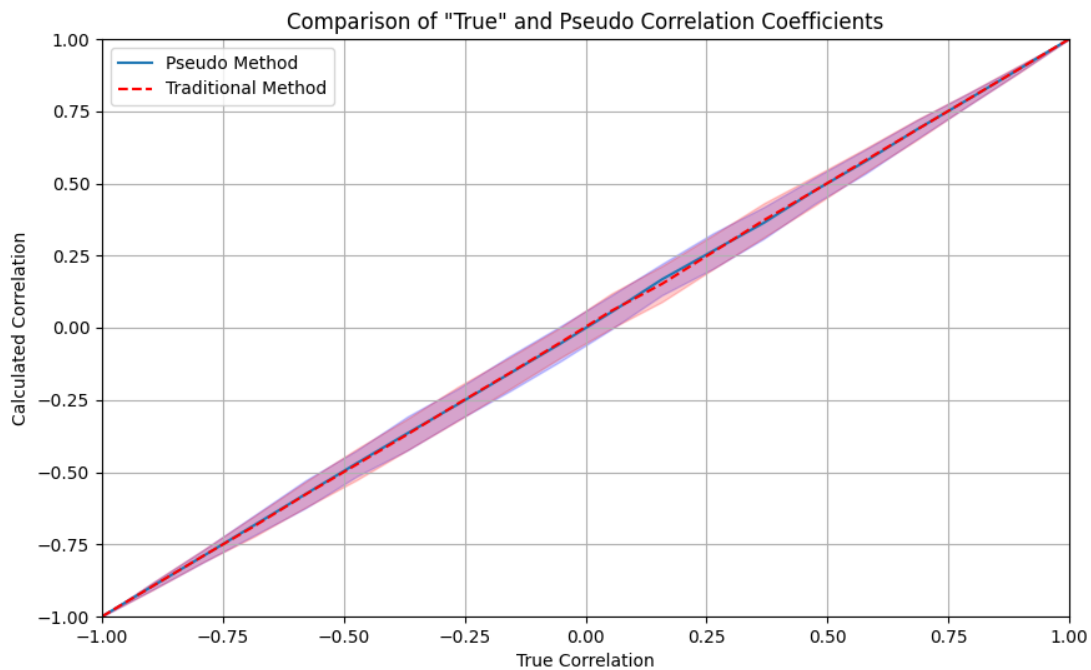


Figure 3.1. Plot of the determined correlations coefficients via the traditional (Pearson) and proposed pseudo methods. The plot displays the mean values, and 1-sigma distribution of repeated randomly generated, but correlated, test cases

While more work can be done to further investigate this phenomenon, for the purposes of this study, we can reasonably conclude that the pseudo method sufficiently estimates the Pearson coefficient, and that the pseudo method is not likely to significantly over or underestimate the correlation between MCNP tallies. Further, while Neutron Capture Ion Algorithm secondary particles are correlated in MCNP, the purpose of this method is to validate avoidance of transporting these computationally expensive particles [4]. Thus, the creation of the secondary particles transported in the model implemented in this study should be uncorrelated. This effort than simply serves as a backstop to any unforeseen correlation, in place of a conservative no-correlation assumption.

3.3 Dosimeter Response

In the study, it was observed that the ambient dose equivalent remained consistent across various angles due to the consistent positioning of the dose point relative to the radiation field during each measurement. This was the expected outcome and serves to further increase confidence in the results. Conversely, the personal dose equivalent showed a decline as the angle increased, which can be logically linked to the increased volume of material the radiation had to penetrate to reach the $H_p(10)$ dose point at steeper angles, effectively acting as a shield with the ICRU soft tissue model.

This shielding effect, while anticipated, is crucial to highlight within the context of this research, as this study's focus is on dosimeter responses compared to operational quantities. The behavior of the dosimeter response to increased angles of incidence was complex, showing an overall reduction in heavy charged particle energy deposition as the angle increased. The patterns of this decrease did not follow a simple linear, logarithmic, or any predictable trend that might mirror the reduction seen in personal dose equivalents. This finding underlines the nuanced

interaction between radiation and the dosimetry device, accentuating the need for further exploration into how dosimeters quantify exposure in varying conditions and how these measurements correlate with established operational quantities.

3.3.1 Energy Response

The neutron albedo energy response of the OSL-N dosimeter was expected to increase with the increase in neutron energy, mirroring proportional rises in both the ambient and personal dose equivalents. While cross-section decreases with increasing neutron energy, the dosimeters are expected to quantify deposition of thermalized neutrons scattered from the phantom and not incident neutrons on the phantom. Therefore, the expectation arises from the idea that as neutron energy increases, the mean free path increases and less neutrons are absorbed before reaching the dosimeter. Surprisingly, within the energy range of 2.5×10^{-9} MeV to 20 MeV, such an increase was not observed. When observed in comparison with the ambient dose equivalent, or the personal dose equivalent at near 0-degree incident angles, the energy response decreases as the energy of incident neutrons increases. Figure 3.2 displays the energy results at 10 degrees. The decrease of the energy response becomes exaggerated at energies over 1 MeV. This response is more like the energy dependency observed in previous research on albedo dosimeters for photons [26]. Additionally, discrepancies arise at lower neutron energies, where both the dose equivalents and dosimeter responses diverge markedly from expected patterns.

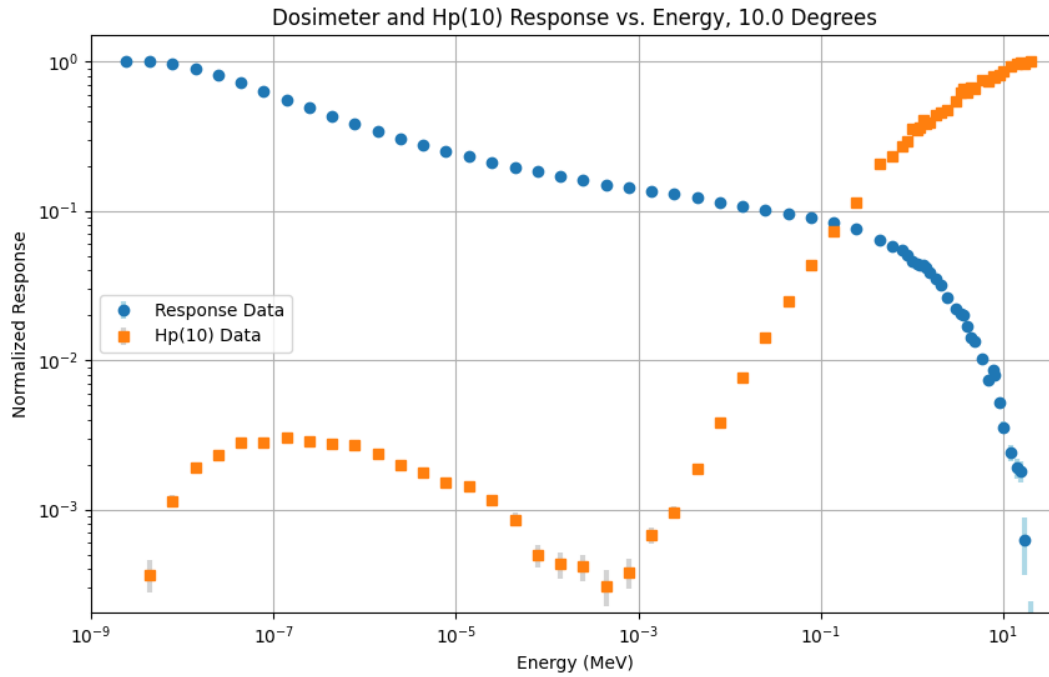


Figure 3.2. Simulated dosimeter neutron energy response vs simulated $H^*(10)$, with incident neutrons at 10 degrees off normal.

The decrease in energy response with increasing neutron energy may be attributable to the diminishing cross-section for the ${}^6\text{Li}(n, t){}^4\text{He}$ reaction with increasing in neutron energy, as depicted previously Figure 1.4. Between approximately 2 MeV to 20 MeV, where cross-section data from CENDL and EAF nuclear databases indicate the dominance of the ${}^6\text{Li}(n, n + d){}^4\text{He}$ reaction above 2 MeV, a steady or even slight increase in the response was anticipated given the relative consistency of these cross-sections [26], [27]. However, this was also absent in the observed results. Despite an increase in the MCNP tallies' error margin across this range, the actual dosimeter values continued to decline beyond what could be attributable to just error. This suggests that the expected contribution of the ${}^6\text{Li}(n, n + d){}^4\text{He}$ reaction to the dosimeter's responsiveness might be minimal or not as influential as previously thought.

Another important observation was the notable change in the relationship between the response of the dosimeter and the observed dose at exaggerated incident angles. Within approximately 0 to 90 degrees, the relationship maintained its characteristic attributes. Above 90 degrees, those attributes broke down and the dosimeter response and the observed dose features, over energy, became more similar. Error also increased in both the response and observed dose. Above 1 MeV, the observed personal dose equivalent sporadically jumped to be in approximate proportional alignment with the dosimeter response. Figure 3.3 displays the comparison of the energy response and the personal dose equivalent at 170 degrees. The spike in observed dose may be due to the increase in the mean free path of neutrons in hydrogenous materials when compared to the fixed depth of ICRU soft tissue for a given angle. The sporadic nature of the spike may also be explained by the strong elastic scattering resonances overserved with oxygen, carbon, and nitrogen above 1 MeV, which are the remaining components of the ICRU tissue model. Figure 3.4 displays the elastic scattering cross sections for each material with the elastic scattering cross section of hydrogen, as reported in ENDF/B-VIII.0 [28]. As can be observed, each material other than hydrogen displays strong resonance behavior over the 1 to 20 MeV energy range.

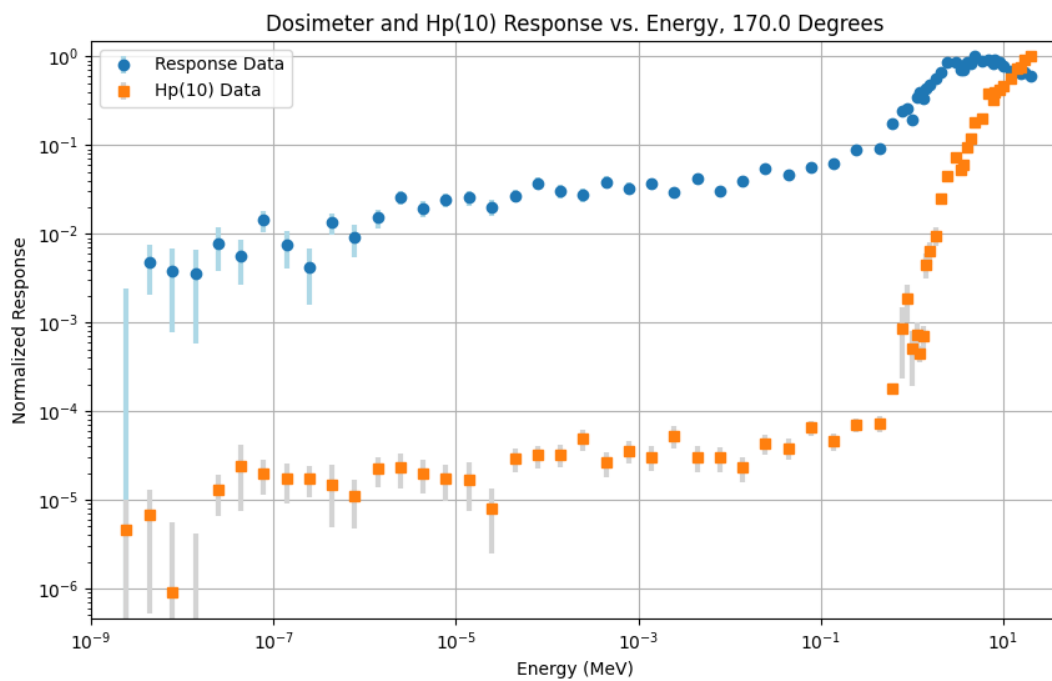


Figure 3.3. Comparison of Normalized Response and $H_p(10)$ at 170 Degrees

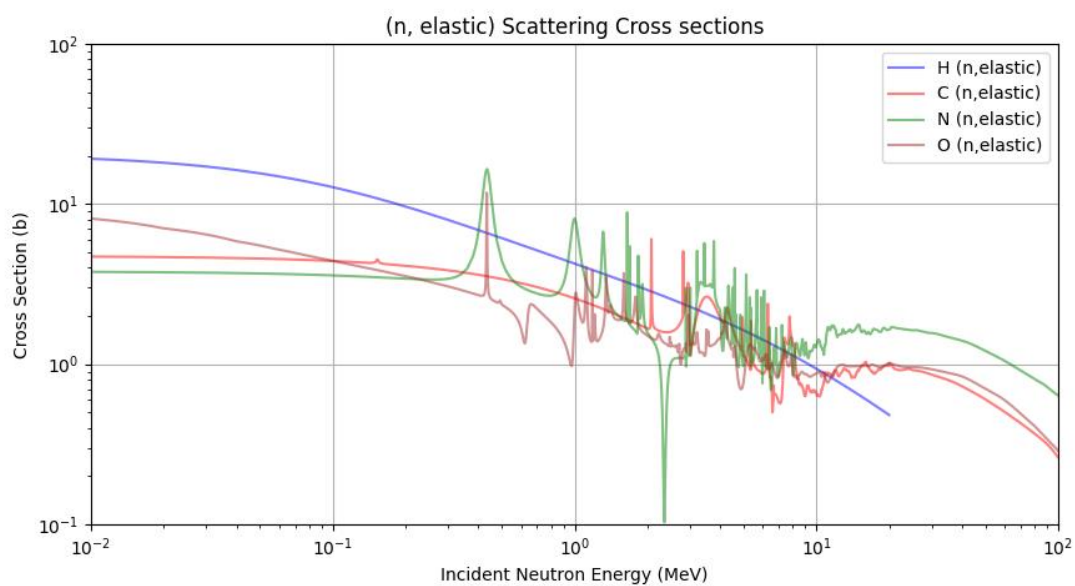


Figure 3.4. Comparison of Elastic Scatter Cross Sections for ICRU Soft tissue

3.3.2 Angular Response

The dosimeter's angular response was analyzed against the cosine of the incident neutron field's angle, where a value of 1.0 represents a direct Anterior to Posterior (AP) orientation (0 degrees), and -1.0 indicates a Posterior to Anterior (PA) orientation (180 degrees). Responses were standardized to the maximum response observed, typically seen in the AP or slightly deviated (10 degrees) positions and plotted across all angles. To accommodate the variation in response across the energy spectrum, a sigmoidal regression was applied, providing a robust 2-sigma fit for the dataset. The general equation for the logistic function used is displayed in (31). This choice was informed by the response's tendency to shift across the energy range, which the logistic function modeled acceptably. It should be noted that the intention of the regression applied here is tied to the goal of this study. It is not providing a perfect fit representing all singular values exactly, but to provide a reasonably accurate correction factor that can be applied to a dosimeter's measurement to account for the angular response.

$$f(x) = \frac{L}{1 + e^{-k(x-x_0)}} + b \quad (31)$$

For energies below approximately 1 keV, the dosimeter's sensitivity diminishes sharply, falling below 20% of the maximum response once the angle exceeds 90 degrees ($\cos(\theta) = 0$). The response to thermal incident neutrons is depicted in

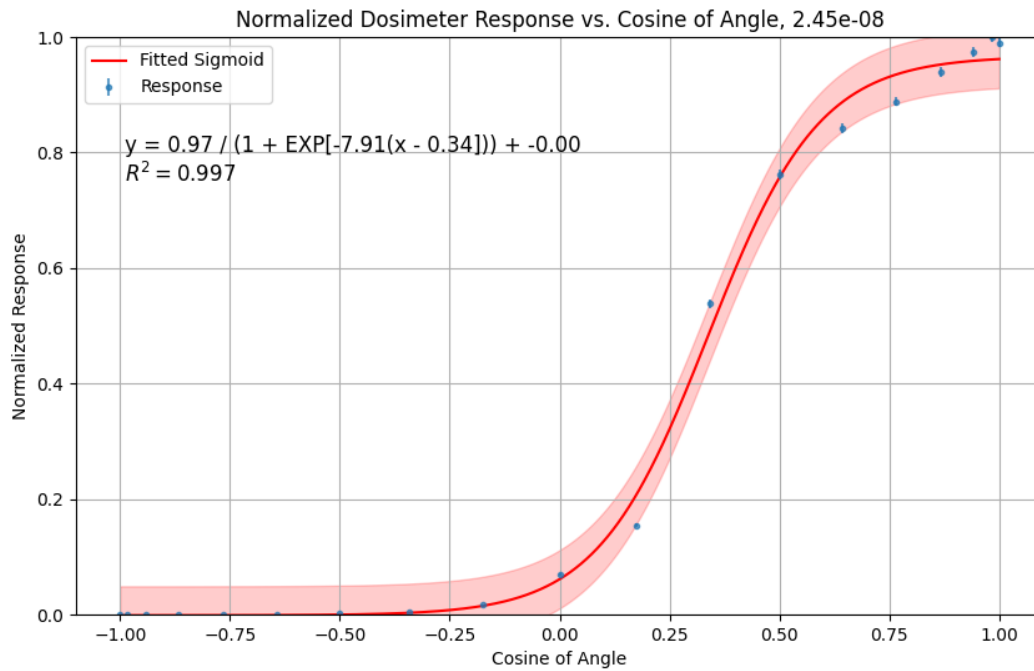


Figure 3.5. This suggests that at lower energies, without the application of an angular correction, the dosimeter may not effectively quantify doses for individuals positioned perpendicularly or away from the neutron source. The sensitivity reduction is also still pronounced at angles between 45 and 90, dropping below 80% of the maximum response, indicating that without angular correction, dose measurements from the dosimeter could substantially diverge from the true dose imparted on an individual or phantom.

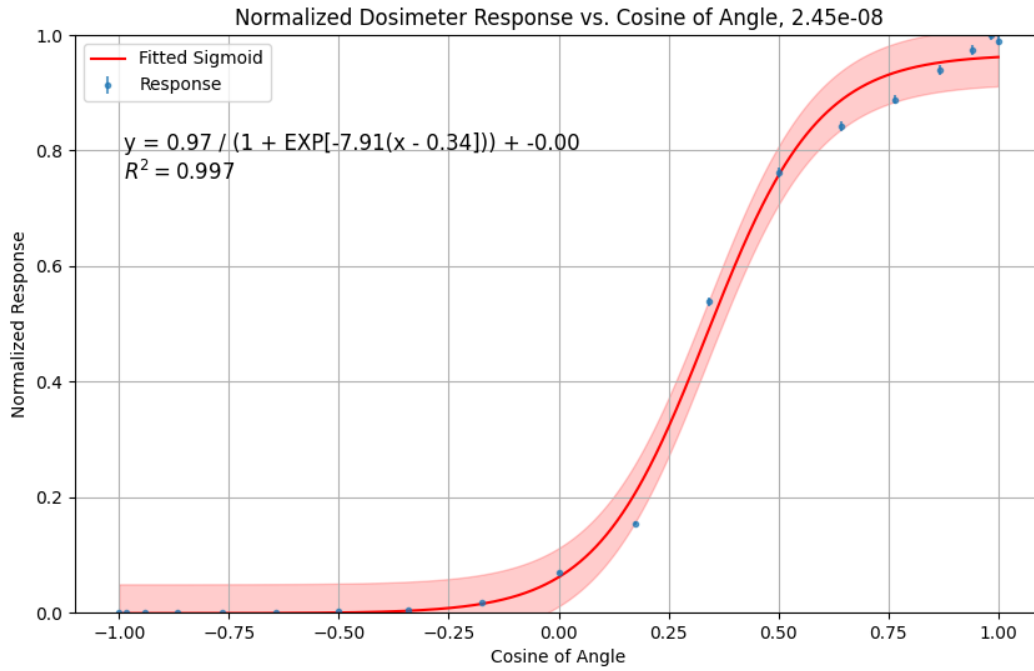


Figure 3.5. Normalized Response at 2.45×10^{-8} MeV

As the energy of the field increases beyond about 1 keV, the normalized response to the cosine relationship becomes more linear. This is somewhat intuitive as the soft tissue has less attenuation effect on the higher energy neutrons, which will have increased probability of interacting with the dosimeter element. The observed effect is the tail of the curve increasing in relative value as the energy increases, while the top end of the normalized range stays relatively constant, due to the normalization. Figure 3.6 shows the results of the angular response at 2.45 keV. Above 1 MeV, the tail end begins to become more on scale with the Anterior-Posterior position as the more graduated attenuation effects begin to even out. Figure 3.7 shows this effect at 2.45 MeV. As energy is increased more though, especially above 6 MeV, the error in the model increases and the solution becomes more erratic.

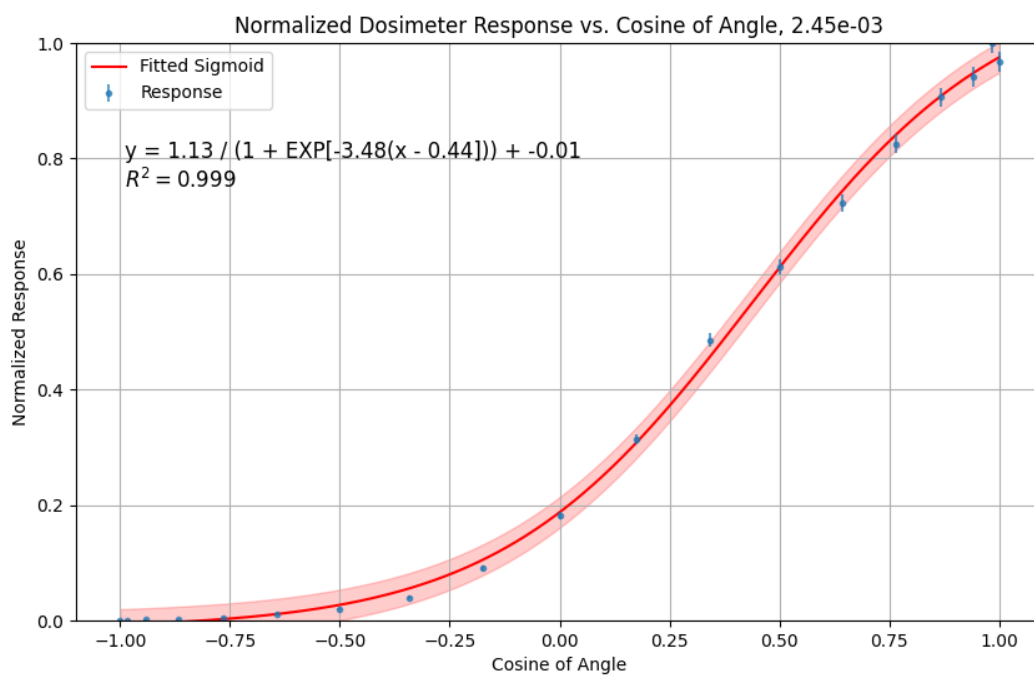


Figure 3.6. Normalized Response at 2.5×10^{-3} MeV

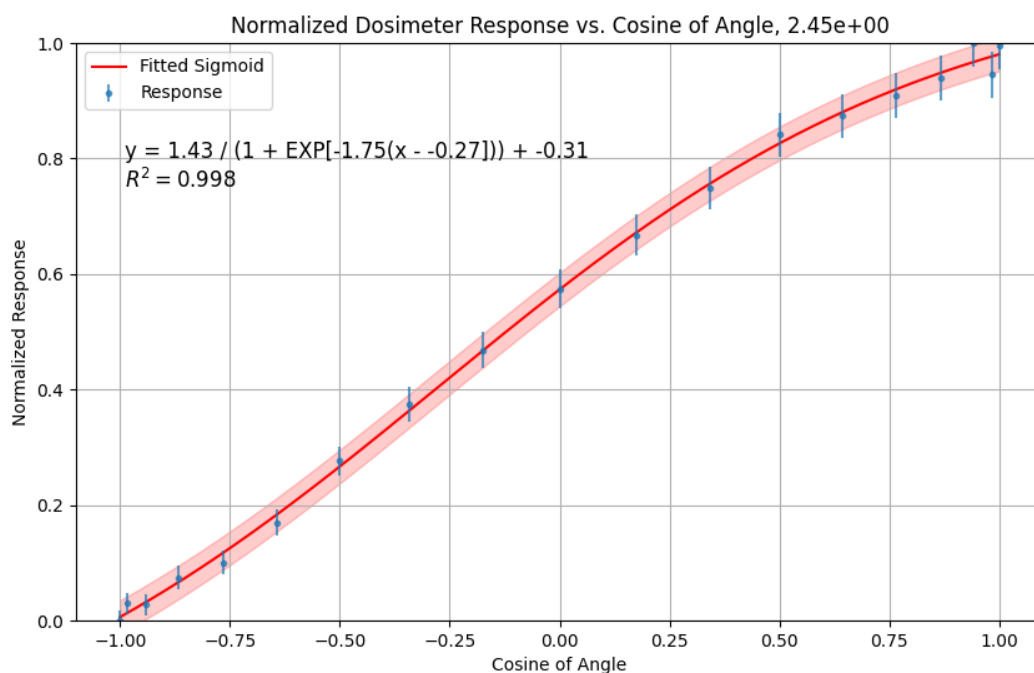


Figure 3.7. Normalized Response at 2.45 MeV

On plausible explanation for the dosimeter's variable response at higher energy levels is the role of inelastic scattering, particularly concerning Oxygen-16, which constitutes the majority of the ICRU 4-component tissue model at a 76.2% mass fraction. As energy surpasses 6 MeV, the threshold for inelastic scattering is passed and the reaction becomes increasingly prevalent, aligning more closely with the elastic cross section. This shift is documented in ENDF/B-VIII.0 data for energy ranges from 1 MeV to 20 MeV and displayed in Figure 3.8 [28]. The plot illustrates a significant increase in cross-section at energies above 6 MeV. Figure 3.9 shows the dosimeter response at 20 MeV.

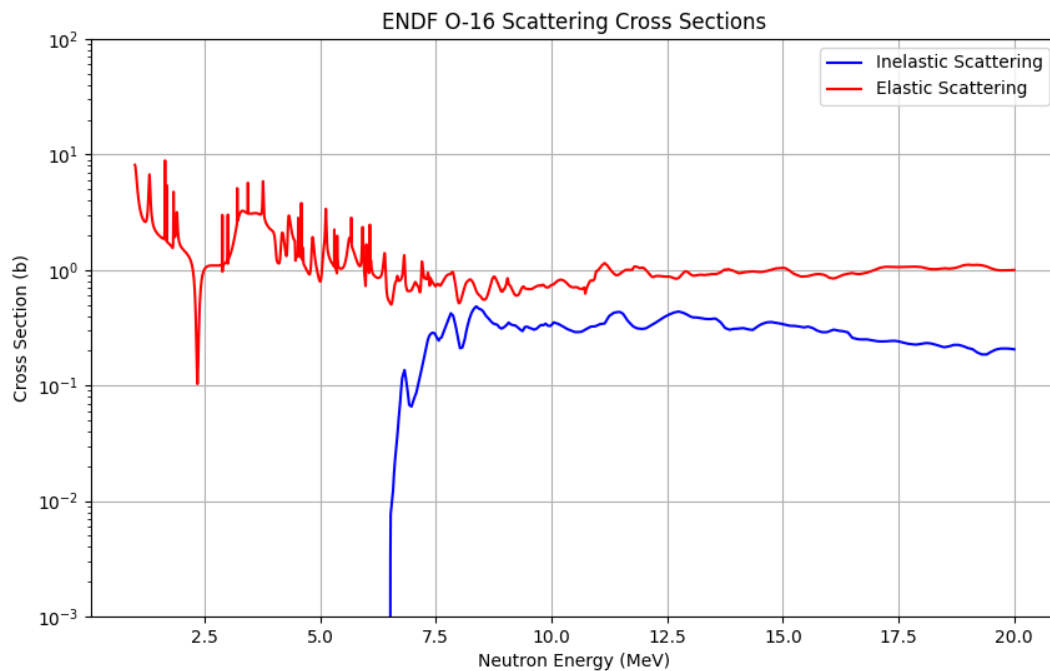


Figure 3.8. Scattering Cross-section of Oxygen-16 (JANIS)

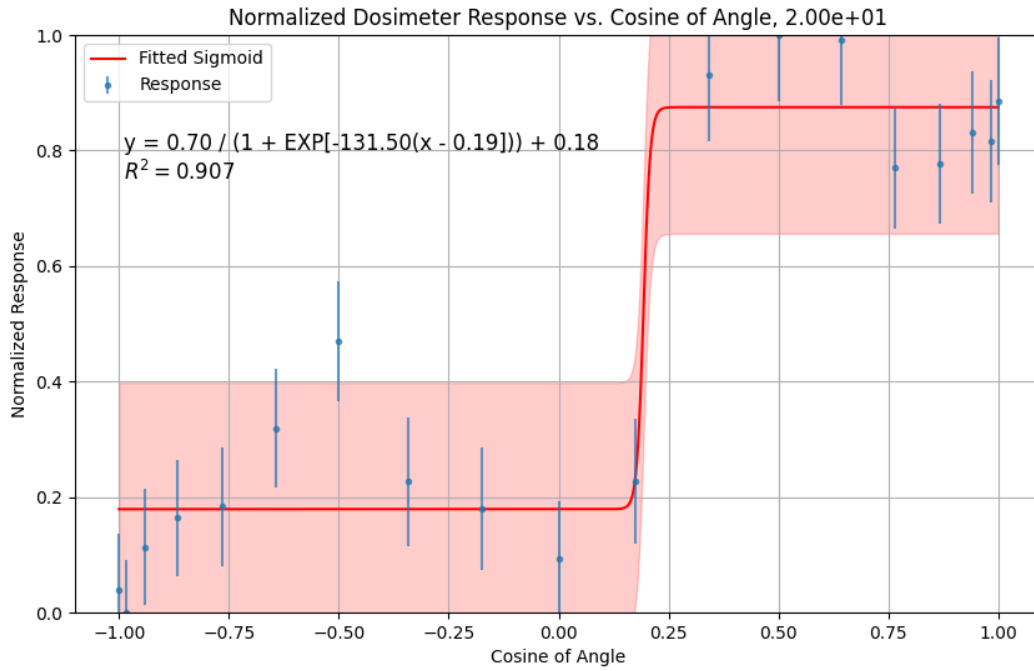


Figure 3.9. Normalized Response at 20 MeV

This increase in inelastic scattering at higher energies suggests neutrons slow down more rapidly than would have been expected by elastic scattering alone, and possibly making certain neutron energy sub-ranges more likely to be directed towards the dosimeter. However, beyond a 90-degree incidence, the likelihood of these neutrons reaching the dosimeter before being absorbed by the tissue decreases. This aspect is crucial since, at these angles and energy levels, the dosimeter's response may not proportionally reflect the actual radiation dose an individual receives with angular response is not corrected.

The adaptation of a sigmoid model for dosimeter response at higher energies above 6 MeV may be generally effective, capturing the overall trend within the bounds of overlapping residuals and a 2-sigma error margin of observed values, here achieving an R^2 of 0.91 at 20 MeV. However, considering the complex behavior of neutron interactions, a stepwise function

might offer a more precise modeling approach for these scenarios, potentially enhancing the accuracy of dose quantification and the predictive capability.

Based on these differing results between energy ranges, this approach underscores the importance of considering both the energy-dependent characteristics of neutron interactions and the geometric orientation of dosimeters relative to the radiation source for accurate dose measurement and assessment in radiological protection and research. It is evident that if angular corrections are not made to dosimeter data following processing of the measurement, that the result may not reflect the true dose imparted on an individual, within the acceptable error guidelines outlines in applicable accreditation standards.

3.3.3 Coupled Response

The goal of this research was to examine and detail the dosimeter's energy and angular responses in tandem. By synthesizing the findings from both energy and angular analyses, a three-dimensional data contour was created, linking these variables to the dosimeter's normalized response. Figure 3.10 displays the contour in two-dimensional space with a color gradient representing the third dimension, or the normalized response. As can be observed, over all energies as the cosine of the angle of incidence decreases to -1, the normalized response drops rapidly. At incident angles above 90 degrees, the dosimeter response is below 60% of what otherwise would have been observed in a field normal to the dosimeter.

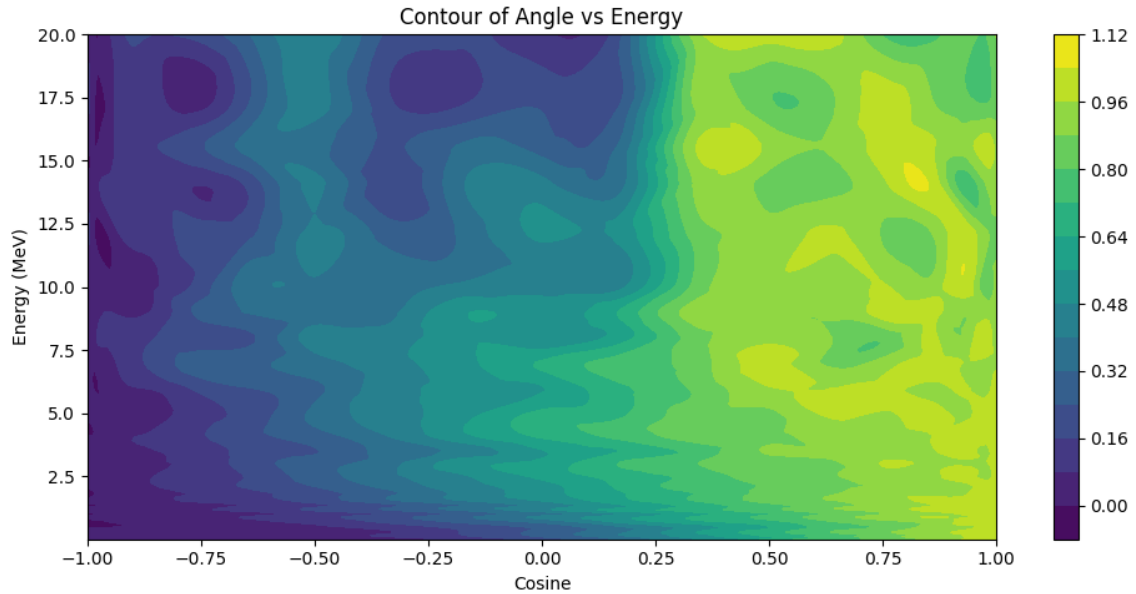


Figure 3.10. Coupled Energy and Angular Dependence Response. Changes in color represent the normalized dosimeter response where the breakpoints are values 0 to 1.

A multiple regression analysis of this comprehensive dataset was attempted, in order to predict the dosimeter's behavior across any given combination of energy and angle within the study's scope. If applicable, this predictive capability would be pivotal for calculating the necessary correction factors to adjust dosimeter readings accurately for any radiation field encountered within the researched energy spectrum.

Adopting a sigmoidal model was in line with the standard approach for fitting data in this study, achieving an R^2 value of 0.93. This indicates a generally sufficient fit for most scenarios, though the exploration of higher-degree polynomial models might yield more nuanced interpretations of the data. Particularly, the angular response analysis highlighted that while the sigmoid fit provides a solid approximation within a two-sigma confidence interval, discrepancies become more pronounced at elevated energy levels, suggesting potential areas for refining the

model to enhance prediction accuracy. Appendix C - contains the three-dimensional data products using the multiple regression technique.

3.4 Comparison to Operational Quantities

The findings on the estimated dose per source particle aligned well with the characteristics and values of energy detailed in existing literature. At lower incidence angles, as illustrated in Figure 3.11, the operational dose quantities, ambient and personal dose equivalents, appeared indistinguishable—which was the anticipated result. This observation is crucial though for validating facility area monitoring practices. It suggests that the ambient dose equivalent can be applied directly in estimating personal dose equivalents, at lower incident angles. This is a common industry practice and informs the need for this study.

However, as the predominant angle of radiation increases, especially for energies below 1 MeV, notable discrepancies between these dose measurements emerge. Figure 3.12 displays the results of comparing the operational quantities at a PA configuration. The use of the ambient dose equivalent in such scenarios offers a "conservative" approach, often yielding higher readings than those observed through personal dosimetry. In regular operational settings, where passive dosimetry is employed, and personnel orientations vary widely, such differences might not be critically impactful. But for facilities where personal dosimeters are utilized, this difference can become critically impactful as the ambient dose equivalent is not applied, in favor of the measured personal dose equivalent. Reliance on uncorrected personal dosimeter readings in these instances could lead to significantly lower estimations of the actual dose received. For lower-energy radiation fields, the discrepancy could span several orders of magnitude, underscoring the necessity for adjustments in dose assessment methodologies to ensure accuracy and safety in radiation protection practices.

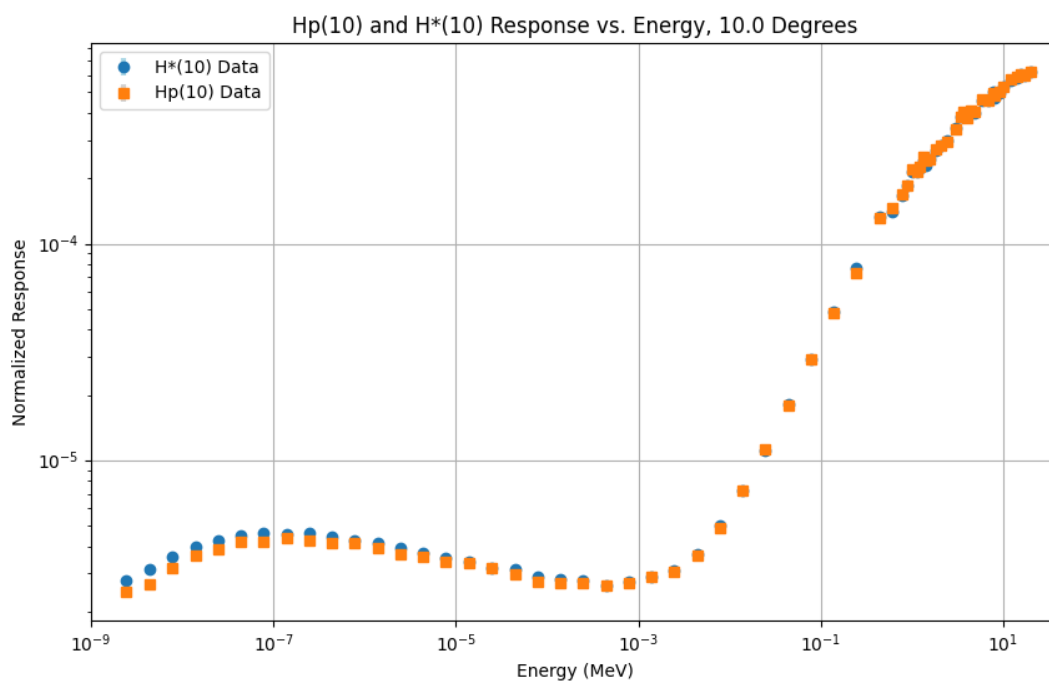


Figure 3.11. Comparison of $H_p(10)$ and $H^*(10)$ at 10 Degrees

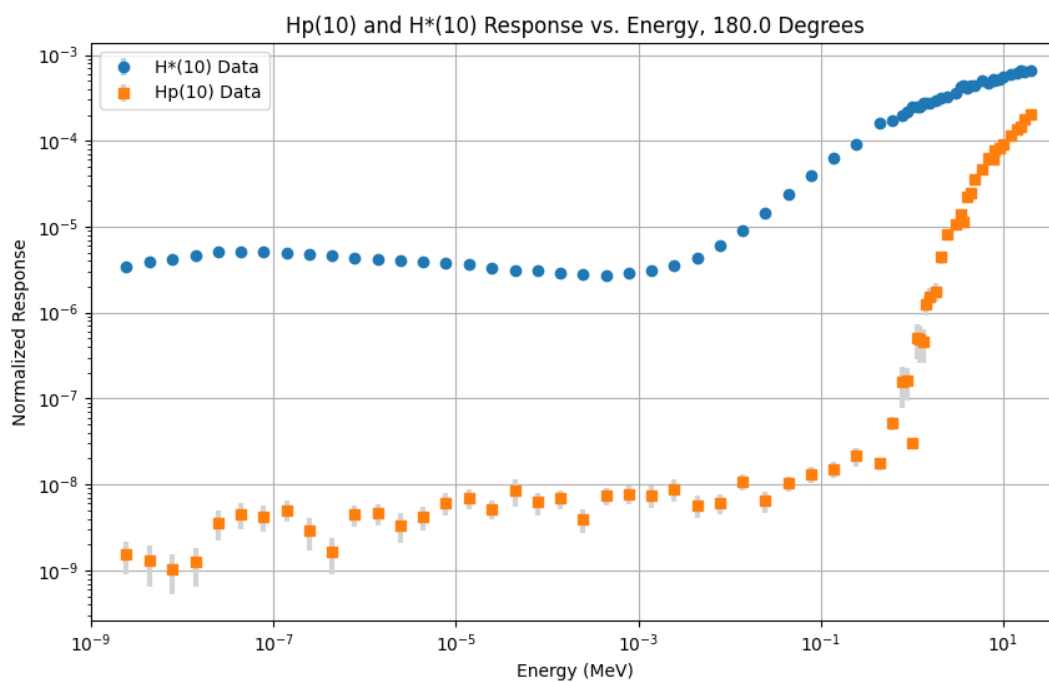


Figure 3.12. Comparison of $H_p(10)$ and $H^*(10)$ at 180 Degrees

Chapter 4 - Conclusions

4.1 General Conclusions

For most of the single variable data, the results were expected to align with previously published data. Indeed, for the energy dependent dose in both the ambient and personal dose equivalent quantification, this was the case. The energy dependent dosimeter response was also more in line with previously published energy characterizations. At higher energies, there were some deviations in the expected response as the observed response decreased rapidly above 1 MeV. This may be due to a lack of nuclear data required to represent all the interaction mechanisms and may signal a need to include the CENDL and EAF cross-sections within the ACE databases that MCNP utilizes. Upon further investigation, the $(n, n' + d)$ reaction for Li-6 was not found to be included in the current MCNP databases. This is a significant finding, as when evaluating dosimeter responses computationally, the lack of the cross-section information would depress the expected response of the dosimeter leading to inaccurate computationally derived correction factors. Therefore, practitioners should rely on experimentally derived correction factors within their applicable boundary conditions, or work to include the Li-6 $(n, n' + d)$ ACE data for their computational modeling efforts.

The relation of the energy and angular dependent responses results in an inverse relationship. As angle of incidence increases and the energy decreases, the response decreases. Inversely, as the energy increases and the angle decreases, the response increases. While intuitive in nature, this multivariable relationship has not been previously quantified in published material, especially for newer OSL whole-body albedo dosimeters. In fact, it is observed that at exaggerated angles for the personal dose equivalent, if the neutron energy spectrum is dominated by incident neutrons above 1 MeV, the lower energy spectrum neutrons are likely to be shielded

by the body and the ambient dose equivalent may provide a more accurate quantification of the dose delivered. The importance of this cannot be understated, as in practice, most dosimeter readings are taken at face value, or only minor corrections are applied. This emphasizes the need for licensees to empower qualified radiation protection practitioners with the resources required to evaluate doses beyond that of a common dosimetry user-facility program.

4.2 Future Work

There were three main observations from the results of this study that call for further analysis. The first unexpected observation was the lack of effect of the ${}^6\text{Li}(n, n + d){}^4\text{He}$ reaction on dosimeter response at higher energies. The $(n, n' + d)$ reaction cross-sections are only published in CENDL and EAF (JEFF) nuclear databases, with each differing quite drastically. Generally, the development of cross-section databases has been robust and currently most databases tend to have good agreement across specific reactions. To find a significant disagreement between the two, and a lack of inclusion in the ENDF and JENDL databases, shows a space for improvement in nuclear data. Indeed, as a follow-up, the ENDF-based ACE databases that MCNP uses were researched and there was found no explicit mention of the use of the ${}^6\text{Li}(n, n + d){}^4\text{He}$ cross-section. Though not all reactions are explicitly mentioned in legacy databases and most updates include read-me files that only mention the newly-introduced reactions.

The second finding was that at higher energies, the angular dependent dosimeter response becomes far more erratic than expected. Further work needs to be completed to characterize this erratic behavior and possibly decrease the error found in this study. This is especially important for energies encountered in high energy work environments such as spaceflight, defense, or accelerator research. When calculating dose, commonly used linear or logarithmic interpolations

are not appropriate at these higher energies. If these methods are used, there is significant and unnecessary error introduced in personal or facility area dose quantifications due to this phenomenon.

Finally, the multiple regression model may be adequate in most circumstances, as demonstrated in this study, but there is a need to further refine the responses and the relationship to the operational quantities. With the increasing order of fit, it is more appropriate to establish standardized vectors or resultant matrices based off of the ICRU-sphere, or other computational phantoms, in order to provide radiation protection practitioners a more easily accessible and accurate method of dose correction for their workplaces.

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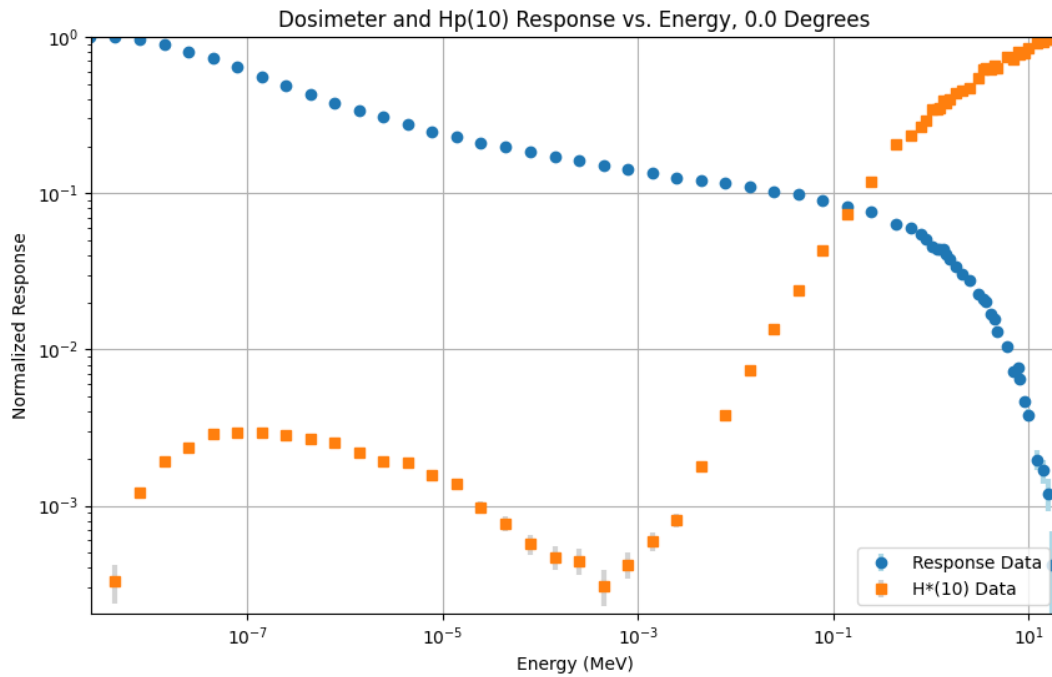
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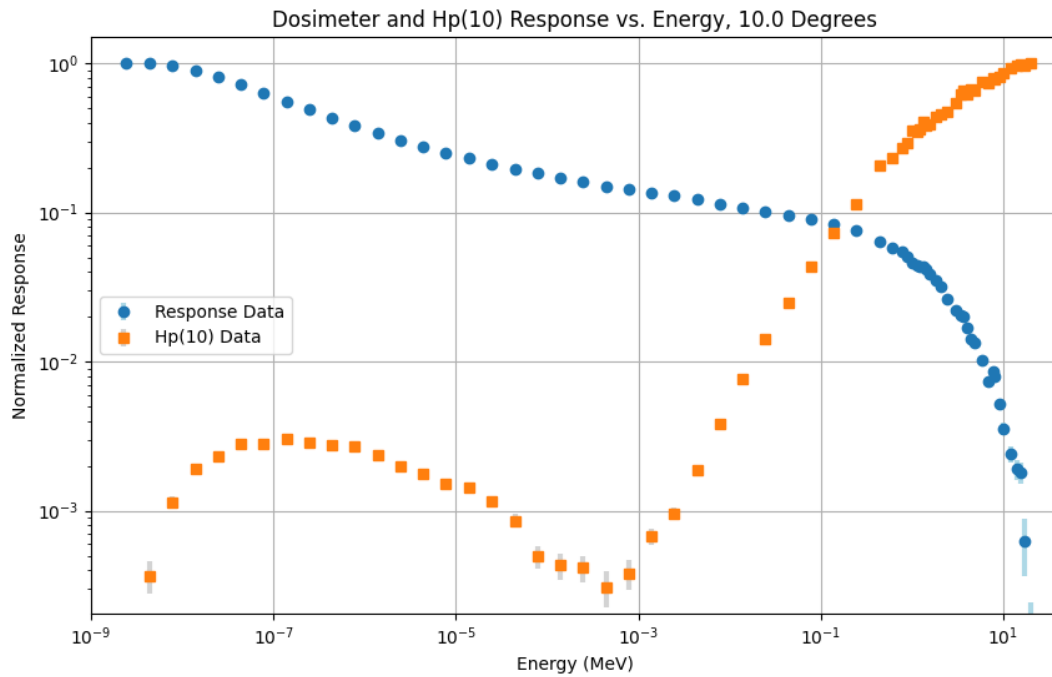
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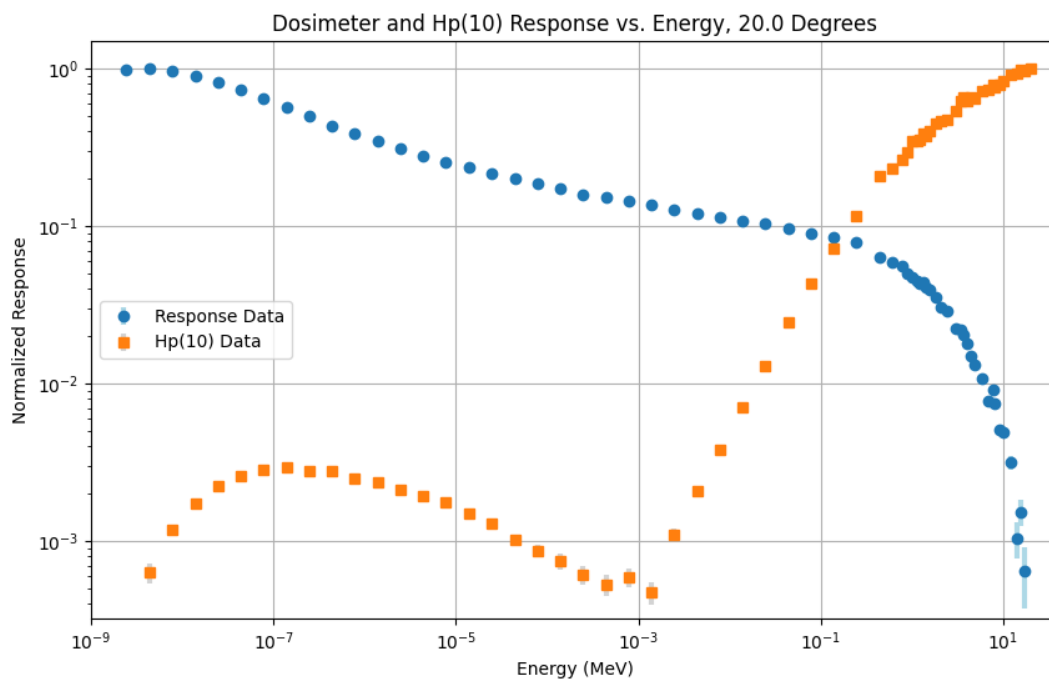
Appendix A - Energy Dependent Responses



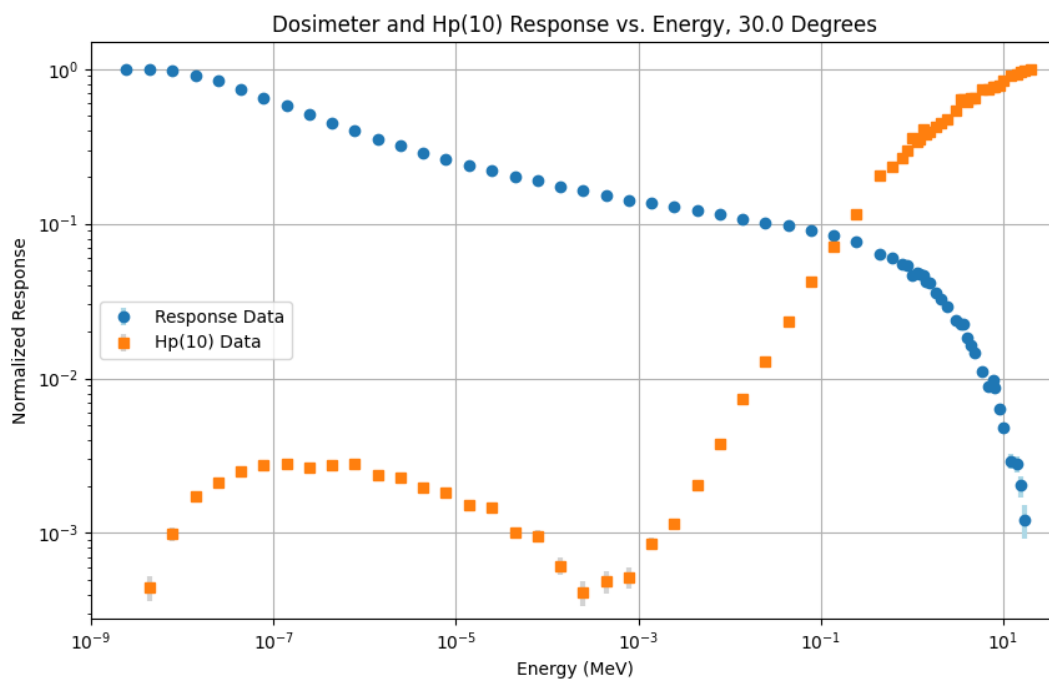
Appendix A. Figure A.1. Normalized Energy Dependent Response at 0 Degrees



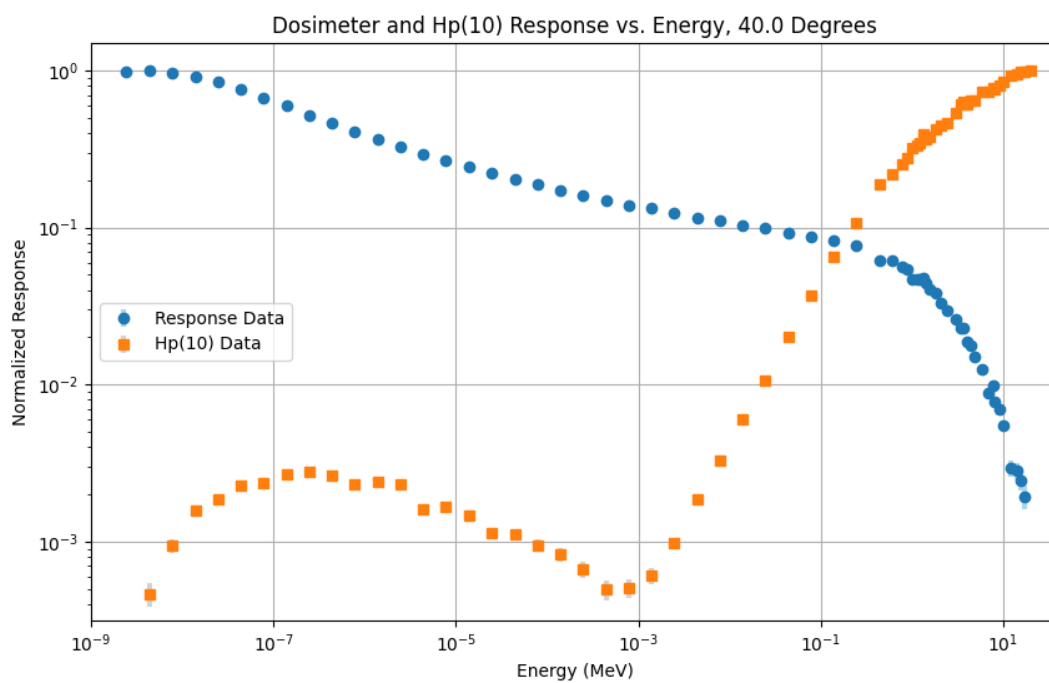
Appendix A. Figure A.2. Normalized Energy Dependent Response at 10 Degrees



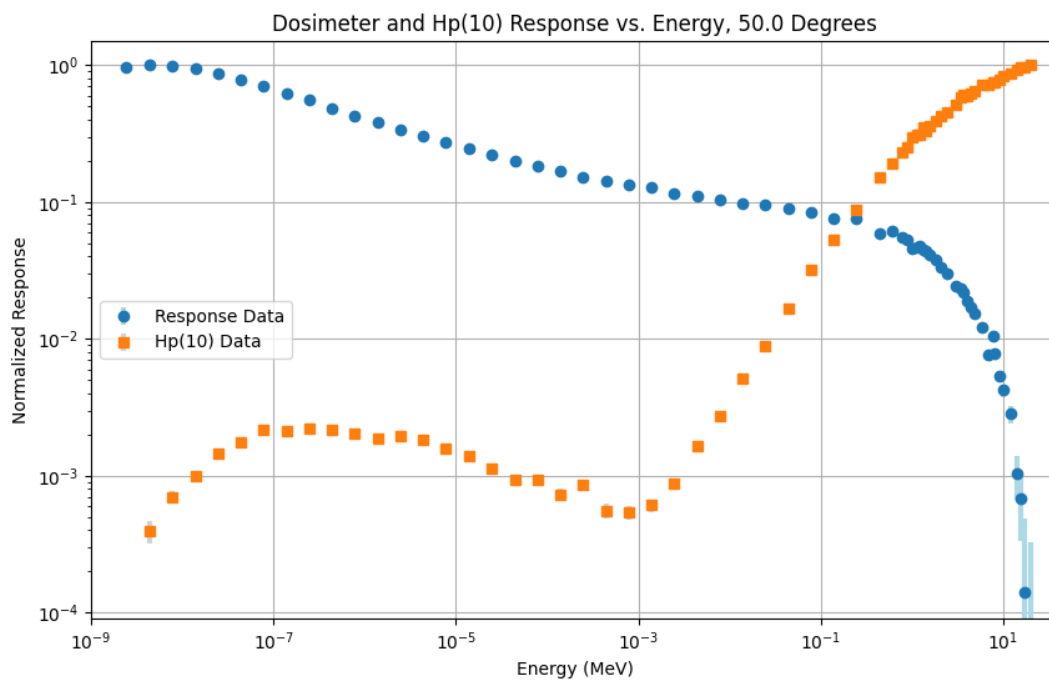
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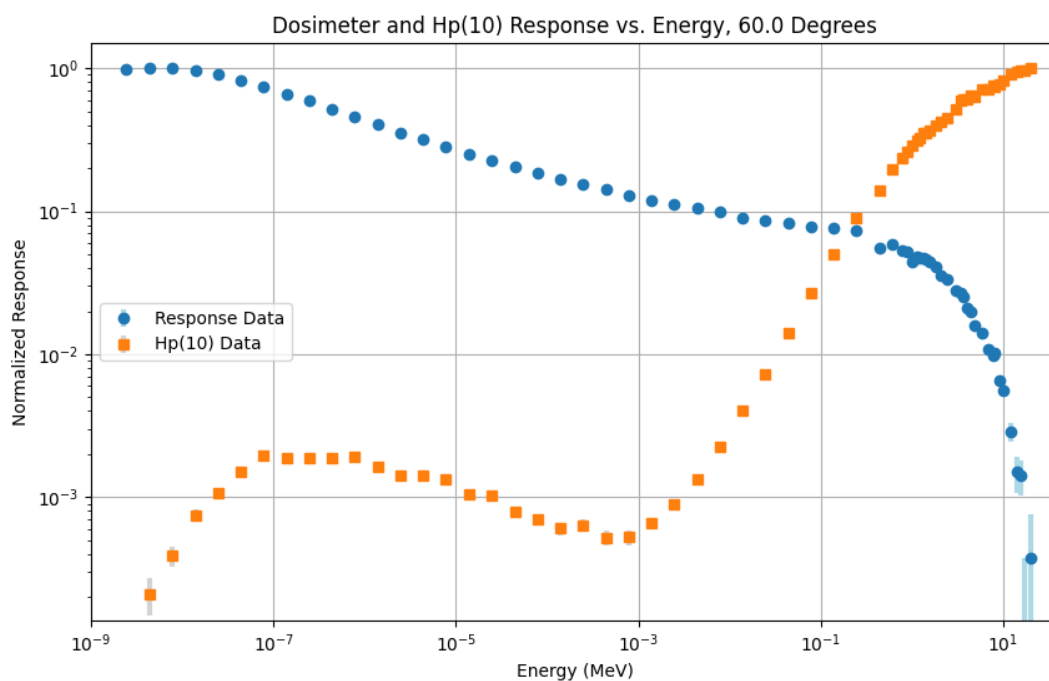
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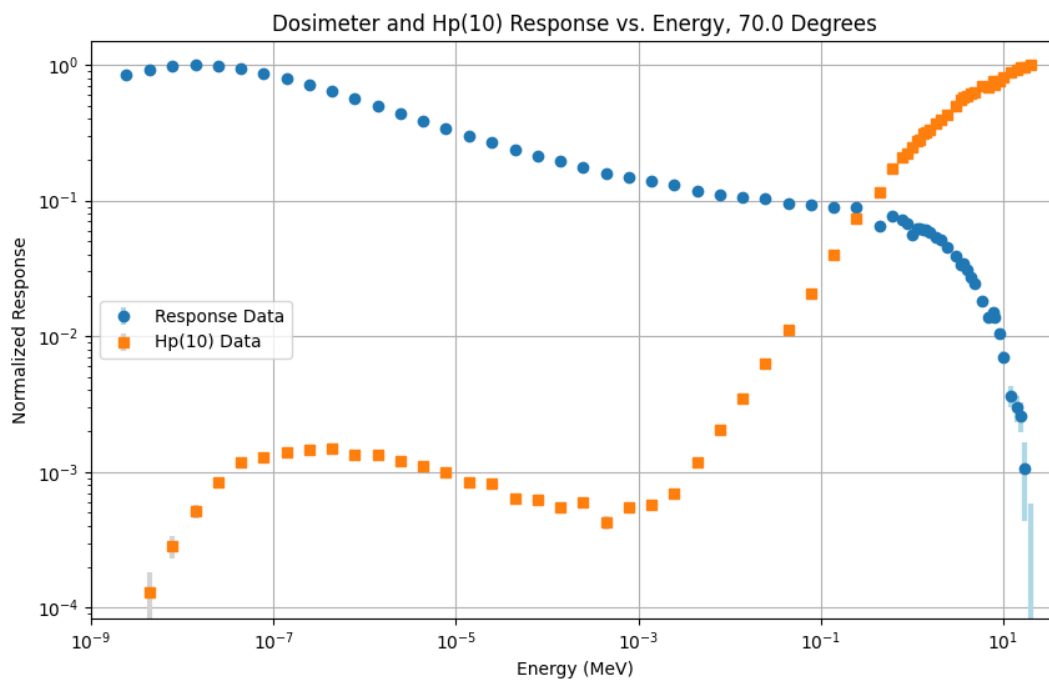
Appendix A. Figure A.5. Normalized Energy Dependent Response at 40 Degrees



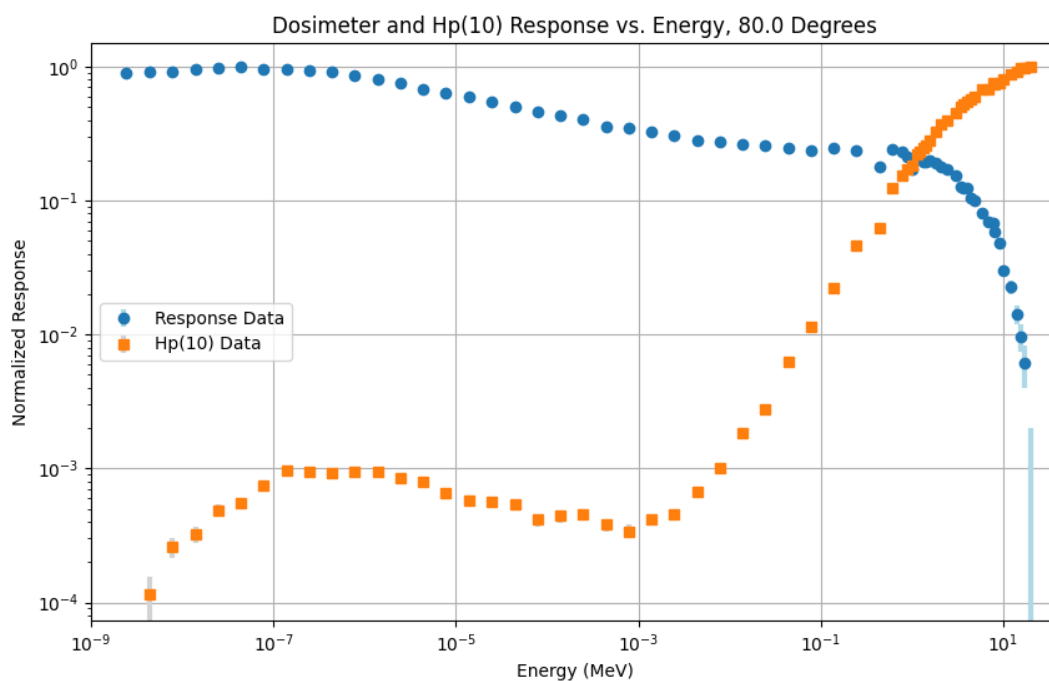
Appendix A. Figure A.6. Normalized Energy Dependent Response at 50 Degrees



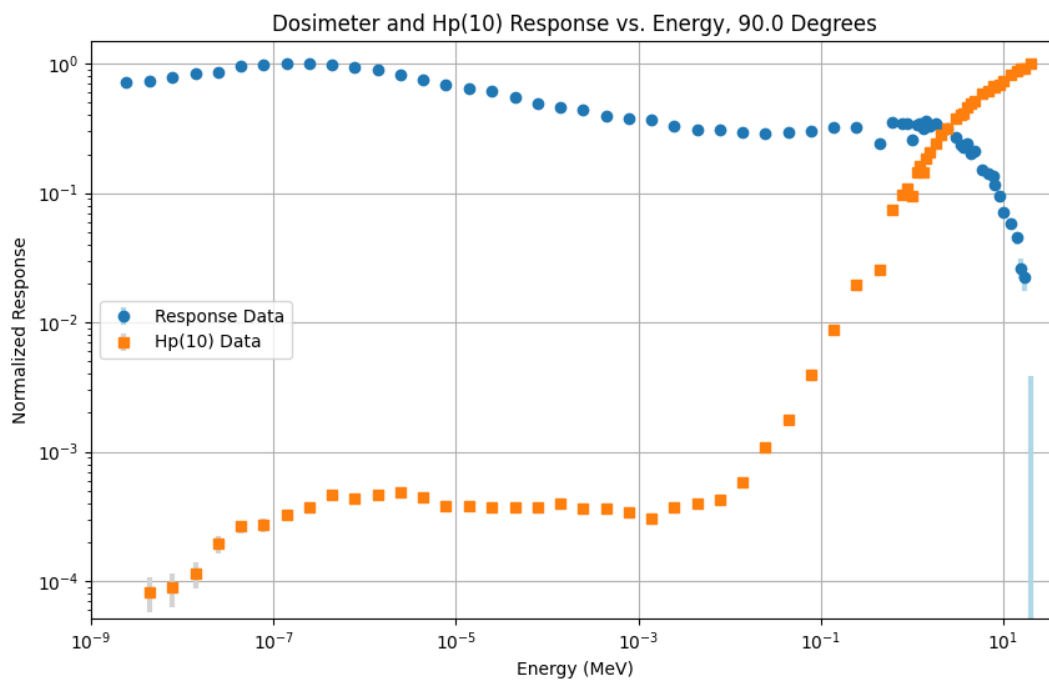
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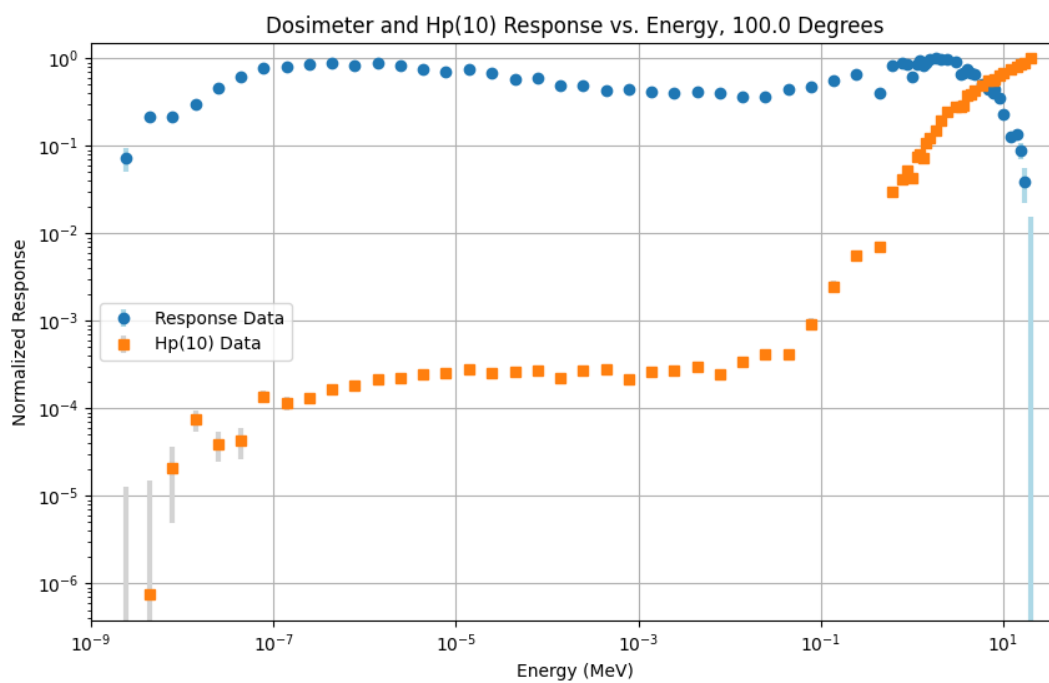
Appendix A. Figure A-8. Normalized Energy Dependent Response at 70 Degrees



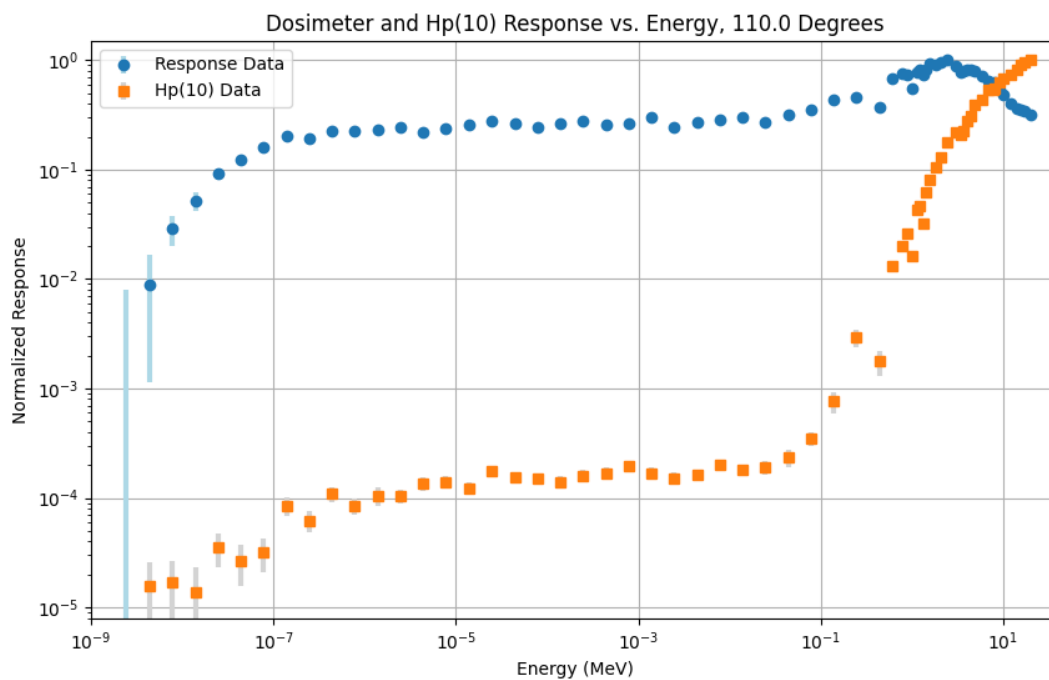
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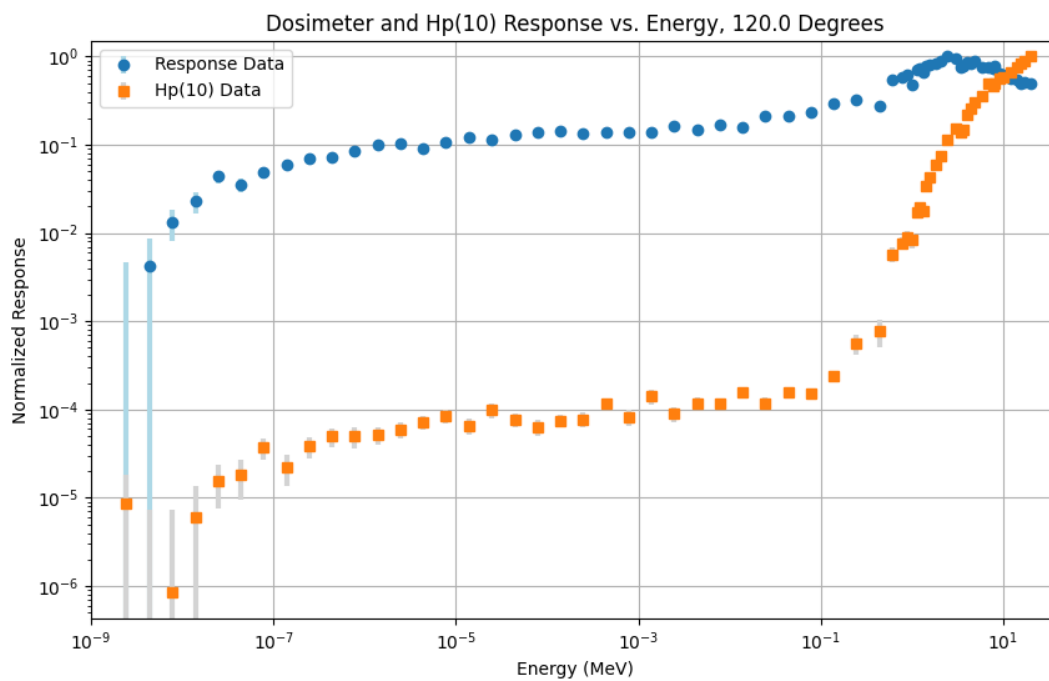
Appendix A. Figure A.10. Normalized Energy Dependent Response at 90 Degrees



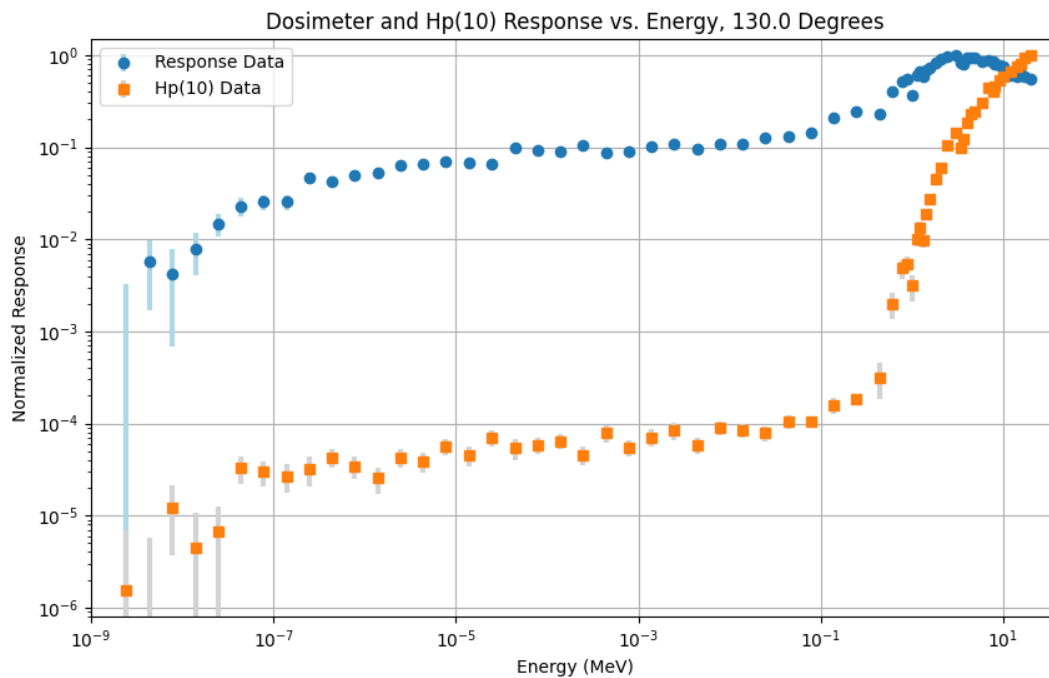
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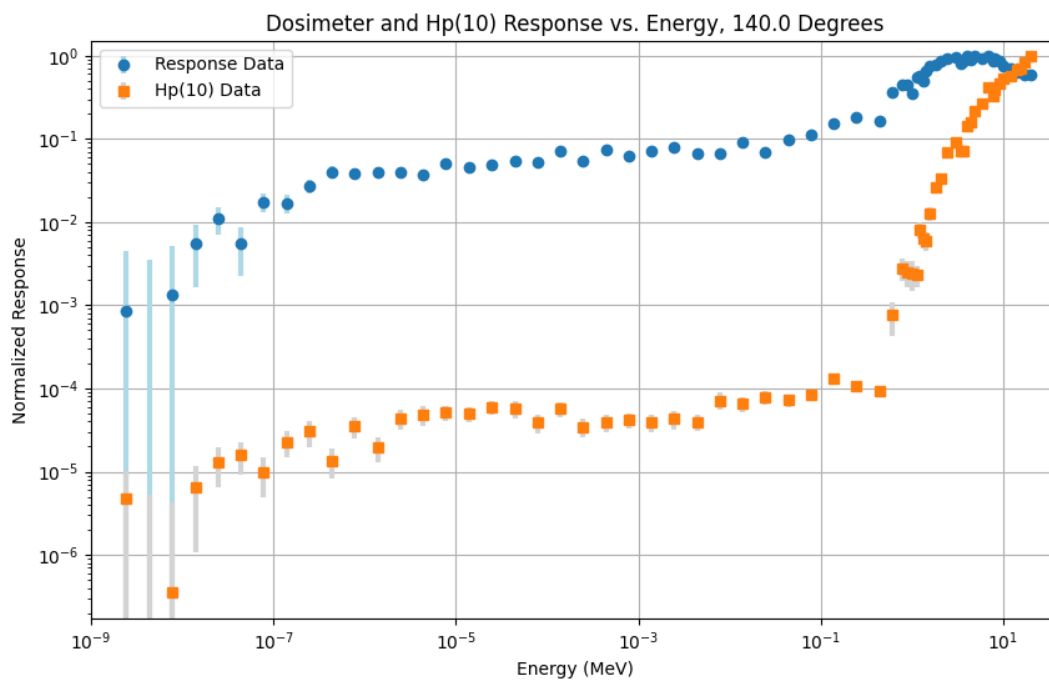
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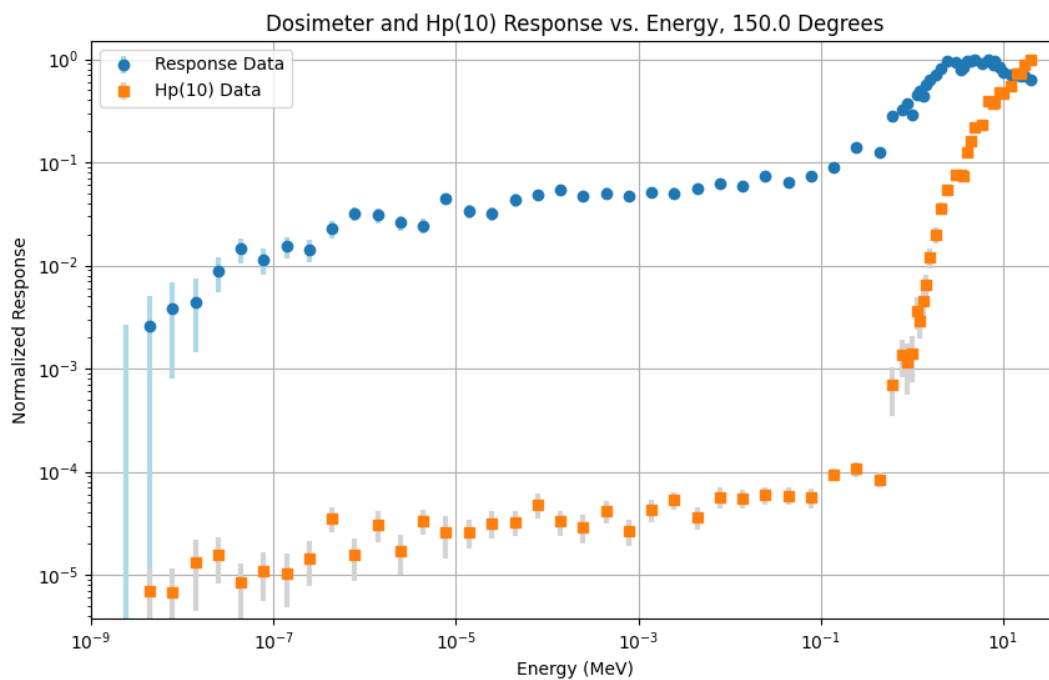
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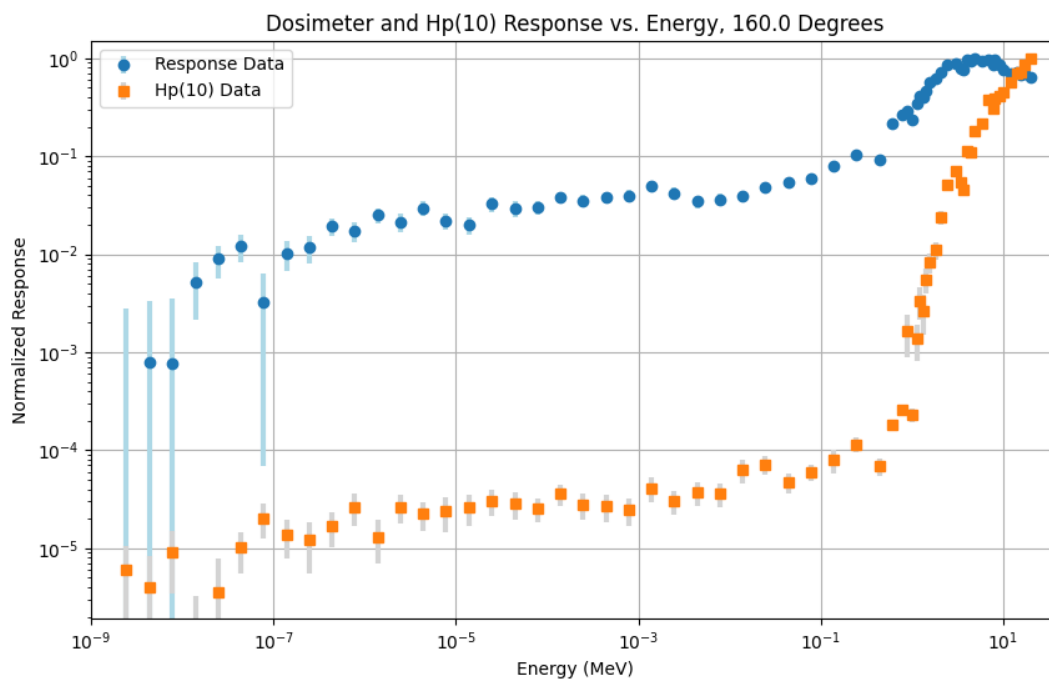
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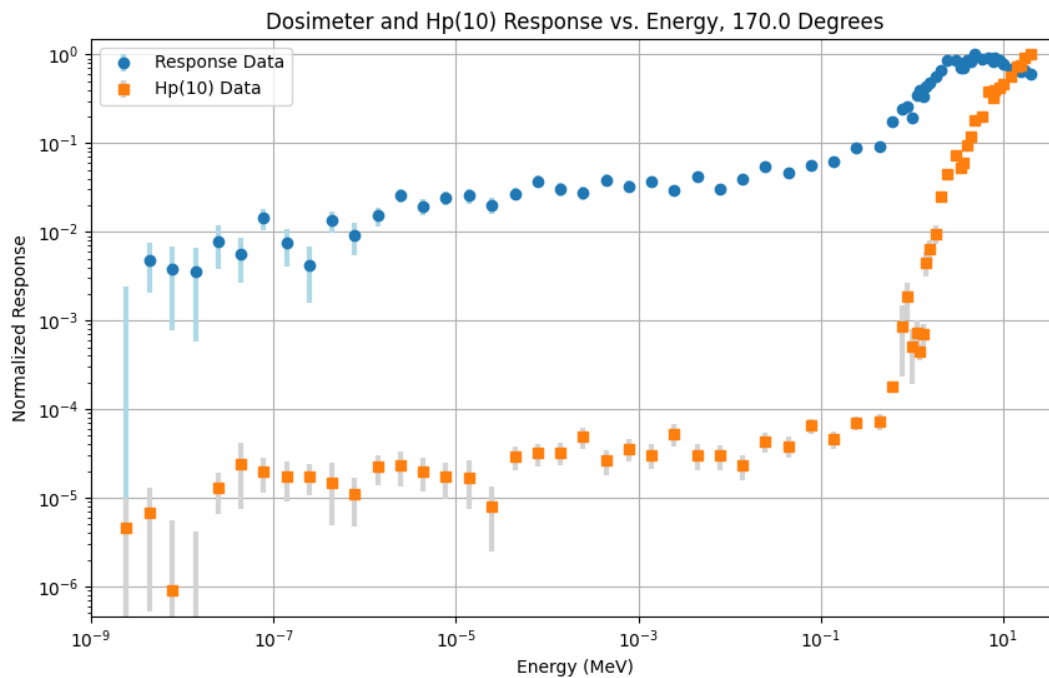
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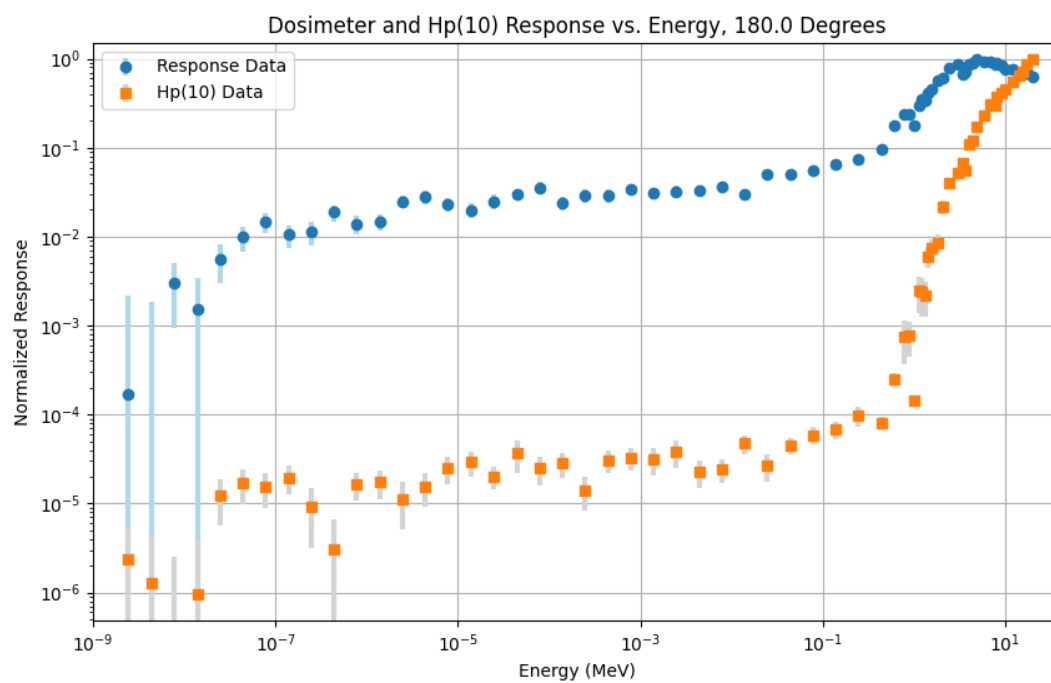
Appendix A. Figure A.16. Normalized Energy Dependent Response at 150 Degrees



Appendix A. Figure A.17. Normalized Energy Dependent Response at 160 Degrees

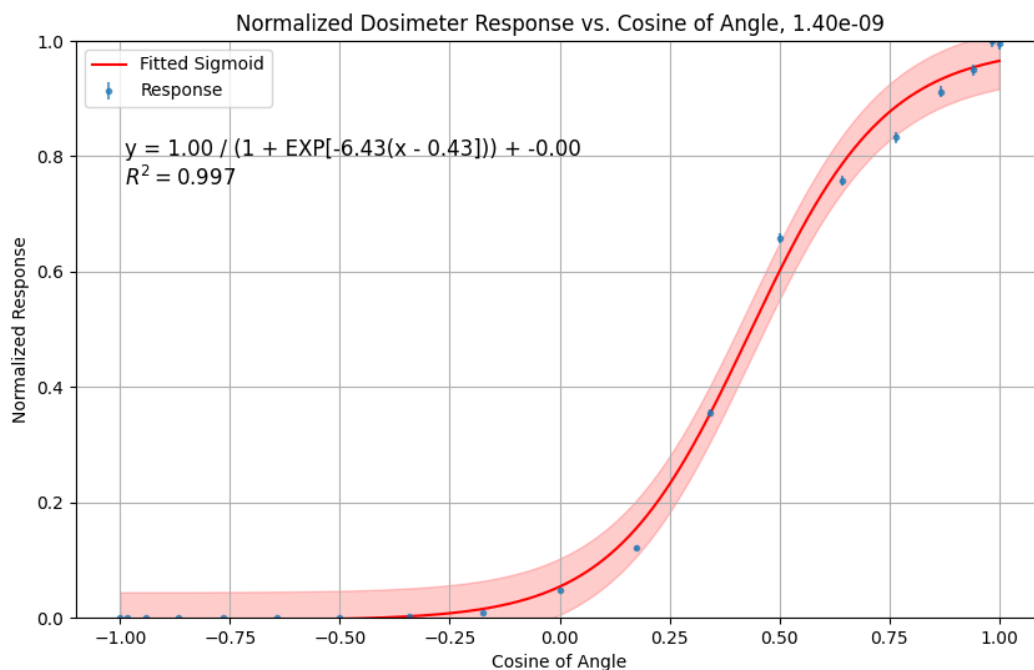


Appendix A. Figure A.18. Normalized Energy Dependent Response at 170 Degrees

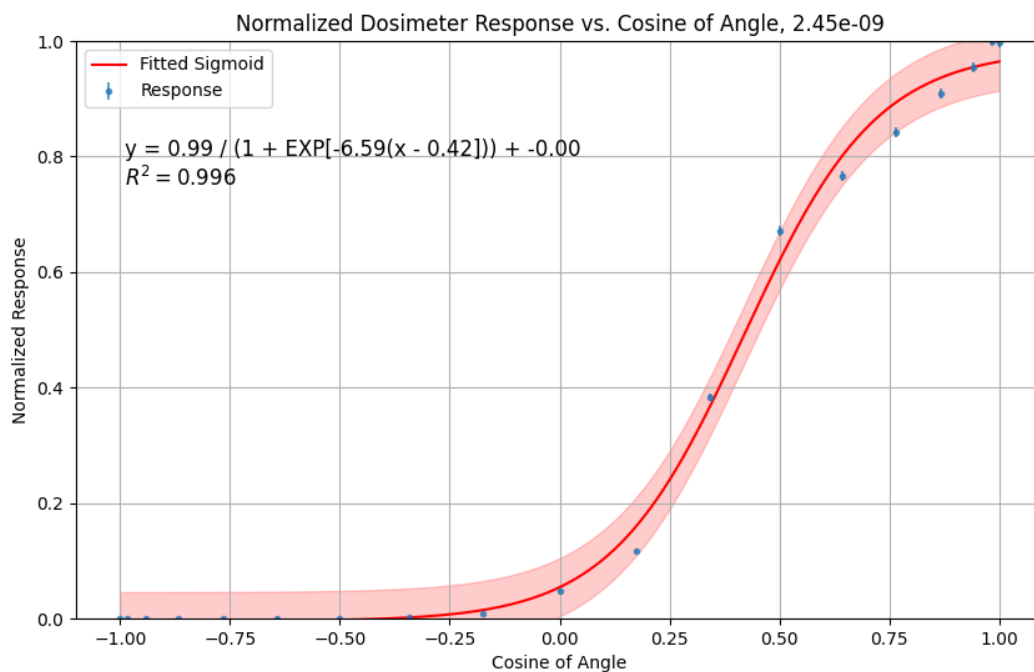


Appendix A. Figure A.19. Normalized Energy Dependent Response at 180 Degrees

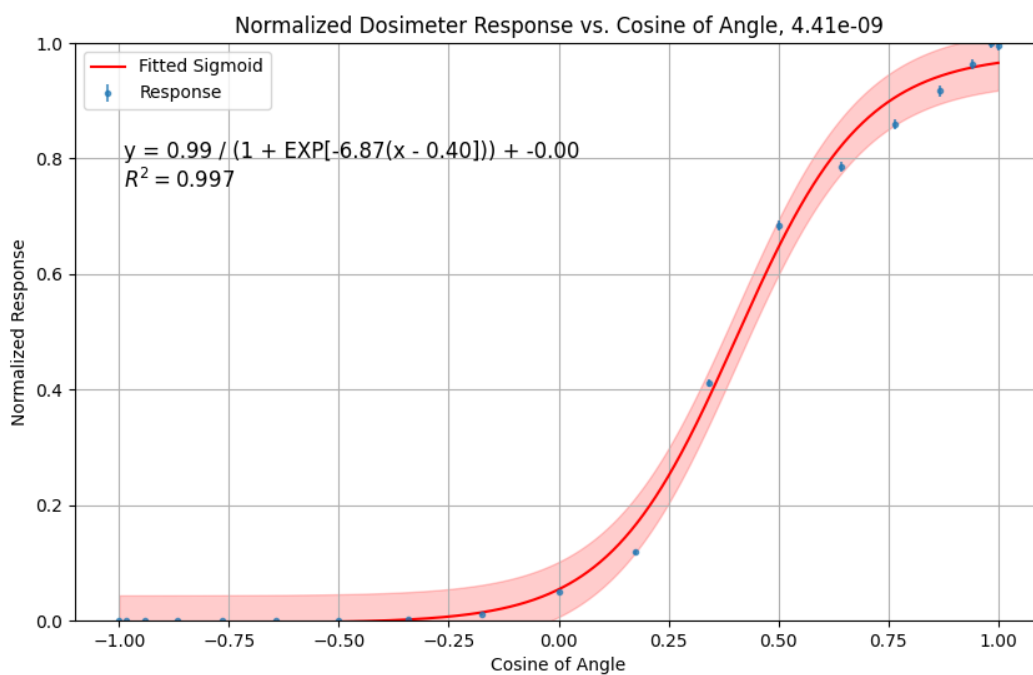
Appendix B - Angular Dependent Responses



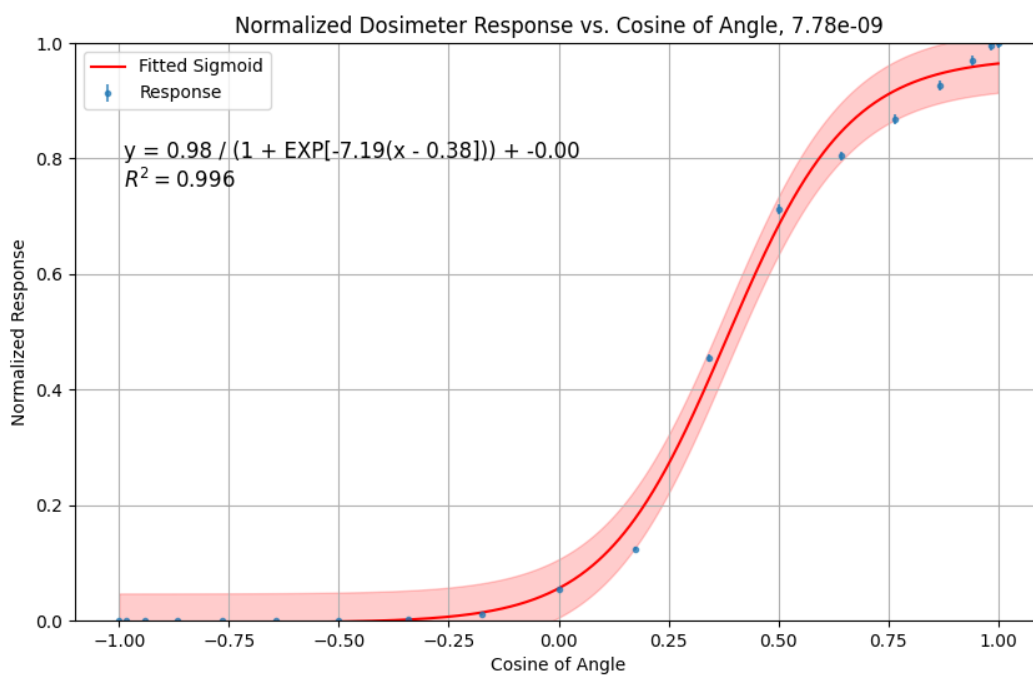
Appendix B. Figure B.1. Normalized Angular Dependent Response at 1.40×10^{-9} MeV



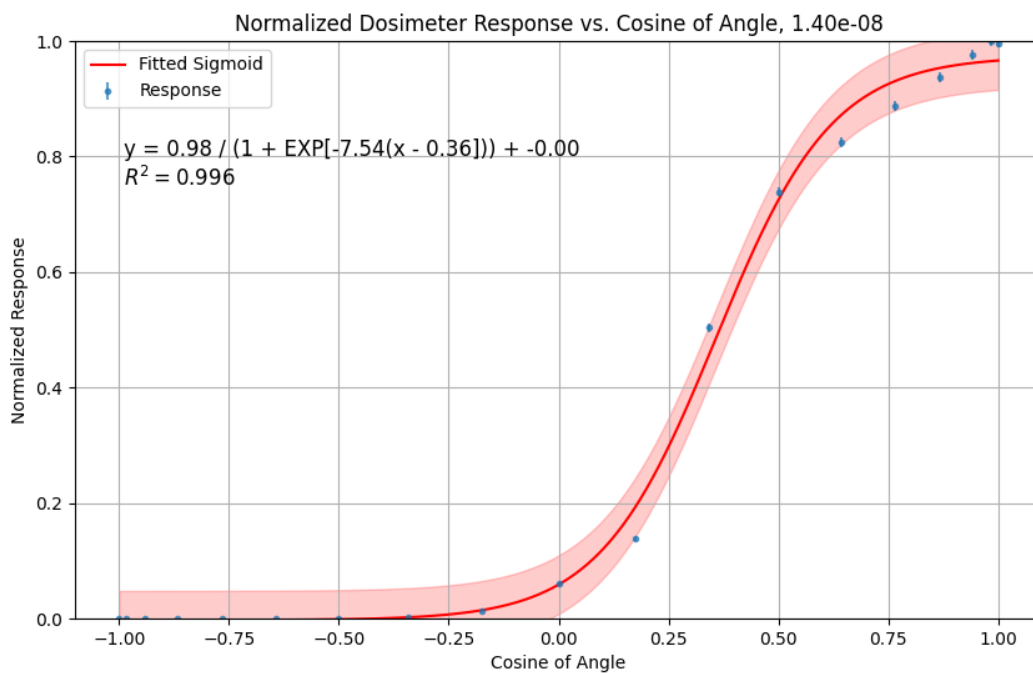
Appendix B. Figure B.2. Normalized Angular Dependent Response at 2.45×10^{-9} MeV



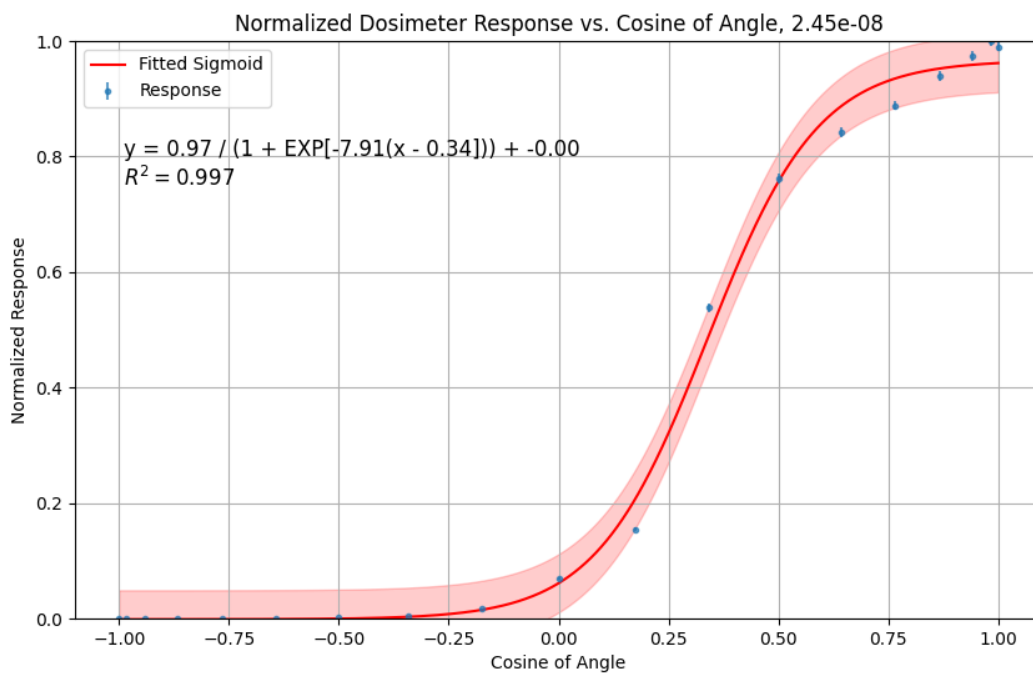
Appendix B. Figure B.3. Normalized Angular Dependent Response at 4.41×10^{-9} MeV



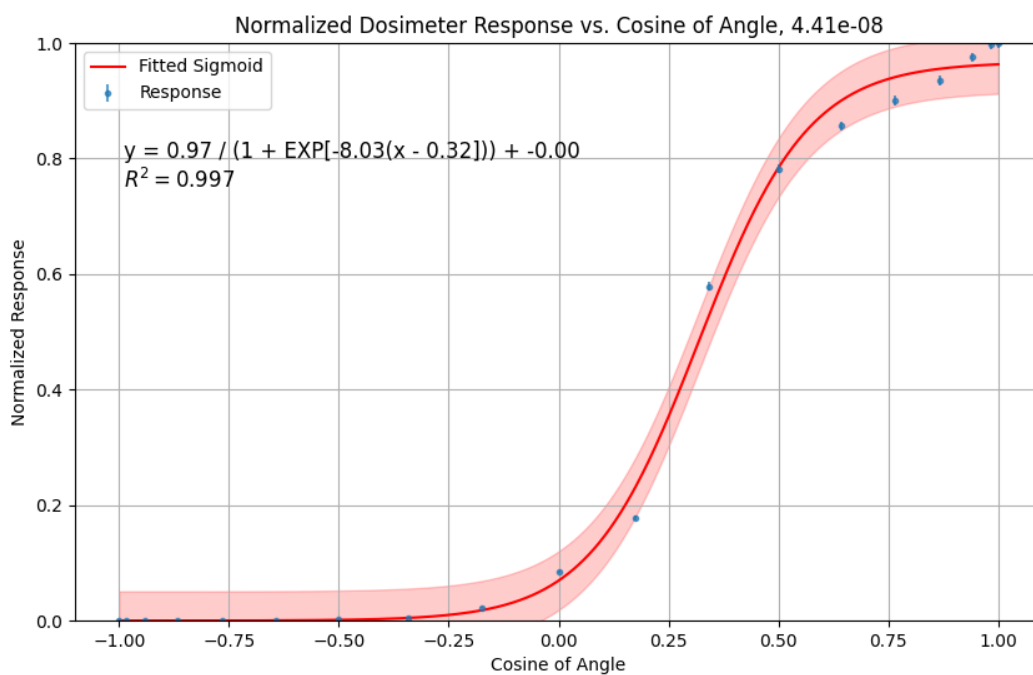
Appendix B. Figure B.4. Normalized Angular Dependent Response at 7.78×10^{-9} MeV



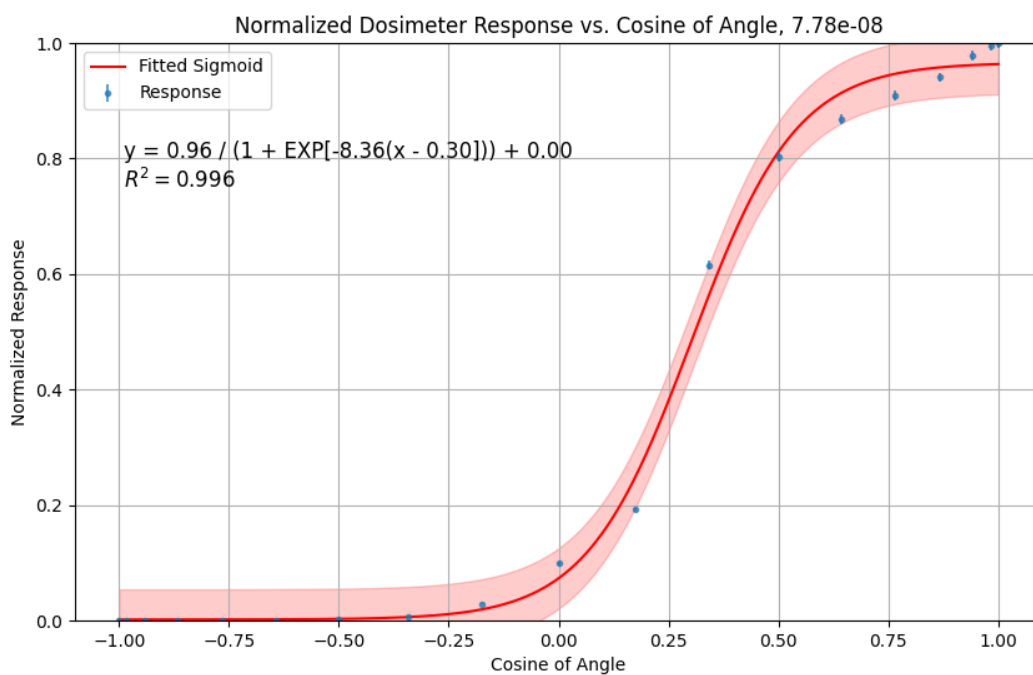
Appendix B. Figure B.5. Normalized Angular Dependent Response at 1.40×10^{-8} MeV



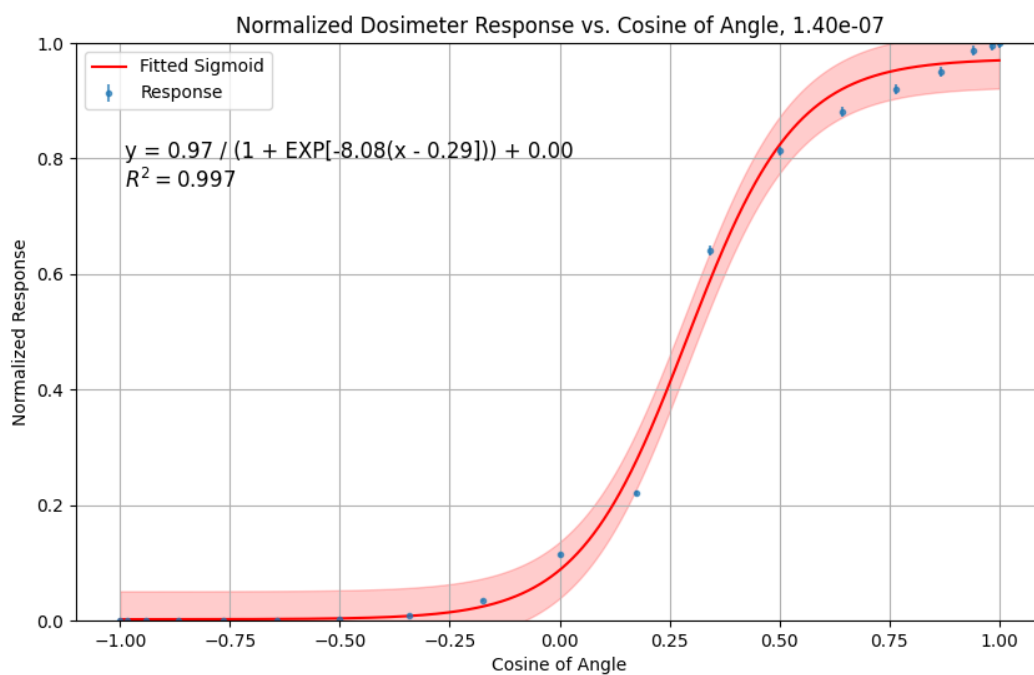
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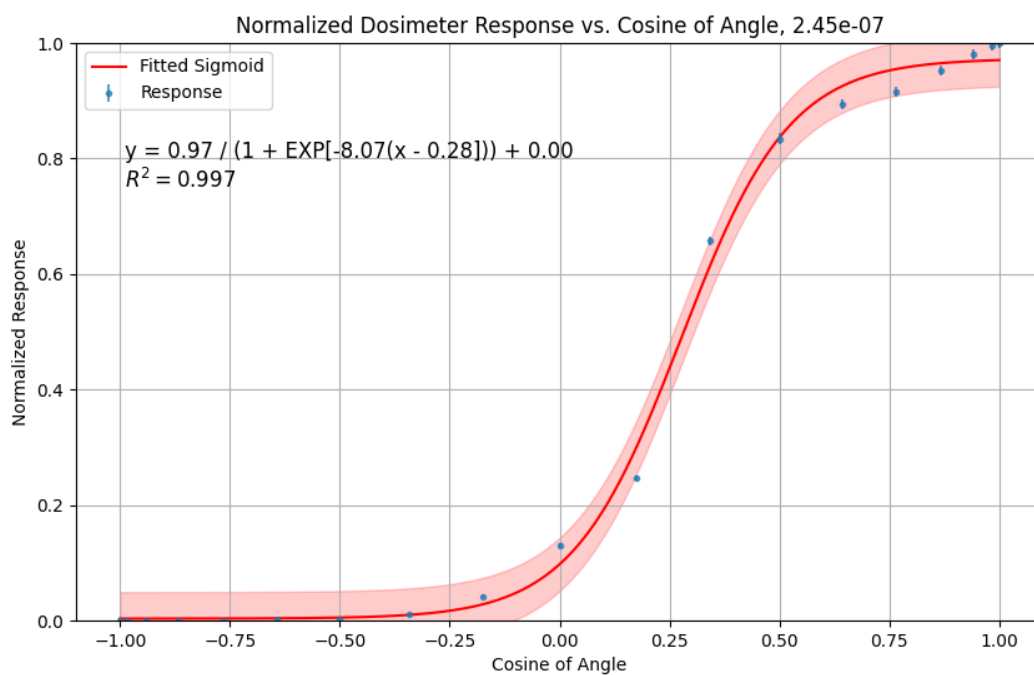
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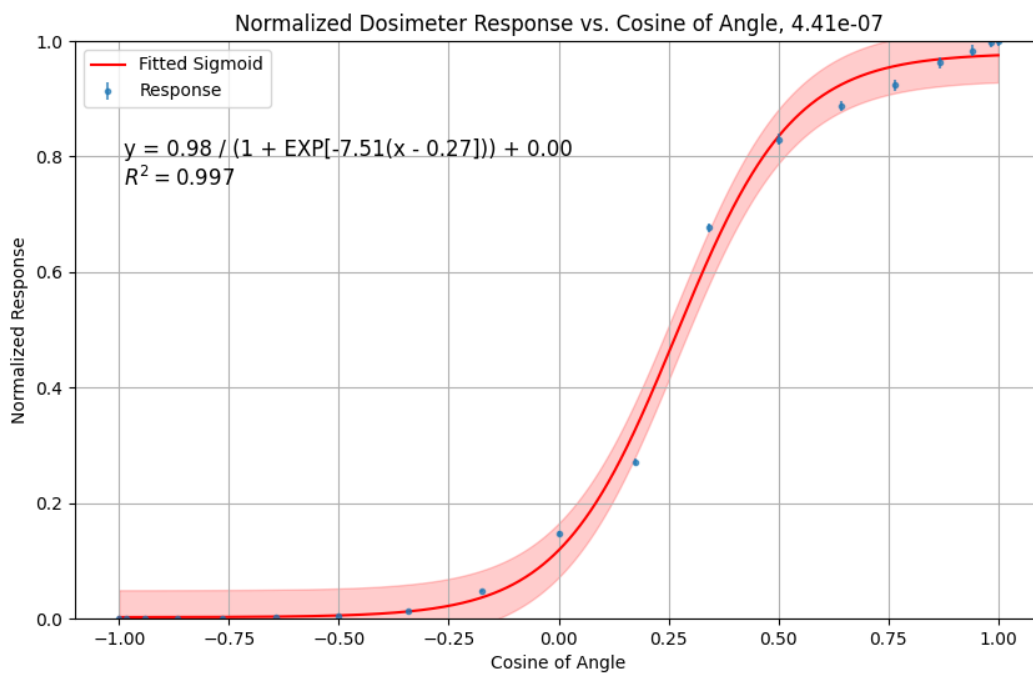
Appendix B. Figure B.8. Normalized Angular Dependent Response at 7.78×10^{-8} MeV



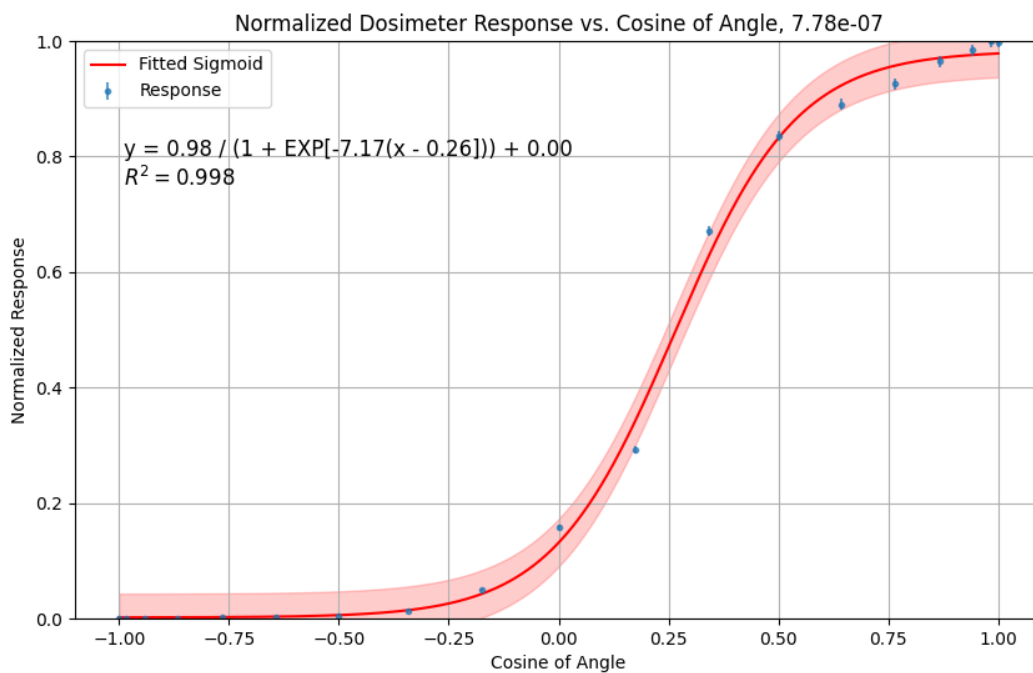
Appendix B. Figure B.9. Normalized Angular Dependent Response at 1.40×10^{-7} MeV



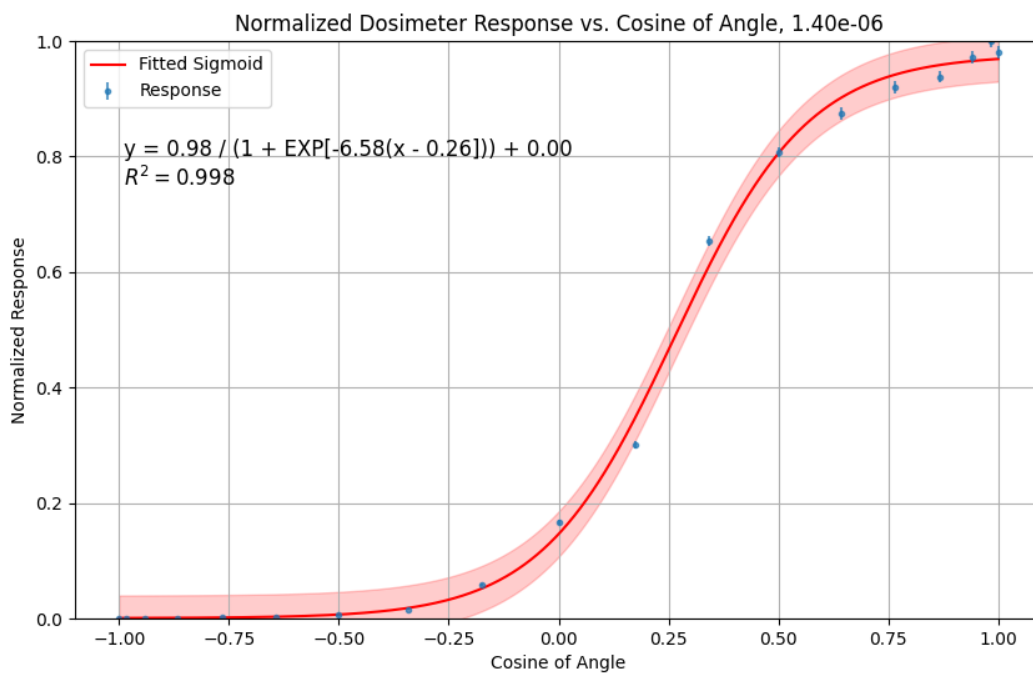
Appendix B. Figure B.10. Normalized Angular Dependent Response at 2.45×10^{-7} MeV



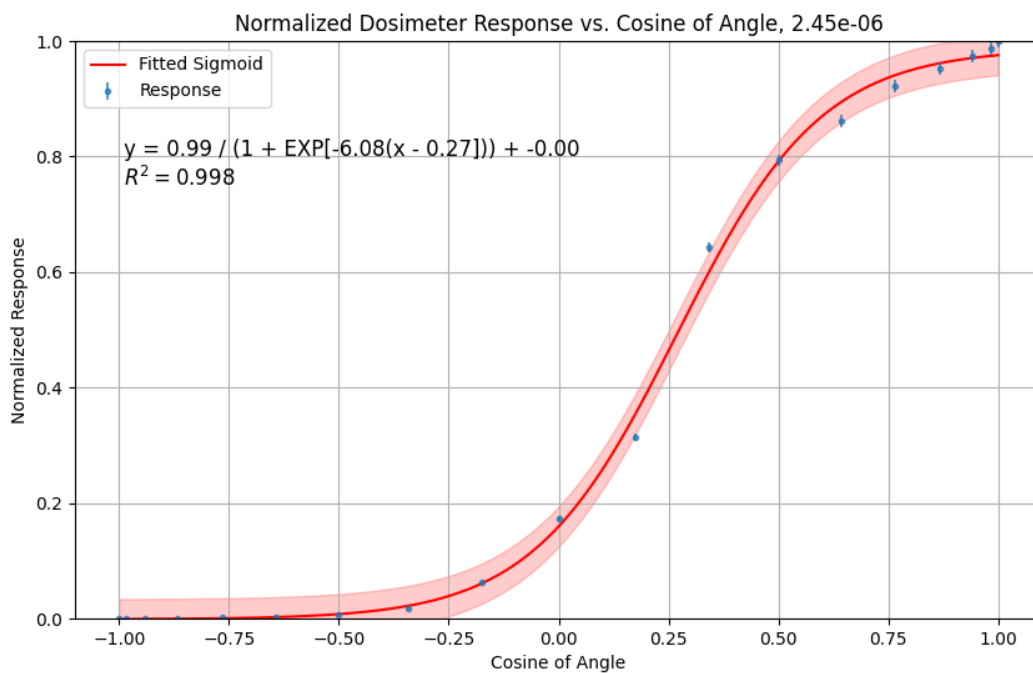
Appendix B. Figure B.11. Normalized Angular Dependent Response at 4.41×10^{-7} MeV



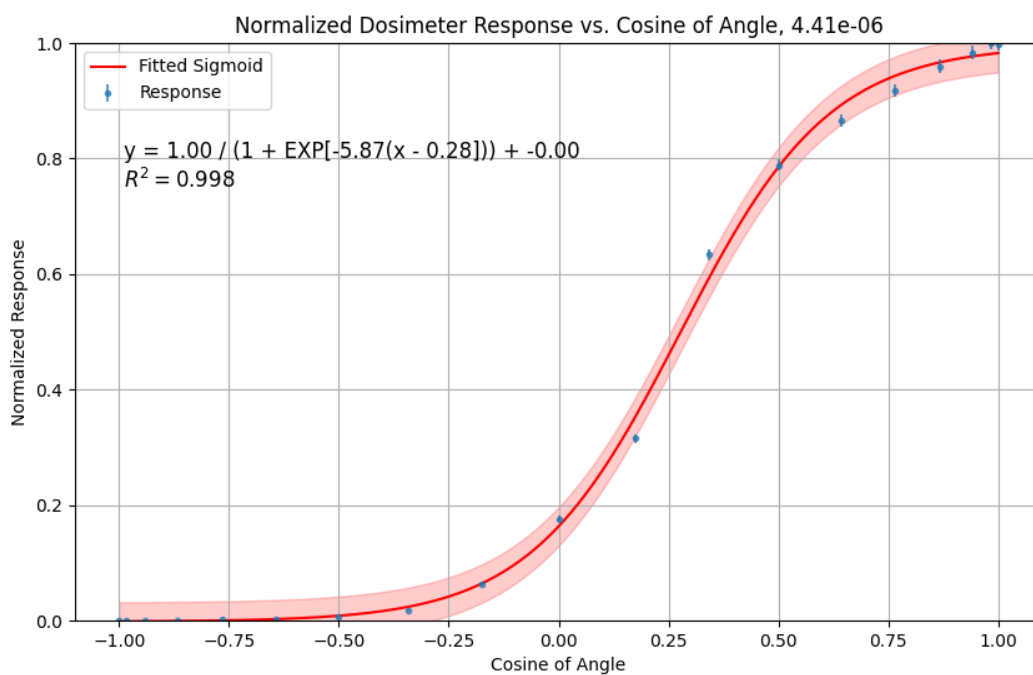
Appendix B. Figure B.12. Normalized Angular Dependent Response at 7.78×10^{-7} MeV



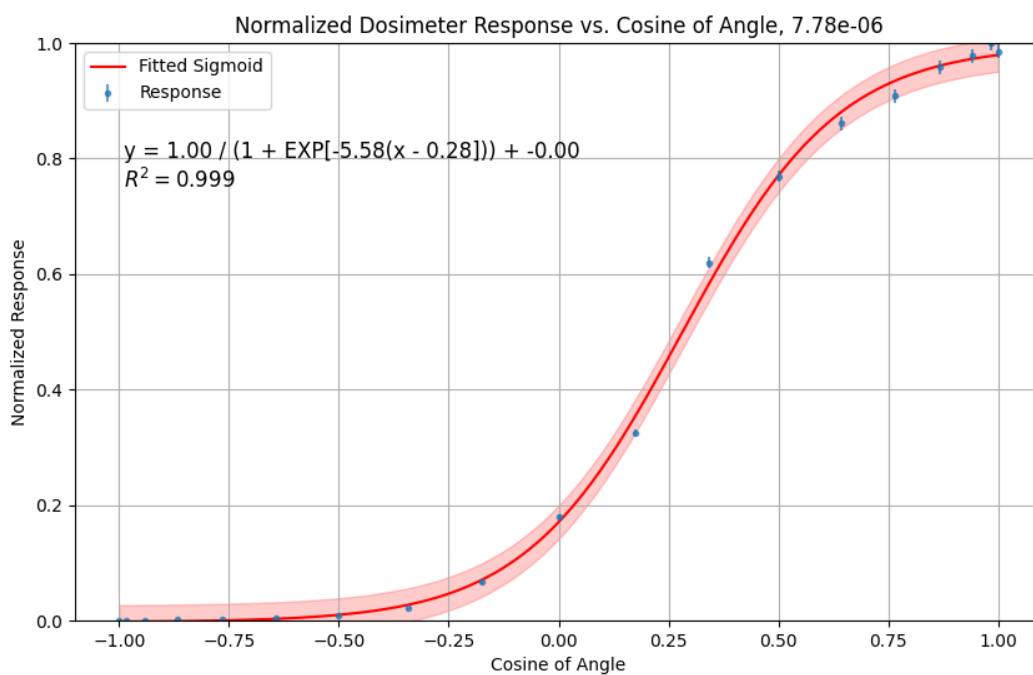
Appendix B. Figure B.13. Normalized Angular Dependent Response at 1.40×10^{-6} MeV



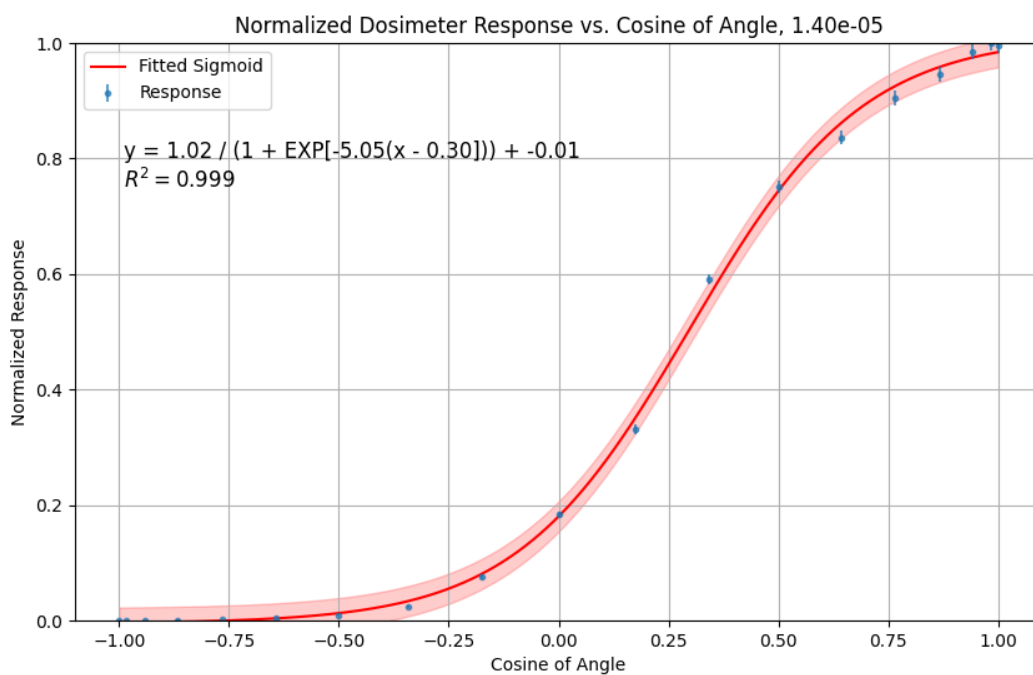
Appendix B. Figure B.14. Normalized Angular Dependent Response at 2.45×10^{-6} MeV



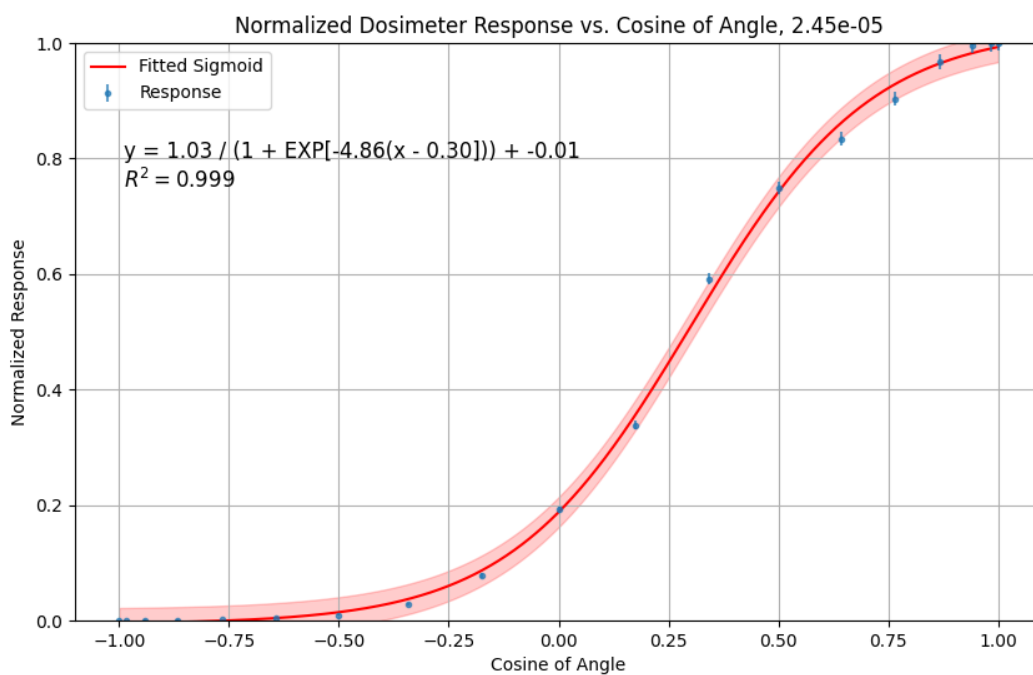
Appendix B. Figure B.15. Normalized Angular Dependent Response at 4.41×10^{-6} MeV



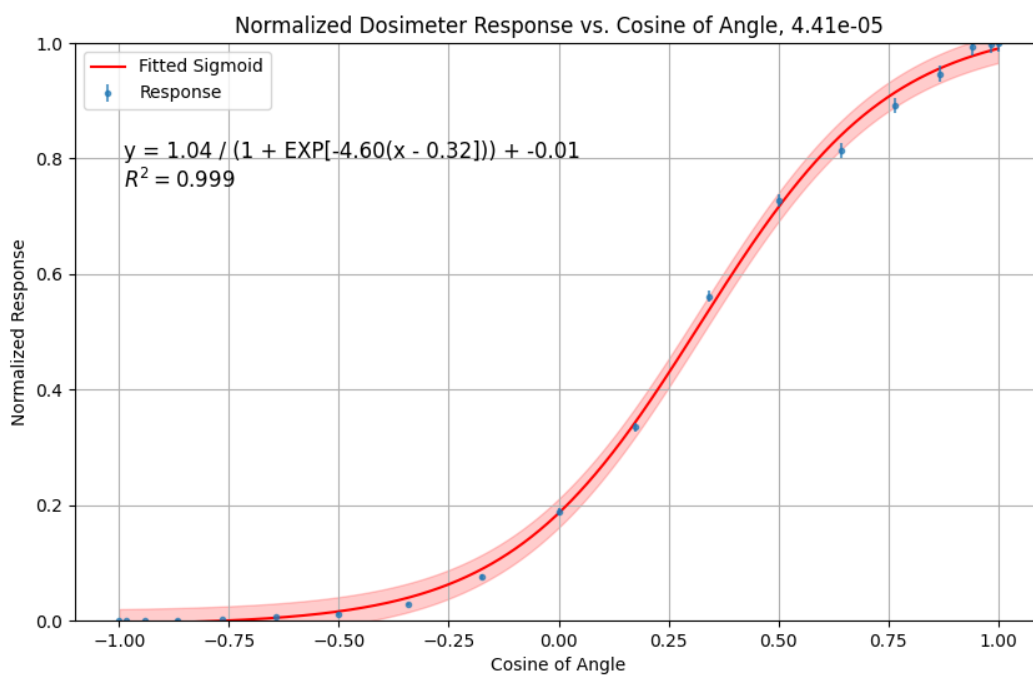
Appendix B. Figure B.16. Normalized Angular Dependent Response at 7.78×10^{-6} MeV



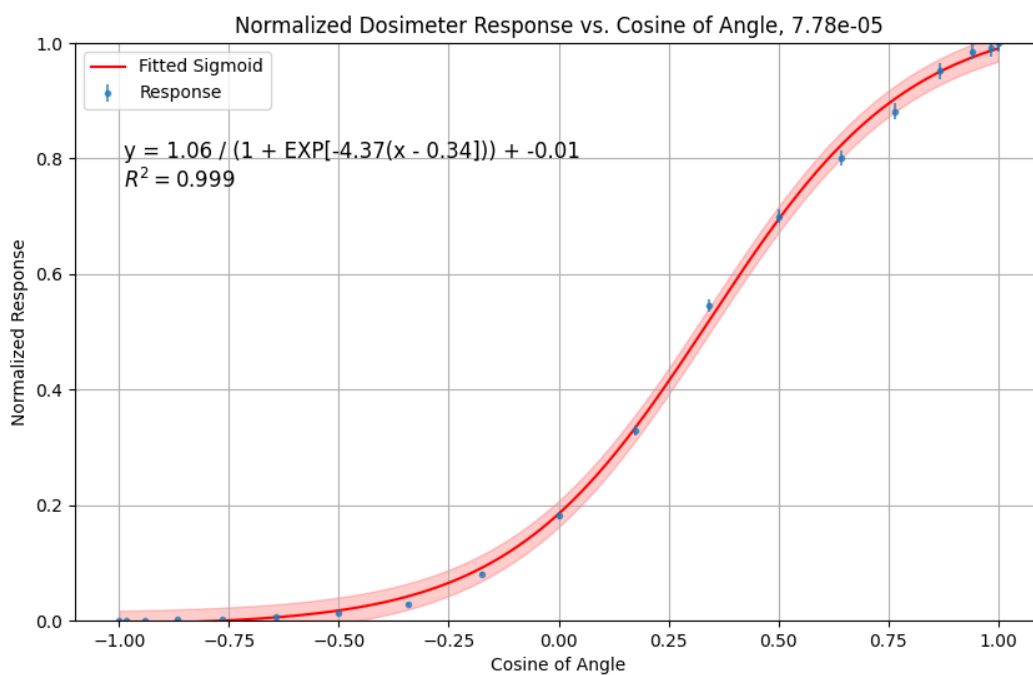
Appendix B. Figure B.17. Normalized Angular Dependent Response at 1.40×10^{-5} MeV



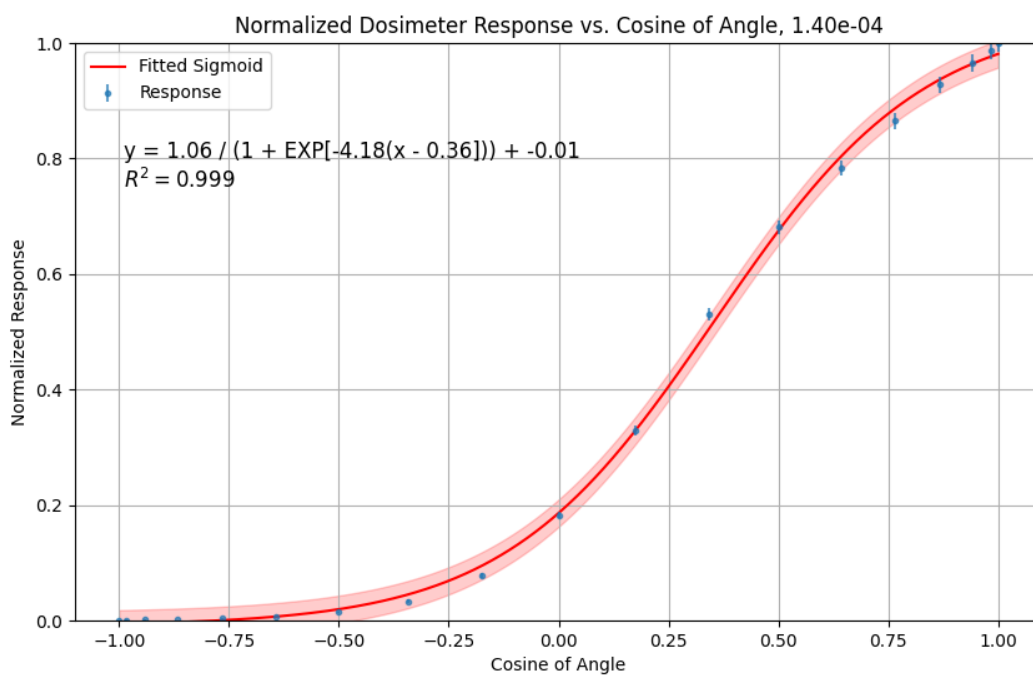
Appendix B. Figure B.18. Normalized Angular Dependent Response at 2.45×10^{-5} MeV



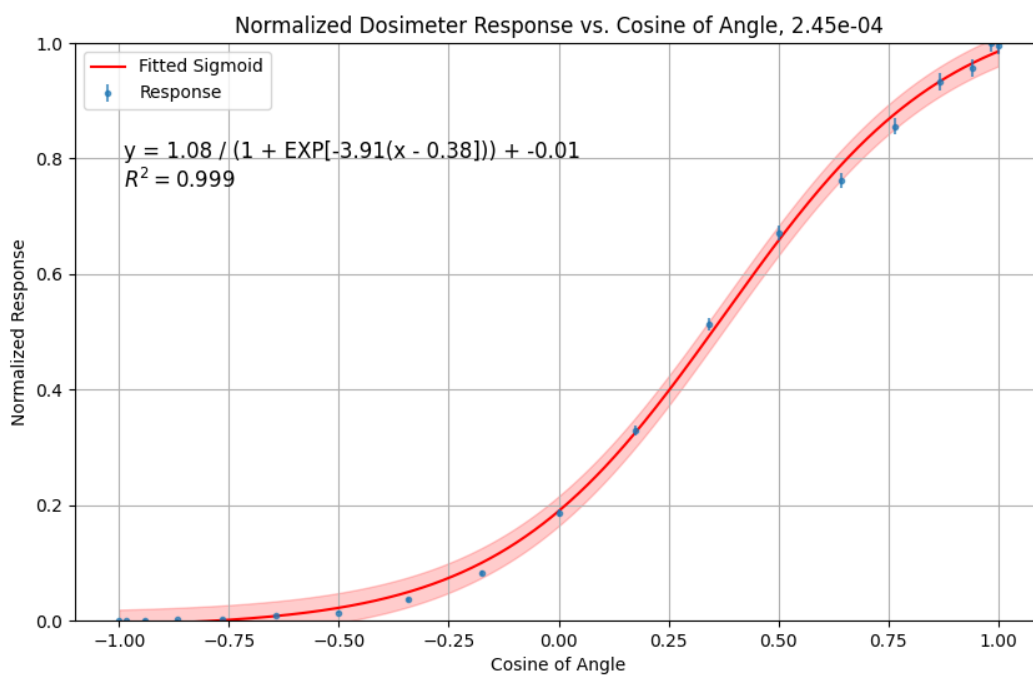
Appendix B. Figure B.19. Normalized Angular Dependent Response at 4.41×10^{-5} MeV



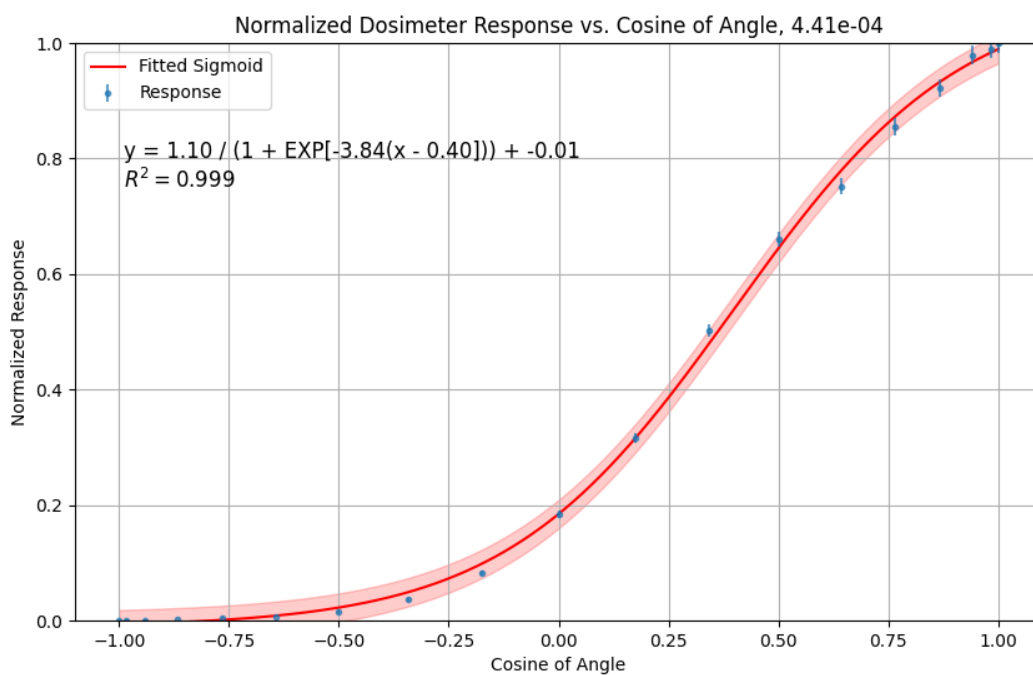
Appendix B. Figure B.20. Normalized Angular Dependent Response at 4.41×10^{-5} MeV



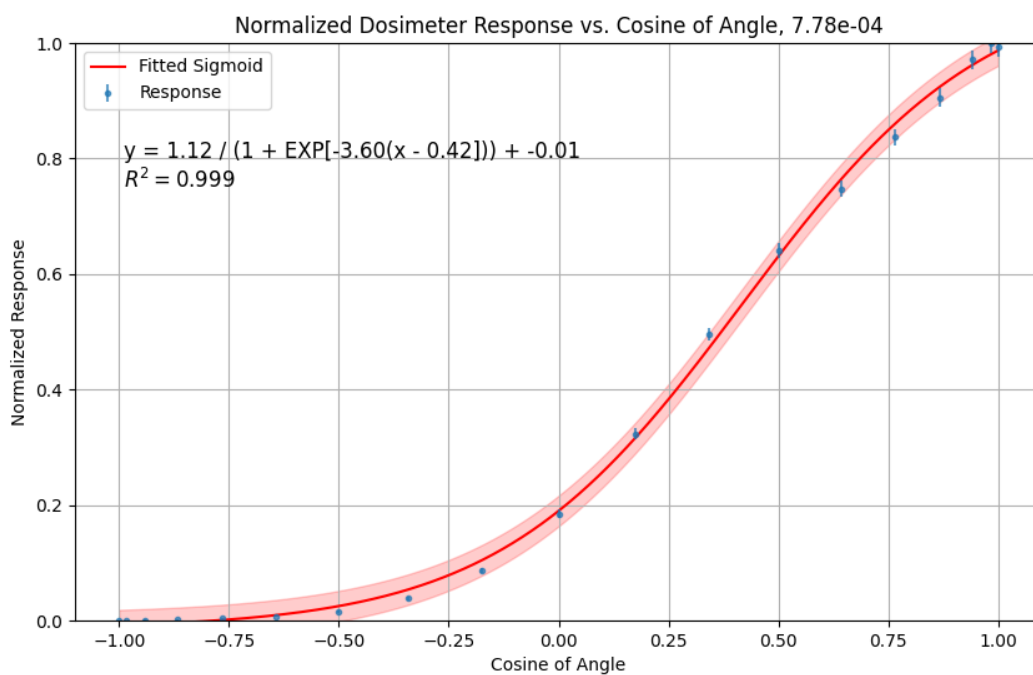
Appendix B. Figure B.21. Normalized Angular Dependent Response at 1.40×10^{-4} MeV



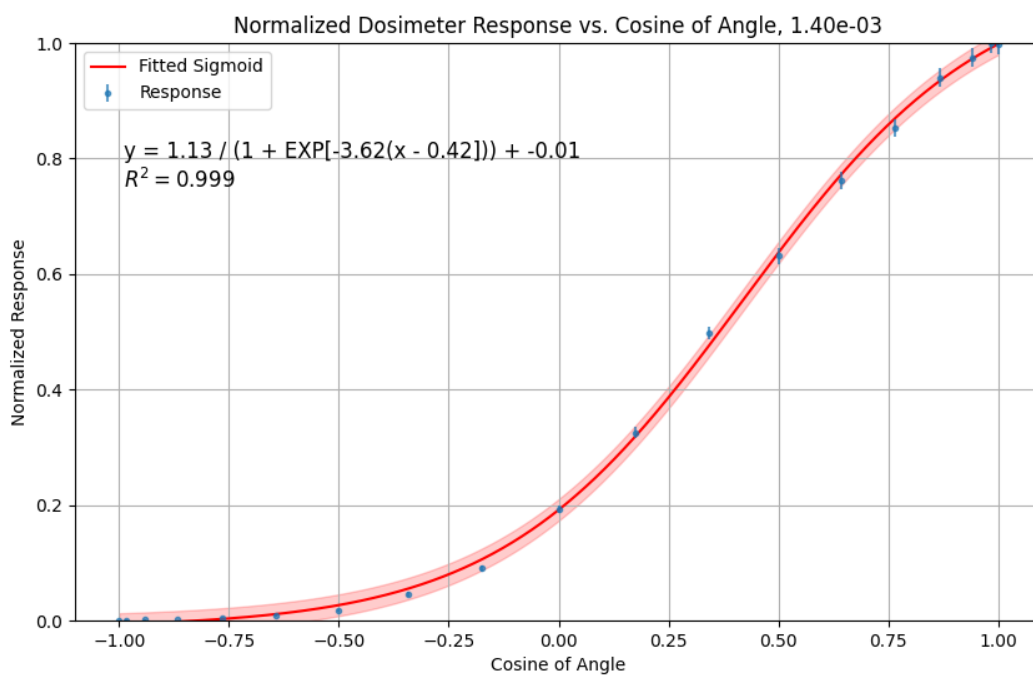
Appendix B. Figure B.22. Normalized Angular Dependent Response at 2.45×10^{-4} MeV



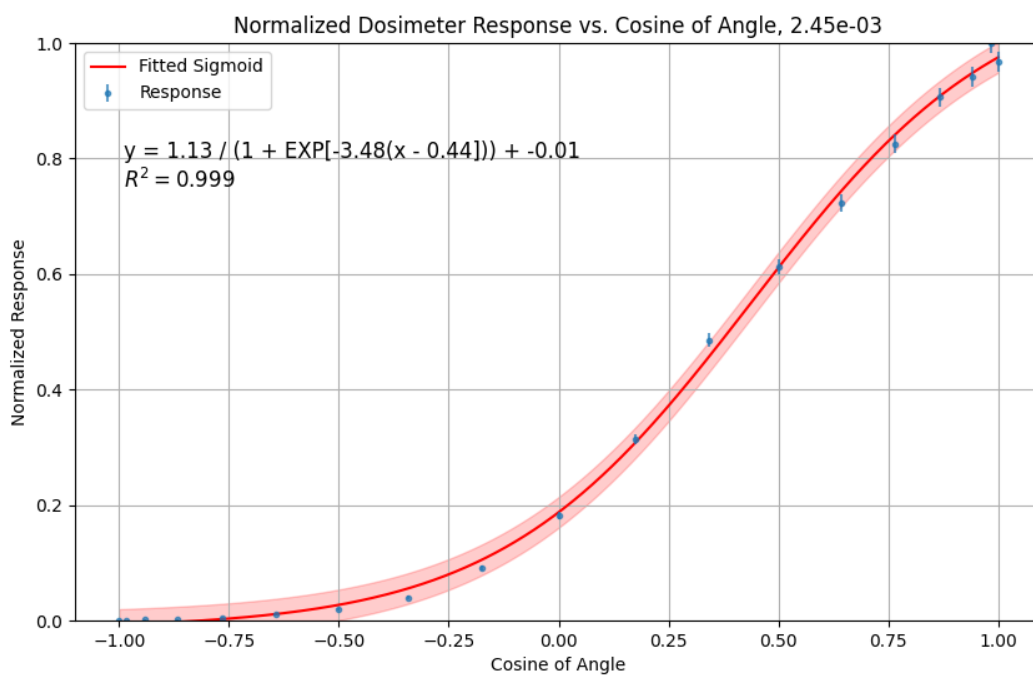
Appendix B. Figure B.23. Normalized Angular Dependent Response at 4.41×10^{-4} MeV



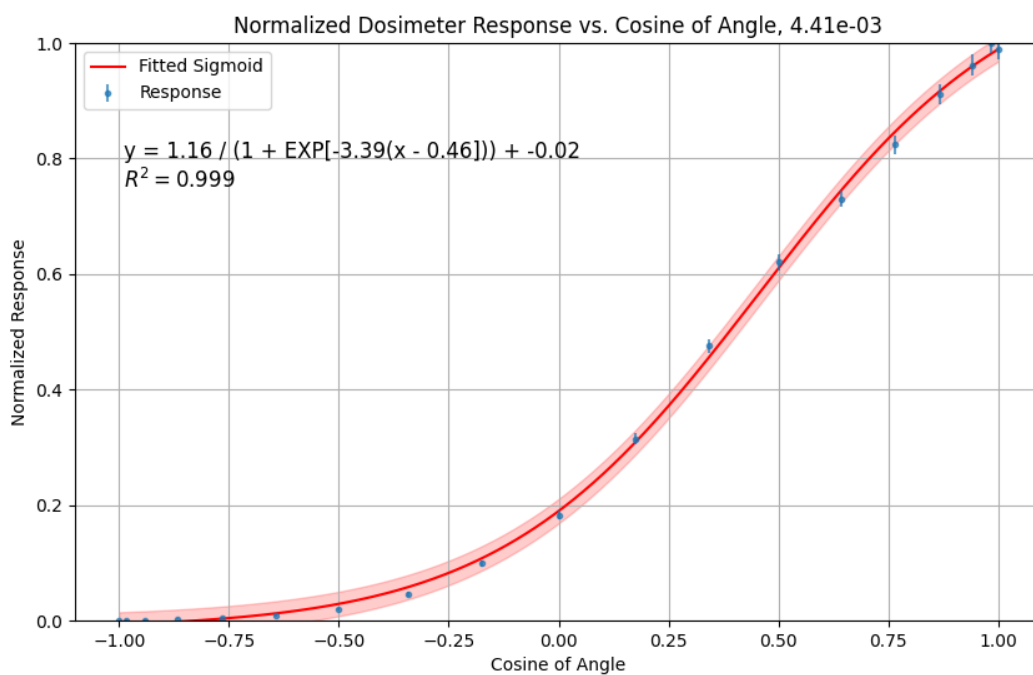
Appendix B. Figure B.24. Normalized Angular Dependent Response at 7.78×10^{-4} MeV



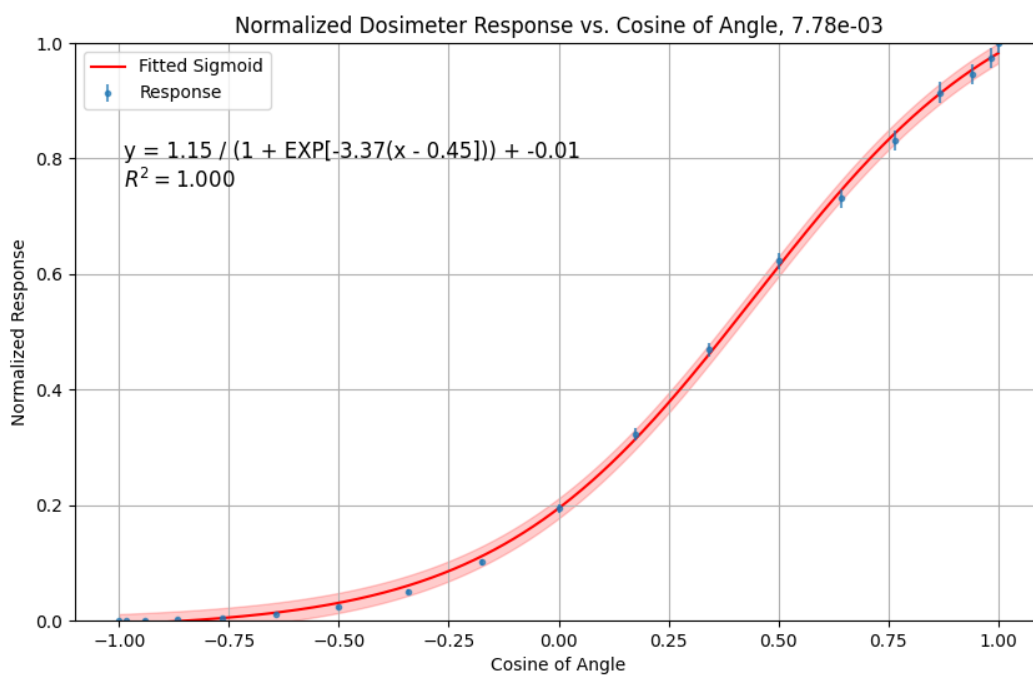
Appendix B. Figure B.25. Normalized Angular Dependent Response at 1.40×10^{-3} MeV



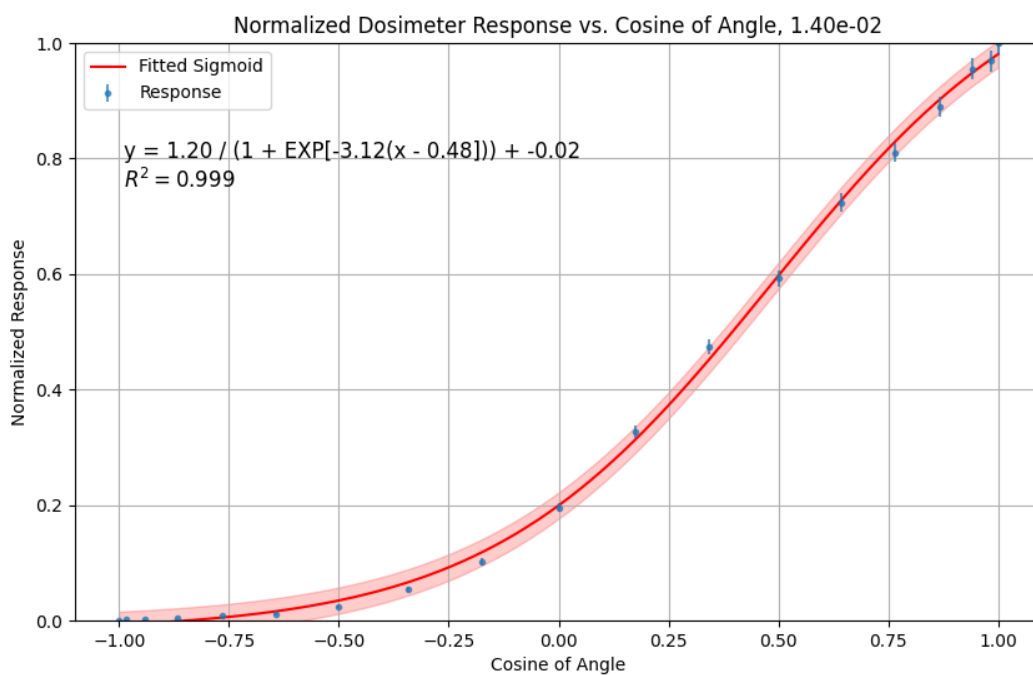
Appendix B. Figure B.26. Normalized Angular Dependent Response at 2.45×10^{-3} MeV



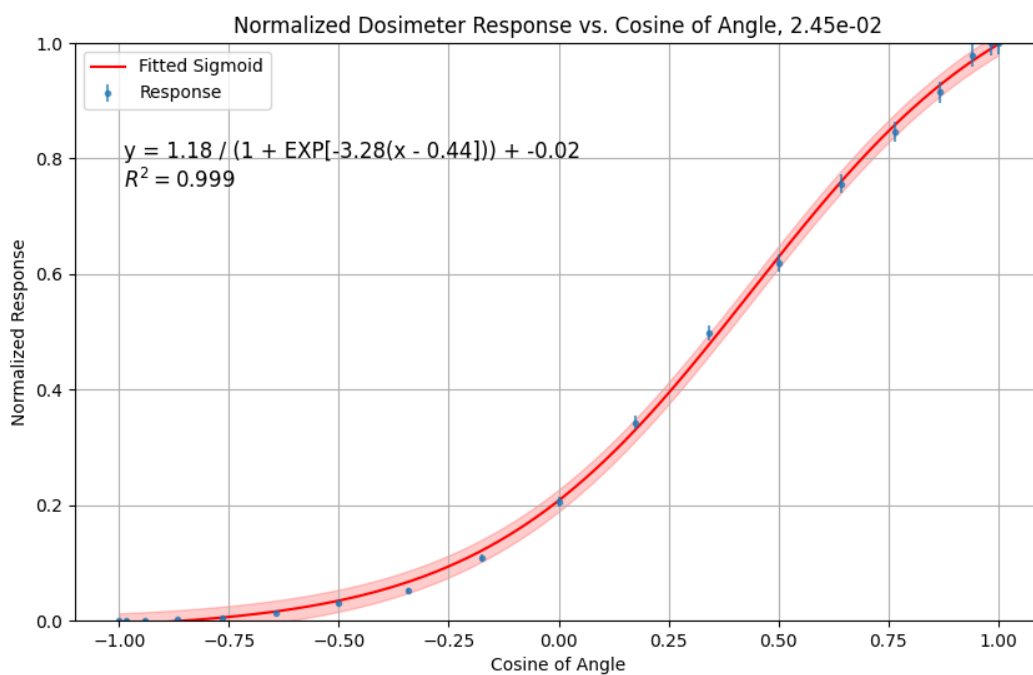
Appendix B. Figure B.27. Normalized Angular Dependent Response at 4.41×10^{-3} MeV



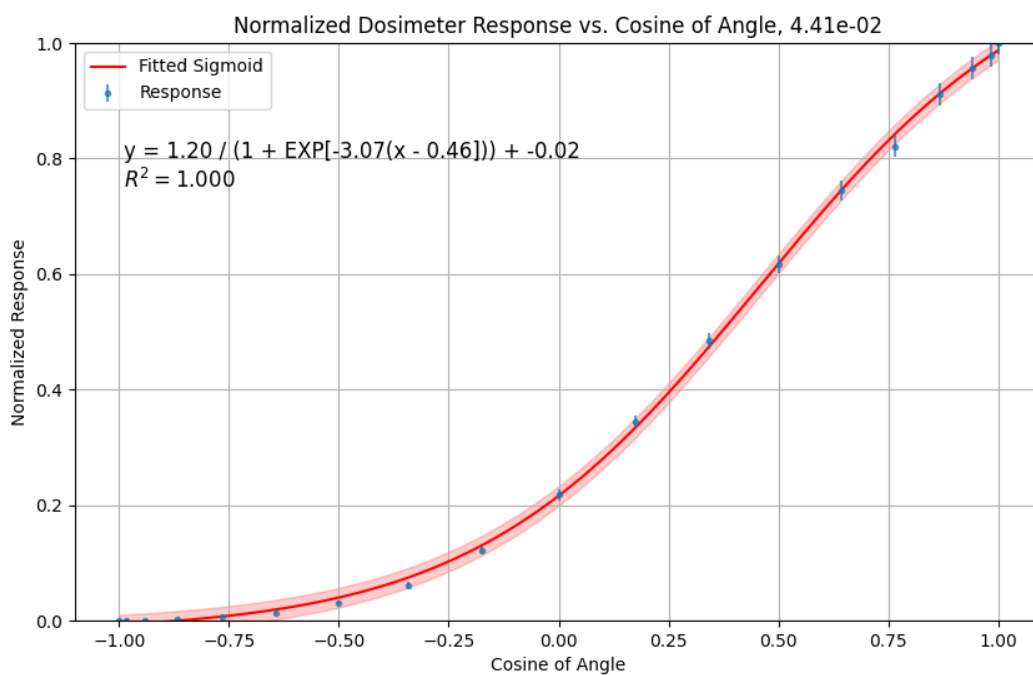
Appendix B. Figure B.28. Normalized Angular Dependent Response at 7.78×10^{-3} MeV



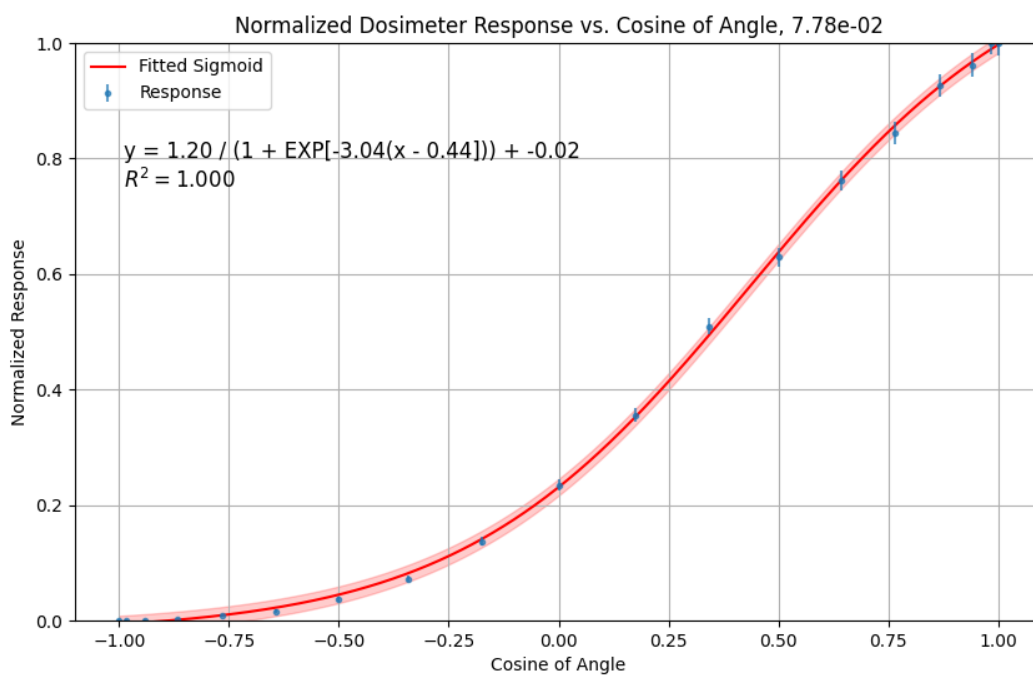
Appendix B. Figure B.29. Normalized Angular Dependent Response at 1.40×10^{-2} MeV



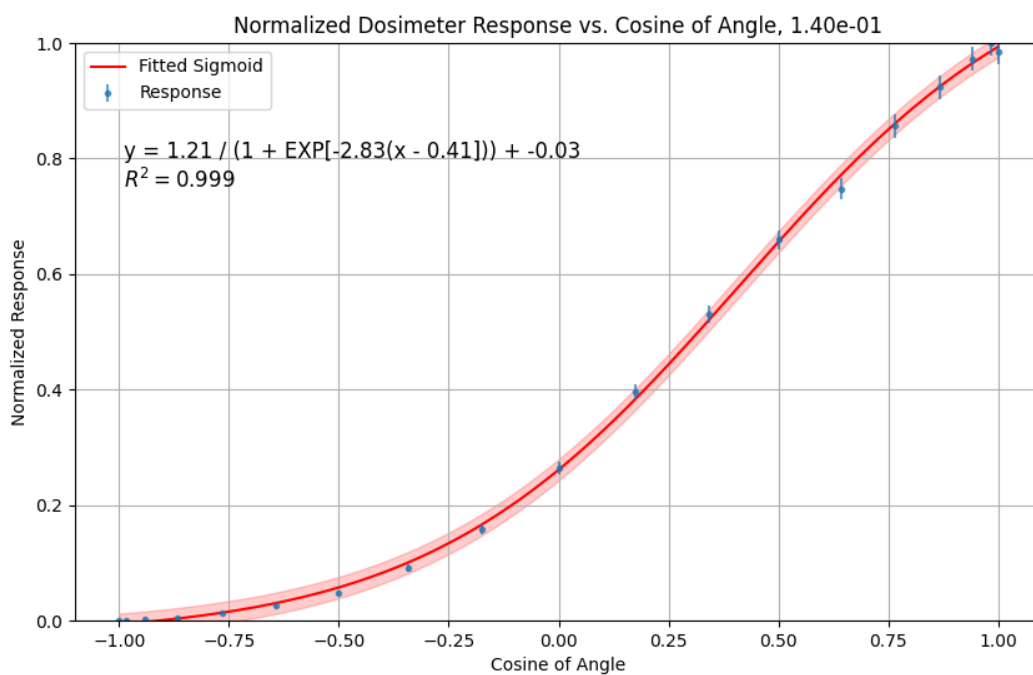
Appendix B. Figure B.30. Normalized Angular Dependent Response at 2.45×10^{-2} MeV



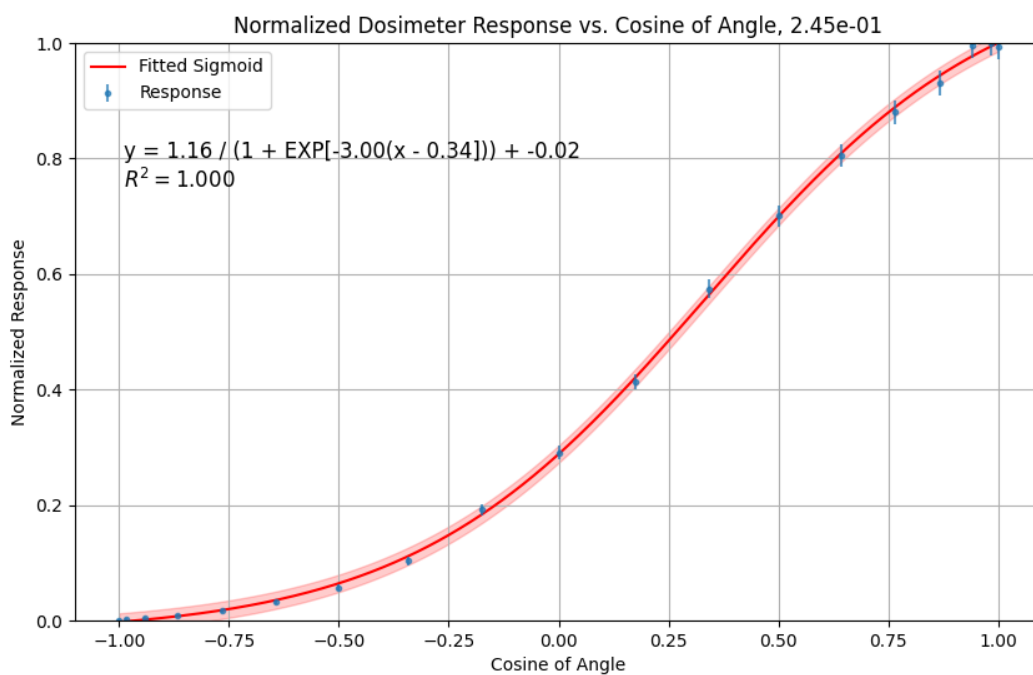
Appendix B. Figure B.31. Normalized Angular Dependent Response at 4.41×10^{-2} MeV



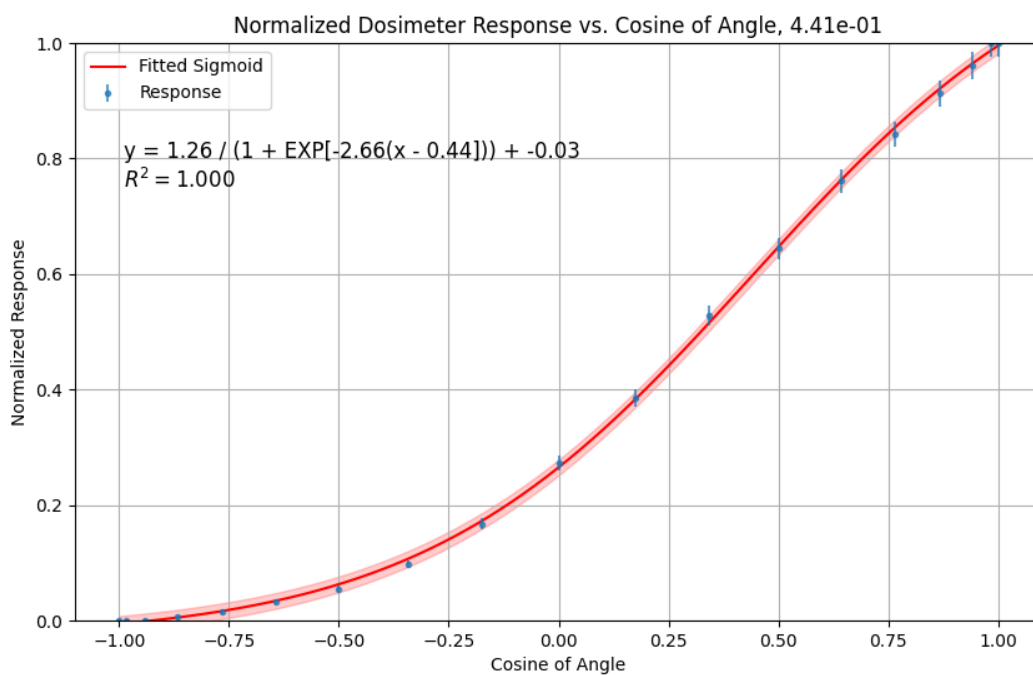
Appendix B. Figure B.32. Normalized Angular Dependent Response at 7.78×10^{-2} MeV



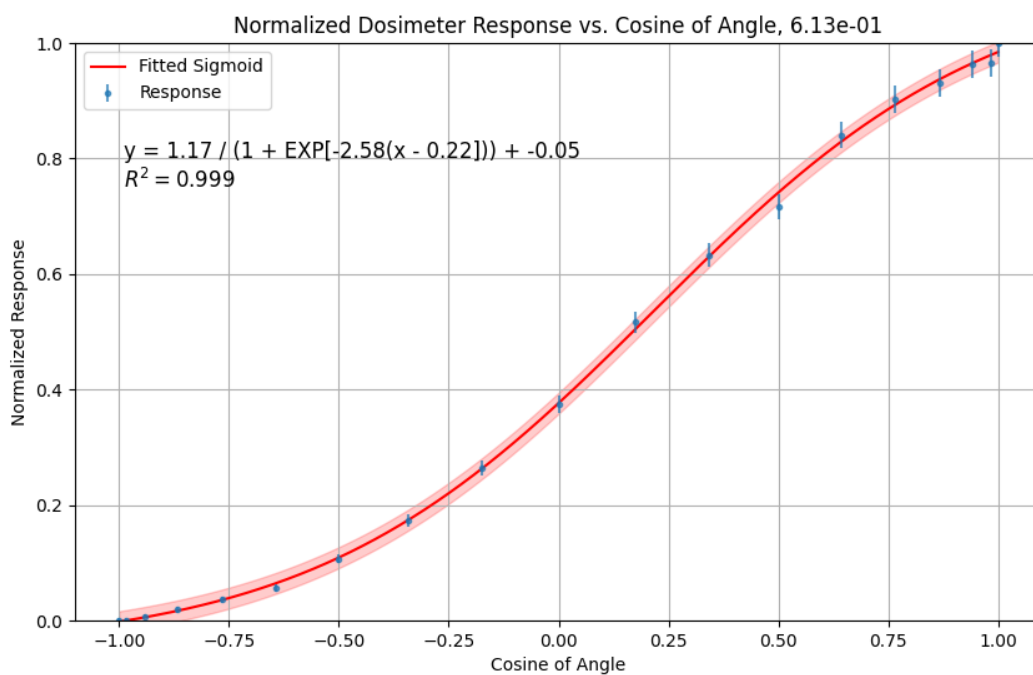
Appendix B. Figure B.33. Normalized Angular Dependent Response at 1.40×10^{-1} MeV



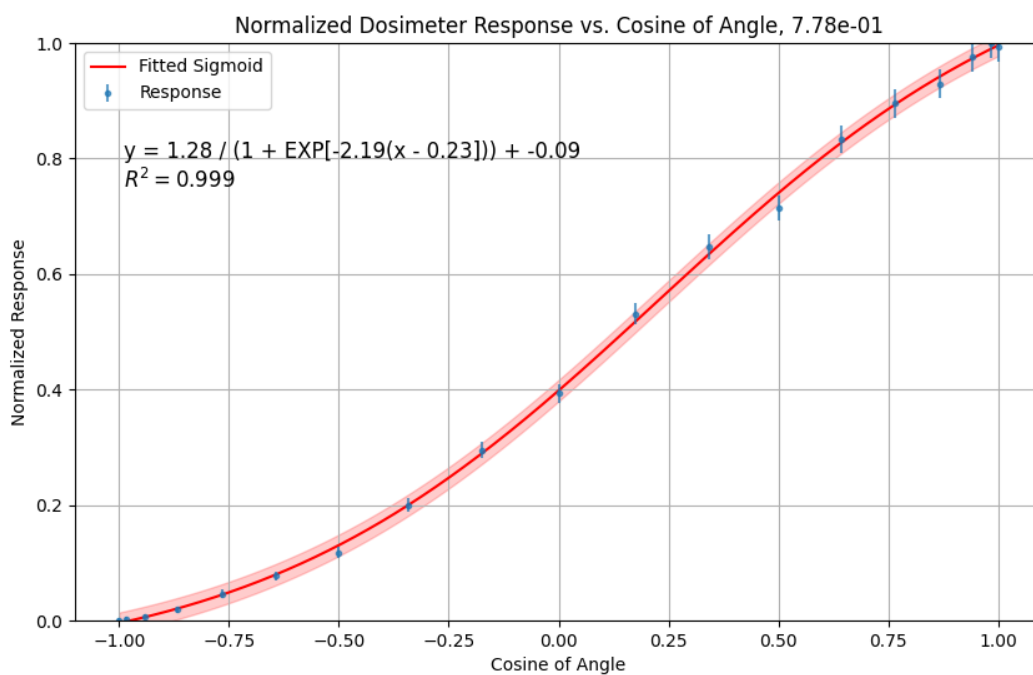
Appendix B. Figure B.34. Normalized Angular Dependent Response at 2.45×10^{-1} MeV



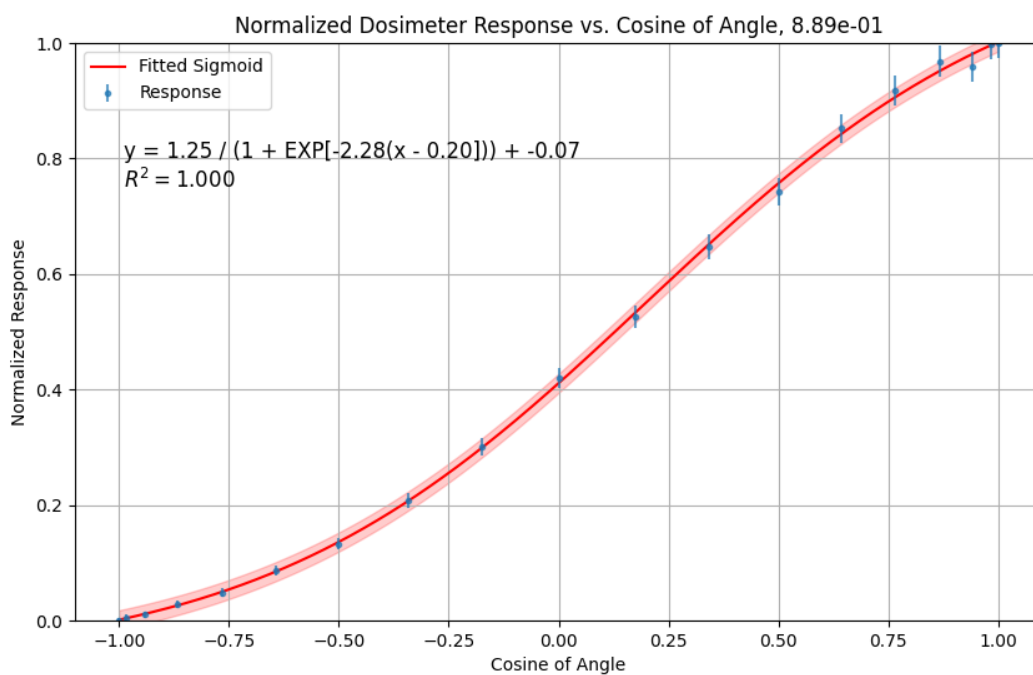
Appendix B. Figure B.35. Normalized Angular Dependent Response at 4.41×10^{-1} MeV



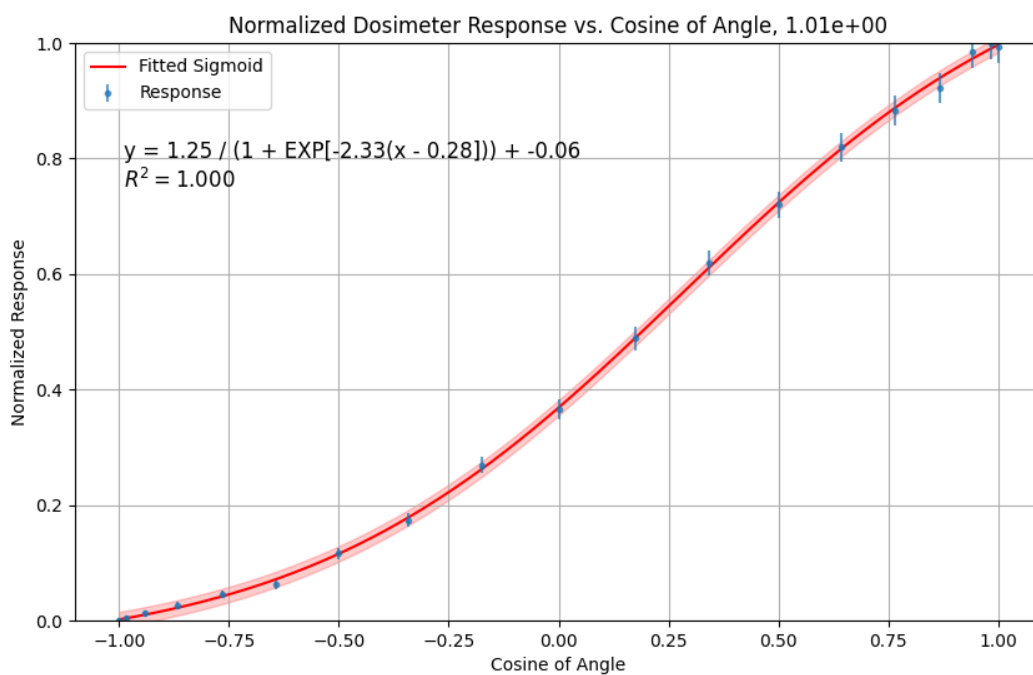
Appendix B. Figure B.36. Normalized Angular Dependent Response at 6.13×10^{-1} MeV



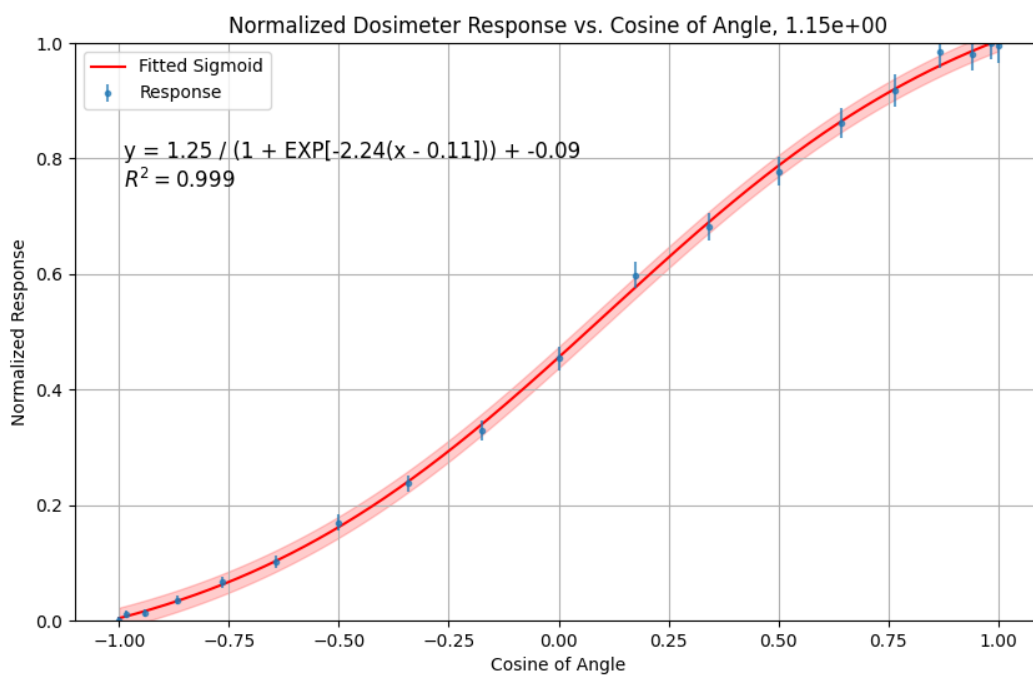
Appendix B. Figure B.37. Normalized Angular Dependent Response at 7.78×10^{-1} MeV



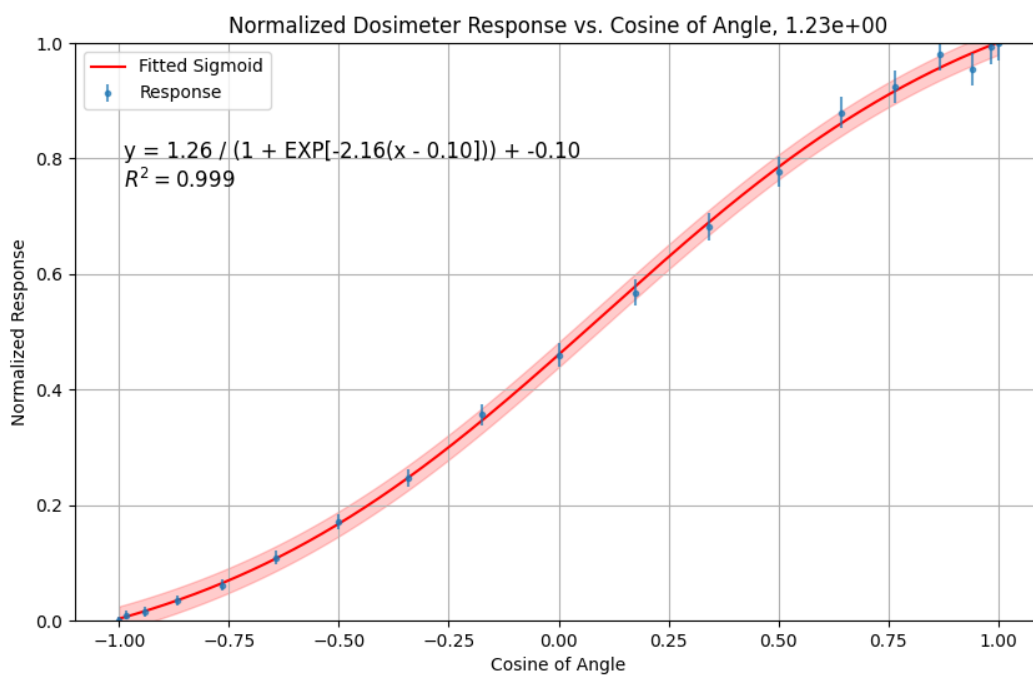
Appendix B. Figure B.38. Normalized Angular Dependent Response at 8.89×10^{-1} MeV



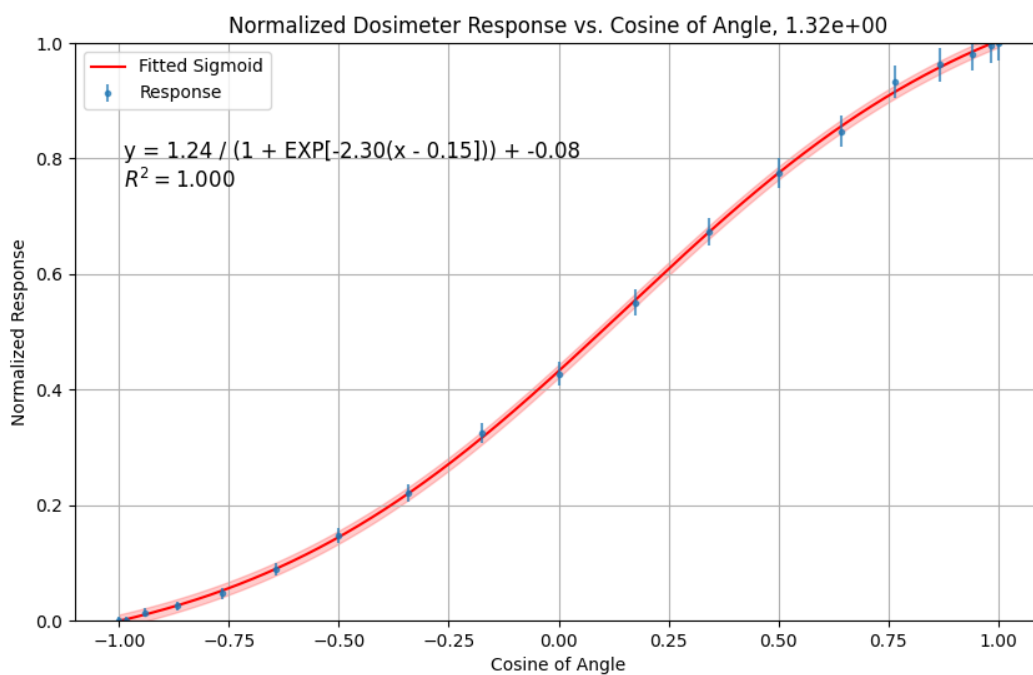
Appendix B. Figure B.39. Normalized Angular Dependent Response at 1.01 MeV



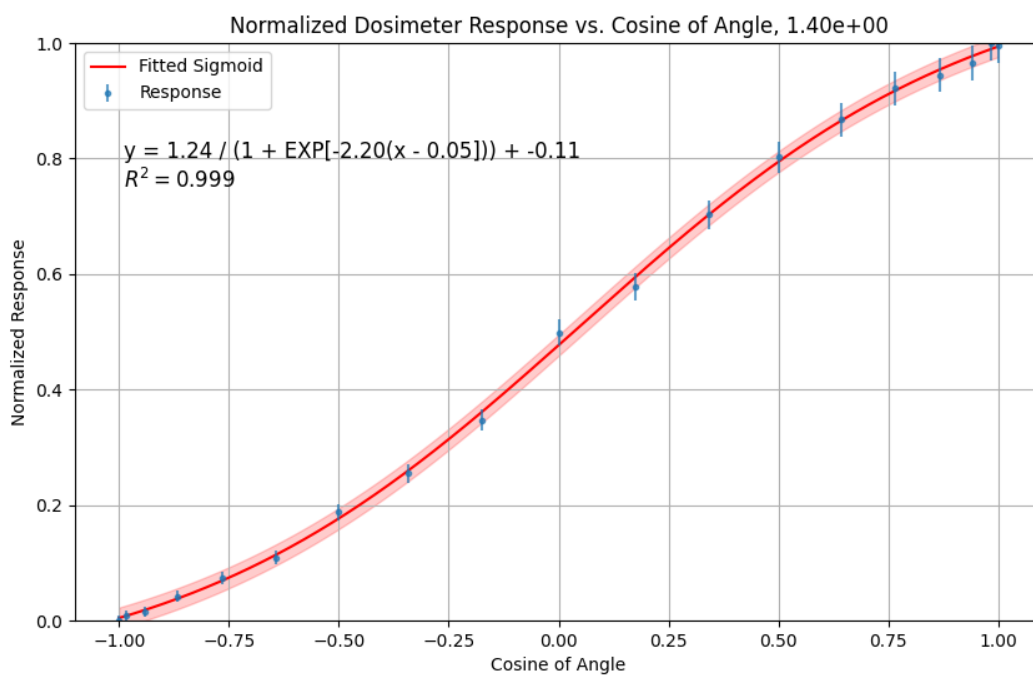
Appendix B. Figure B.40. Normalized Angular Dependent Response at 1.15 MeV



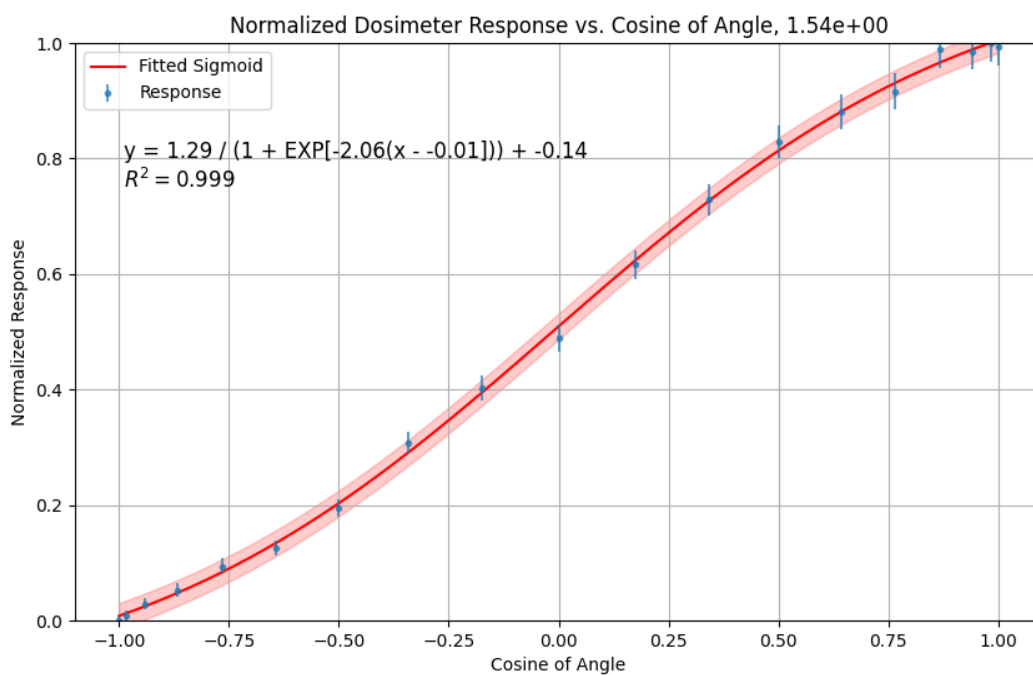
Appendix B. Figure B.41. Normalized Angular Dependent Response at 1.23 MeV



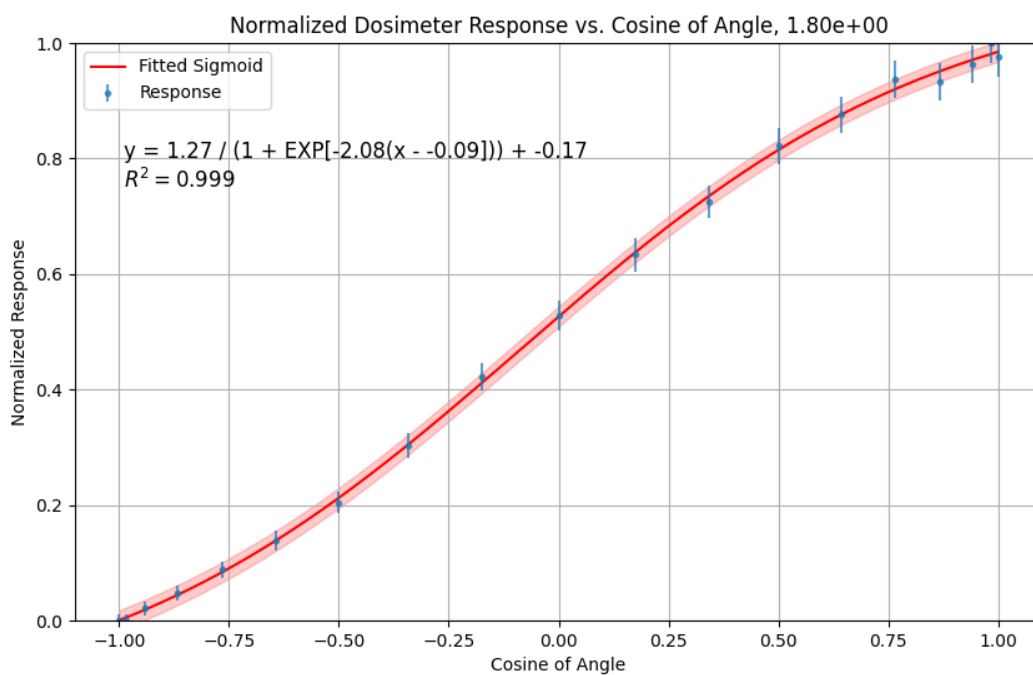
Appendix B. Figure B.42. Normalized Angular Dependent Response at 1.32 MeV



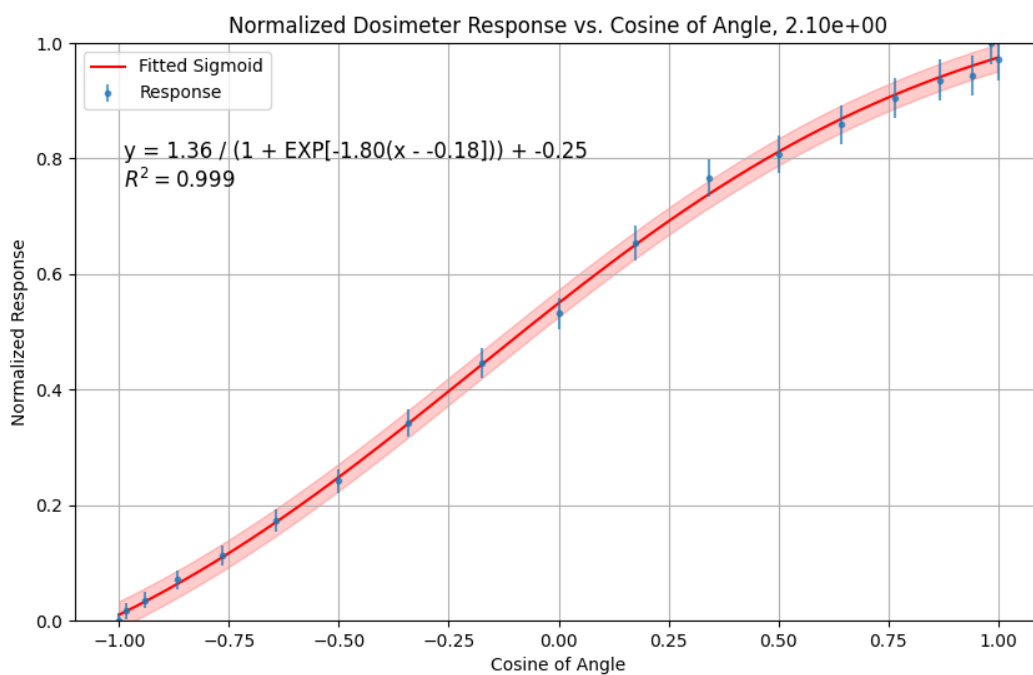
Appendix B. Figure B.43. Normalized Angular Dependent Response at 1.40 MeV



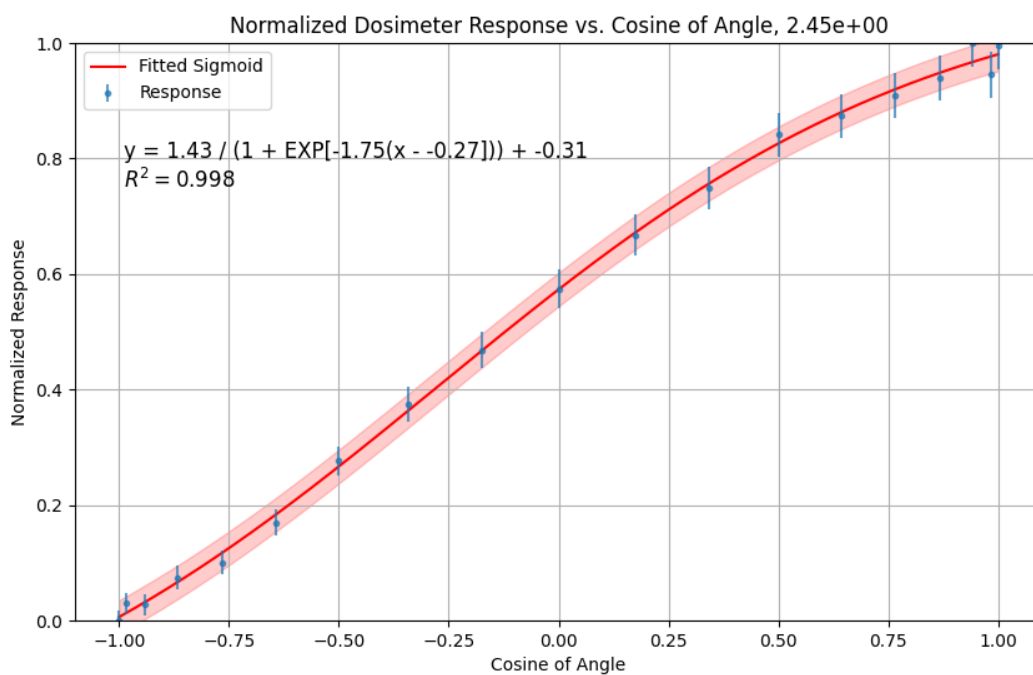
Appendix B. Figure B.44. Normalized Angular Dependent Response at 1.54 MeV



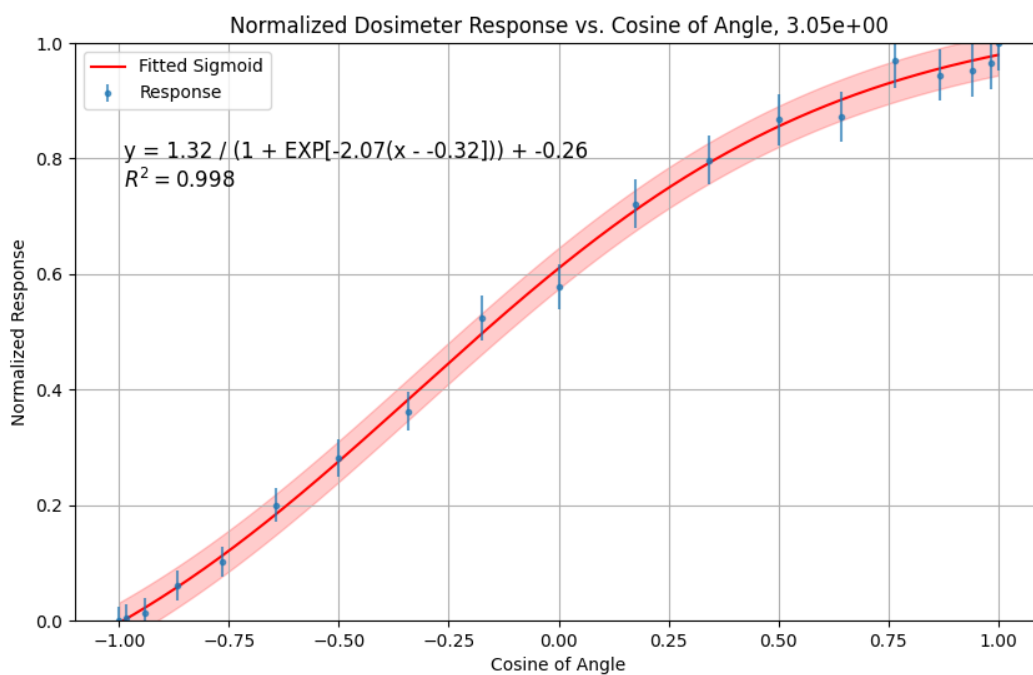
Appendix B. Figure B.45. Normalized Angular Dependent Response at 1.80 MeV



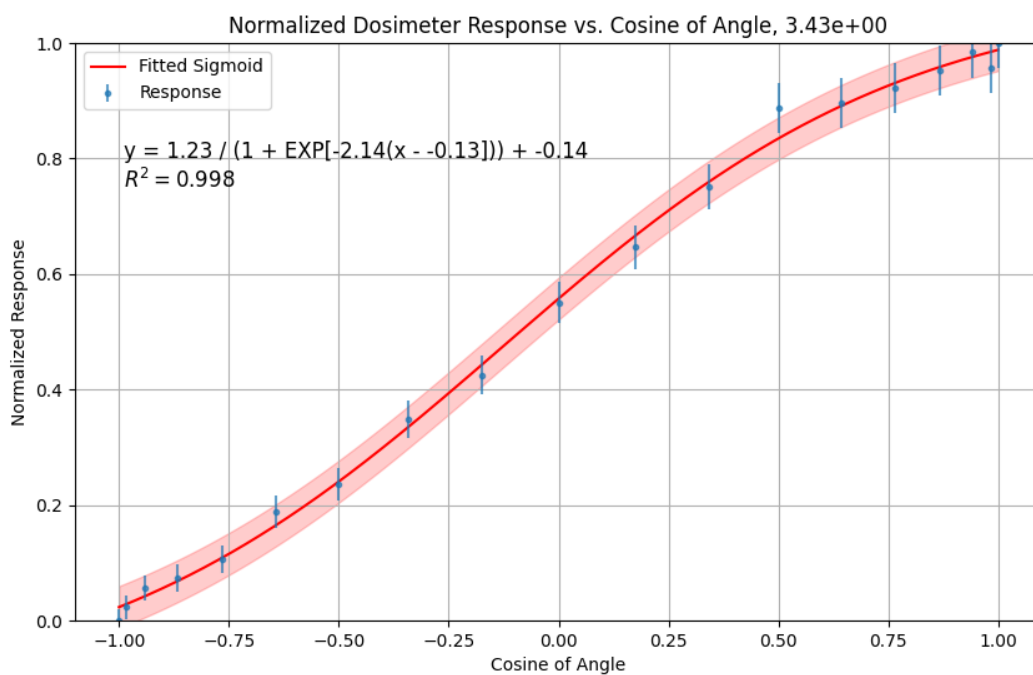
Appendix B. Figure B.46. Normalized Angular Dependent Response at 2.10 MeV



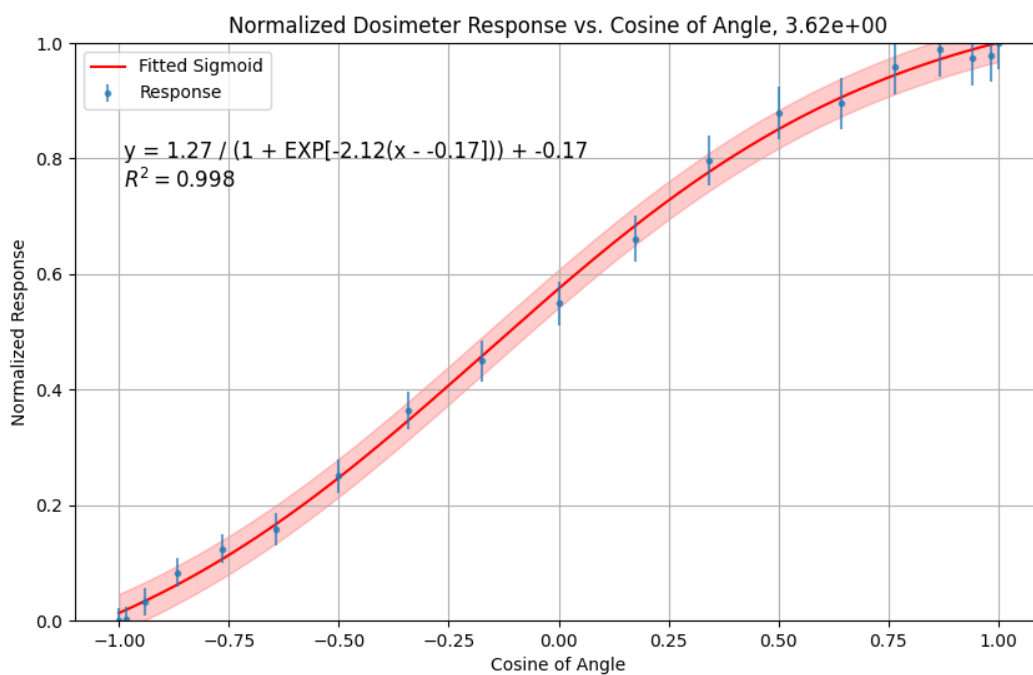
Appendix B. Figure B.47. Normalized Angular Dependent Response at 2.45 MeV



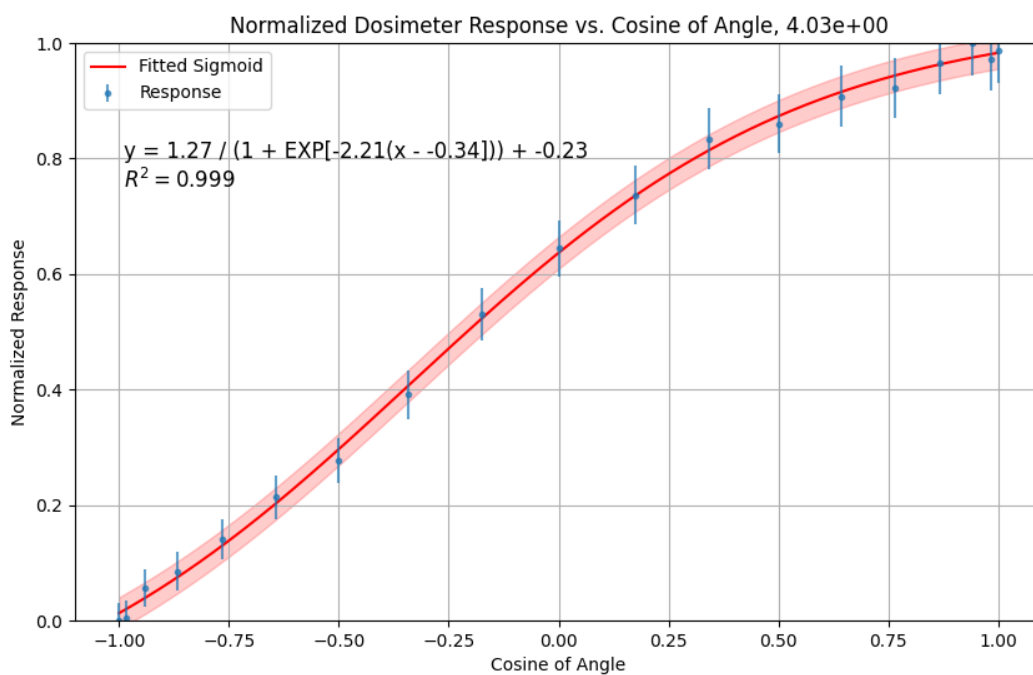
Appendix B. Figure B.48. Normalized Angular Dependent Response at 3.05 MeV



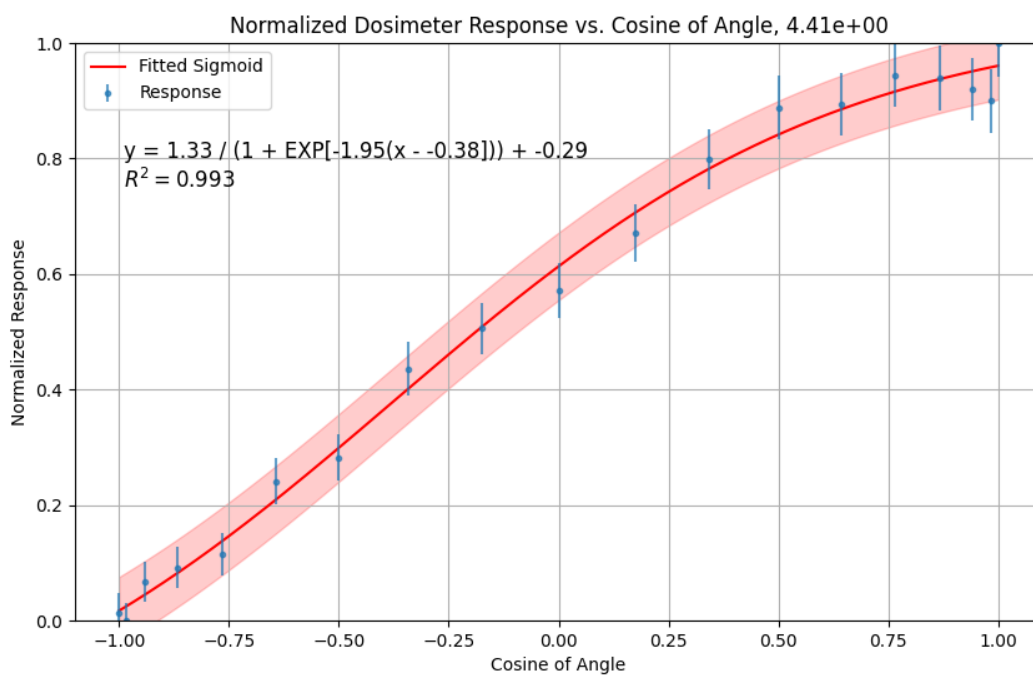
Appendix B. Figure B.49. Normalized Angular Dependent Response at 3.43 MeV



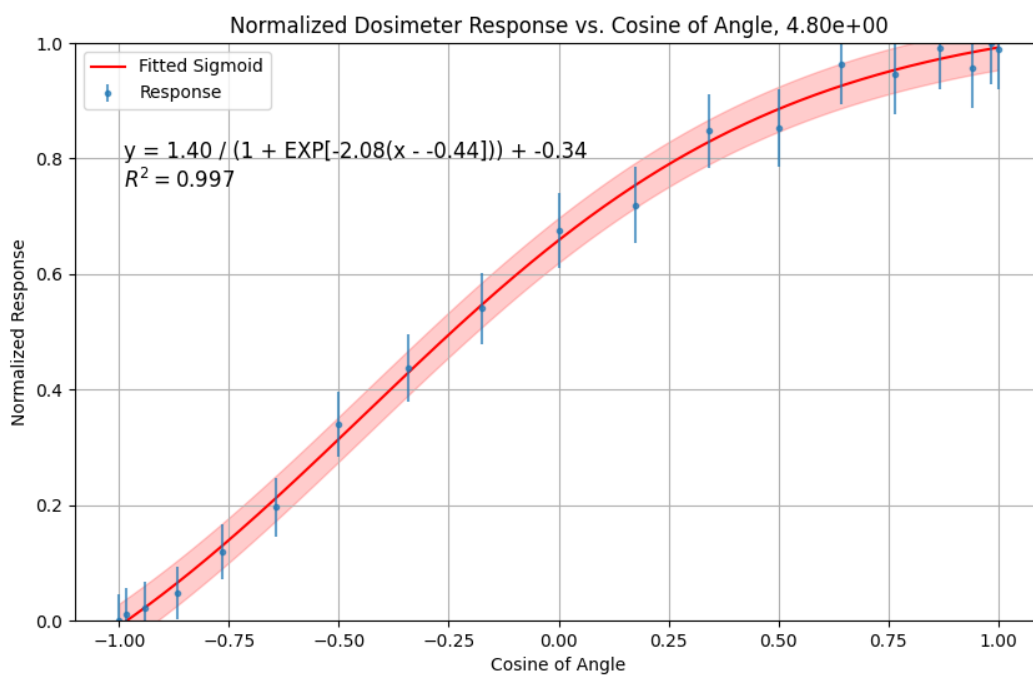
Appendix B. Figure B.50. Normalized Angular Dependent Response at 3.62 MeV



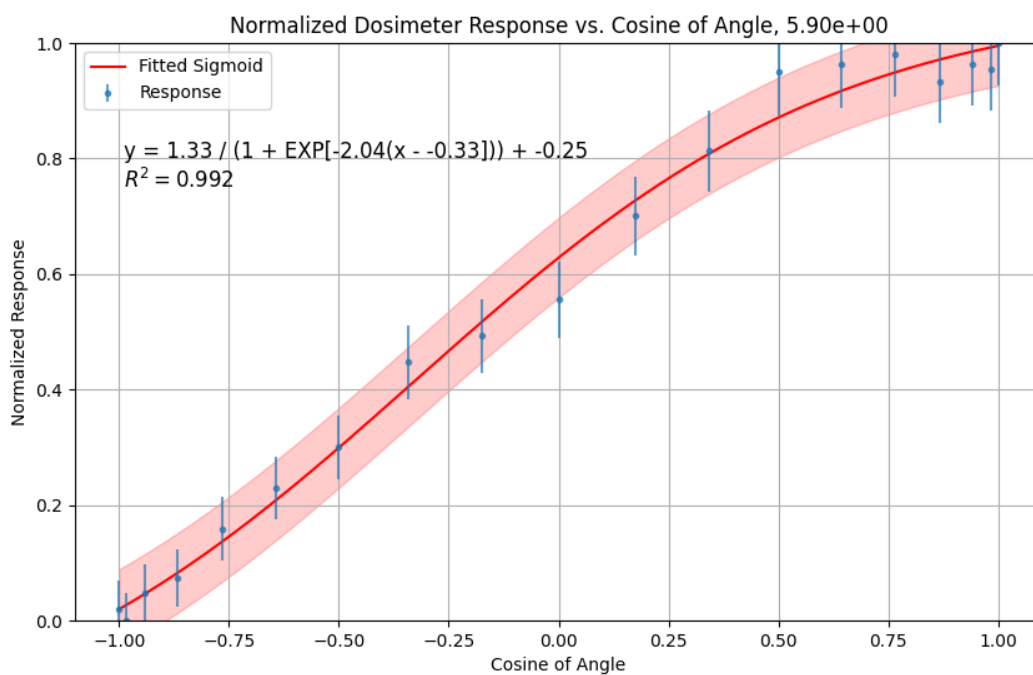
Appendix B. Figure B.51. Normalized Angular Dependent Response at 4.03 MeV



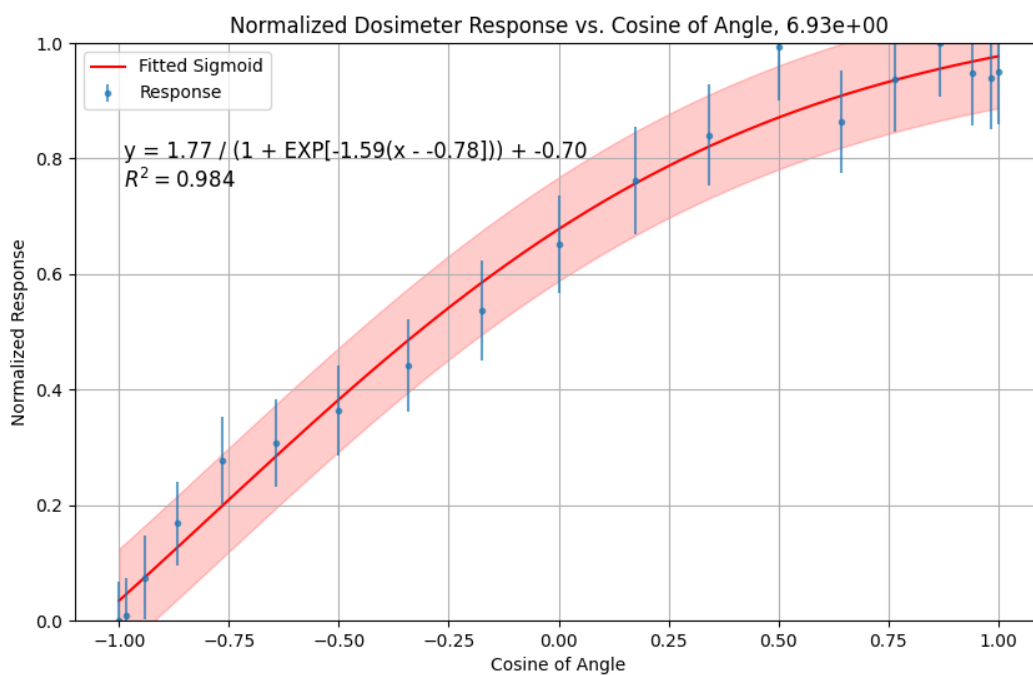
Appendix B. Figure B.52. Normalized Angular Dependent Response at 4.41 MeV



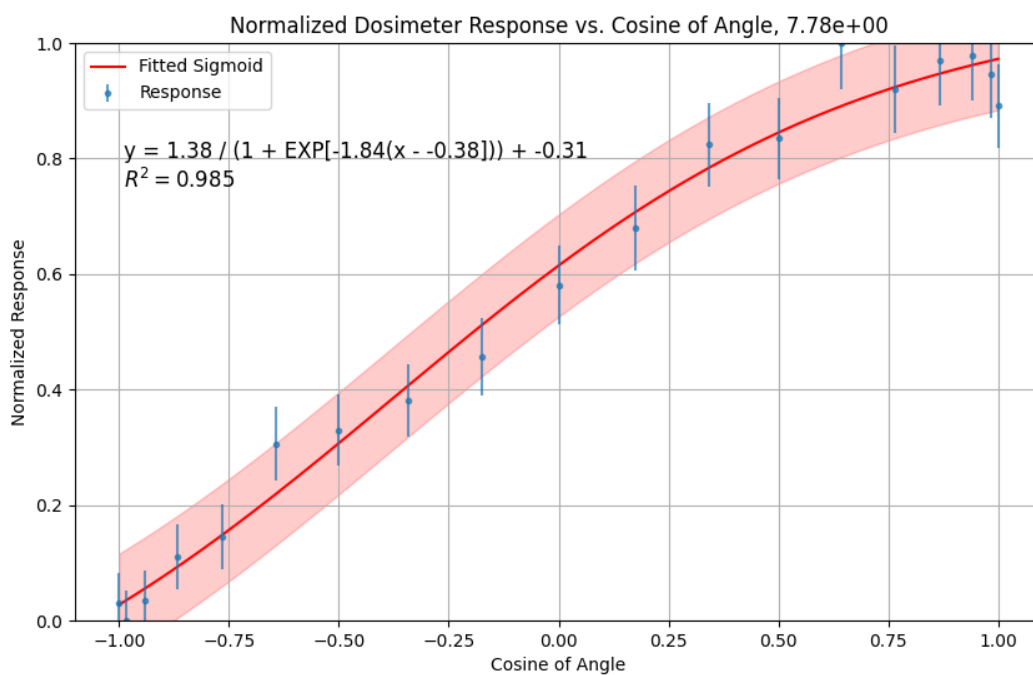
Appendix B. Figure B.53. Normalized Angular Dependent Response at 4.80 MeV



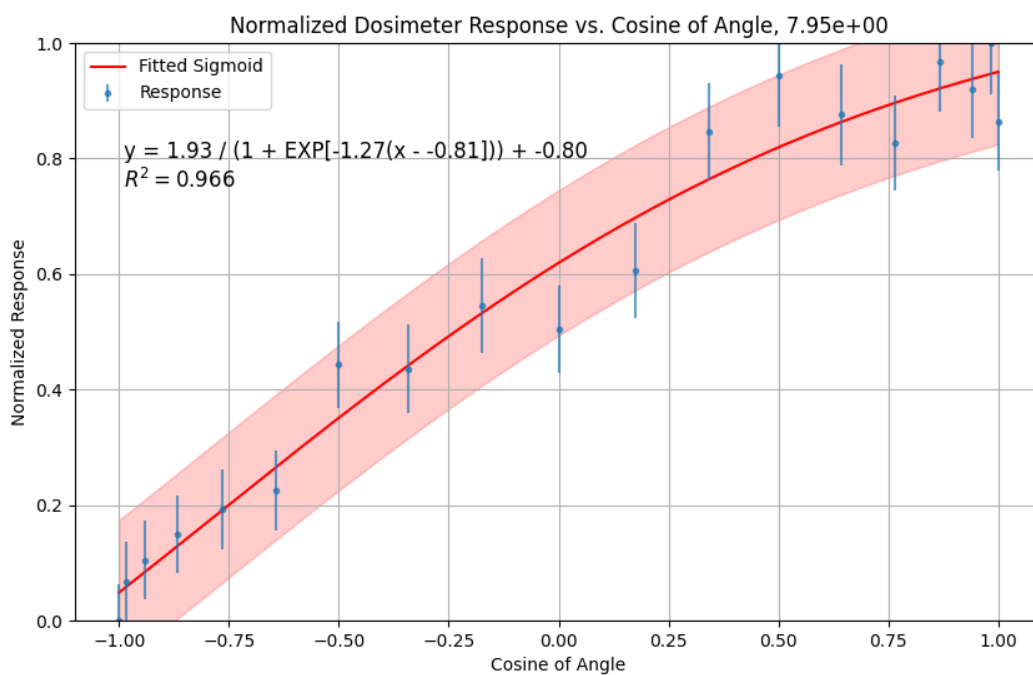
Appendix B. Figure B.54. Normalized Angular Dependent Response at 5.90 MeV



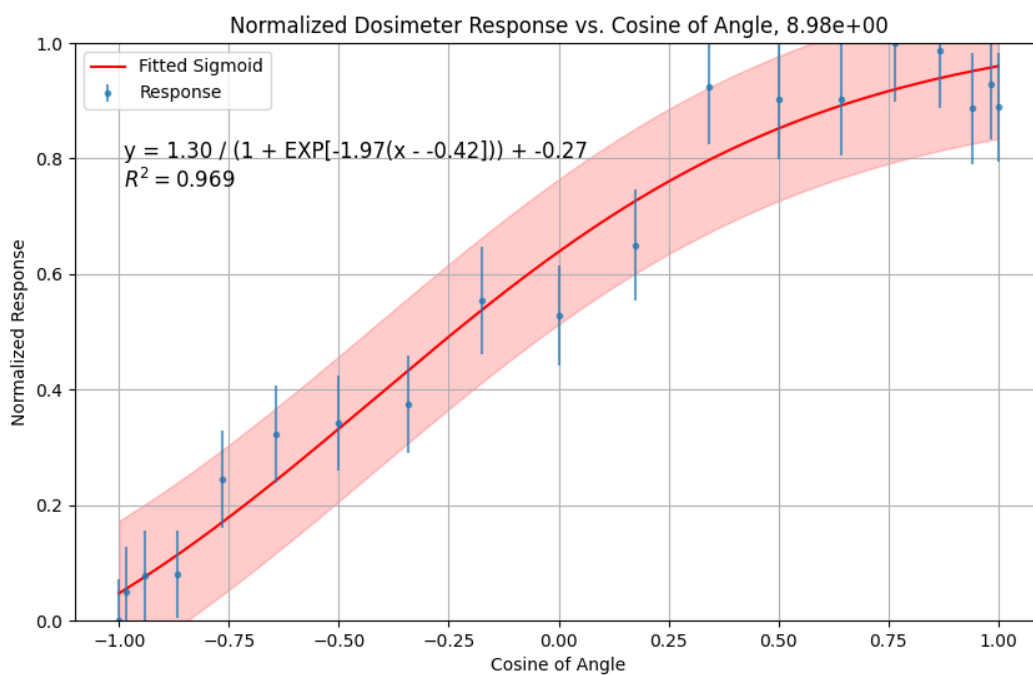
Appendix B. Figure B.55. Normalized Angular Dependent Response at 6.93 MeV



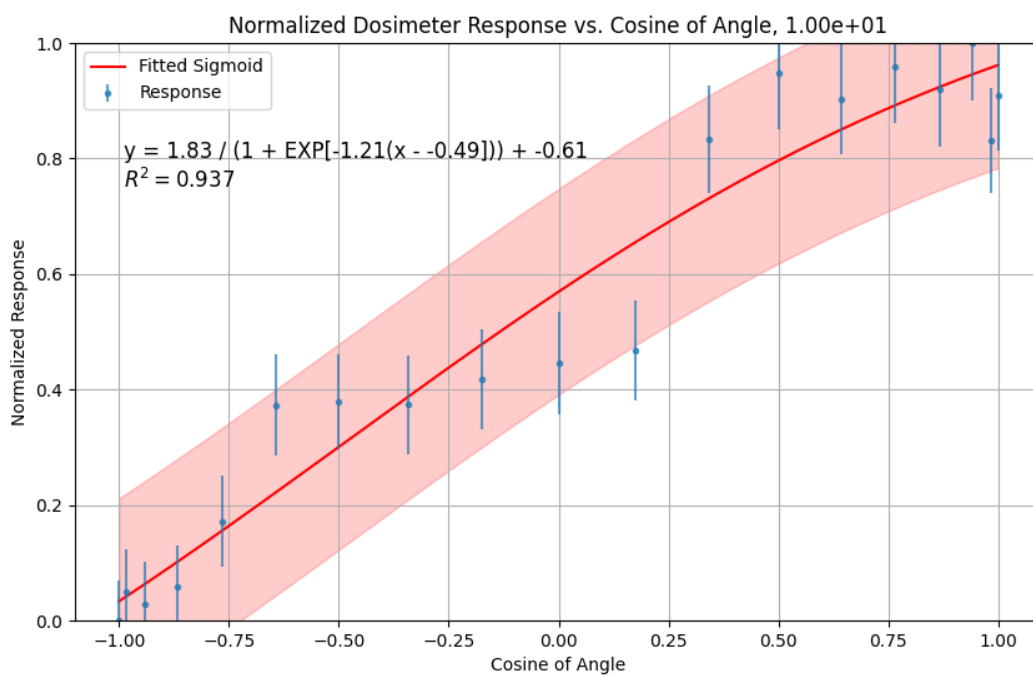
Appendix B. Figure B.56. Normalized Angular Dependent Response at 7.78 MeV



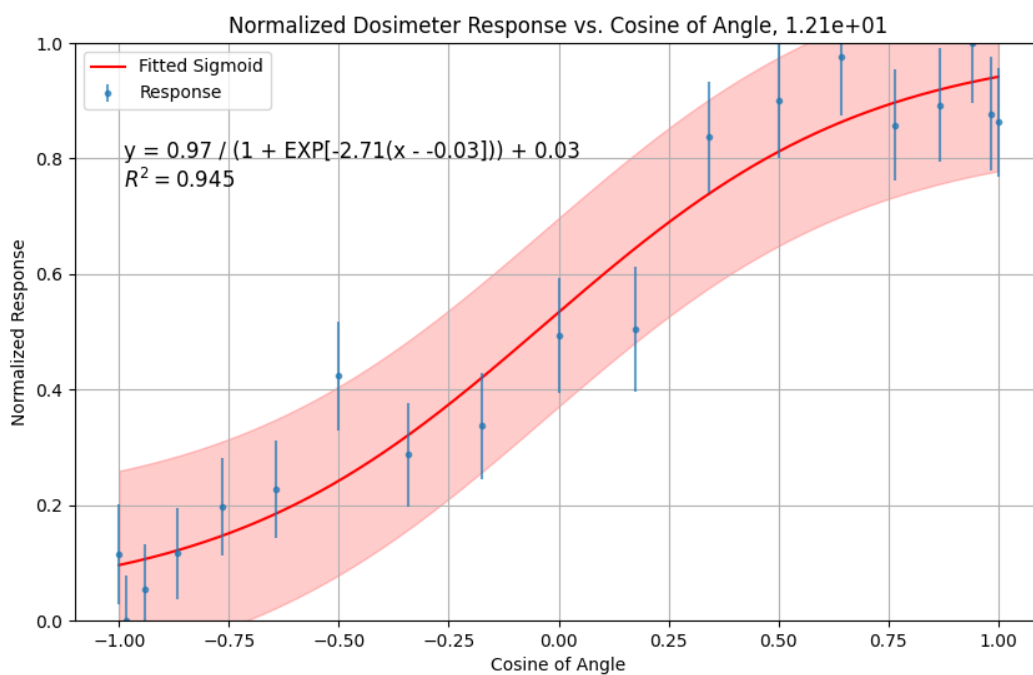
Appendix B. Figure B.57. Normalized Angular Dependent Response at 7.95 MeV



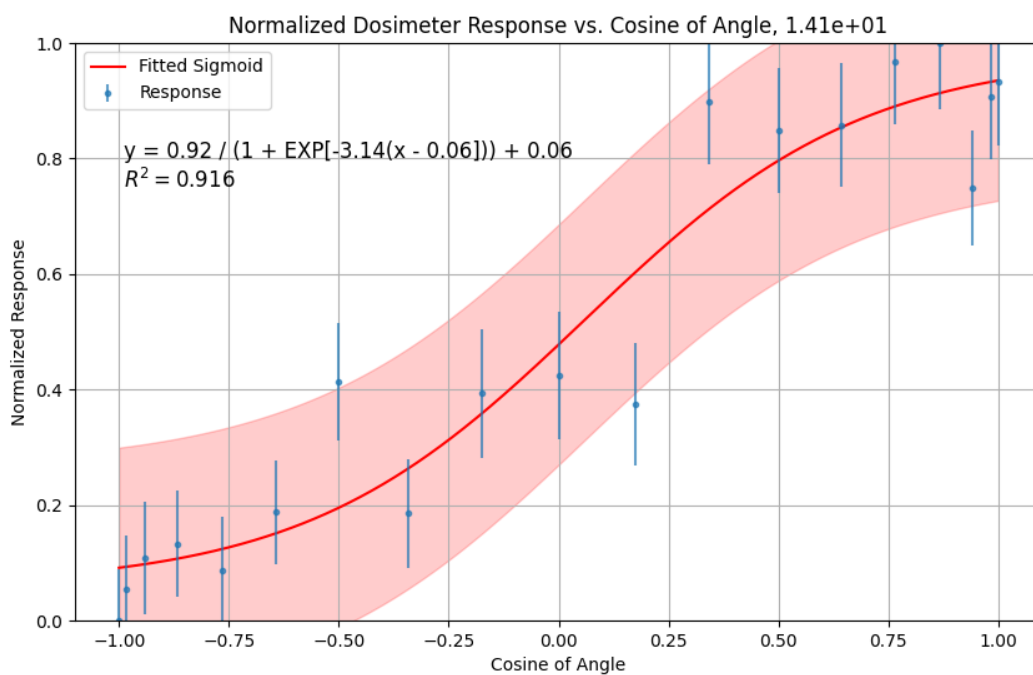
Appendix B. Figure B.58. Normalized Angular Dependent Response at 8.98 MeV



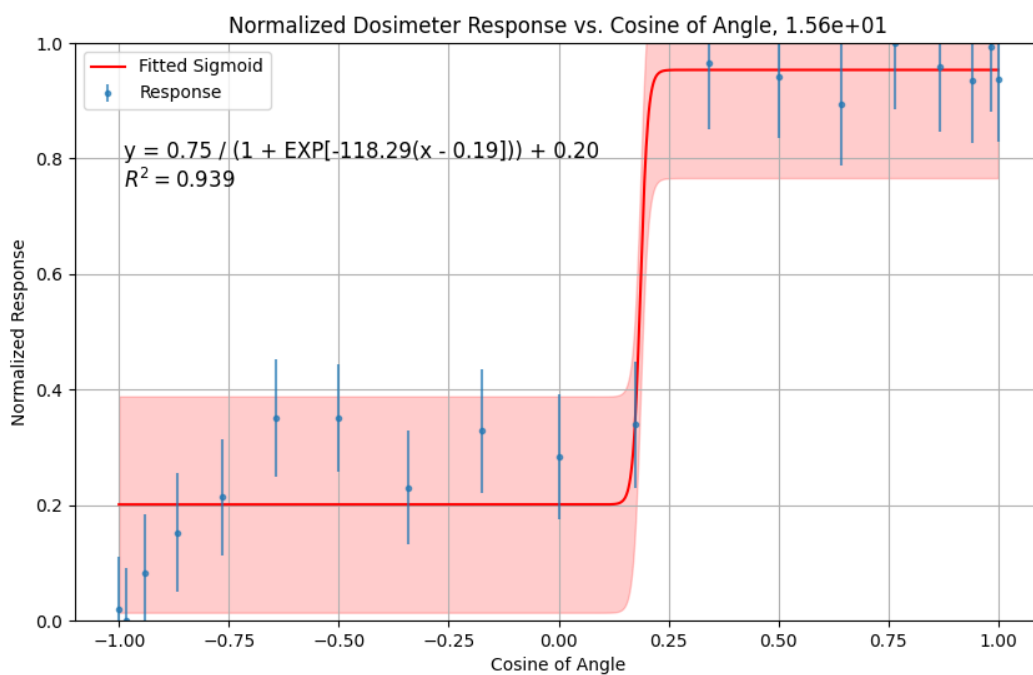
Appendix B. Figure B.59. Normalized Angular Dependent Response at 10.0 MeV



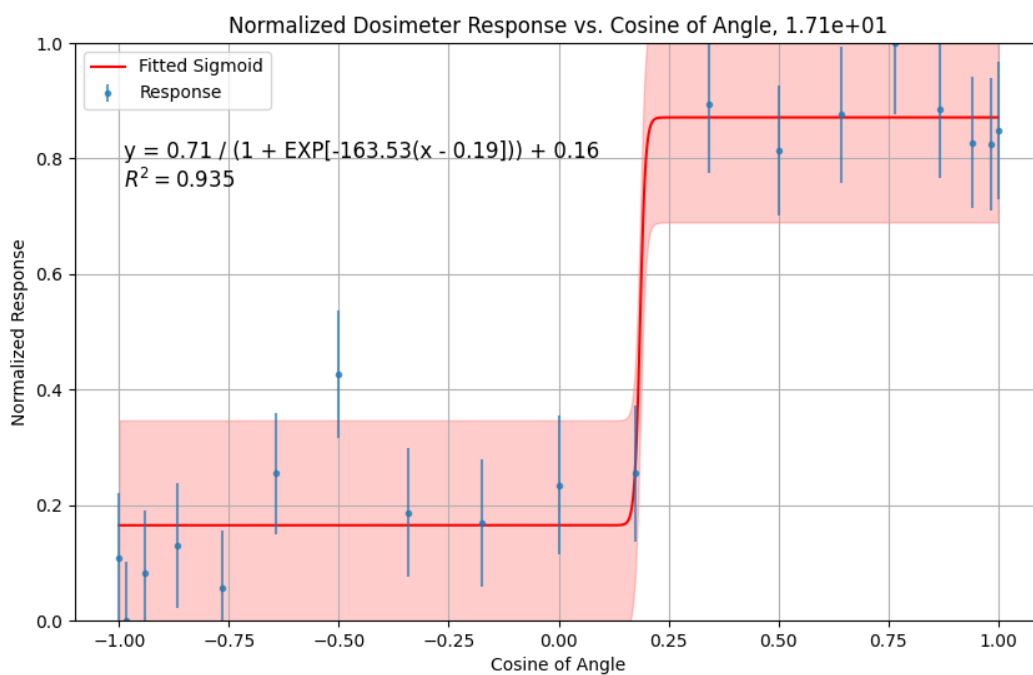
Appendix B. Figure B.60. Normalized Angular Dependent Response at 12.1 MeV



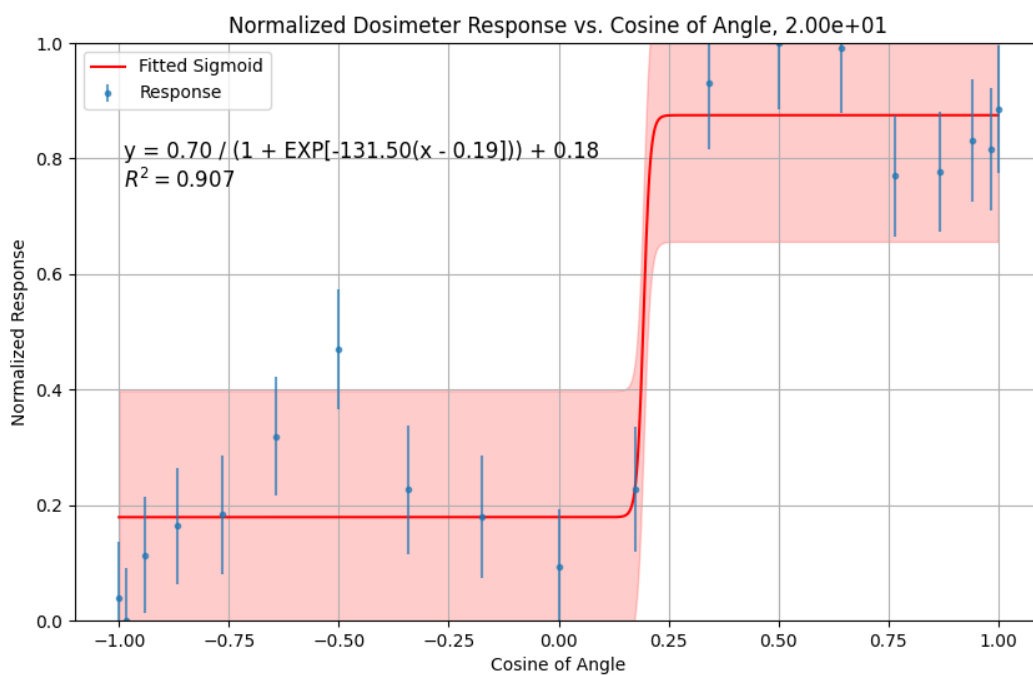
Appendix B. Figure B.61. Normalized Angular Dependent Response at 14.1 MeV



Appendix B. Figure B.62. Normalized Angular Dependent Response at 15.6 MeV



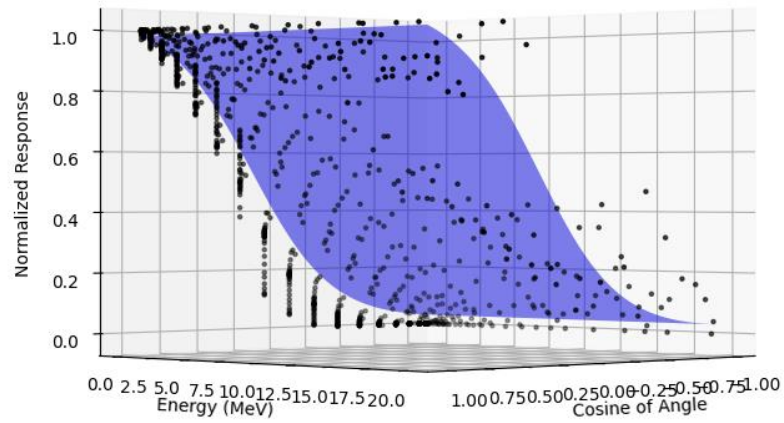
Appendix B. Figure B.63. Normalized Angular Dependent Response at 17.1 MeV



Appendix B. Figure B.64. Normalized Angular Dependent Response at 20.0 MeV

Appendix C - Multivariable Comparison and Regression

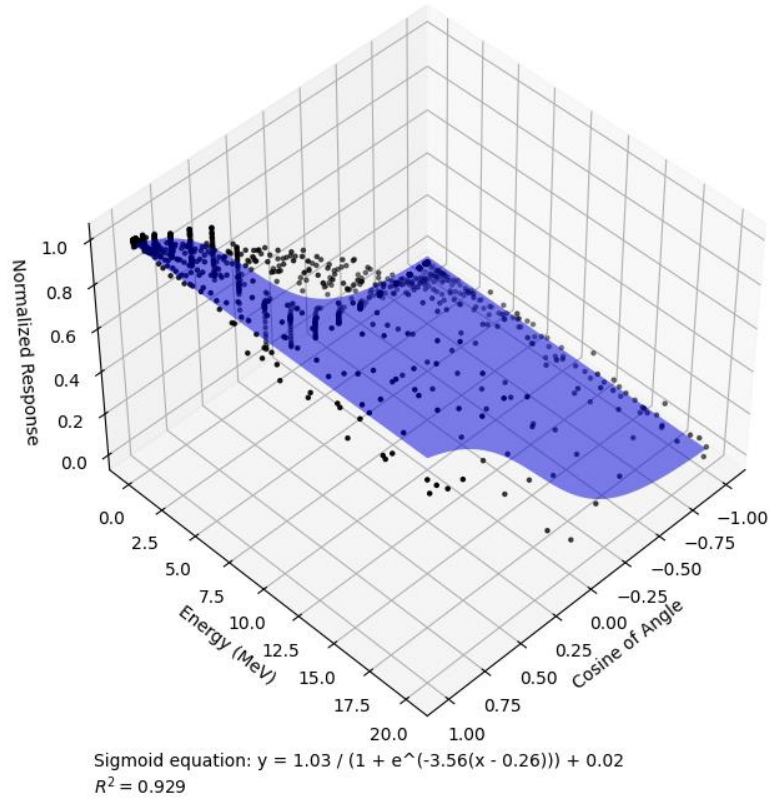
Sigmoid 3D Regression View: Elevation 0, Azimuth 45



Sigmoid equation: $y = 1.03 / (1 + e^{(-3.56(x - 0.26))}) + 0.02$
 $R^2 = 0.929$

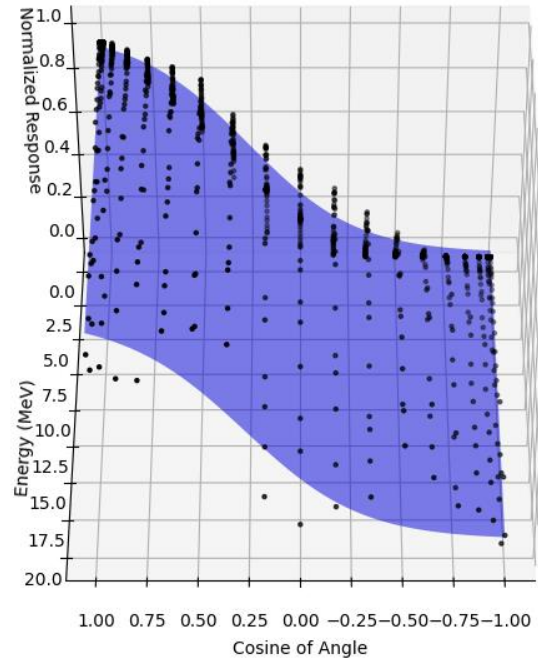
Appendix C. Figure C.1. Multiple Regression, Energy v Angle v Response (0, 45)

Sigmoid 3D Regression View: Elevation 45, Azimuth 45



Appendix C. Figure C.2. Multiple Regression, Energy v Angle v Response (45, 45)

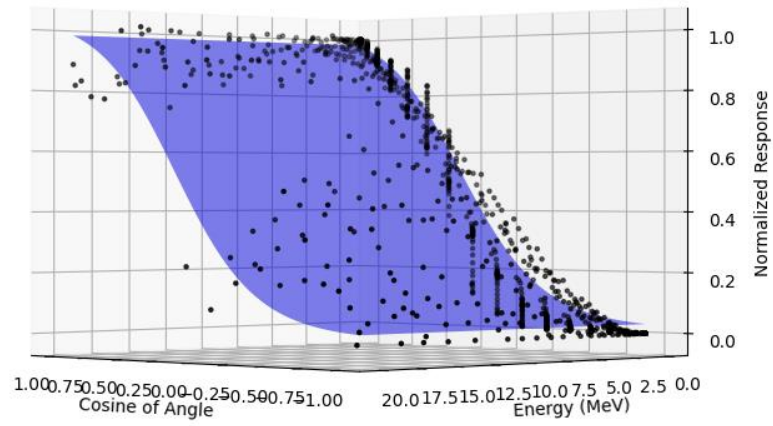
Sigmoid 3D Regression View: Elevation 45, Azimuth 90



Sigmoid equation: $y = 1.03 / (1 + e^{(-3.56(x - 0.26))}) + 0.02$
 $R^2 = 0.929$

Appendix C. Figure C.3. Multiple Regression, Energy v Angle v Response (45, 90)

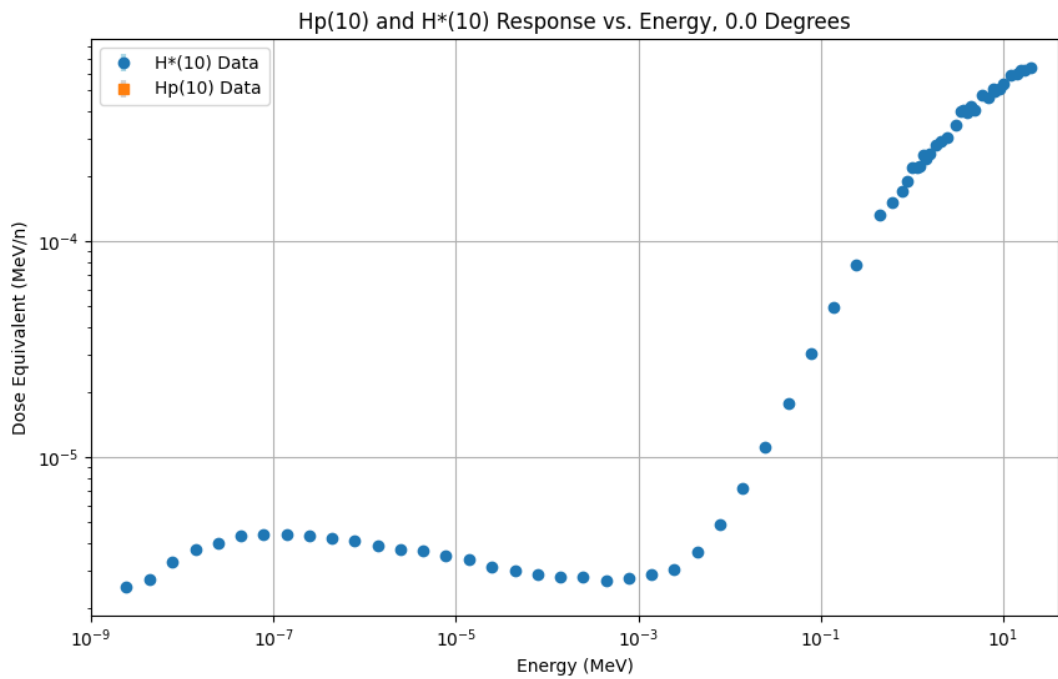
Sigmoid 3D Regression View: Elevation 0, Azimuth 135



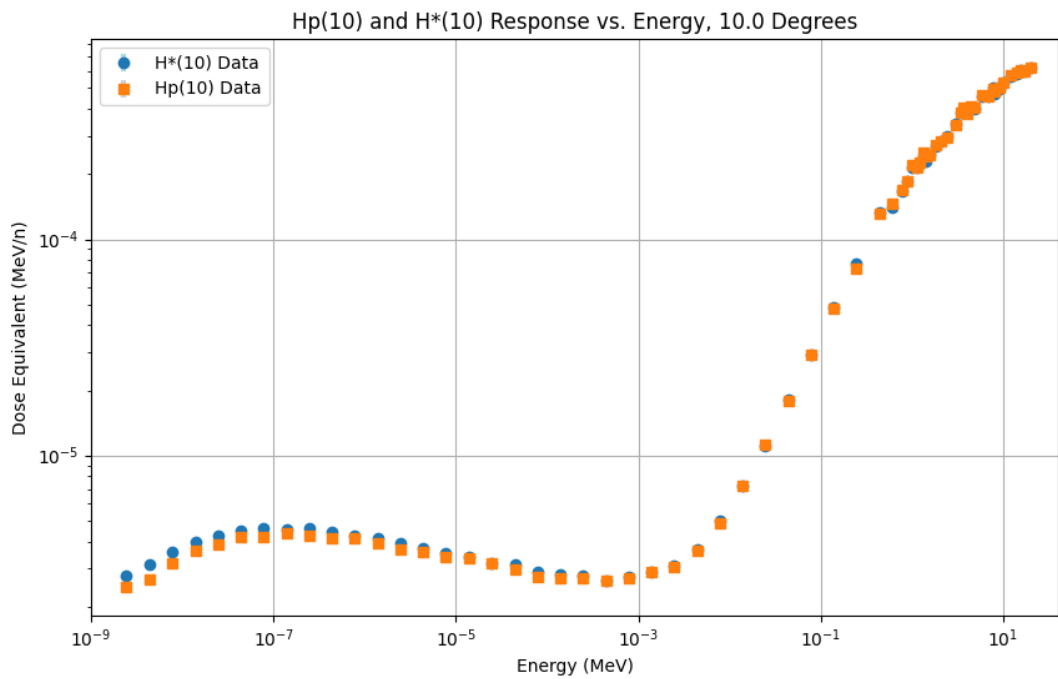
Sigmoid equation: $y = 1.03 / (1 + e^{(-3.56(x - 0.26))}) + 0.02$
 $R^2 = 0.929$

Appendix C. Figure C.4. Multiple Regression, Energy v Angle v Response (0, 135)

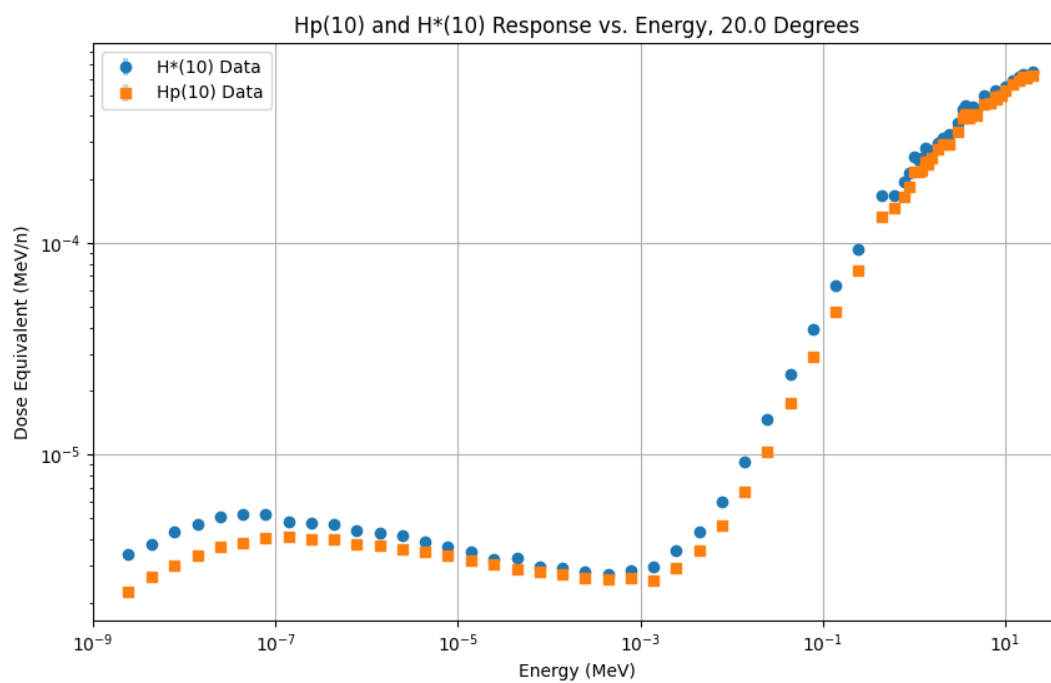
Appendix D - Comparison of PDE and ADE



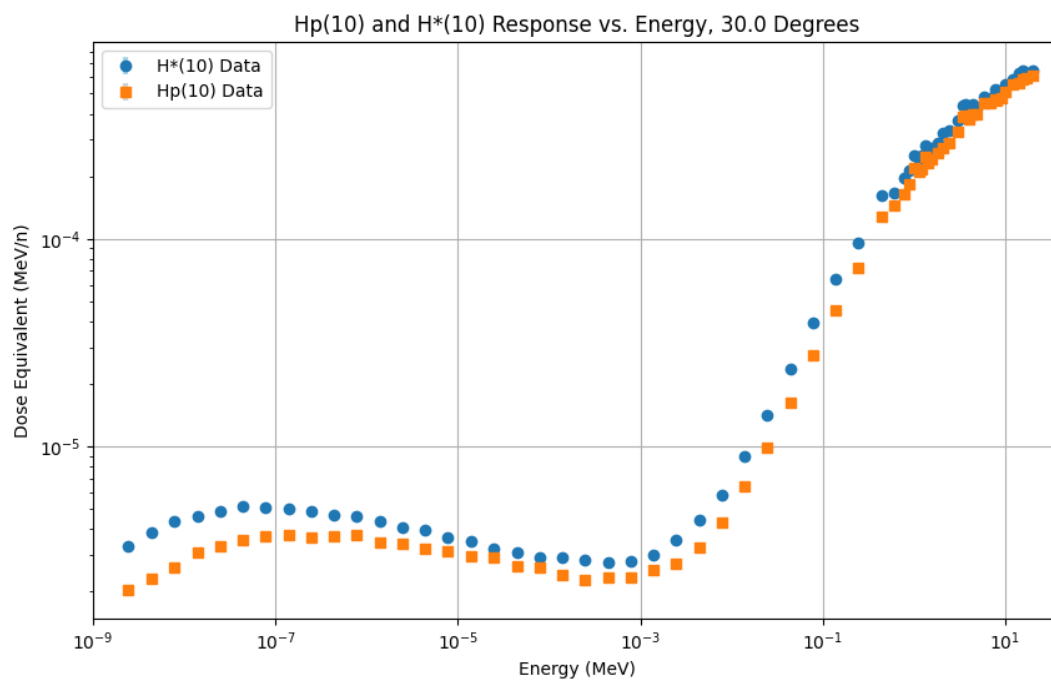
Appendix D. Figure D.1. Comparison of Personal and Ambient Dose Equivalents at 0 Degrees



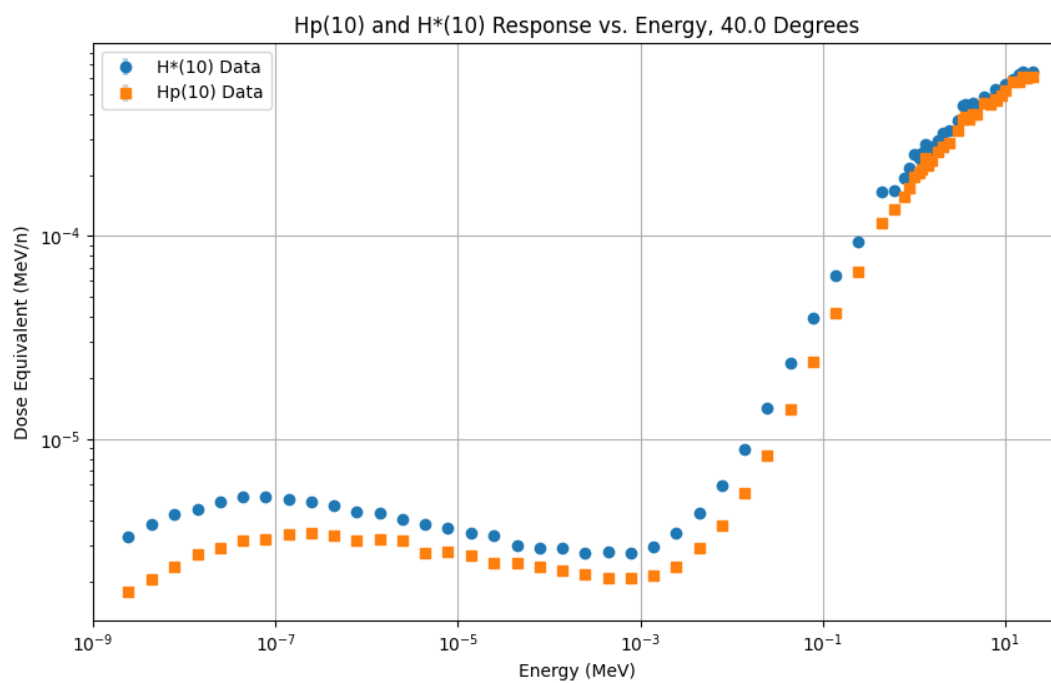
Appendix D. Figure D.2. Comparison of Personal and Ambient Dose Equivalents at 10 Degrees



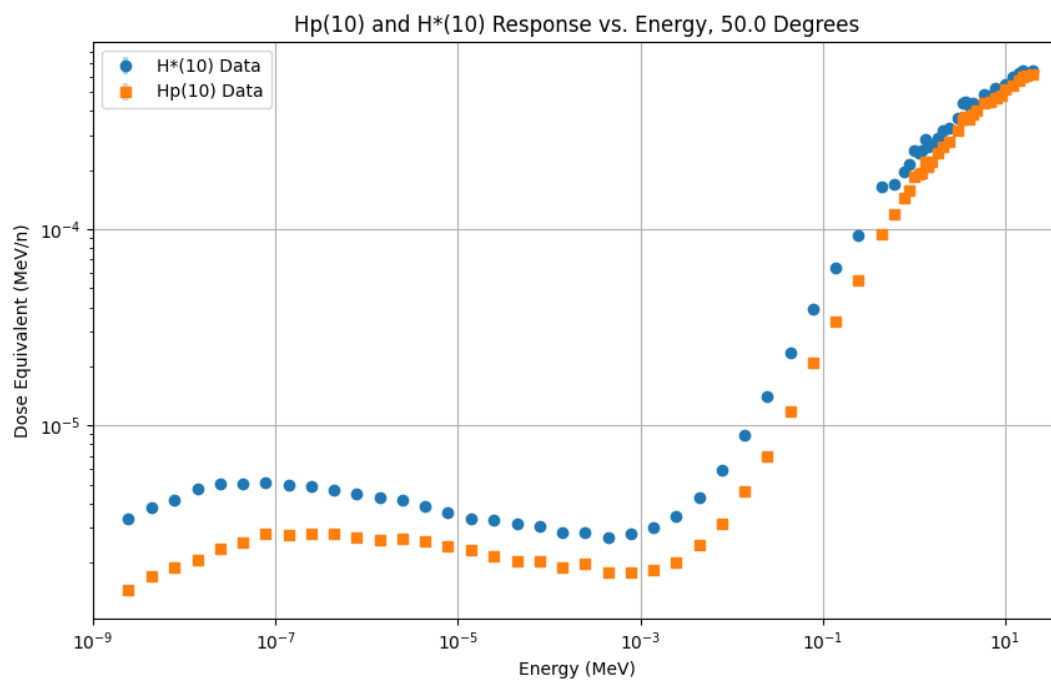
Appendix D. Figure D.3. Comparison of Personal and Ambient Dose Equivalents at 20 Degrees



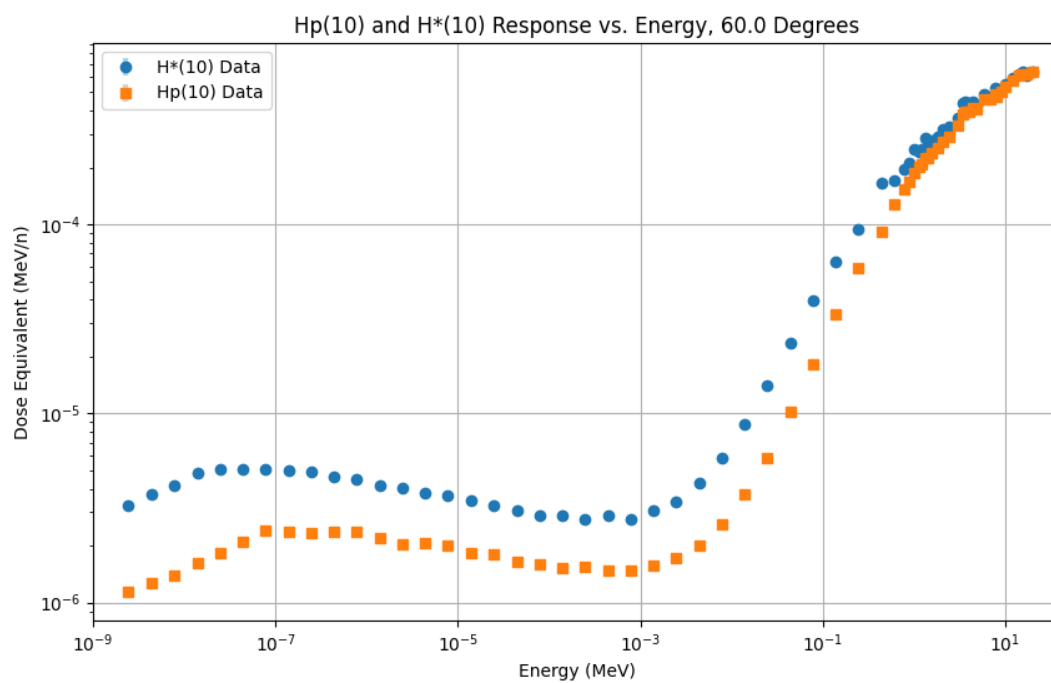
Appendix D. Figure D.4. Comparison of Personal and Ambient Dose Equivalents at 30 Degrees



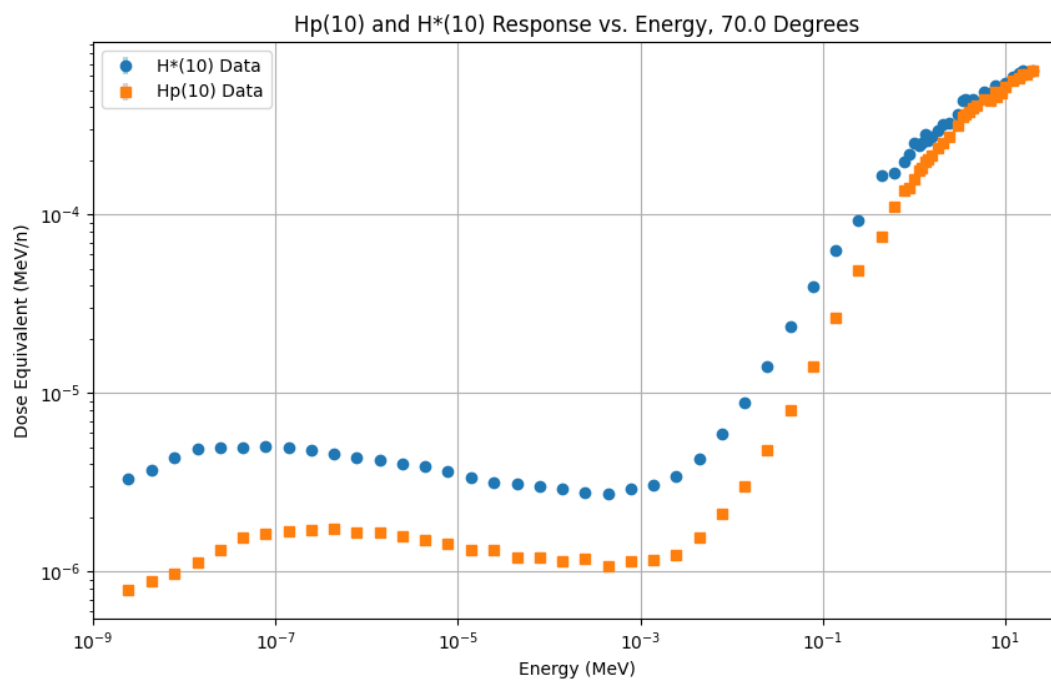
Appendix D. Figure D.5. Comparison of Personal and Ambient Dose Equivalents at 40 Degrees



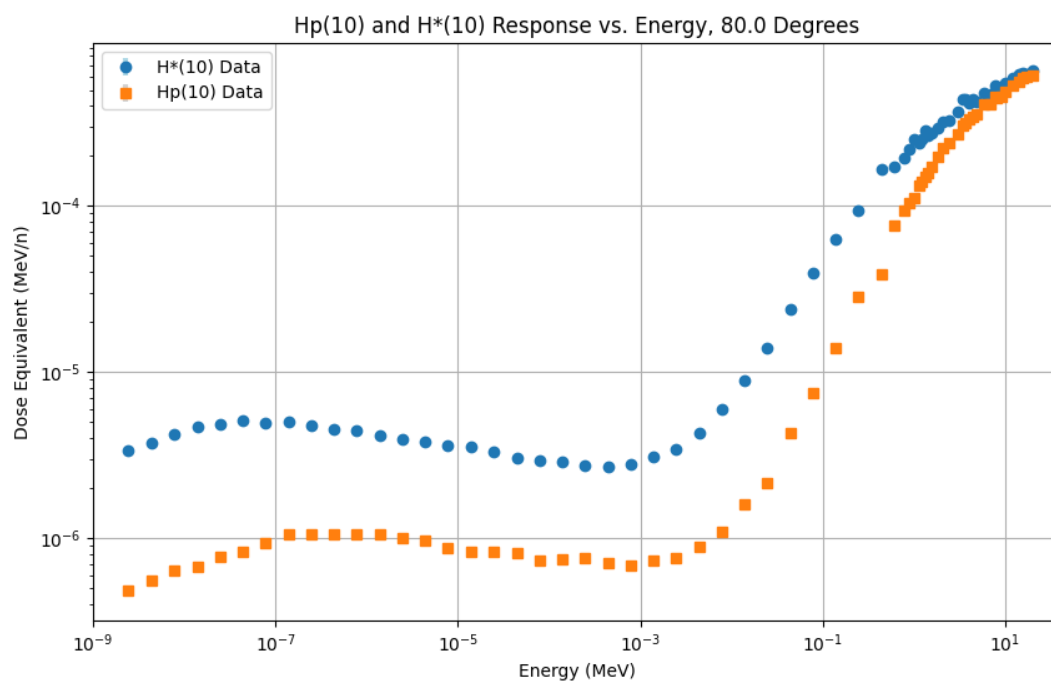
Appendix D. Figure D.6. Comparison of Personal and Ambient Dose Equivalents at 50 Degrees



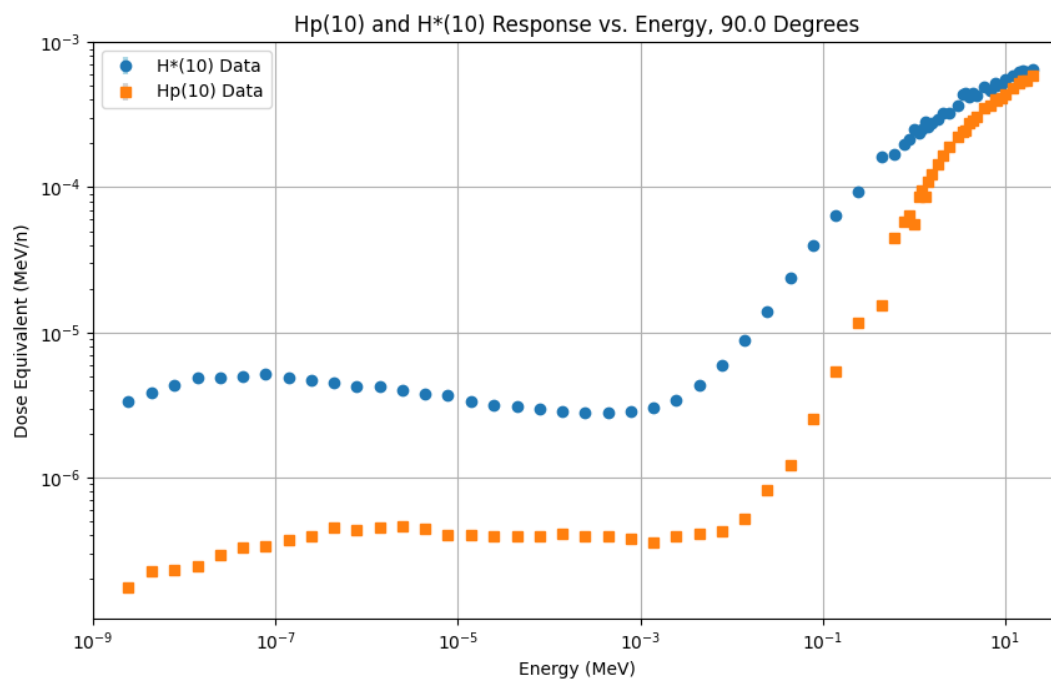
Appendix D. Figure D.7. Comparison of Personal and Ambient Dose Equivalents at 60 Degrees



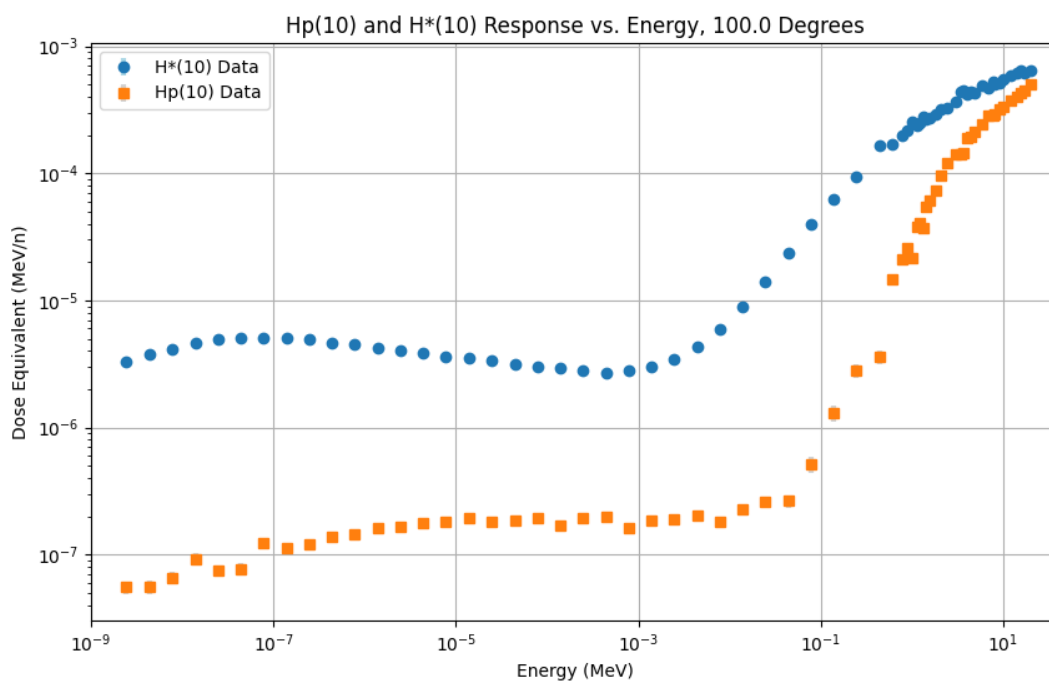
Appendix D. Figure D.8. Comparison of Personal and Ambient Dose Equivalents at 70 Degrees



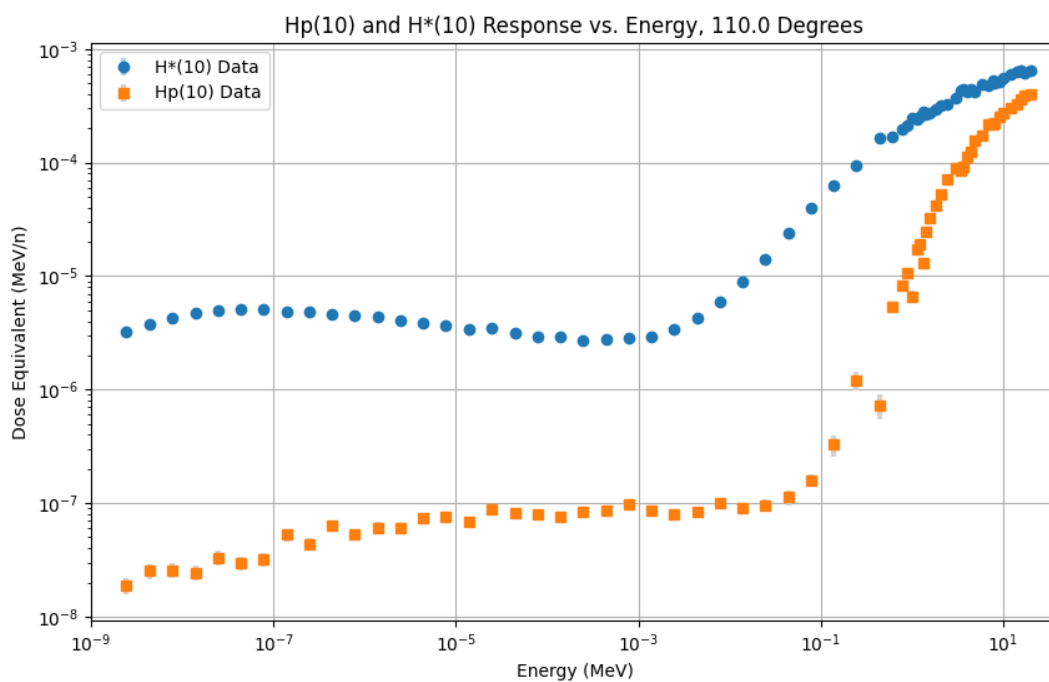
Appendix D. Figure D.9. Comparison of Personal and Ambient Dose Equivalents at 80 Degrees



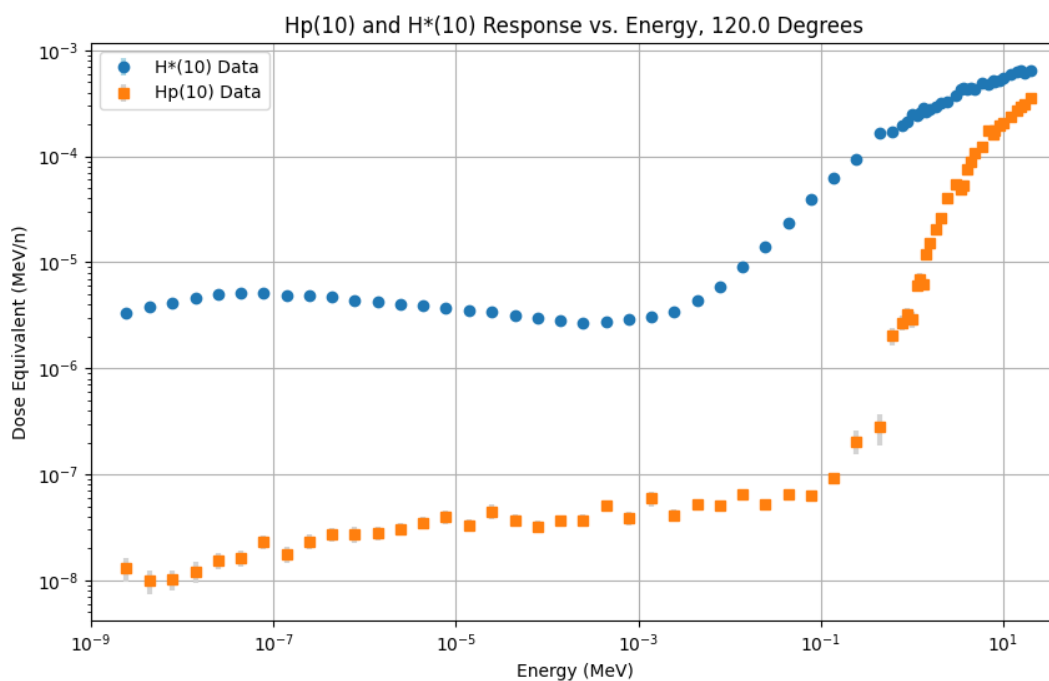
Appendix D. Figure D.10. Comparison of Personal and Ambient Dose Equivalents at 90 Degrees



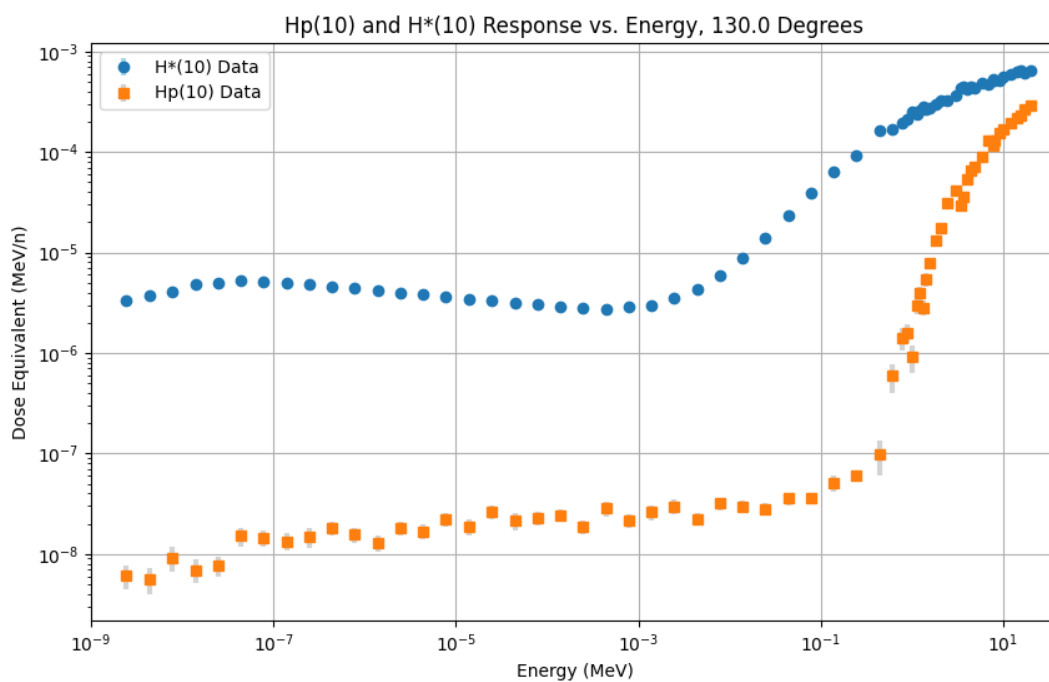
Appendix D. Figure D.11. Comparison of Personal and Ambient Dose Equivalents at 100 Degrees



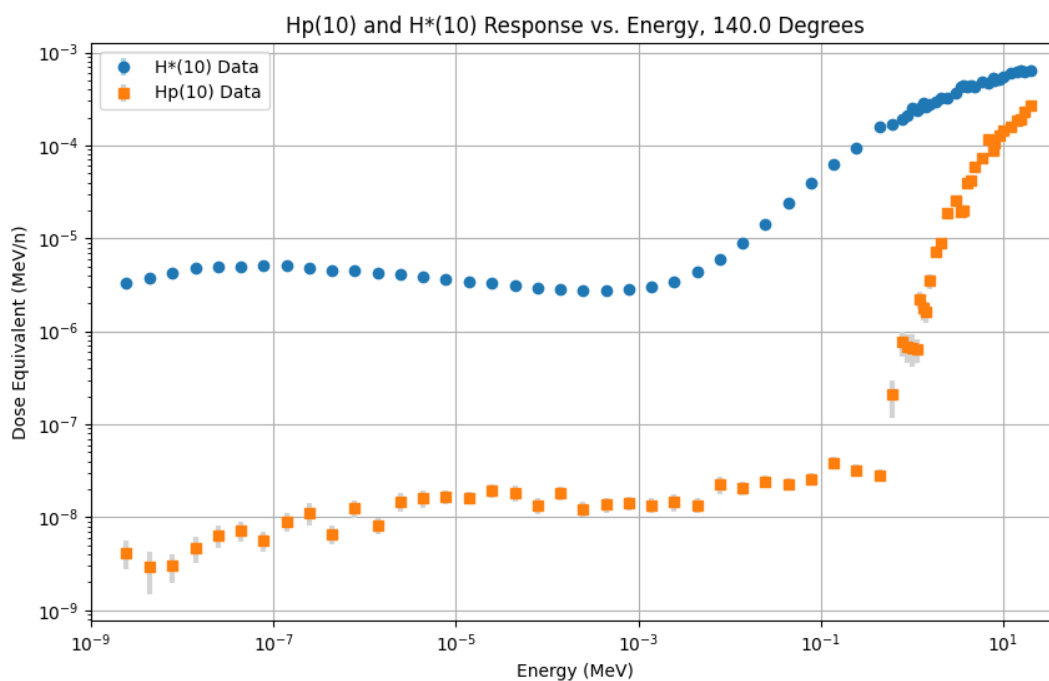
Appendix D. Figure D.12. Comparison of Personal and Ambient Dose Equivalents at 110 Degrees



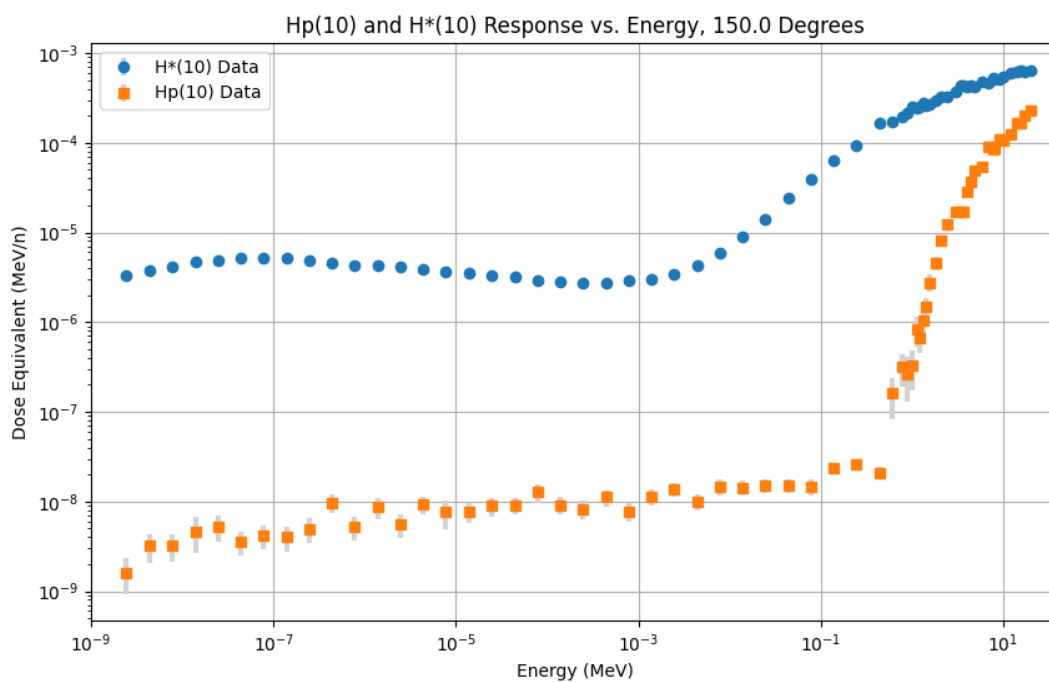
Appendix D. Figure D.13. Comparison of Personal and Ambient Dose Equivalents at 120 Degrees



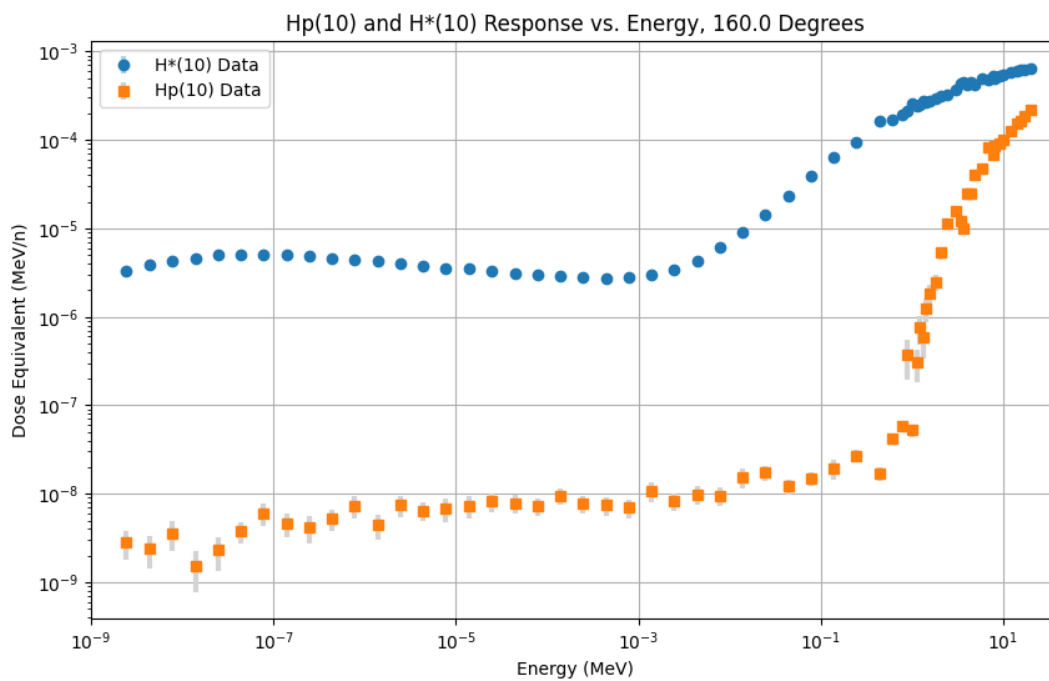
Appendix D. Figure D.14. Comparison of Personal and Ambient Dose Equivalents at 130 Degrees



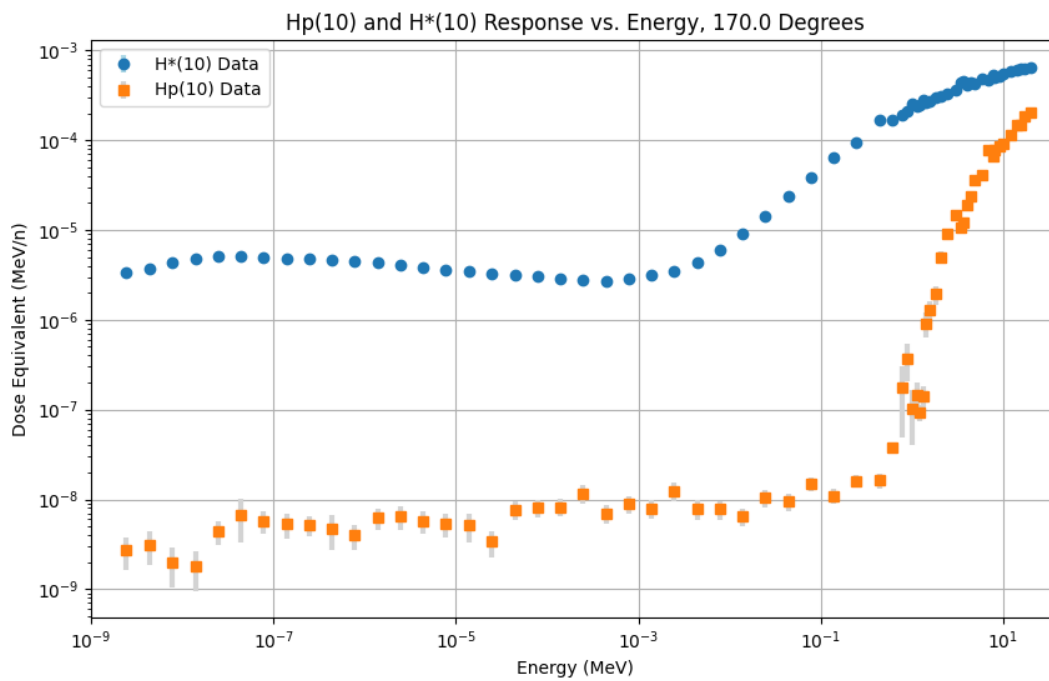
Appendix D. Figure D.15. Comparison of Personal and Ambient Dose Equivalents at 140 Degrees



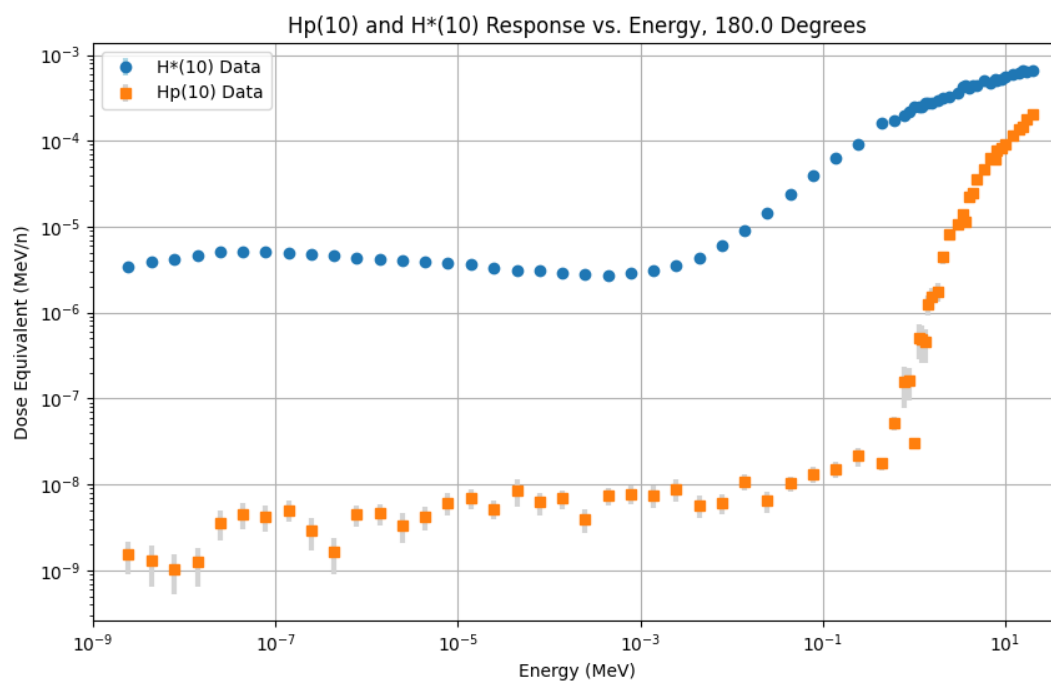
Appendix D. Figure D.16. Comparison of Personal and Ambient Dose Equivalents at 150 Degrees



Appendix D. Figure D.17. Comparison of Personal and Ambient Dose Equivalents at 160 Degrees

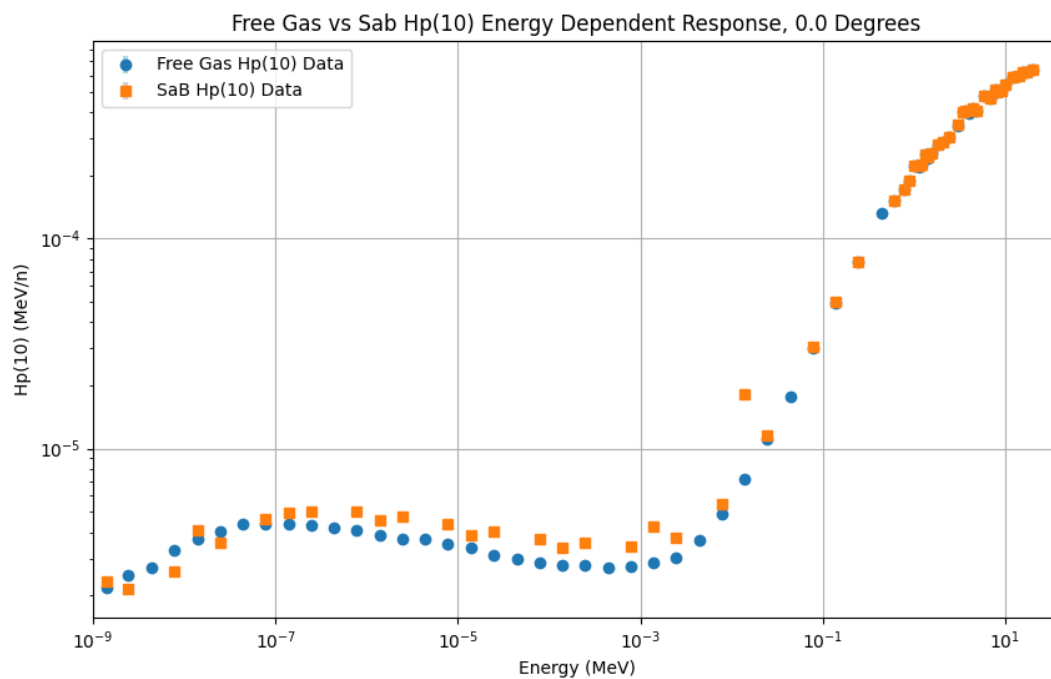


Appendix D. Figure D.18. Comparison of Personal and Ambient Dose Equivalents at 170 Degrees

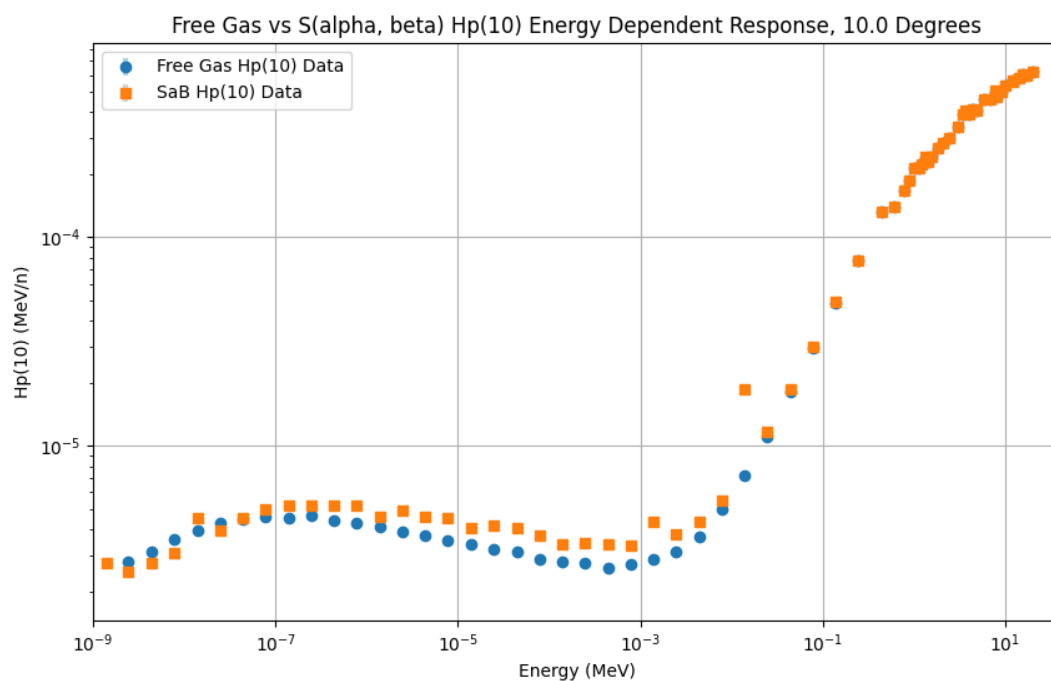


Appendix D. Figure D.19. Comparison of Personal and Ambient Dose Equivalents at 180 Degrees

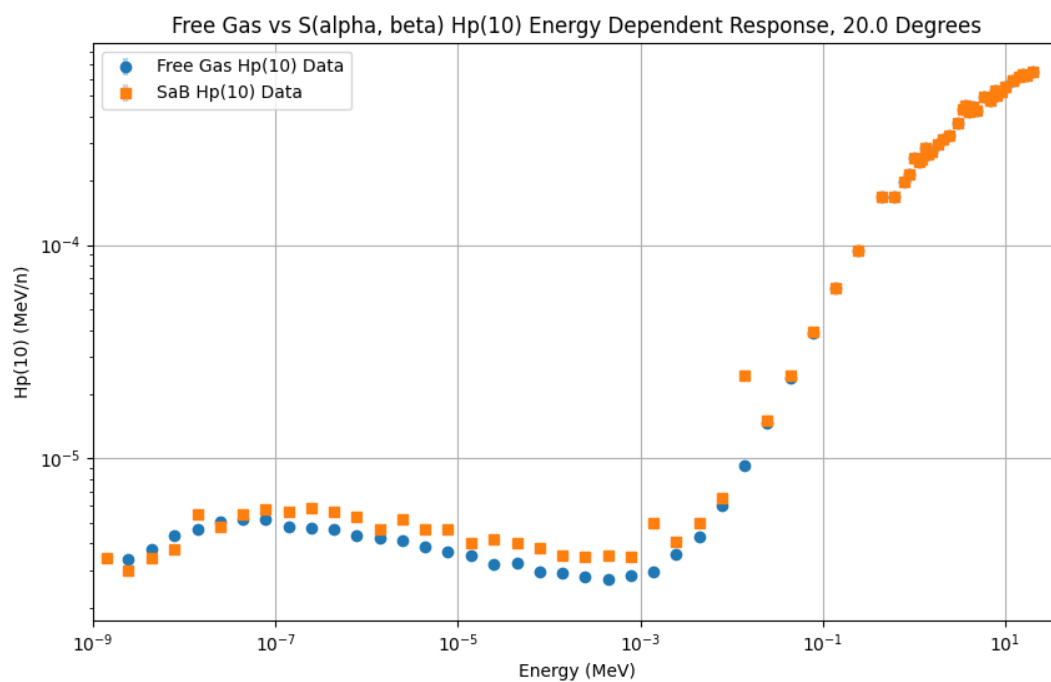
Appendix E - Free-gas vs $S(\alpha, \beta)$ Scattering on $H_p(10)$



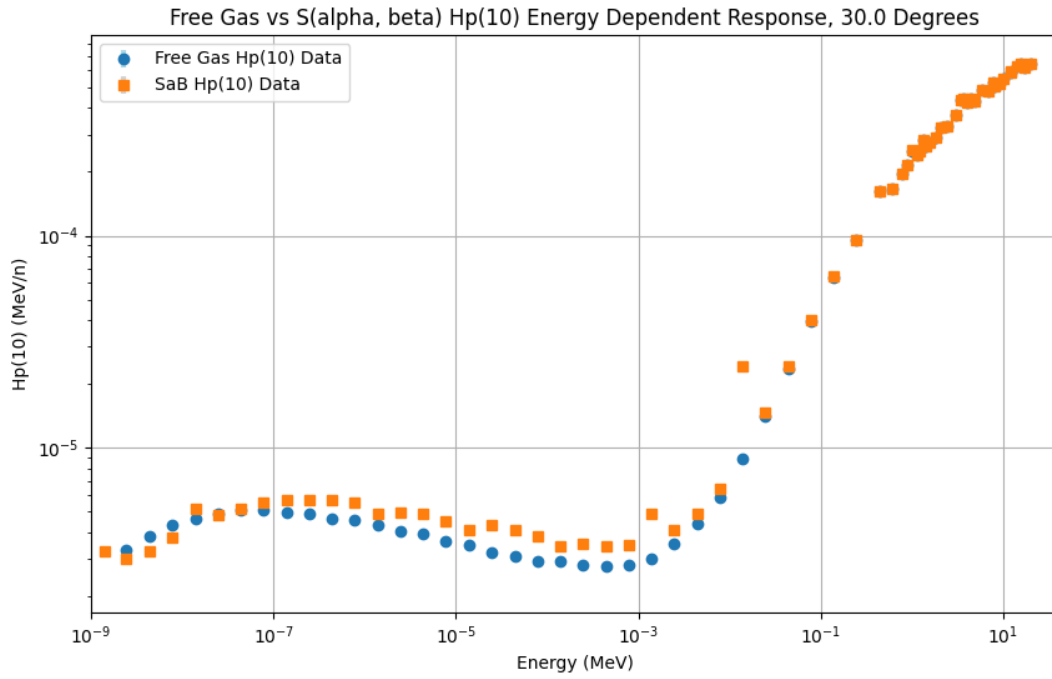
Appendix E. Figure E.1. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 0 Degrees



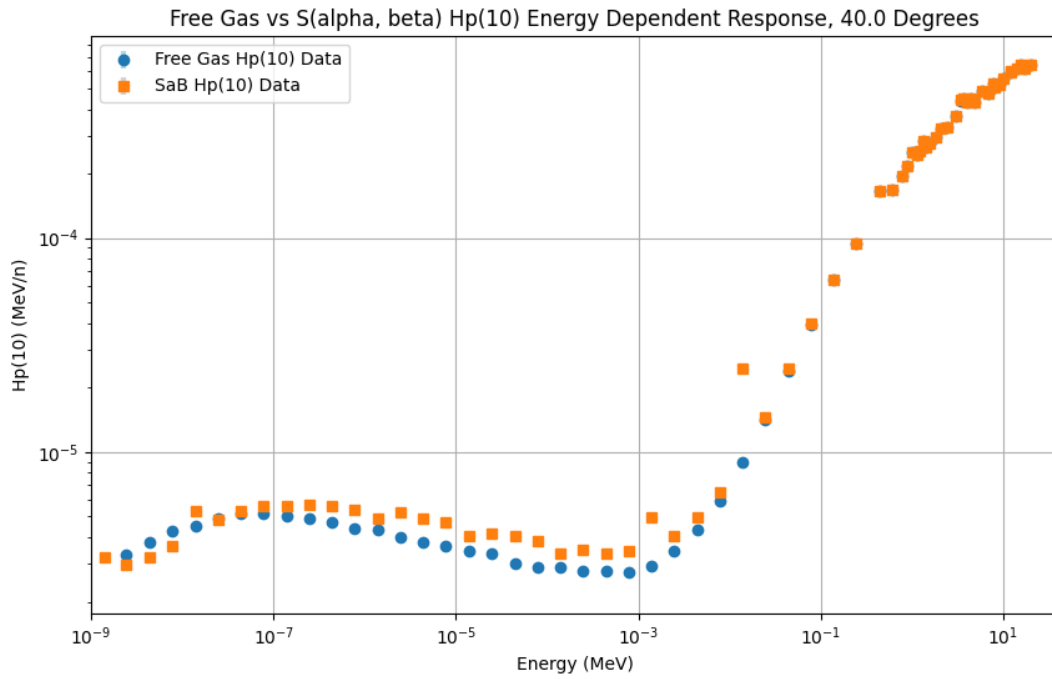
Appendix E. Figure E.2. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 10 Degrees



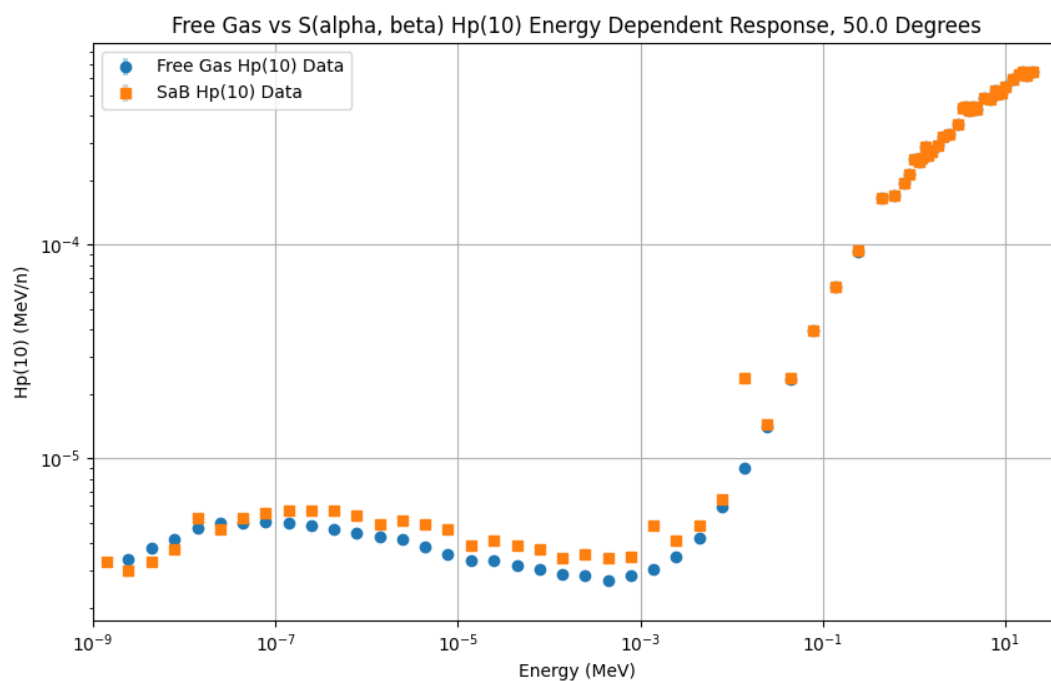
Appendix E. Figure E.3. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 20 Degrees



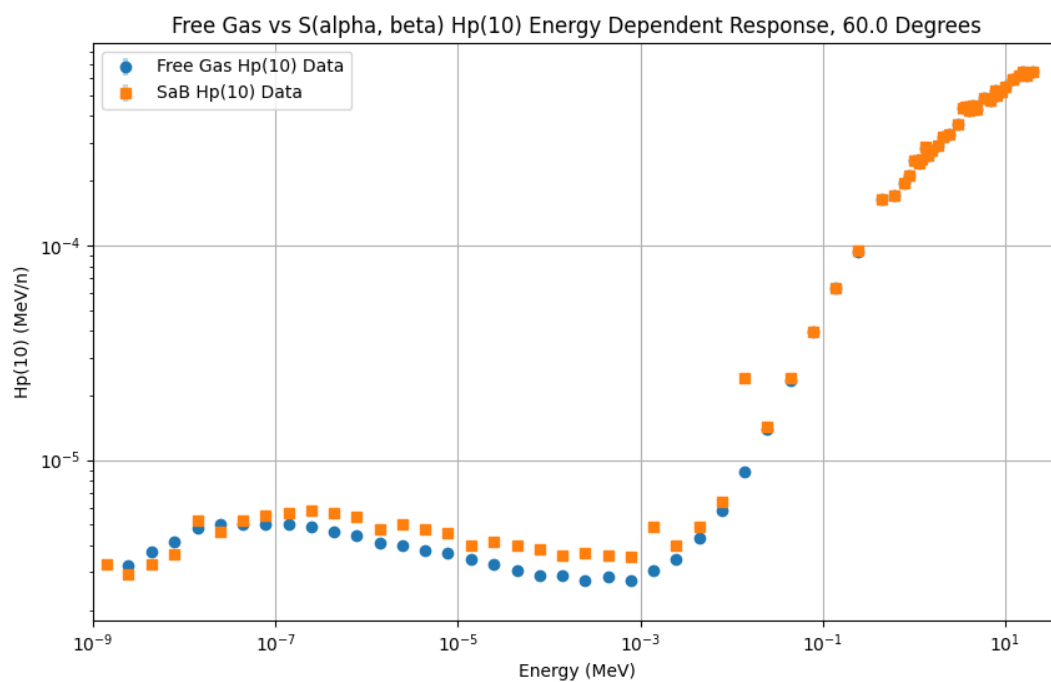
Appendix E. Figure E.4. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 30 Degrees



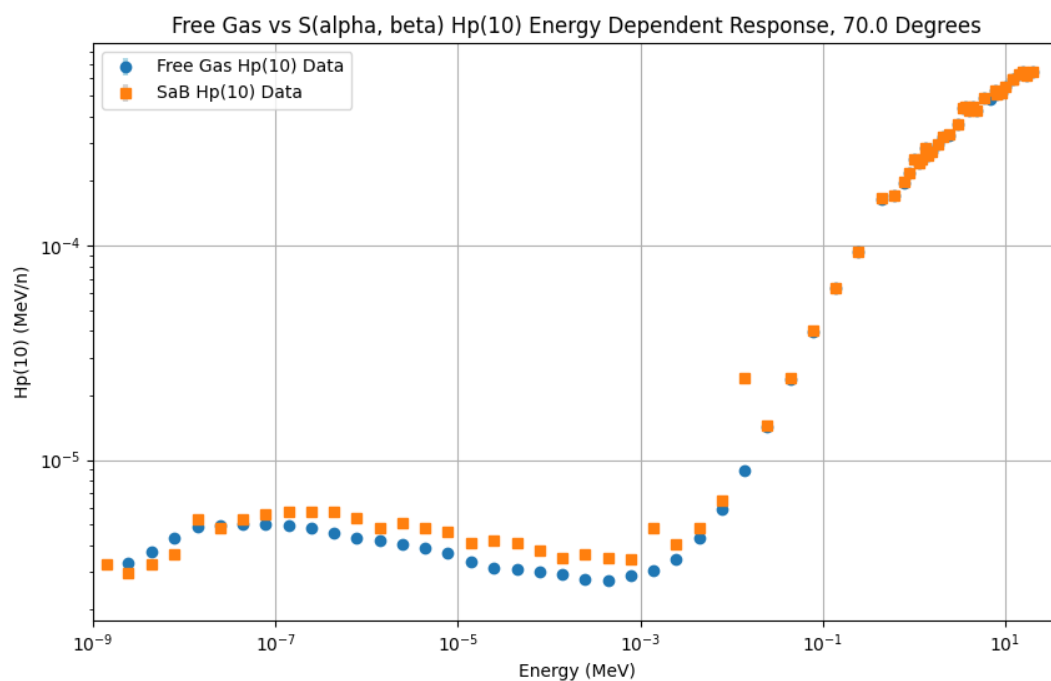
Appendix E. Figure E.5. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 40 Degrees



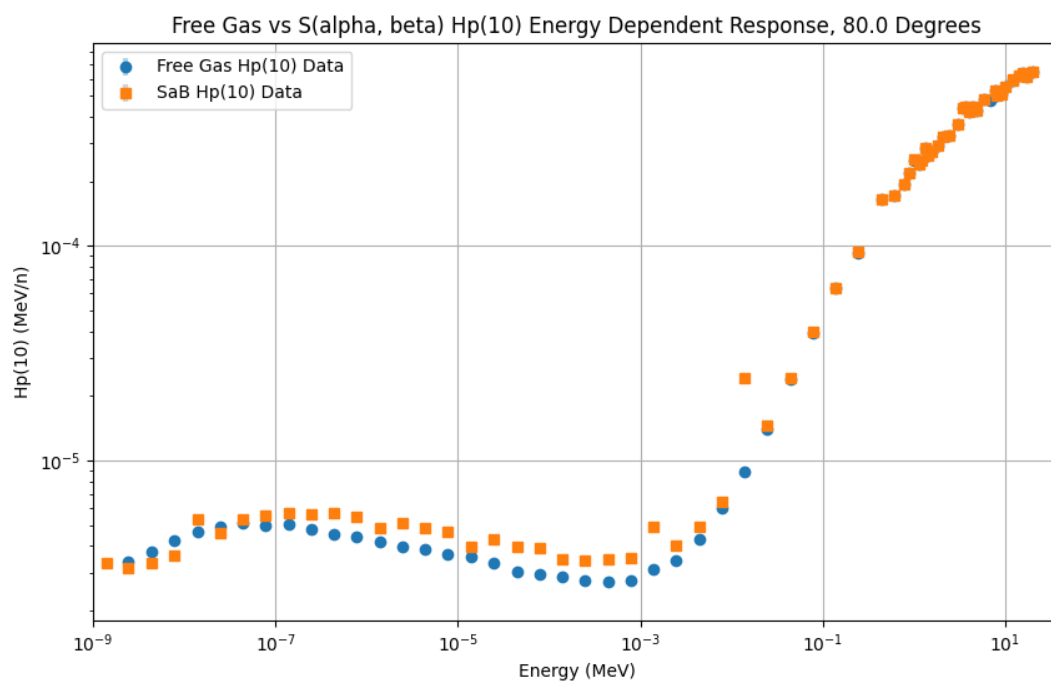
Appendix E. Figure E.6. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 50 Degrees



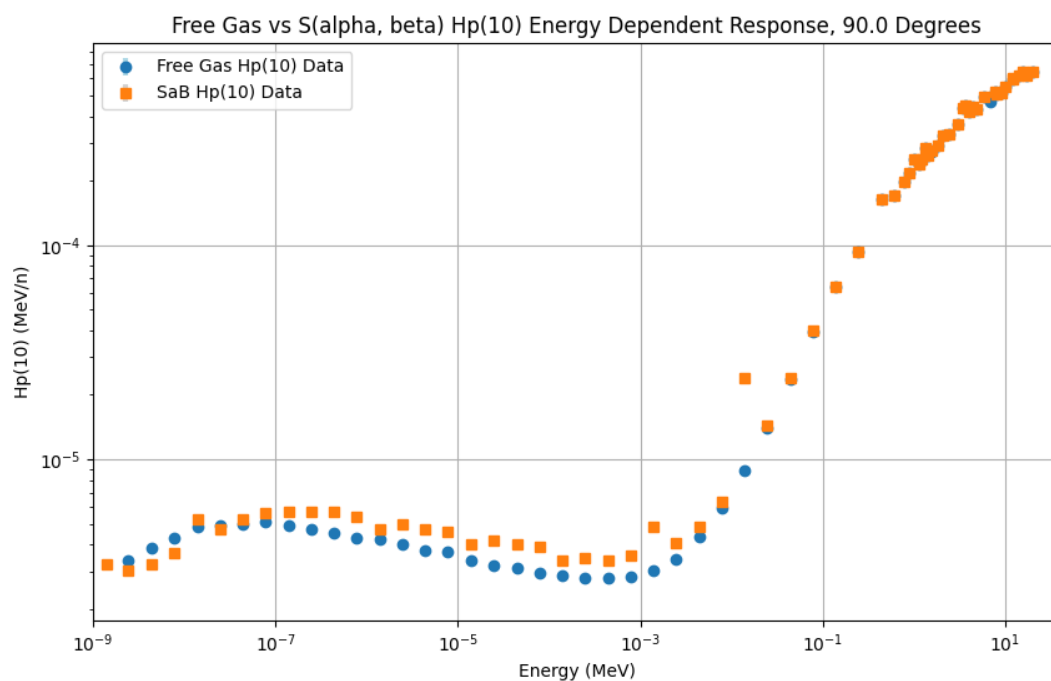
Appendix E. Figure E.7. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 60 Degrees



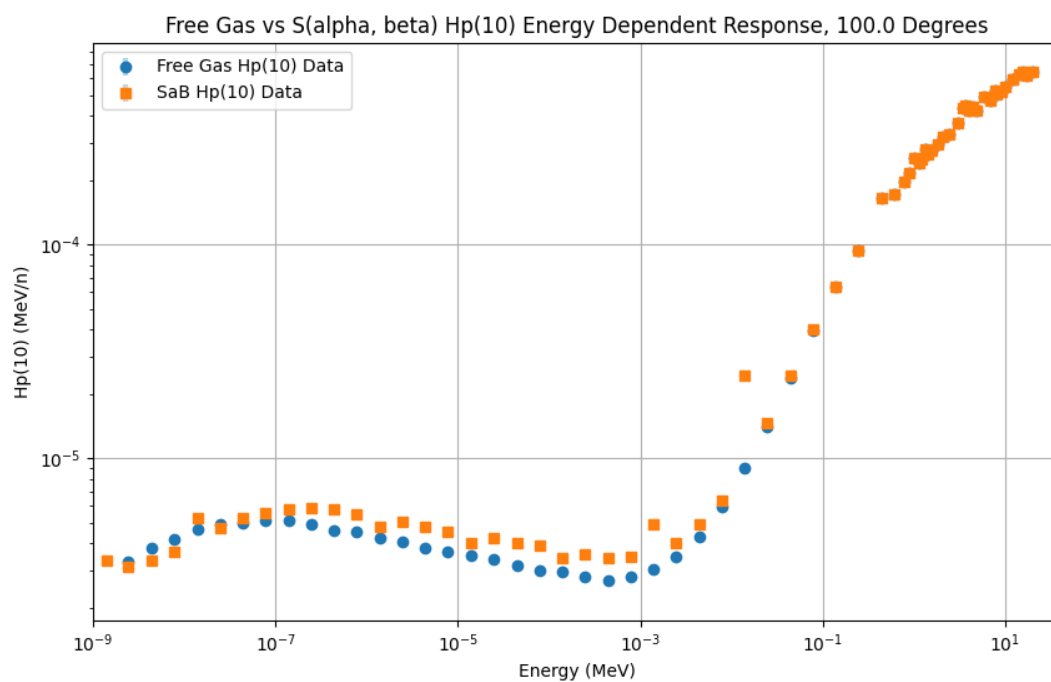
Appendix E. Figure E.8. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 70 Degrees



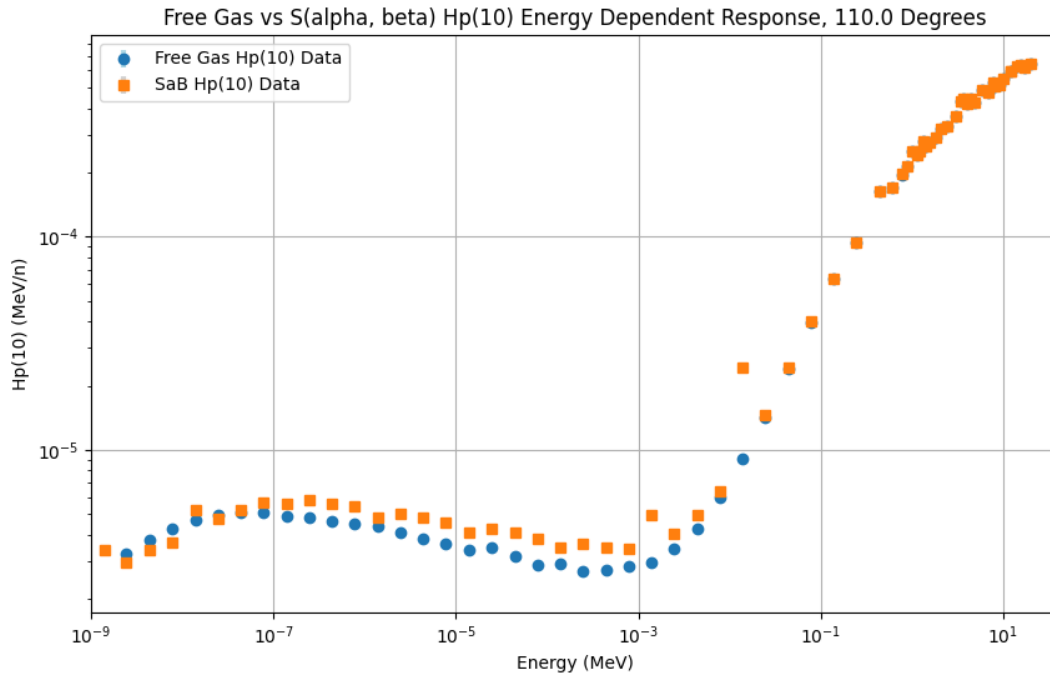
Appendix E. Figure E.9. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 80 Degrees



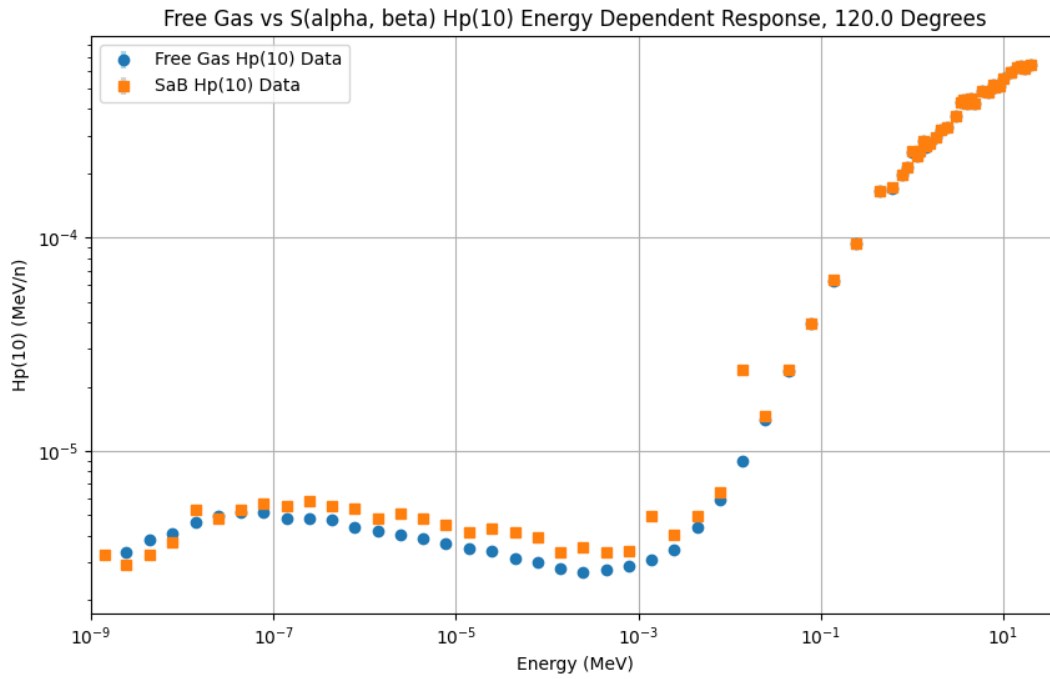
Appendix E. Figure E.10. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 90 Degrees



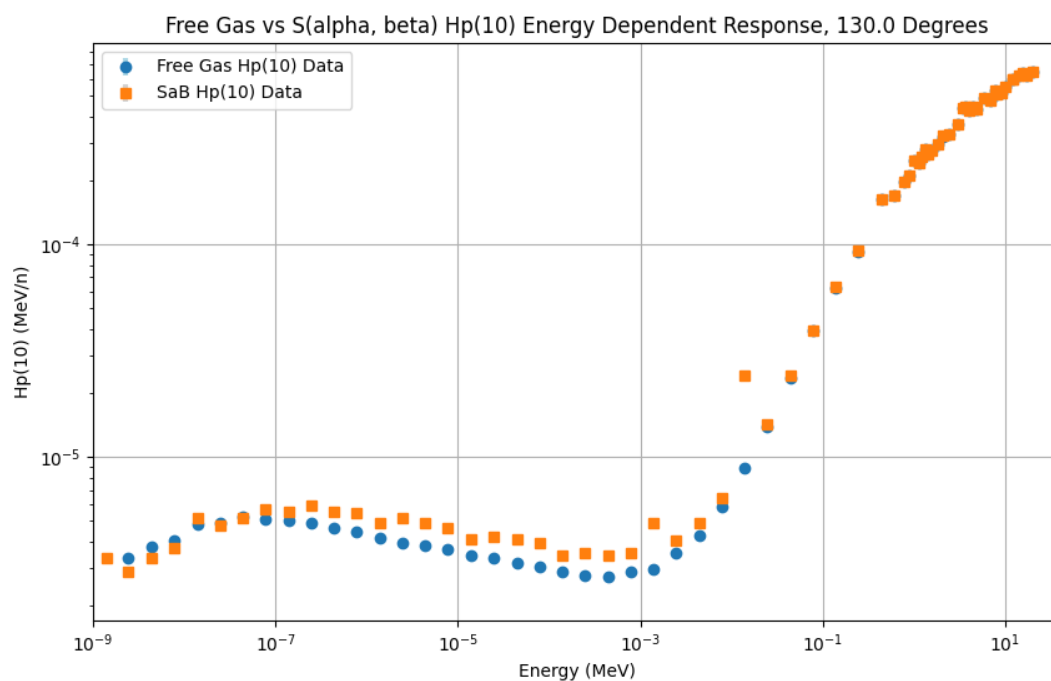
Appendix E. Figure E.11. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 100 Degrees



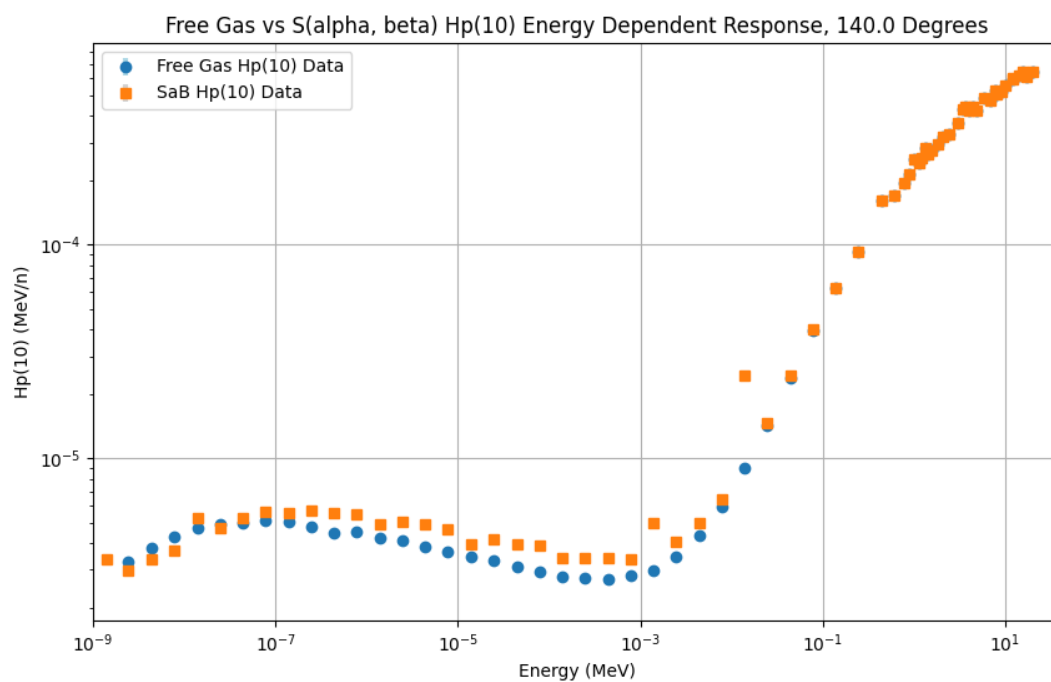
Appendix E. Figure E.12. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 110 Degrees



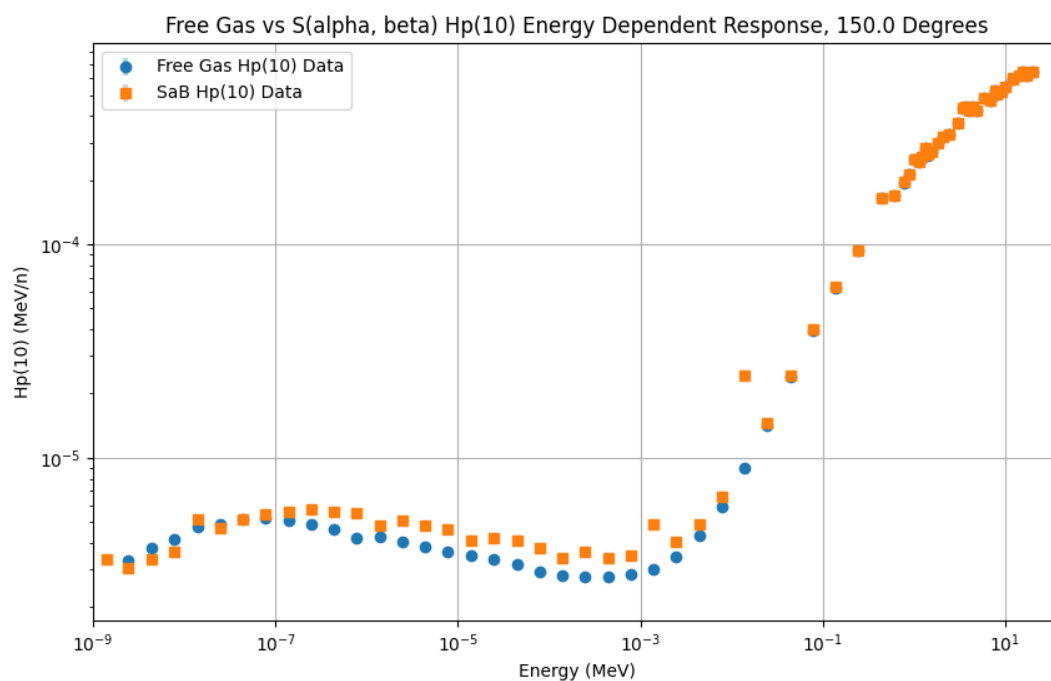
Appendix E. Figure E.13. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 120 Degrees



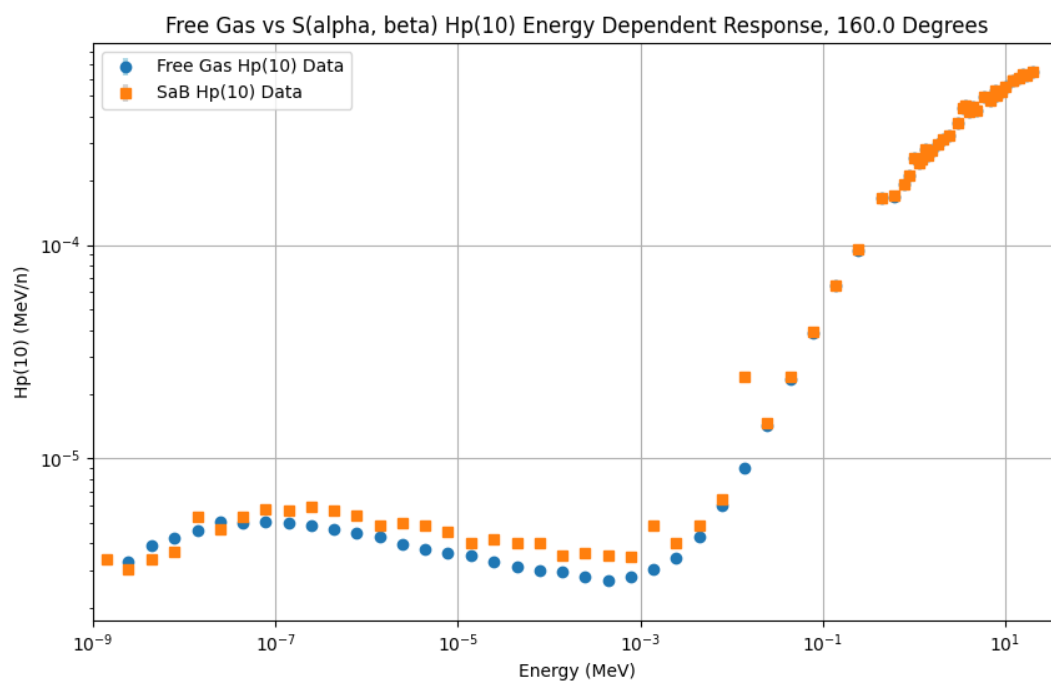
Appendix E. Figure E.14. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 130 Degrees



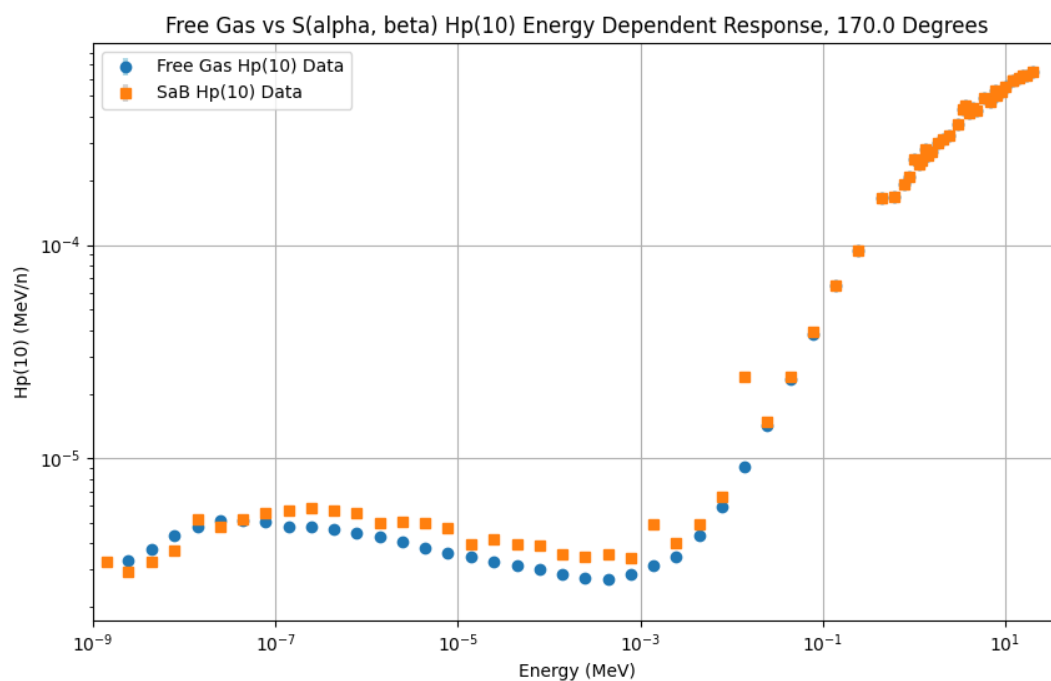
Appendix E. Figure E.15. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 140 Degrees



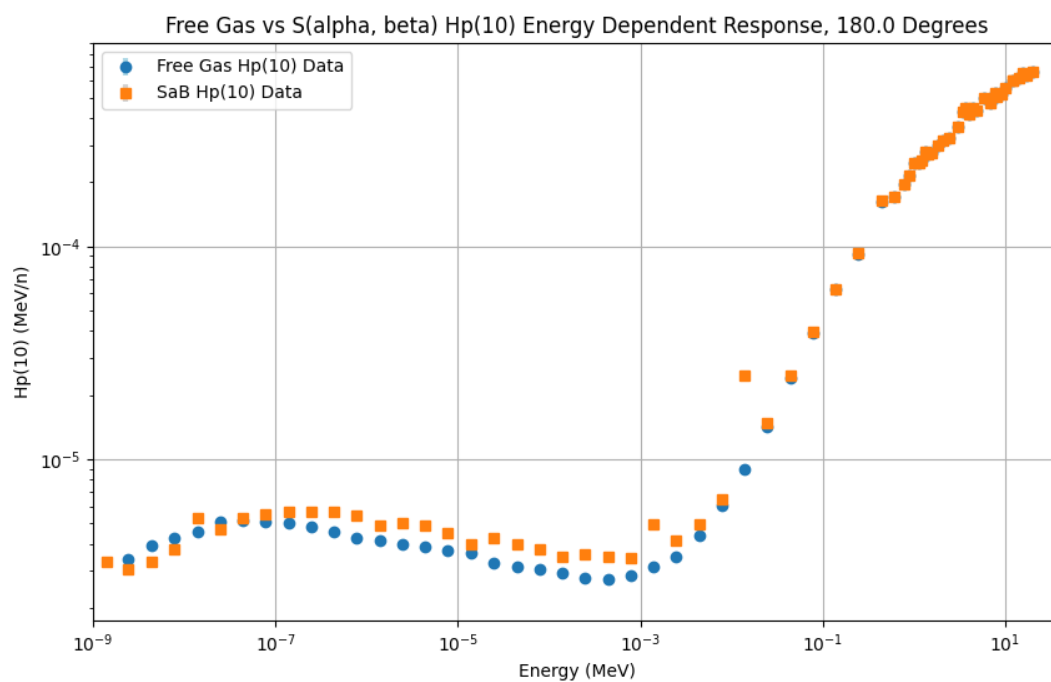
Appendix E. Figure E.16. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 150 Degrees



Appendix E. Figure E.17. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 160 Degrees



Appendix E. Figure E.18. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 170 Degrees



Appendix E. Figure E.19. Comparison of Personal Dose Equivalents for a Free-gas and $S(\alpha, \beta)$ models at 180 Degrees

Appendix F - Free-gas Scattering Compiled Data files

Note: To open the files (1) select the icon to open the attachments side-bar, (2) download the .zipX files, (3) re-name the files as .zip, (4) extract the files from the .zip folder.



Free-gas Compiled
Energy Results.zip

Appendix F. File F.1. Compiled results using a Free-gas scattering model, by incident neutron energy



Free-gas Compiled
Angle Results.zip

Appendix F. File F.2. Compiled results using a Free-gas scattering model, by incident neutron angle

Notation for data file headings:

- ADE = ambient dose equivalent
- PDE = personal dose equivalent
- E2 = Element 2 (neutron sensitive element)
- t = total (+F6 tally)
- n = neutron (F6:n tally)
- p = photon (F6:p tally)
- e = electron (F6:e tally)
- err = tally relative error

Example: tPDEerr = total (+F6 tally) Personal Dose Equivalent relative tally error

Appendix G - $S(\alpha, \beta)$ Scattering Compiled Data Files

Note: To open the files (1) select the icon to open the attachments side-bar, (2) download the .zipX files, (3) re-name the files as .zip, (4) extract the files from the .zip folder.



$S(\alpha, \beta)$
Compiled Energy Res

Appendix G. File G.1. Compiled results using a $S(\alpha, \beta)$ scattering model, by incident neutron energy



$S(\alpha, \beta)$
Compiled Angle Res

Appendix G. File G.2. Compiled results using a $S(\alpha, \beta)$ scattering model, by incident neutron energy

Notation for data file headings:

- ADE = ambient dose equivalent
- PDE = personal dose equivalent
- E2 = Element 2 (neutron sensitive element)
- t = total (+F6 tally)
- n = neutron (F6:n tally)
- p = photon (F6:p tally)
- e = electron (F6:e tally)
- err = tally relative error

Example: tPDEerr = total (+F6 tally) Personal Dose Equivalent relative tally error