Neutron properties and definitions (supplement). Heilbronn L. https://three.jsc.nasa.gov/articles/Heilbronn_Neutron_Supplement.pdf.
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Neutron Properties and Definitions

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This document is meant as a supplement and guide to the slides currently used for the neutron lectures in the NASA summer school's sessions on radiation physics. Those slides may be found at http://three.usra.edu/articles/NeutronPropertiesDefinitions.swf. Because not all of the slides will be shown in this document, it is recommended that a copy of the slides be available as the reader goes through this text.

Neutron Properties

There are basic properties of the neutron that must be presented in order to understand how and why neutrons are unique relative to the other components of the radiation environment in space. One of the most important properties is the lifetime of the neutron. A free neutron has a *half life* of approximately *15 minutes* (the neutron decays via beta decay to a proton and an antineutrino). What this effectively means is that there are no neutrons in the primary Galactic Cosmic Ray (GCR) environment. By the time GCR enters our solar system, any neutron created at the source of the GCR has decayed away. There are some neutrons emitted from the sun that do live long enough to reach the vicinity of Earth, but those are very few in number compared to the other radiations emitted from the sun.

If there are essentially no neutrons in the primary radiation environment in space, why do neutron monitors on the ISS and spacecraft indicate a significant number of neutrons inside those environments? Those *neutrons are created by nuclear interactions* between the primary GCR and any material it comes in contact with, including the spacecraft hull, other structural materials, and humans. As the amount of material increases, the number of generated neutrons also increases.

As one increases the amount of material and shielding in a spacecraft, the dose from the charged particle component of GCR decreases because those particles slow down (see "stopping power" discussion in other parts of the course) and stop or break up due to nuclear interactions. However, compared to charged particles, neutrons are much more *highly penetrating* and can go through shielding without interacting because *they have no charge*. As a result, neutrons can become more significant in terms of their contribution to the total dose and effective dose behind thick shielding in space.

The dose and subsequent biological damage delivered by neutrons depends greatly on the energy of the neutron. On Earth, radioactive sources of neutrons generate neutrons between 0 and 10 MeV. In space, the energy spectrum of neutrons is much different, with neutron energies going well beyond 10 MeV, up to several TeV. The presence of high energy neutrons poses problems not only for dosimetry and monitoring, but also for radiobiologists seeking to understand the biological effects from high-energy neutrons.

Neutron Energy Classification

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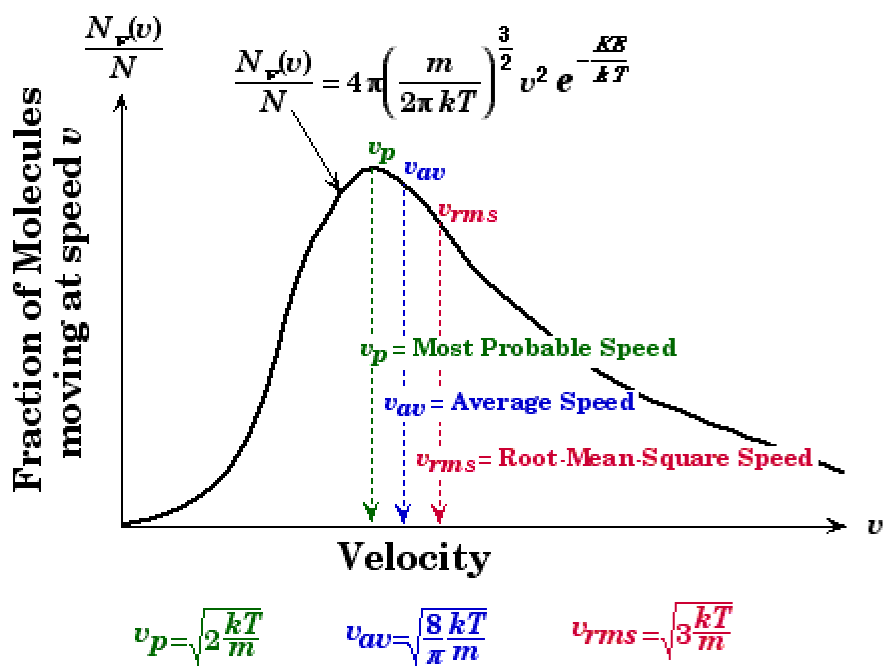
The followings terms used to describe neutron energy ranges were created primarily to distinguish how neutrons interact with materials used for applications utilizing fission, such as nuclear power, weapons development, and isotope production. They are relevant for the discussion of biological effects from neutrons, as well, because again how neutrons interact in the body depends very much on their energies.
Note that the energy ranges shown in the definitions below are not absolute, but are “ballpark” figures meant to convey information on the general range of energies.

1. *Cold neutrons* – energies below thermal energies (see next definition), typically corresponding to meV and sub meV energies, i.e., from 0 to 0.025 eV. Cold neutrons are used in a variety of applications, including studies on the structure of bio molecules. Both scattering and absorption reactions can occur at these energies (scattering and absorption are defined later), although for most atoms that compose biological material, the dominant reaction is scattering.

2. *Thermal neutrons* – the energy of neutrons that are in equilibrium with the motion of the atoms that make up the medium in which the neutron is found. Neutrons colliding with atomic nuclei either pick up energy if they are moving slower than the colliding nucleus, or lose energy if they are moving faster. This constant slowing down and picking up of energy by free neutrons leads to a distribution of neutron energies centered about the most likely energy (thermal energy). The distribution is called a Maxwell – Boltzmann distribution, which is shown in the plot below. For biological materials, scattering is the dominant reaction, although absorption of thermal neutrons by hydrogen and nitrogen is also present and can deliver dose to the biological material.



3. *Epithermal neutrons* – energies between thermal (~.025 eV) and a few hundred eV. This represents the transition region between thermal and slow neutrons where

"resonances" present in the nuclei that interact with the neutron begin to present themselves. Resonances are very important in the next energy range, slow neutrons.

4. _Slow neutrons_ – generally have energies between 100's of eV to 0.5 or 1 MeV. In this energy range, many of the nuclei that the neutron interacts with have nuclear structure within the combination of neutrons and protons that make up the nucleus, and that structure leads to an enhanced probability of interaction between the neutron and nucleus. The atoms than make up most biological materials – H, C, N and O – don't have the significant resonances that heavier atoms such as the actinides have.

5. _Fast neutrons_ – generally between 0.5 and 10 - 20 MeV. These are the energies of neutrons emitted by fission sources. This represents the upper limit of where most of the research on neutron radiobiology and fundamental neutron interaction cross sections has been done. This is not to say that there has been no research above these energies, it's just that the amount of information and data at energies above 20 MeV is not as well established as the data below.

6. _High energy neutrons_ – above 20 MeV. A significant fraction of the dose and effective dose from neutrons in space is delivered by high energy neutrons and represents the region of greatest uncertainty in the biological effects from neutrons in space.

Neutron Interactions

Neutron interact via two main reaction mechanisms: scattering and absorption. Both types of interactions can occur at any energy, but in general scattering reactions dominate once the neutron energy is above a few hundred keV. Scattering can also dominate at energies below a few hundred keV, but at those lower energies absorption can become significant, especially at thermal and slow energies.

Most terrestrial applications of neutron radiobiology have dealt with fission neutrons with energies above 0.5 MeV. At those energies, scattering is the main mechanism for delivering dose is neutron scattering, and in the human body, the main interaction is neutron-hydrogen elastic scattering.

A. Scattering interactions

In a scattering interaction, the neutron remains free after the interaction, but loses energy by transferring energy to the nucleus it strikes. An _elastic scattering interaction_ is one in which the total energy, momentum, and kinetic energy are conserved. An _inelastic interaction_ is one in which only total energy and momentum are conserved. Typically, in an inelastic interaction, the energy transferred to the struck nucleus can break up the nucleus into two or more pieces or leave the nucleus in an excited state. In all scattering interactions, the neutron delivers dose via _indirect ionization_, meaning that first the neutron transfers energy to a charged recoil particle (such as a nucleus, proton, deuteron, etc..), and then that charged recoil particle deposits its energy, leading to possible biological and material effects.

The power point slides accompanying this document show crude animations of elastic and inelastic neutron interactions with various nuclei. Note that in addition to the charged particle recoils that are produced and can deliver dose to the surrounding material, uncharged radiation such as gamma rays and scattered neutrons can also be produced. The important distinction between charged and uncharged secondary radiation is that the uncharged radiation can be deeply penetrating through the material, whereas the charged secondary particles have much shorter ranges in the material and are not considered deeply penetrating. These charged particles, however, typically have high values of Linear Energy Transfer (LET, see previous lectures), and as such have high associated quality factor (Q) values.

In the human body, fission energy neutrons interact primarily via elastic scattering with tissue components (H, C, N and O). There are well defined kinematical limits for the energies of the scattered recoil nuclei, which are shown in terms of the maximum fraction of the incoming neutron's energy given to the recoil nucleus in the table below.

Nucleus	Maximum fraction of neutron energy transferred to recoil nucleus
^{1}H	1.000
^{12}C	0.284
^{14}N	0.249
^{16}O	0.221

For example, if a 5 MeV neutron elastically scattered off of a ^{12}C nucleus, the maximum energy the recoil ^{12}C could have is:

(max. fraction) x (neutron energy) = (0.284) x (5 MeV) = 1.42 MeV.

Note that this isn't the only scattered energy the recoil ^{12}C can have, it's the maximum. The recoil ^{12}C can have any energy between zero and the maximum, with every energy in that range equally likely. Thus, on average, the recoil nucleus has ½ of the maximum recoil energy. In the example used above, for a 5-MeV neutron scattering off of ^{12}C, the average energy the ^{12}C has is:

(1/2) x (max recoil energy) = (½) x (1.42 MeV) = 0.71 MeV.

Once the neutron scatters, it can continue on and interact again and again until the neutron stops and is absorbed, or punches through the material. Generally, in the human body, the neutron scatters just once, with a small probability of additional scattering. The "**First Collision Dose**", the dose delivered to the human body by the first scatterings between neutrons and tissue, accounts for 80 to 90 percent of the total dose delivered to the body from neutrons.

The first collision dose can be calculated with the following equation:

$$D = \frac{\Phi N \sigma Q_{average}}{\rho}$$

Where D = dose (energy deposited per unit mass)
Φ = neutron fluence (number of neutrons per unit area)
N = # of nuclei per cm^3

σ = interaction cross section (unit of cm^2, related to the probability of an interaction)
ρ = density of the material (i.e., density of tissue, water,…)
$Q_{average}$ = Average energy transferred to the recoil nucleus (= ½ max energy transferred)

As an example, the following is a calculation of the first collision dose from a fluence of 5-MeV neutrons scattering off of the hydrogen in the human body. The cross section for elastic scattering of 5-MeV neutrons off of protons (hydrogen nucleï) is approximately 1.6 barns (= $1.6x10^{-24}$ cm^2). Since the maximum energy transfer to a proton from a 5-MeV neutron is 5 MeV, the average energy transferred to the proton is 2.5 MeV. The number of hydrogen nuclei per cm^3 in tissue is $5.98x10^{22}$. The density of tissue varies, but in this case a value of 0.92 g/cm^3 is used. Assuming that the neutron flux is 10^8 neutron/cm^2, the dose is:

$$D = \frac{(10^8 cm^{-2})(5.98x10^{22} cm^{-3})(1.6x10^{-24} cm^2)(2.5\ MeV)}{0.92\ g\ cm^{-3}} = 2.6x10^7\ \frac{MeV}{g}$$

In the SI unit of dose from radiation exposure,

$$D = 2.6x10^7 \frac{MeV}{g} \times \frac{1.6x10^{-13} J}{MeV} \times \frac{1000\ g}{kg} = 0.00416 \frac{J}{kg} = 0.00416\ Gy$$

B. Absorption interactions

In an absorption reaction, neutrons are absorbed by the nucleus, and as a result can deposit a significant amount of energy into that nucleus. The combined neutron + nucleus system, referred to as a compound nucleus, is often in an excited state and can de-excite through a number of different pathways, including the emission of charged secondary particles and uncharged (neutrons and gammas) radiation. As with scattering interactions, the charged secondaries have high LET and high Q values. The following figures show the basic steps in one such absorption reaction, neutron absorption in a ^{10}B nucleus, leading to the emission of an alpha and a ^7Li particle.

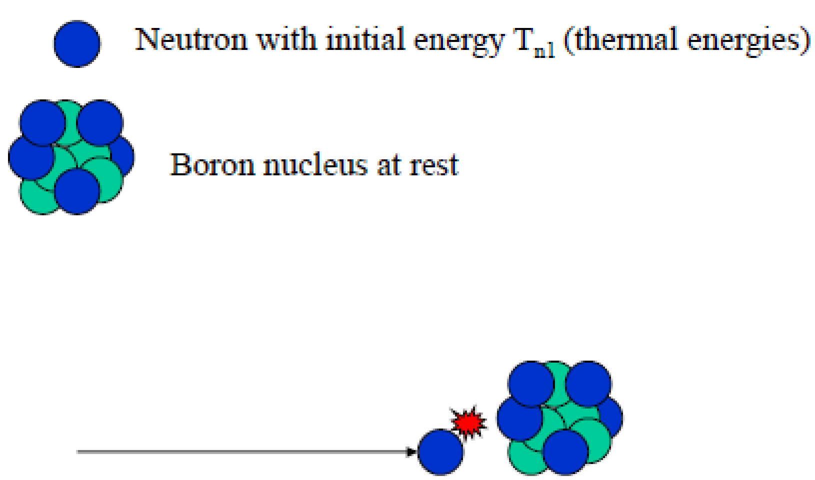


Figure 1a: An incoming thermal neutron strikes a ^{10}B nucleus.

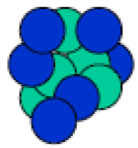


Figure 1b: The neutron from fig. 1a is absorbed in the ^{10}B nucleus, forming a compound ^{11}B nucleus.

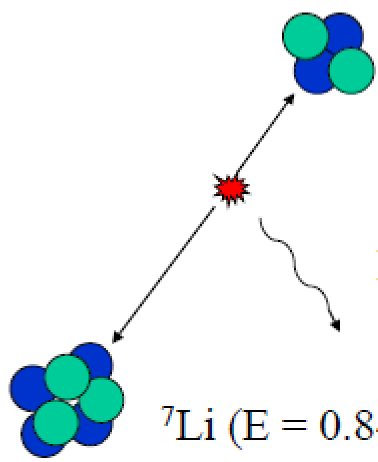


Figure 1c: The compound ^{11}B nucleus breaks up, emitting a ^{4}He nucleus, ^{7}Li nucleus, and a gamma ray.

In these types of absorption reactions, energy is often released via the emission of charged particle secondaries, such as the He and Li particles that are emitted in ^{10}B neutron capture. Typically, these charged secondary particles have very high LET values, and as a result deliver a large dose in a very small volume, with a high accompanying quality factor Q due to the high LET values.

To summarize neutron interactions, we've seen that neutrons interact via elastic scattering, inelastic scattering, and absorption. All of these interactions are nuclear interactions (interactions with atomic nuclei), and because of this the pattern of energy deposition by a neutron in a material is much different than how an incoming charged particle deposits energy. Whereas a charged particle deposits energy continuously through a material, a neutron deposits energy randomly in confined areas in the material, in a stochastic manner. As a result, the determination of the dose and dose equivalent delivered by neutrons in a material requires a different approach than what you have learned about charged particle dosimetry, and that is the subject of the next section.

Neutron Dosimetry Concepts

In the previous section we learned that a neutron can interact in a number of different ways in a material, and that each interaction can emit a variety of particles over a wide range of energies. Each interaction has an associated probability of that interaction, and as such it is possible to average over all of the possible interactions to give an average energy released to secondary charged particles per incoming neutron, as well as determine an average LET value. Note that these average quantities are a function of the incoming neutron energy as well as the material it goes through.

One average quantity that can be calculated is the KERMA (Kinetic Energy Released per unit MAss), which described the average energy released to charged secondary particles per unit mass. Note that KERMA will have the same units of dose (energy per unit mass), but is NOT the dose due to one subtle difference: KERMA is energy transferred to charged particles per unit mass, whereas dose is the energy absorbed from charged particles per unit mass. It is not necessarily true that all of the energy given to a charged secondary particle will be absorbed in the mass in which it is created.

The KERMA can be calculated with the following:

$$K(T_n) = \sum_i N_i \left[\sum_j \varepsilon_{ij}(T_n)\sigma_{ij}(T_n) \right]$$

i refers to the constiuent element, j is the type of reaction, σ is the reaction cross section, and ε is the average energy deposited in that reaction.

If all of the energy transferred to charged particles is absorbed in the medium (or if the energy not absorbed is taken into account), then one can use KERMA values to determine the dose delivered per unit fluence of neutrons. The table shown below gives one determination of the dose per unit fluence of neutrons, which in this case assumes an isotropic fluence of neutrons incident upon an adult male.

Neutron energy	Fluence for 1 cGy
thermal	2.9×10^9
5 keV	2.1×10^9
20 keV	2.0×10^9
100 keV	9.6×10^8
500 keV	4.3×10^8
1 MeV	2.7×10^8

5 MeV	1.8×10^8
10 MeV	1.6×10^8
20 MeV	9.0×10^7
50 MeV	5.2×10^7
100 MeV	4.2×10^7
200 MeV	2.6×10^7
500 MeV	1.1×10^7
1000 MeV	5.4×10^6

As mentioned above, in addition to calculating the average energy deposited by neutron interactions, one can determine the average LET value per neutron interaction, and consequently determine a radiation weighting factor associated with the dose delivered by neutrons. The radiation weighting factor is similar to the quality factor Q in that the dose is multiplied by a weighting factor to determine the equivalent biological effectiveness of that dose, allowing for a more direct comparison of the different types of radiation that can deliver dose to the body. In this case, multiplying the neutron dose by its radiation weighting factor yields the Equivalent Dose (as opposed to Dose Equivalent when multiplying charged particle dose by the quality factor Q). The figure below shows the old and new neutron radiation weighting factors as a function of neutron energy. The new factors were proposed after a reassessment of the dosimetry in the Nagasaki and Hiroshima nuclear weapon detonations.

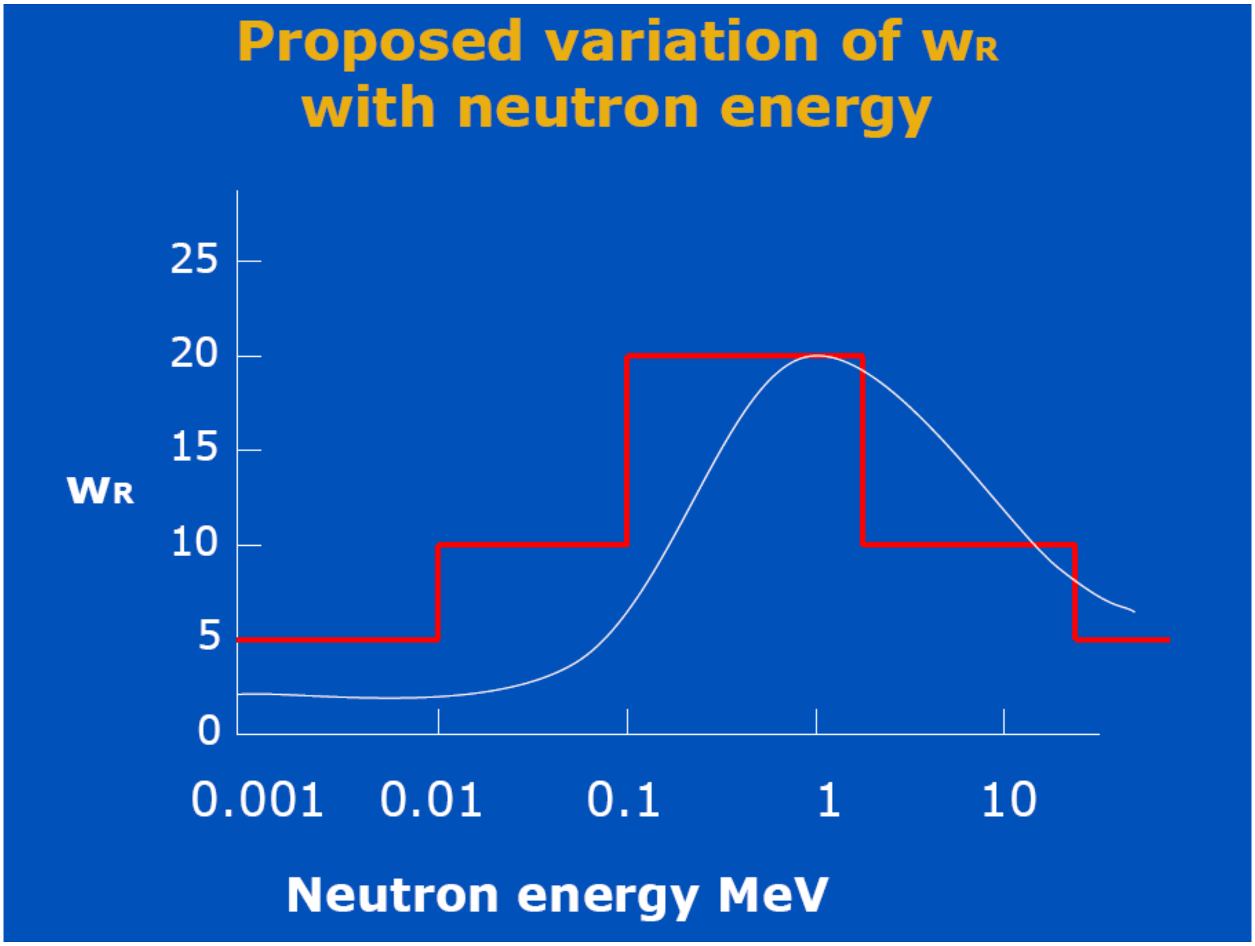


Figure 2. Neutron radiation weighting factors.

The physics behind these neutron dosimetric concepts is continually being improved with information gathered from new measurements and better modeling. In much the same manner, understanding the radiobiological effects of neutron irradiation is a field that needs more data and is open to wide range of possible experiments. One critical requirement for neutron radiobiology experiments is a reliable dosimetry system that can deliver accurate information on the dose, neutron energy (or energies), and neutron field shape. The next section covers some of the standard neutron dosimetry methods currently being used.

<u>Neutron Dosimetry Methods and Instrumentation</u>

When performing neutron radiobiology experiments, the neutron dosimetry system will most likely be required to deliver all or some of the following:

1. Monitor and determine the dose on target
2. Control the shape and size of the neutron beam (or neutron field)
3. Determine the neutron energy spectrum
4. Determine the level of contaminants in the neutron field (if any)
5. Minimize the amount of room scattered neutron striking the target

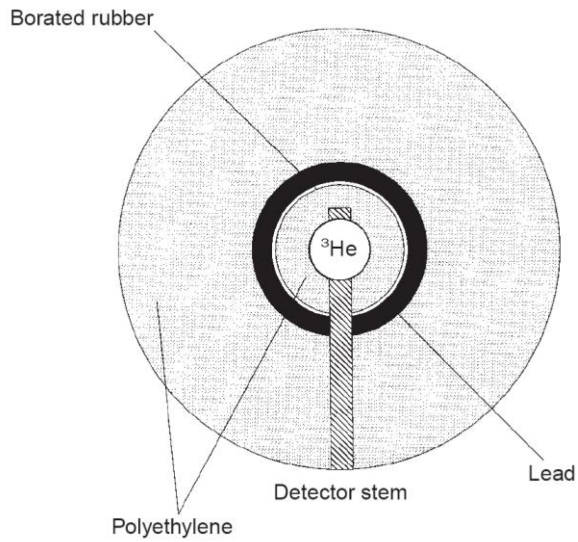
Most important of all of these is #1, the ability to monitor and determine the dose on target. The monitoring system should be able to:

a. Have a real time response
b. Distinguish neutrons from possible contaminants, such as gamma rays
c. Respond to a wide range of neutron energies, and have a known response to those energies.

There are a number of instruments and detectors that have been developed that meet these criteria, and they'll be briefly described here. One commonly used instrument that's found in accelerator environments is the REM counter, shown below

Typical REM counter/monitor used at accelerators.

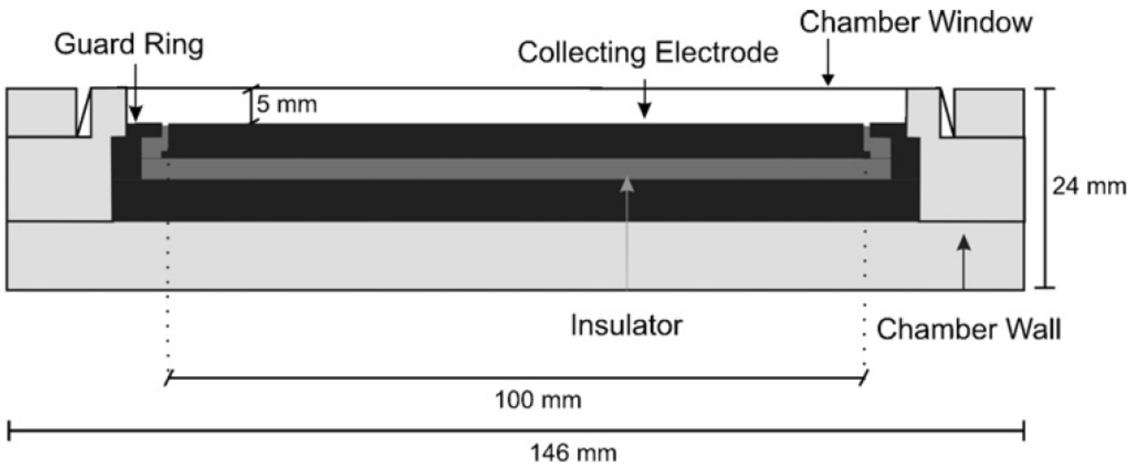
The REM counter works by surrounding a detector that is very sensitive to thermal neutrons with a large amount of moderating material, such as polyethylene. The poly will moderate (slow down) fast neutrons to thermal energies, where they are then detected by the thermal neutron detector placed in the center. The next figure shows a schematic of a typical REM counter.



Cut away view of a REM counter. The ^{3}He detector in the center is very sensitive to thermal neutrons

The REM counter has a real time response, will not respond to a gamma ray field up to 200 R/hr, has a uniform response to low energy neutrons (such as fission neutron energies), and can give an equivalent dose response if given information about the incident neutron energy spectrum. The disadvantages of the REM counter is that it responds to a limited range of neutrons (generally below 20 MeV, although the SWENDII detectors claim responses up to 100 MeV), the equivalent dose response is not accurate in a non-standard field, and incident charged particles can also give an unwanted response in the detector.

The fission chamber is an instrument used at high energy neutron facilities, such as the white neutron source at Los Alamos (LANSCE). It has the advantage that is responds to a much wider range of energies than the REM counter, is generally unresponsive to gamma rays, and is ideal for determining the total fluence of neutrons incident on a sample. The disadvantage is that one needs to know the neutron energy spectrum beforehand in order to determine the neutron dose on target. Established facilities such as LANSCE have standard, well characterized fields that they deliver, allowing for the determination of neutron dose, if needed. The figure below shows a schematic of the fission chamber. The chamber works by depositing a thin layer of ^{252}Cf on one of the parallel plates. When neutrons hit Cf, they can create a fission event that deposits a large amount of charge into the ionization chamber. The probability of creating a fission event is uniform with neutron energy, and as such the fission chamber is used in neutron fields containing a wide range of neutron energies, yielding information on total neutron flux.



Fission chamber. Neutrons are incident from the top, and the ^{252}Cf is deposited on the top layer.

The thimble chamber (ionization chamber) that is used at NSRL and many other accelerators can be used for neutron dosimetry. Because it has been widely used, its response to neutron fields below 20 to 50 MeV has been well characterized. However, the detector is also sensitive to charged particles and gamma rays, and as such may not be as useful in a neutron field that has associated gammas or charged particles.

Assuming that the dosimetry system is in place, the radiobiologist will want to choose a facility that can deliver the neutron energies needed for the experiment. The final section deals with the types of neutron facilities and sources that are available.

Sources of Neutrons

1. Accelerator-based systems

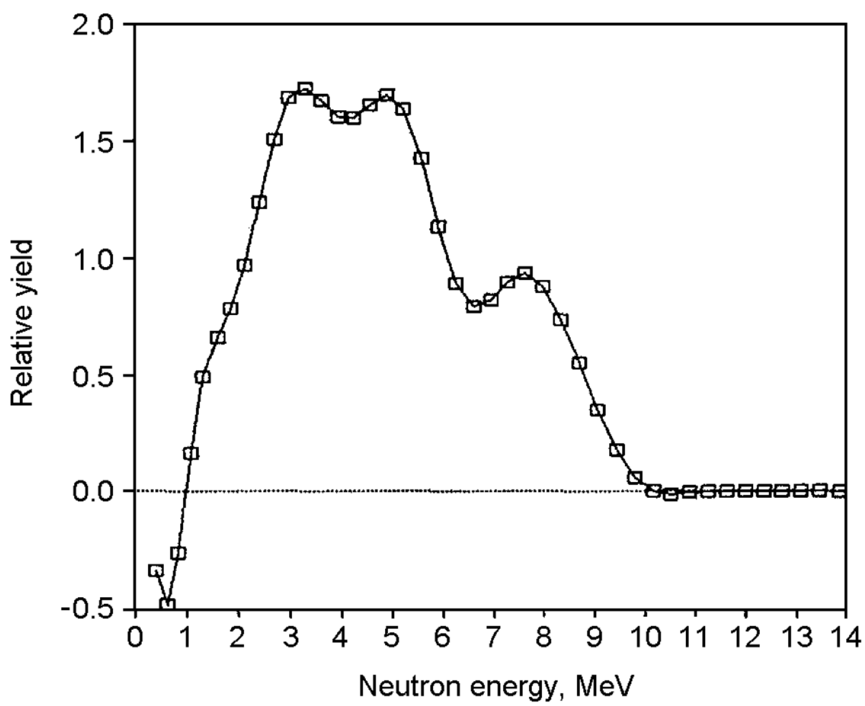
Accelerators can deliver mono-energetic beams of neutrons using the (d,T) and (d,D) interactions where a deuteron beam is delivered onto either a triton or deuteron target. The (d,T) reaction yields 14.1 MeV neutrons, whereas the (d,D) reaction yields 2.5 MeV neutrons. The dosimetry is greatly simplified due to the fact that the neutron just has one energy. Also, these accelerators are capable of delivering large fluences of neutrons over a short period of time.

Accelerators can also deliver quasi-monoenergetic beams via $^{7}Li(p,n)$, $^{9}Be(p,n)$, Ti(d,n) reactions. With those reactions, most of the neutron energies reside within a small window, but there usually is an associated flux of low energy neutrons that is also present and cannot be avoided. Some facilities create a beam of neutrons with a large range of energies, and through the use of timed shutters, can cut out a large dynamic range of neutron energies from the beam and deliver a narrow window of neutron beam energies. The total fluences at these facilities is somewhat limited, however, especially if one desires a narrow range of neutron energies. Quasi-monoenergetic facilities include the Crocker Cyclotron (Davis, CA), TSL (Uppsala U, Sweden), Cyric (Tohoku U., Japan), and the US Naval academy.

Facilities such as LANSCE and the Spallation Neutron Source (SNS, located at Oak Ridge National Laboratory) are called white neutron sources because they deliver beams of neutrons over a very wide range of energies. LANSCE will deliver neutrons from a few MeV up to 800 MeV, and the SNS has neutrons up to 1 GeV. If needed, small windows of energy can be selected out of the white beam through shuttering and timing techniques, but in those cases the background created by the neutrons outside the window must be dealt with either through a second experiment that just measures the background, or through online vetoing techniques that stop data collection when background is present.

Reactors are capable of delivering thermal neutron fluences as high as 10^{15} neutrons per second per cm^2. Thus, dose rates are high, albeit at thermal energies, which may not be a great concern in space. Also, reactor neutron beams have relatively high gamma ray backgrounds that must be taken into account with the dosimetry.

Radioactive neutron sources such as PuBe, AmBe, SbBe, ^{252}Cf deliver a “white” beam of fission energy neutrons, typically up to 8 or 10 MeV with a peak around 3 MeV (see figure below). The advantages of these sources is that they represent the neutron energies in most terrestrial environments where sources are present, and these sources are portable and can be used most any place that has a Radiation Safety Officer. The disadvantages to using these sources is that the fluences are very low, requiring long run times in order to collect enough dose. These sources also have an appreciable gamma ray background that must be taken into account.



Relative neutron yield as a function of neutron energy from a AmBe source.