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14 MeV Neutron Generator Dose Modeling

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14 MeV Neutron Generator Dose Modeling

by

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Abstract

14 MeV Neutron Generator Dose Modeling

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The University of Texas at Austin, 2013

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Modeling and understanding the doses around the neutron generator provides insightful data in regard to radiation safety and protection precautions. Published data can be used to predict doses, but realistic data for the Nuclear Engineering Teaching Laboratory's Thermo MP 320 Neutron Generator helps health physicists more accurately predict dose rates and protect experimenters against exposure. The goal was to create a model inclusive of the entire setup and room where the neutron generator is housed.

Monte Carlo N-Particle (MCNP) Code reigns as the preferred method for modeling radiation transport and was utilized to model the transport of neutrons within the current configuration of the 14 MeV neutron generator facility. This model took into account all shielding materials and their respective dimensions and locations within the concrete room. By utilizing tallies and tally modifiers, the model predicts dose rates that can be used with experimental factors such as irradiation time and flux to predict a dose in millirem.

Validation experiments were performed in the current setup using Landauer Luxel®+ with Neutrak dosimeters placed in strategic locations to record the neutron dose

received as well as a Ludlum Model 42-41 PRESCILA neutron probe to predict dose rates. The dosimeters and PRESCILA measurement locations matched the positions of the point detector tallies in MCNP. After laboratory analysis, a comparison was performed between the model output and the dosimeter and PRESCILA values to successfully validate the accuracy of the model.

Table of Contents

List of Tables

List of Figures

Chapter 1: Introduction

The neutron generator facility at the Nuclear Engineering Teaching Laboratory (NETL) at the Pickle Research Campus houses a Thermo Scientific MP 320 Neutron Generator with many applications including neutron activation analysis, shielding experiments, basic research, and nuclear forensics. With such a variety of experiments occurring, it is necessary to follow certain processes when it comes to safety. The University of Texas at Austin maintains a Radioactive Material License from the Texas Department of State Health Services, which contains certain guidelines for what types of elements are allowed to be utilized and what kinds of experiments are allowed to be performed. When an entity on campus is looking to do an experiment utilizing radioactive sources in a method different from an already approved experimental setup, an "Application to Use Radioactive Material" must be completed, submitted, reviewed, and approved by the on-campus Radiation Safety Committee before proceeding [1, 2].

Part of that application requires the submitter to be able to quantify the dose rates that a user could potentially encounter if unknowingly exposed to an active source as part of the experimental setup. In past experiments and applications for experiments using the neutron generator, submitters derived dose rate estimates from published values found in literature. Figure 1 shows the comparison of dose rates versus distance using the NCRP (circles) and ICRP (squares) conversion factors for a 1 x 10^8 n/s DT neutron source one meter above a concrete pad without scattering (dashed lines, calculated using $1/r^2$ scaling) and with scattering (solid lines, calculated using MCNP5) taken from the Chichester 2008 manuscript on Radiation Fields in the Vicinity of Compact Accelerator Neutron Generators [3].

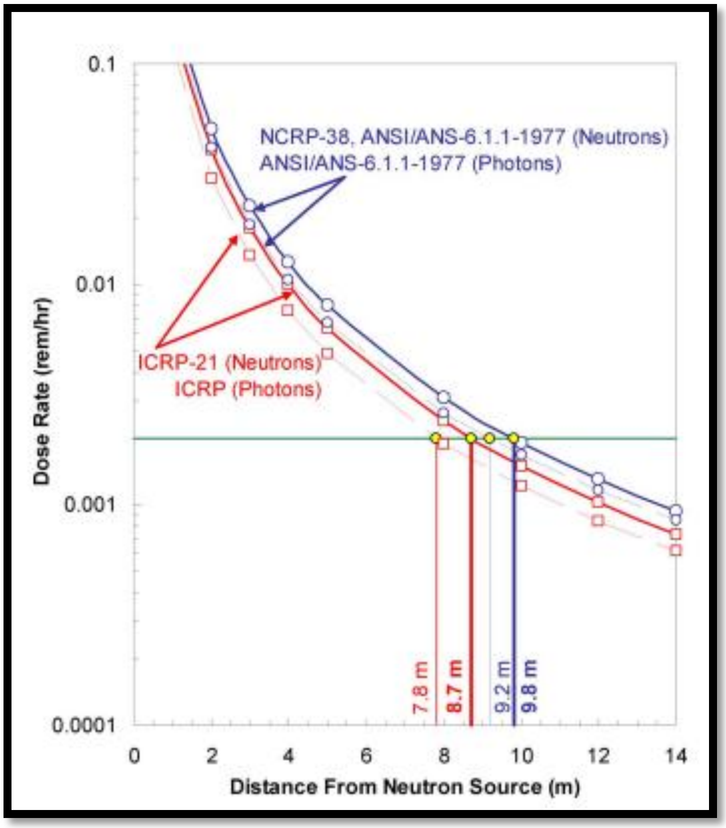

Figure 1: Dose Rates versus Distance [3]

Figure 2 shows the neutron, photon, and total dose rates 100 cm from a DT point source generating 1 x 10^8 n/s for polyethylene, concrete, and Bi/Poly shields ranging from 1 cm to 100 cm thick taken from the 2007 Chichester manuscript on Radiation Field from Neutron Generators Shielded with Different Materials [4].

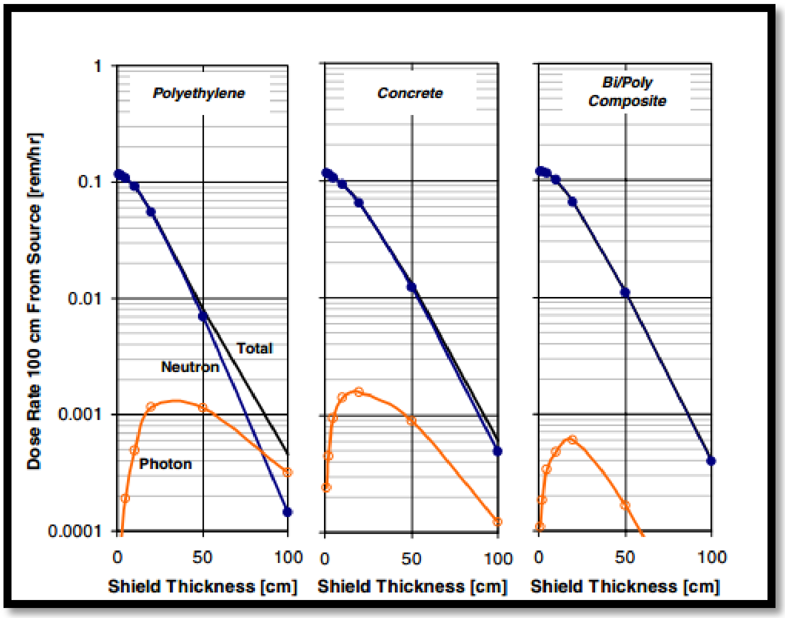

Figure 2: Neutron, photon, and total dose rates with varying shields [4]

The radiation safety staff currently utilizes the graphs shown in Figure 1 and Figure 2 to predict worst case scenario exposures as well as exposures with a shielded neutron generator setup. The radiation safety staff seeks a more realistic dose mapping for NETL's neutron generator for use in future predictions for different experimental setups as well as seeks to gain more insight into the dose rates researchers could be exposed to during an accident where safeguards fail.

The goals of this research include:

- Create an MCNP model mimicking the current setup of the neutron generator
- Use the MCNP model to predict neutron dose rates at various locations around the setup
- Use a PRESCILA neutron probe to collect predicted dose rates at various locations

- Perform an experiment to collect dosimeter data over a given amount of time
- Compare the output of the three methods to validate the MCNP model output

Chapter 2: Theory

NEUTRON GENERATION

For applications requiring neutrons, there are three popular choices for neutron production: nuclear reactors, radioisotopes, and accelerator-based neutrons sources. Depending on the application, one choice may be more pertinent than the other. When using a reactor, one must consider the cost, size, and complexity, which sometimes hinders their use for anything except power generation [5]. Radioisotopes also have their cons because of proliferation concerns about use in dirty bombs. Particle accelerators are commonly used, vary in size, and the most mainstream ones are compact light ion accelerators utilizing hermetically sealed tubes using deuterium-deuterium (D-D) or deuterium-tritium (D-T) [5]. The following reactions occur for the D-D and D-T neutron generators, respectively:

$$D + {}^{2}H \rightarrow {}^{3}H + n \quad Q = 3.270 \text{ MeV}$$
$$D + {}^{3}H \rightarrow {}^{4}H + n \quad Q = 17.590 \text{ MeV}$$

These reactions generate approximately 2.5 MeV and 14.1 MeV neutrons, respectively.

A modern neutron generator consists of a source used to generate positively charged ions, structures to accelerate the ions, metal hydride target filled with deuterium, tritium, or a mixture, and a metal hydride gas control reservoir [5]. Figure 3 is a schematic outlining the common parts of neutron generators.

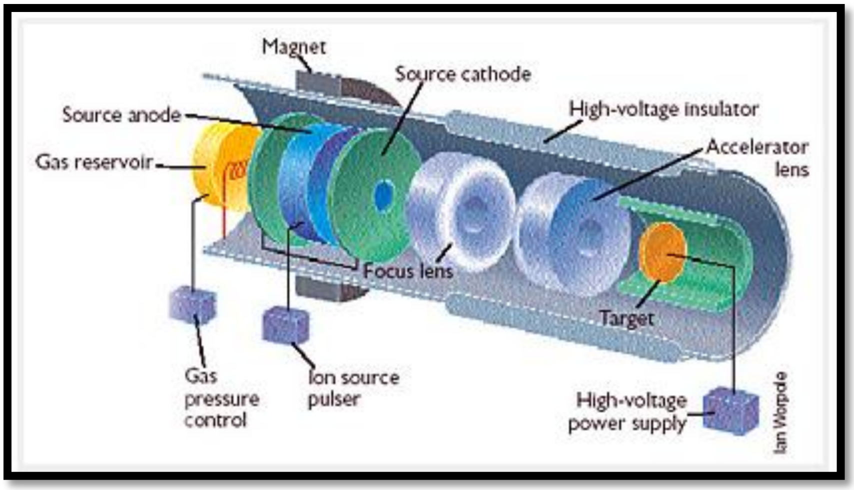

Figure 3: Sealed tube neutron generator with Penning Ion Source [5]

The Penning ion source is the most commonly used ion source. A hollow cylindrical anode is surrounded by cathode plates at each end. A coaxial field is generated by arranging an external magnet [5]. This field ionizes the deuterium and/or tritium gas when it is introduced into the anode. Ions are extracted through the exit cathode where they are accelerated through the potential difference of exit cathode and the accelerator electrode. These accelerated ions strike the deuterium and/or tritium target, where fusion occurs and produces the 14.1 MeV neutrons [5].

The neutron generator at NETL is a Thermo Scientific MP 320 utilizing D-T reactions to generate 14.1 MeV neutrons with a neutron yield of 1.0×10^8 n/s. It has a pulse rate of 250 Hz to 20 KHz, and a duty factor varying from 5% to 100% [6]. The tube used in the neutron generator is a sealed deuterium-tritium neutron generator tube that contains a nominal quantity of three Curies of tritium in the form of a metal tritide [7].

MONTE CARLO METHOD

Monte Carlo methods are used to theoretically mimic statistical processes often unable to be modeled with other deterministic methods. In Monte Carlo, the probabilistic processes of an event are simulated sequentially [8]. In order to obtain the whole picture, the event's probability distributions are statistically sampled based on the selection of random numbers. Monte Carlo N-Particle Transport Code, or MCNP can be used to model radiation transport for neutrons, photons, and/or electrons. In particle transport, each particle is followed from its birth at a source to its death [8]. Probability distributions are randomly sampled using transport data to determine a particle's fate [8]. Figure 4 shows an example of a particle history for a neutron incident on a slab of fissionable material.

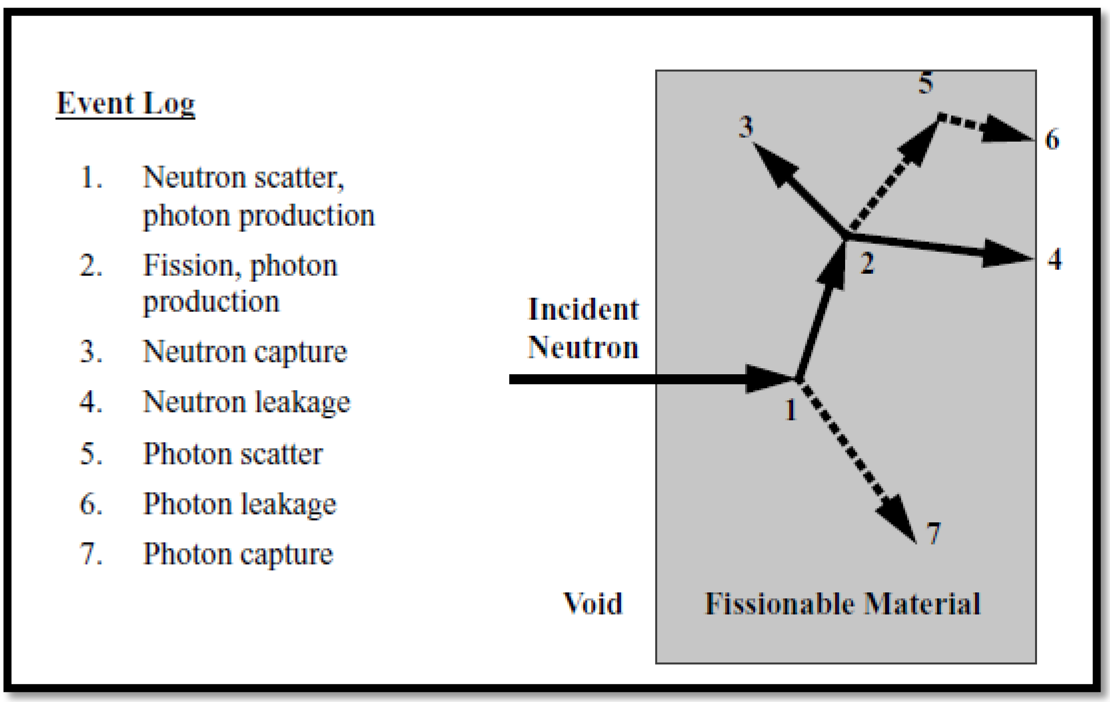

Figure 4: Particle History for Neutron Incident on Fissionable Slab [8]

MCNP randomly samples between 0 and 1 to determine what, if any, event will occur based on physics (rules) and transport data (probabilities). In the case of Figure 4, the incident neutron undergoes a collision at event 1. A photon is created and banked for later analysis. The direction of the neutron after scattering is randomly selected from a physical scattering distribution. Fission occurs at event 2 resulting in the termination of the neutron and the production of two new neutrons and a photon. One of the neutrons and the photon are banked for later analysis. At event 3, the first fission neutron is captured and terminated. At event 4, the banked neutron is retrieved for analysis and via random sampling is determined to leak out of the slab. The banked photon from fission is now retrieved, undergoes a collision at event 5, and leaks out at event 6. The photon generated from the incident neutron's collision event undergoes a capture at event 7 [8]. The tracing of all seven events is one particle history. As more histories are performed, the distributions more accurately represent actuals [8].

Inherent to MCNP are powerful standard features that provide users versatility including variety of sources, geometry plotters, variance reduction techniques, and an extensive set of cross sections [8]. The tallies are very important because they offer the flexibility to measure data such as surface flux, volume flux, and pulse heights. Additionally, tally modifiers can be used to manipulate tallies to provide a wide range of data including doses [8].

Using these features, users utilize MCNP for many real life applications such as reactor safety and shielding designs. Often times, MCNP is used to validate proposed design before investing capital into a project. MCNP is also very useful in predicting dose maps around a specific setup, which are often required in facility licensure, experiment approval, and general lab safety.

MCNP requires certain standard information on the input file including geometry specifications, description of materials, location/characteristics of source, tallies desired, and any variance reduction techniques [8]. MCNP was utilized in this project to model the transport of neutrons through a laboratory setup of a neutron generator.

LUXEL®+ WITH NEUTRAK DOSIMETERS

The Landauer Luxel®+ Dosimeter for X, Gamma, Beta, and Neutron Radiation provides X, gamma, and beta radiation monitoring using optically stimulated luminescence (OSL) technology [9]. Figure 5 shows a breakout of the different pieces that can be selected to build the dosimeter.

Figure 5: Luxel®+ with Neutrak Dosimeter [9]

The Al_2O_3detector is used to monitor X, gamma, and beta radiation. Neutron monitoring requires the addition of the CR-39 to the dosimeter. A CR-39 is an allyl diglycol carbonate based, solid-state nuclear track detector, and it is sealed inside the blister pack to prevent tampering [9]. It can detect fast neutrons with energies 40 keV to 40 MeV and thermal/intermediate neutrons 0.25 eV to 40 keV. For fast neutron detection, a polyethylene radiator records recoil protons resulting from neutron interactions in the dosimeter [9]. The thermal/intermediate neutrons are detected using a boron loaded Teflon® radiator to record alpha particles resulting from neutron interactions in the dosimeter [9]. When CR-39, or Neutrak® 144 detector, is included in the dosimeter, an analysis method named Track-Etch® is used. A 15 hour chemical etching takes place to enlarge exposure tracks. Fast neutron dose is determined by counting the tracks generated by proton recoil in the polyethylene radiator [9]. Thermal/intermediate dose is determined by counting the tracks generated by the alphas in the boron radiator [9].

PRESCILA NEUTRON PROBE

The Proton Recoil Scintillator (PRESCILA) was designed by Los Alamos National Laboratory to replace the traditional rem-ball design for neutron radiation detection. The PRESCILA has a sensitivity of 35 cpm per $\mu Svh^{-1}$for a ^{241}AmBe source and has an extended energy response to over 20 MeV [10]. It also uniform directional response (±15%) over a wide range of energies. Another advantage of the PRESCILA is that is lightweight in comparison to traditional tools [10]. Figure 6 shows an exploded view of the PRESCILA.

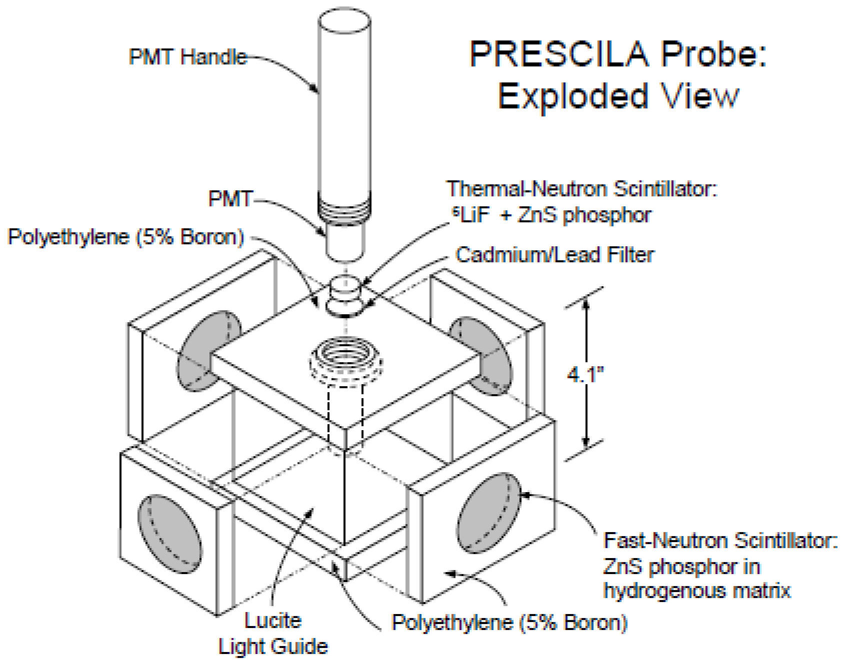

Figure 6: Exploded View of PRESCILA Probe [10]

The proton recoils occurring in the hydrogenous matrix of the fast neutron scintillator are used to measure the fast neutron signal. The phosphor used in the fast neutron scintillator is ZnS(Ag) powder [10]. In order to minimize self-absorption in the ZnS(Ag) phosphor, five rings of phosphor were used. The thermal neutron scintillator uses a mixture of ^{6}LiF and ZnS(Ag) powders that are hot pressed onto the back of a Lucite™ disc [10]. Any light produced in the scintillators is routed by the Lucite™ light guide to the photomultiplier tube to be converted into an electrical signal. The light guide and the borated polyethylene casing are used to moderate neutrons for the thermal scintillation detector [10]. Ludlum manufactures the model 42-41 PRESCILA neutron probe, which was used in this project to predict dose at various locations around the neutron generator.

Chapter 3: Development of an MCNP Model

As discussed previously, MCNP is a powerful tool for modeling radiation transport. Since modeling the transport of neutrons and calculating dose rates was the goal, an MCNP model was developed to provide dose rates for comparison with experimental dosimeter and PRESCILA neutron probe data.

NETL Neutron Generator Facility Geometry

The neutron generator is kept in the basement of the NETL facility, which is surrounded by three foot thick concrete walls. The room is approximately a 30' x 30' x 30' cube with a small hallway. Figure 7 shows the layout and dimensions of the facility as viewed from above, which was used for the MCNP model.

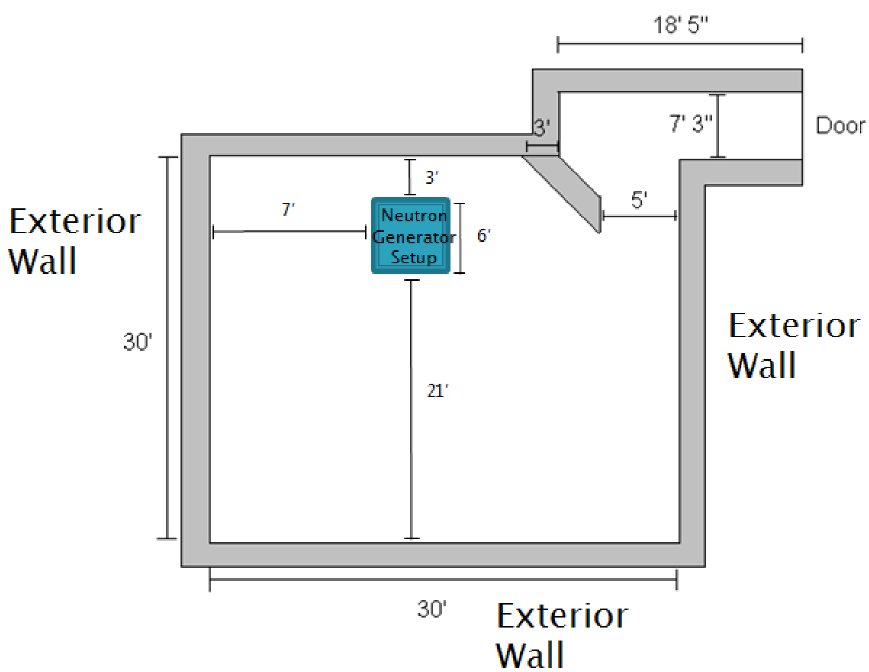

Figure 7: NETL Neutron Generator Facility Geometry [11]

MCNP MODEL GEOMETRY

The configuration of the shielding and the neutron generator in the lab as well as the facility geometry referenced in Figure 5 served as the basis for the model's surfaces and cells. Figure 8 shows two pictures of the setup taken from the lab.

Figure 8: NETL Neutron Generator Configuration (30-09-2013)

The picture on the left shows the main structure, the concrete, and leaded concrete bricks. The picture on the right shows a top view of the borated polyethylene and iron sheets that surround the neutron generator. Note the coordinate axis locations on the pictures. The origin is centered on the beam plane 15.75" above the neutron generator.

The dimensions of the shielding configuration were measured and recorded before running the experiments. Figure 9 shows the results of those measurements in inches.

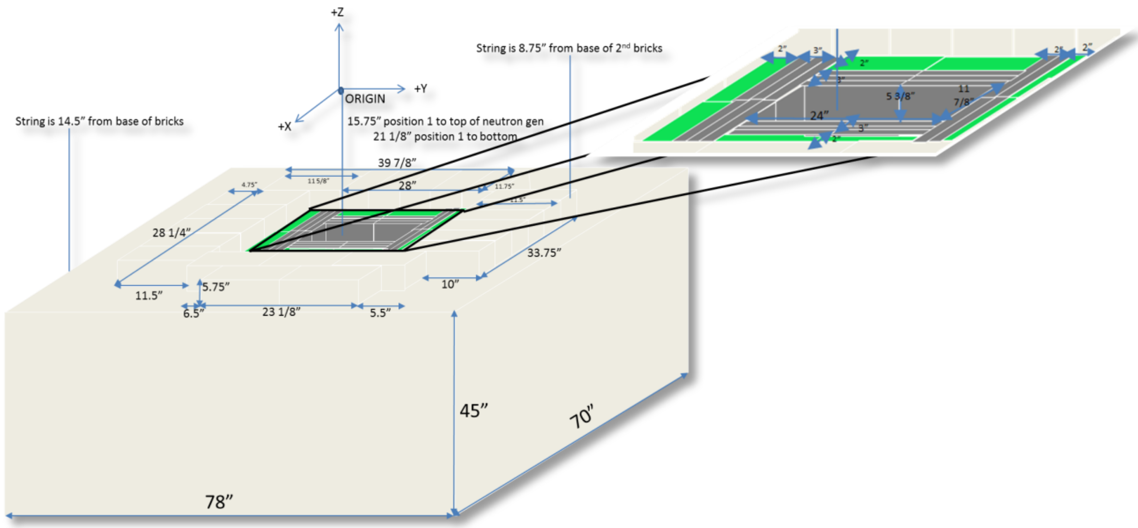

Figure 9: Dimensions of the NETL Neutron Generator Setup (30-09-2013)

In order to create a similar geometry in MCNP, the setup was broken down into 19 surfaces. Seven cells were formed using these surfaces. Figure 10 shows the surface definitions used in this model to create the concrete and leaded concrete bricks. Each of the surfaces was defined using a BOX macrobody and was based on the dimensions taken from the lab setup.

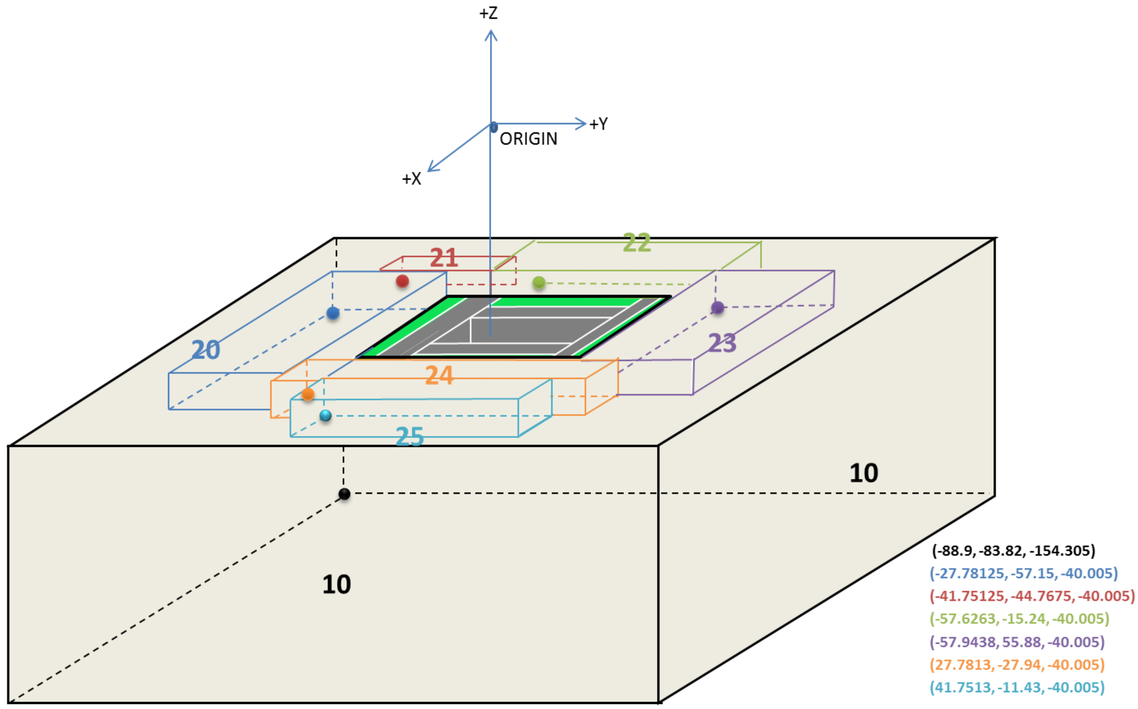

Figure 10: Surface Specification of Concrete Block & Leaded Concrete Blocks

Surfaces 20 – 25 represent the leaded concrete bricks, while surface 10 represents the concrete bricks that make up the majority of the shielding structure.

Figure 11 shows the surface definitions for the borated polyethylene and the iron sheets stacked around the neutron generator.

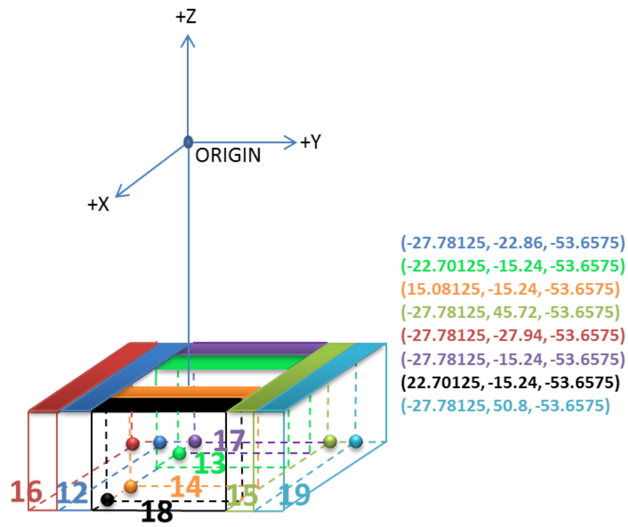

Figure 11: Surface Specification for Borated Polyethylene and Iron Sheets

Surfaces 12 – 15 represent the iron sheets while surfaces 16 – 19 represent the borated polyethylene sheets. Next in order to model the sheet of iron that the neutron generator sits on top of, surface 26 representing an iron sheet 11.8” x 24” x 3” was created and placed beneath the setup shown in Figure 11. Lastly, two concentric concrete cubes were modeled to create the three foot thick concrete walls of the facility. Surface 11 represented the inside of the concrete wall and was 30’ x 30’ x 30’. Surface 27 represented the outside of the concrete wall and was 36’ x 36’ x 36’. Figure 12 shows surface 11 and 27 as a small and large cube labeled inside the red box.

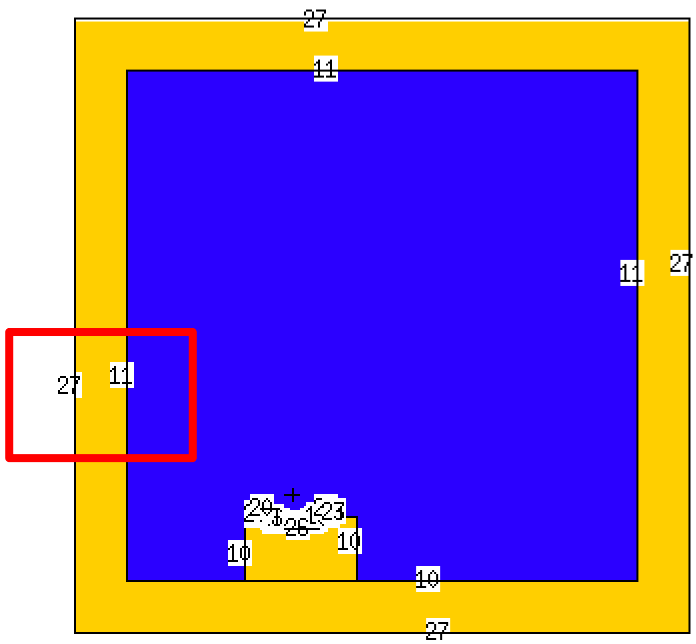

Figure 12: Surface Specification for Concrete Facility Walls

When placed on inside the other, the two create a surrounding wall three feet thick. After the creation of the surfaces, the surfaces were used to create seven cells. Figure 13 shows the x = 0 section cut of the model with the cells numbered 1 -7 and their corresponding material.

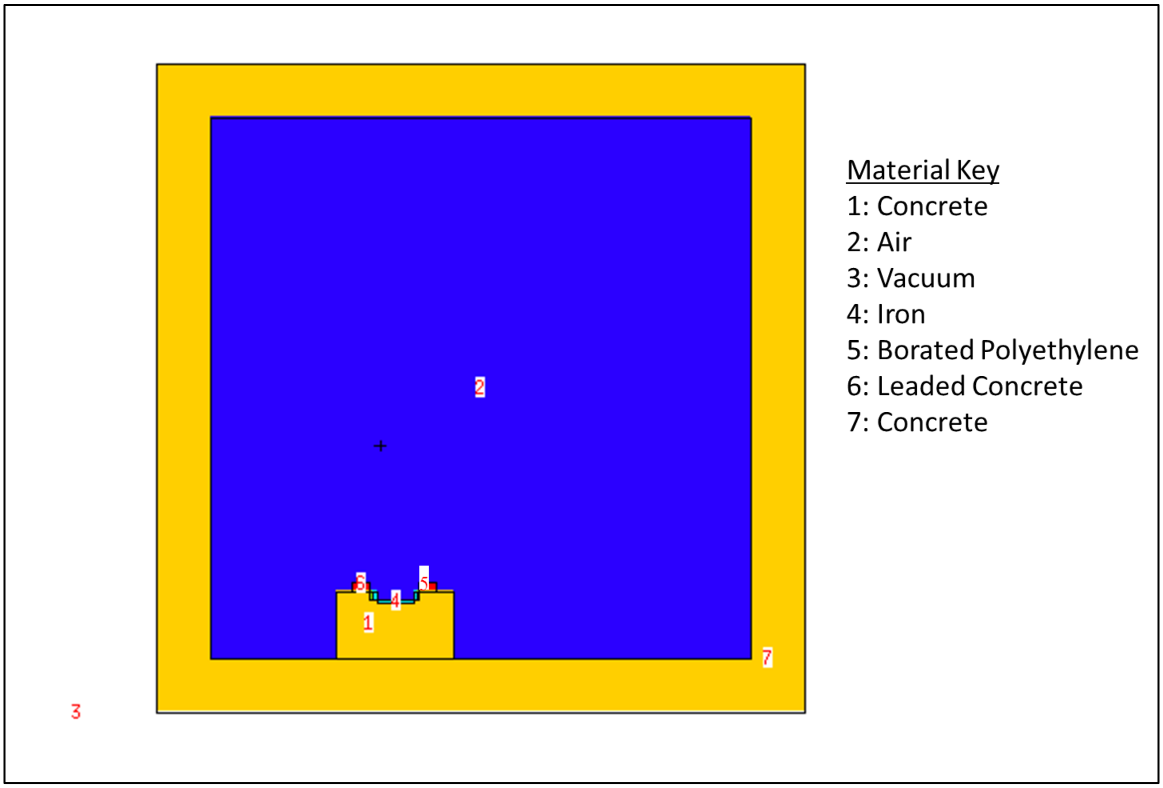

Figure 13: MCNP Overall Cell and Material Specification

Cell 1 was filled with concrete and represented the main shielding structure. Cell 2 was filled with air. Cell 3, the area extending beyond surface 27, was a vacuum representing the problem boundary. Cell 4 was the combination of the iron sheet surfaces 12 – 15 and surface 26. Cell 5 was the combination of the borated polyethylene surfaces 16 – 19. Cell 6 was a combination of the leaded concrete surfaces 20 -25. Lastly, Cell 7, designated as the overlap of 11 and 27, was filled with concrete.

Figure 14 shows the zoomed in top view from MCNP of the concrete structure. The red represents the leaded concrete bricks, the yellow represents the ordinary concrete, the light blue represents the iron, the green represents the borated polyethylene, and the dark blue represents air.

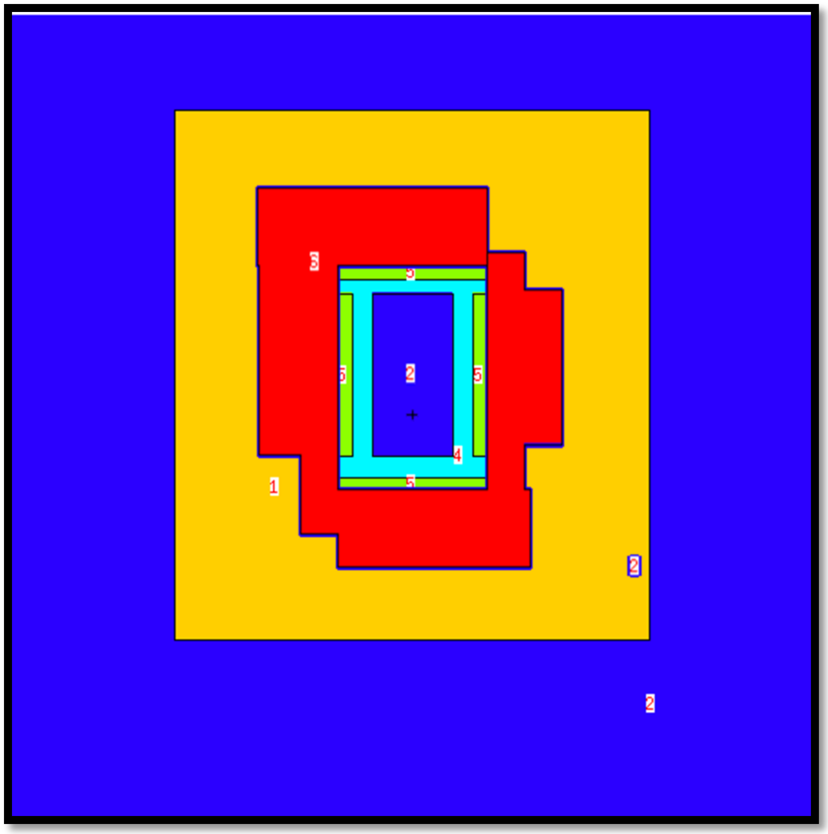

Figure 14: Top View of MCNP Neutron Generator Shielding Setup

This view matches the lab setup and provides a way to validate the geometry of the model. Figure 15 shows the MCNP code corresponding to the specification of surfaces and cells.

```
c ***  Cell Cards ***
c
c # mat#   mat_rho    surface_relations                   importances  comments
1   400   -2.300000  -10 12 13 14 15 16 17 18 19 26 28 imp:n=1      $concrete cube of bricks
2   100   -0.001205   (10 -11 20 21 22 23 24 25):-28    imp:n=1      $air
3   000               27                                imp:n=0      $problem boundary (outside concrete wall)
4   200   -7.874000   -12:-13:-14:-15:-26 28            imp:n=1      $iron sheets
5   300   -1.000000   -16:-17:-18:-19                   imp:n=1      $borated poly sheets
6   600   -4.110000   -20:-21:-22:-23:-24:-25           imp:n=1      $leaded concrete
7   400   -2.300000   -27 11                            imp:n=1      $concrete surrounding walls

c *** Surface Cards ***
c
10  BOX -88.9 -83.82 -154.305 177.8 0 0 0 198.12 0 0 0 114.3        $ concrete cube (bricks)
11 BOX -180.34 -297.18 -154.305 914.4 0 0 0 914.4 0 0 0 914.4       $ inside concrete wall
12 BOX -27.78125 -22.86 -53.6575 55.5625 0 0 0 7.62 0 0 0 13.6525   $ iron sheet
13 BOX -22.70125 -15.24 -53.6575 7.62 0 0 0 60.96 0 0 0 13.6525     $ iron sheet
14 BOX 15.08125 -15.24 -53.6575 7.62 0 0 0 60.96 0 0 0 13.6525      $ iron sheet
15 BOX -27.78125 45.72 -53.6575 55.5625 0 0 0 5.08 0 0 0 13.6525    $ iron sheet
16 BOX -27.78125 -27.94 -53.6575 55.5625 0 0 0 5.08 0 0 0 13.6525   $ poly sheet
17 BOX -27.78125 -15.24 -53.6575 5.08 0 0 0 60.96 0 0 0 13.6525     $ poly sheet
18 BOX 22.70125 -15.24 -53.6575 5.08 0 0 0 60.96 0 0 0 13.6525      $ poly sheet
19 BOX -27.78125 50.8 -53.6575 55.5625 0 0 0 5.08 0 0 0 13.6525     $ poly sheet
20 BOX -27.78125 -57.15 -40.005 71.755 0 0 0 29.21 0 0 0 14.605     $ leaded concrete
21 BOX -41.75125 -44.7675 -40.005 13.97 0 0 0 29.5275 0 0 0 14.605  $ leaded concrete
22 BOX -57.6263 -15.24 -40.005 29.845 0 0 0 71.12 0 0 0 14.605      $ leaded concrete
23 BOX -57.9438 55.88 -40.005 85.725 0 0 0 29.21 0 0 0 14.605       $ leaded concrete
24 BOX 27.7813 -27.94 -40.005 13.97 0 0 0 88.9 0 0 0 14.605         $ leaded concrete
25 BOX 41.7513 -11.43 -40.005 13.97 0 0 0 58.42 0 0 0 14.605        $ leaded concrete
26 BOX -15.08125 -15.24 -61.2775 30 0 0 0 60.96 0 0 0 7.62          $ iron sheet (underneath)
27 BOX -271.78 -388.62 -245.745 1097.28 0 0 0 1097.28 0 0 0 1097.28 $ outside concrete wall
28 BOX -15.07 -15.24 -53.6575 30 0 0 0 60.96 0 0 0 13.6525          $ air around neutron gen
```

Figure 15: MCNP Surface and Cells Specification Code

Materials

The materials used in this model were ordinary concrete, 20% lead by mass concrete, borated polyethylene, air, and iron. The first material added to the model, M100, represented air. M100 utilized compositions taken from the National Institute of Technology and Standards (NIST) website for dry air near sea level [12]. Proper material specification in MCNP follows the following format M*n* *ZAID*$_1$ *fraction*$_1$ *ZAID*$_2$ *fraction*$_2$. The *n* corresponds to the material number, ZAID corresponds to either the full ZZZAAA.nn.X or partial ZZZAAA element or nuclear identifier (ZZZ is the atomic number, AAA is the atomic mass, nn is the library identifier, and X is the class of data), and *fraction* corresponds to the atomic fraction (or weight fraction if entered as a negative) [8]. Table 1 shows the partial ZAID used and the corresponding mass fraction for the constituents of air in this model.

Table 1: M100 Air MCNP Material Specification

Element	ZAID	Mass Fraction
Carbon	6000	0.000124
Nitrogen	7014	0.755267
Oxygen	8016	0.231781
Argon	18000	0.012827

The next material added to the model, M200, represented pure iron. Table 2 shows the ZAID and corresponding mass fraction for iron in this model.

Table 2: M200 Iron MCNP Material Specification

Element	ZAID	Mass Fraction
Iron	26000	1.000000

The next material added to the model, M300, represented the borated polyethylene. This was modeled using previously provided data in another analysis [13]. Table 3 shows the ZAID and corresponding mass fractions for the various elements used for the composition of borated polyethylene in this model.

Table 3: M300 Borated Polyethylene MCNP Material Specification [13]

Element	ZAID	Mass Fraction
Hydrogen	1001	0.627759
Boron	5011	0.046690
Carbon	6000	0.325552

The next material added, M400, represented ordinary concrete [13]. Table 4 shows the ZAID and corresponding mass fractions for the elements used in the composition of concrete.

Table 4: M400 Ordinary Concrete MCNP Material Specification [13]

Element	**ZAID**	**Mass Fraction**
Hydrogen	1001	0.305330
Carbon	6000	0.002880
Oxygen	8016	0.500407
Sodium	11023	0.009212
Magnesium	12000	0.000725
Aluminum	13027	0.010298
Silicon	14000	0.151042
Potassium	19000	0.003578
Calcium	20000	0.014924
Iron	26000	0.001605

The last material added, M500, represented leaded concrete that was 20% lead by mass. For this case, the weight fractions were calculated for the MCNP material specification by starting with the NIST listing of constituent densities of ordinary concrete and normalizing them to the ordinary concrete density of 2.3 g/cm^3 [14]. Then each constituent normalized density was multiplied by 80%. Lastly, lead was given a weight fraction of 0.2. Table 5 shows the ZAID and corresponding weight fractions for the elements used in the composition of the leaded concrete.

Table 5: M500 Leaded Concrete MCNP Material Specification [14]

Element	ZAID	Weight Fraction
Hydrogen	1001	-0.0177391
Carbon	6000	-0.0020870
Oxygen	8016	-0.4598261
Sodium	11023	-0.0121739
Magnesium	12000	-0.0010435
Aluminum	13027	-0.0344348
Silicon	14000	-0.0160000
Potassium	19000	-0.2438261
Calcium	20000	-0.0080000
Iron	26000	-0.0052174
Lead	82000	-0.2000000

The MCNP code corresponding to the material specification is shown in Figure 16.

```
c   ----- Materials
c
c      -- AIR
c
M100   6000     0.000124    $ carbon
       7014     0.755526    $ nitrogen
       8016     0.231781    $ oxygen
       18000    0.012827    $ argon
c
c      -- IRON
c
M200   26000  1.000000   $ this one should be obvious
c
c      -- BORATED POLYETHYLENE, 10 wt% B4C
c
M300   1001    0.627759   $ hydrogen (in the poly)
       5011    0.046690   $ boron (in the B4C)
       6000    0.325552   $ carbon (both poly and B4C)
c
c      -- CONCRETE, ORDINARY (NIST)
c
M400   1001    0.305330   $ hydrogen
       6000    0.002880   $ carbon
       8016    0.500407   $ oxygen
       11023   0.009212   $ sodium
       12000   0.000725   $ magnesium
       13027   0.010298   $ aluminum
       14000   0.151042   $ silicon
       19000   0.003578   $ potassium
       20000   0.014924   $ calcium
       26000   0.001605   $ iron
c
c      -- LEADED CONCRETE (20% lead by mass)
c
M500   1001    -0.0177391   $ hydrogen
       6000    -0.0020870   $ carbon
       8016    -0.4598261   $ oxygen
       11023   -0.0121739   $ sodium
       12000   -0.0010435   $ magnesium
       13027   -0.0160000   $ aluminum
       14000   -0.2438261   $ silicon
       19000   -0.0080000   $ potassium
       20000   -0.0344348   $ calcium
       26000   -0.0052174   $ iron
       82000   -0.1996522   $ lead
```

Figure 16: MCNP Material Specification Code

MCNP Source Specification

Looking back at Figure 3, it can be seen that the target, or source, is the shape of a disc; therefore, will be modeled as such in MCNP. In MCNP, a disc is modeled as a degenerate cylinder. The SDEF, SI, and SP cards were used together to define a disc source. Figure 17 shows the code used to model the source.

```
SDEF  POS=0 0 -41.005 AXS=0 1 0 EXT=0 RAD=d1 PAR=1 ERG=14.0
      ARA=3.14159
SI1 0 1
SP1 -21 1
```

Figure 17: MCNP Source Specification Code

This code defines a source centered at 0, 0, -41.005 (POS) with a surface normal in the y direction (AXS). The source is defined as a cylinder with zero axial extent (EXT=0) to create a disc. The source emits neutron (PAR=1) at 14 MeV (ERG) isotropically. The RAD term defines the radius from the axis at which the neutrons are born. RAD=d1 points RAD to the source information and probability cards, SI and SP, respectively. SI and SP work together to define a probability distribution indicating the position of birth for the neutrons. A single radius bin from 0 cm to 1 cm is defined using SI1 0 1. To indicate a uniform probability distribution for the radius, SP -21 1 is used, which defines a power law distribution (-21 is built in) with an exponent of 1within the bin [8]. Lastly, the term ARA was used to give the area of the disc source. Using a radius of 1 cm, the area is just π. [13].

The print table 110 choice was used in the code to print the first 50 particles' position of birth and direction vectors. These were plotted to ensure all particles fell within the disk of radius 1 cm centered about 0, 0, -41.008. Figure 18 shows a plot of the first 50 particles location in the ZX plane plotted inside an appropriately sized disc.

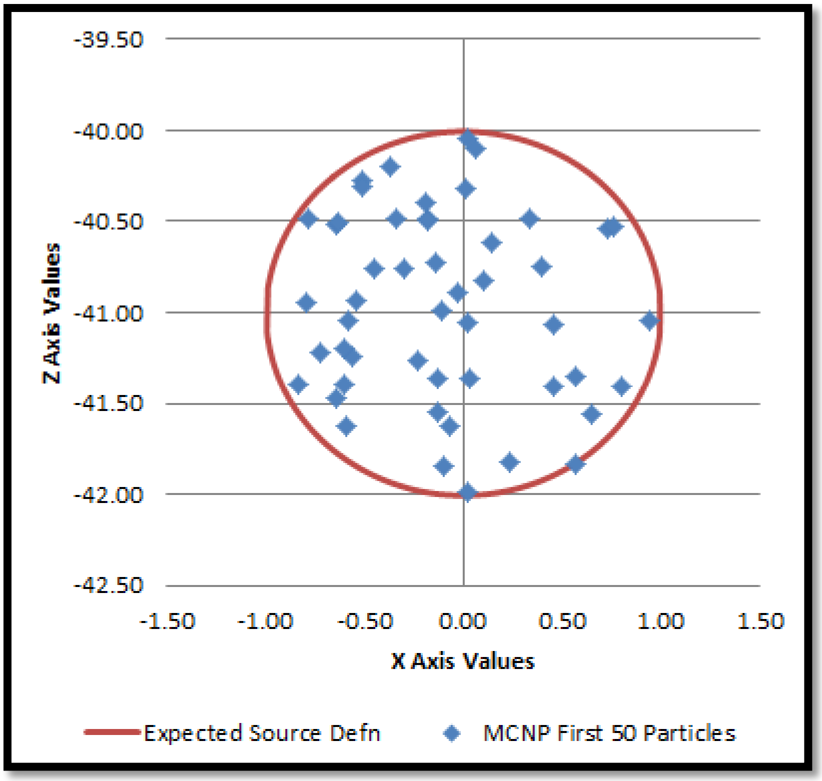

Figure 18: Plot of Birth Location of first 50 particles

The important thing to note about the plot is that all of the 50 particles fall inside the disc as expected, so it was verified that the generation of particles is functioning correctly. Next, a check was performed to ensure that the source was isotropic. The vectors printed in the table 110 were plotted on a MATLAB quiver plot to indicate the direction of travel after the particle was born. Figure 19 shows the quiver plot for the first 50 particles.

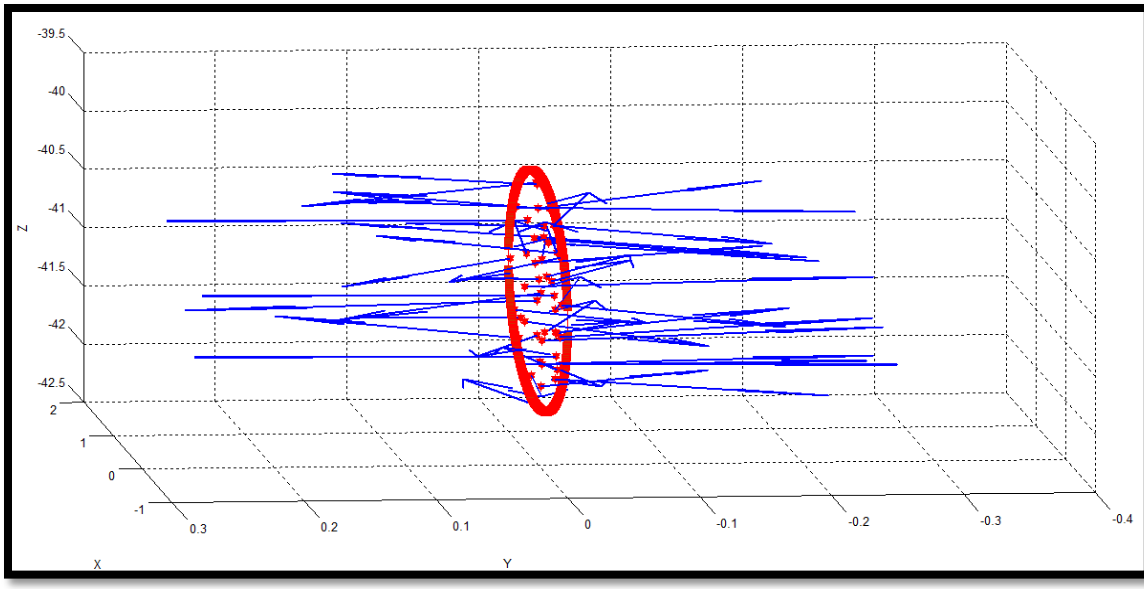

Figure 19: MATLAB quiver plot of first 50 particles

The plot shows no preference to any specific direction, and the first 50 particles vary in direction of birth and travel as expected for an isotropic source.

TALLIES

For this problem, a set of point detector tallies in conjunction with the dose function card were used at each planned experimental dosimeter location to collect the tally values. Since there were 20 dosimeters, 20 point detector tallies were used. In order to define a point detector tally in MCNP, a tally type F5 was used. The F5 type is designated by *F5a:pl x y z r,* where *a* is the tally number, *pl* is the particle type, *x/y/z* are the coordinates, and r is the spherical exclusion zone surrounding the point detector [8]. The output of the F5 is # of particles/ cm^2, which is important to remember for post-processing [8]. In this case, one tally, F15, was used with particle type *N* to designate neutrons for all 20 point detectors. The various position coordinates for each detector were entered one after another and a spherical exclusion zone of 0.5 was applied. Since the goal is to find the dose rates at the various locations, the dose function, or DF, card was used. The DF card for using the built in dose function is designated by *DFn IU=j FAC=f INT IC=I*, where *IU* is the control units (1:rem/h/SP, 2: Sv/h/SP), *FAC* is the normalization factor for dose, *IC* is the standard dose function (provided in Figure 20), and *INT* is the energy interpolation method (Dose interpolation is always linear) [15].

Value of IC	Description
Neutron Dose Function	
10	ICRP-21 1971
20	NCRP-38 1971, ANSI/ANS-6.1.1-1977
31	ANSI/ANS-6.1.1-1991 (AP anterior-posterior)
32	ANSI/ANS-6.1.1-1991 (PA posterior-anterior)
33	ANSI/ANS-6.1.1-1991 (LAT side exposure)
34	ANSI/ANS-6.1.1-1991(ROT normal to length & rotationally symmetric)
40	ICRP-74 1996 ambient dose equivalent
Photon Dose Function	
10	ICRP-21 1971
20	Claiborne & Trubey, ANSI/ANS 6.1.1-1977
31	ANSI/ANS-6.1.1-1991 (AP anterior-posterior)
32	ANSI/ANS-6.1.1-1991 (PA posterior-anterior)
33	ANSI/ANS-6.1.1-1991 (LAT side exposure)
34	ANSI/ANS-6.1.1-1991(ROT normal to length & rotationally symmetric)
35	ANSI/ANS 6.1.1-1991 (ISO isotropic)

Figure 20: Standard Dose Functions [15]

In this case, the neutron dose function chosen was ICRP-21 1971 or *IC=10*, the units chosen were rem/hr/SP, or *IU =1,* and the energy interpolation method chosen was *LOG*. Figure 21 shows the MCNP code written to complete the tallies.

```
F15:N   0 0 0 0.5       & $Dosimeter 1
-121.92 0 0 0.5         & $Dosimeter 15
-91.44 0 0 0.5          & $Dosimeter 16
-60.96 0 0 0.5          & $Dosimeter 5
-30.48 0 0 0.5          & $Dosimeter 4
30.48 0 0 0.5           & $Dosimeter 2
60.96 0 0 0.5           & $Dosimeter 3
91.44 0 0 0.5           & $Dosimeter 18
121.92 0 0 0.5          & $Dosimeter 14
0 30.48 0 0.5           & $Dosimeter 8
0 60.96 0 0.5           & $Dosimeter 9
0 91.44 0 0.5           & $Dosimeter 19
0 121.92 0 0.5          & $Dosimeter 17
0 -30.48 0 0.5          & $Dosimeter 6
0 -60.96 0 0.5          & $Dosimeter 7
0 -91.44 0 0.5          & $Dosimeter 20
0 -121.92 0 0.5         & $Dosimeter 13
91.44 0 -43.815 0.5     & $Dosimeter 11
0 114.3 -43.815 0.5     & $Dosimeter 10
0 -83.82 -43.815 0.5      $Dosimeter 12
DF15 ic=10 iu=1 LOG
```

Figure 21: MCNP Tally Specification Code

In order to pass the statistical checks MCNP performs for point detectors, the relative error must be less than 5% [8]. To achieve this in a timely manner, weight windows were used to reduce the variance in weights of scoring particles [13] The weight window command is given by *WWG* I_t I_c W_g *j j j j* $I_{E,}$ where I_t is the tally number, I_c is the reference cell, W_g value of generated lower weight window bound, the *"j"* represents that the next four terms are 'jumped' over, and $I_{E,}$=0 specifies the generated WWGE card is for energy bins [8]. For this code, the only items requiring explicit specification are I_t and I_c, which are equal to 15 and 2, respectively. Figure 22 shows the WWG card in the MCNP input file. Notice that the WWP is commented out because the first run of the input file is used to create the weight windows.

```
WWG 15 2
c WWP:N 5 3 5 0 0 0
c wwn1:n
```

Figure 22: MCNP Weight Window Specification Code

Lastly, the physics options were added to restrict particle transport to neutrons (MODE N), set an upper limit on neutron energy of 15 MeV and a capture limit of 0 MeV (PHYS:N 15.0 0), and set the number of particle histories to 1,000,000 [13]. Figure 23 shows the MCNP code containing the appropriate physics options.

```
c  ----- Physics Options (With Importances if Necessary)
c
MODE N
PHYS:N 15.0 0
c
NPS  1000000      $-- particle histories
c
```

Figure 23: MCNP Physics Options Specification Code

The file was run from the MCNP command prompt using the MCNPX executable. The desired output of the first run was a *wwout* file specifying the weight windows to be used in the next run to determine actual tally & dose values. Figure 24 shows the output file after running the weight window generator.

```
wwn1:n   1.9967E+00  5.0000E-01 -1.0000E+00  1.1385E+00  4.2574E+00
         2.1246E+00  1.1786E+02
```

Figure 24: Weight Window Generator output file

This output was plugged into the input file by using the weight window parameter (WWP) card. The WWP card is defined as *WWP: p* W_u W_s *MXSP MWHERE SWITCHN*

MTIME, where defaults are *WWP: N* $5\ 0.6{*}W_u\ 5\ 0\ 0\ 0$ [8]. These values were used and the code can be viewed in Figure 25.

```
c WWG 15 2
WWP:N 5 3 5 0 0 0
wwn1:n    1.9967E+00  5.0000E-01 -1.0000E+00  1.1385E+00  4.2574E+00
          2.1246E+00  1.1786E+02
```

Figure 25: MCNP WWP Specification Code

After the WWP card was run, the results were extracted from the output file and a set of post-processing calculations were applied. These methods will be described in Chapter 5. The full code is shown in Appendix A.

Chapter 4: Validation Experiments

An experiment was conducted by exposing neutron and gamma dosimeters to a 14 MeV neutron generator operating in pulse mode for 4 hours. A PRESCILA neutron probe was used to ensure locations selected for dosimeters would provide measureable doses in the allotted time as well as to provide an estimate of dose. The results from this experiment were reviewed against the results of the MCNP model to check for errors.

Radiation Safety Precautions

Administrative controls in place include a locked door with restricted access, a locked fence in the hallway at 5 mrem/hr to block people from entering the experimental area during irradiation, and a safety interlock on the neutron generator configured to shut down the neutron generator if someone enters during operation. [7].

The neutron generator is equipped with several safety features including a key lock, emergency on/off, and a pressure switch. The interlocks are in series, thus breaking if any one shuts down the neutron generator. The key lock is located on the electronics control box. There are three emergency on/off switches: two physical and one virtual. The two physical ones are located on the electronics enclosure and the remote high voltage power supply. The virtual switch exists on the user control interface as a red block located in the lower corner on the computer station. The pressure switch is setup to shut down the neutron generator if the SF_6 pressure inside the housing drops below 80 psi (normal is 120 psi) [7]. Figure 26 shows the location of each of the neutron generators safety features.

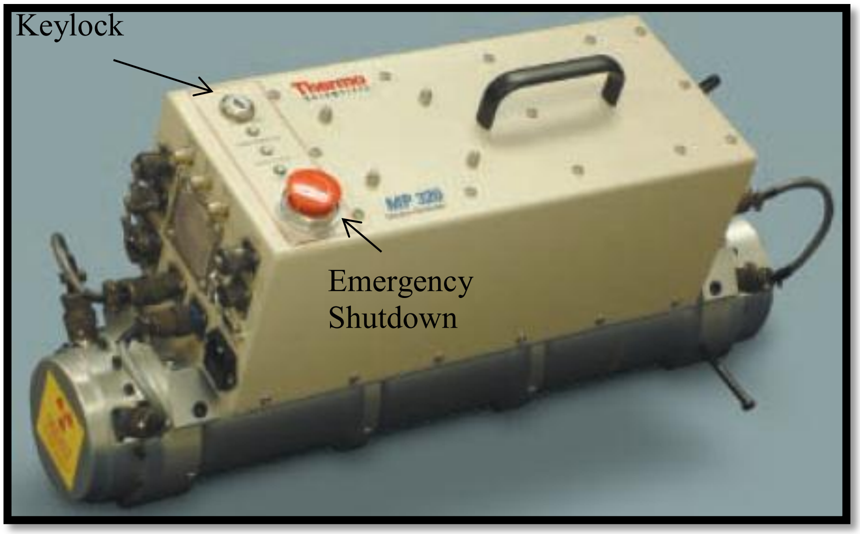

Figure 26: Safety Features of the Thermo MP 320 Neutron Generator [7]

The room is also equipped with a Model 42-41L PRESCILA neutron detector to monitor area dose that is connected to a visual display which sits outside of the enclosed room in the small hallway [7].

EXPERIMENTAL SETUP

The items used in this experiment were 20 Landauer Luxel®+ dosimeters rated at 20 mrem to 2500 mrem, duct tape, a Thermo Scientific MP 320 neutron generator, string, and a Ludlum Model 42-41 PRESCILA neutron detector.

The string was used to create a grid over top of the exposed portion of the neutron generator setup to hang 17 dosimeters at various intervals along it. Duct tape was used to fasten the remaining 3 dosimeters to the outside of the concrete structure. In order to verify the positions of the dosimeters, measurements were taken with the PRESCILA probe to determine dose rates in potential dosimeter locations. The measured dose rate was used to ensure that the dosimeter's minimum readable dose was reached and the dosimeter's maximum readable dose was not exceeded in a four hour time frame. The dosimeters have a readable neutron range from 20 mrem to 25000 mrem [16]. Thus, if irradiating for four hours, the minimum dose rate surveyed needed to exceed 5 mrem per hour and remain under 5000 mrem per hour. Table 6 shows the neutron dose rates measured at the final position of the dosimeters.

Table 6: Initial Dose Rate Measurements Surveyed with PRESCILA Probe

Dosimeter Number	**Neutron dose rate (mrem/hr)**
1	790.0
2	565.0
3	290.0
4	495.0
5	350.0
6	620.0
7	335.0
8	490.0
9	290.0
10	5.5
11	6.0
12	22.0
13	15.0
14	13.0
15	18.0
16	60.0
17	95.0
18	32.0
19	195.0
20	66.0

Each dosimeter came with a clip and was labeled 1 – 20. The clips were used to fasten the 17 dosimeters to the string at 1 meter intervals from the origin. The remaining three dosimeters were placed on the outside of the concrete structure at different locations. Figure 27 shows the position in centimeters of the 17 dosimeters placed on the string grid.

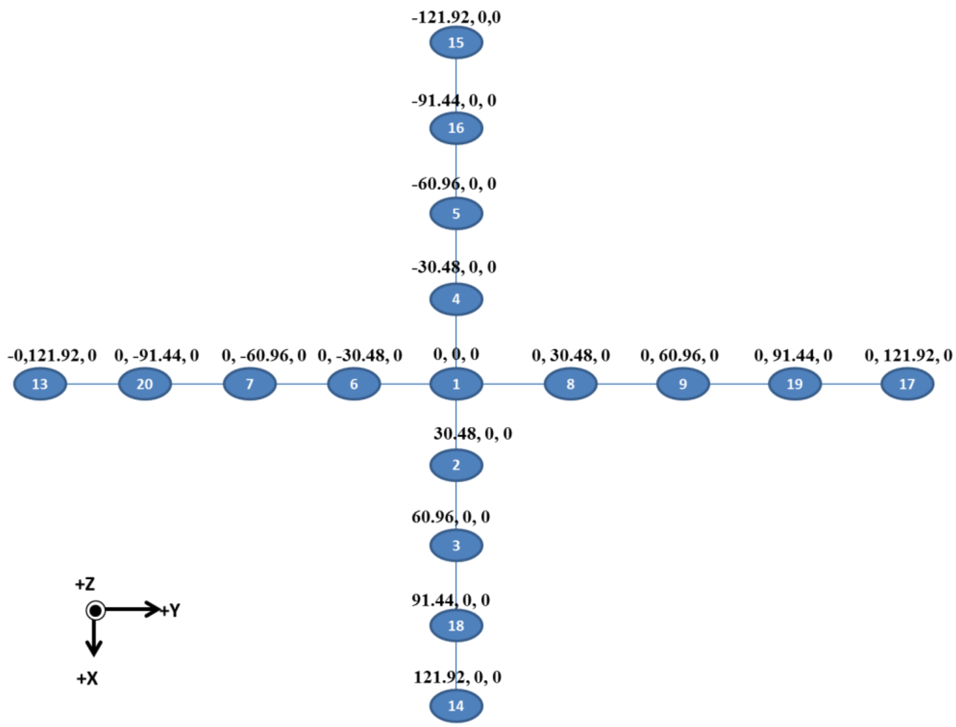

Figure 27: Experimental Dosimeter Positions (cm) from Origin

The remaining three dosimeters were placed on the outside of the concrete at (0, -83.82, -43.815), (0, 114.3, -43.815), and (91.44, 0, -43.815).

RUNNING THE EXPERIMENT

Once the locations of the dosimeters were confirmed, the area was closed off. The neutron generator, set in pulsed operation with 50% duty time at 250 Hz, was turned on for four hours to irradiate the dosimeters to reach detectable measurement limits. At the conclusion of four hours, the dosimeters were removed and shipped off to Landauer for analysis in their labs. An un-irradiated control dosimeter was included in the shipment to monitor for any additional irradiation received in transit. The results were returned and a summarized version is shown in the Chapter 6, while the full report is shown in Appendix B.

Chapter 5: Results

MCNP RESULTS

After running an MCNP model, an output file is generated and the tallies are located at the end of the file. Since a dose modifier, DF, was used to change the original F5 tally into a dose, it is very important to pay close attention to the units. In the background of MCNP, the DF card breaks out the flux of the F5 tally output into energy bins and interpolates against a set of flux-to-dose conversion factors that are functions of energy. By doing this, the correct dose function value to multiply the flux against is obtained. The flux-to-dose conversion factors used in the DF card have units of (rem/hr)/(n/cm^s-s), which are multiplied by the tally output that has units of n/cm^2/SP. After multiplying the two together, the final units become (rem/hr)*(s/SP). Finally, experimental parameters like neutron rate and time are multiplied to obtain a dose. The calculation performed to convert the F5 tally output into a dose in millirem is shown below.

$$\begin{aligned} \textit{Modified Tally Ouput}\ (\text{rem/hr})(\text{s/SP}) = \quad & \\ & \textit{Tally Output}\ (\text{particles/cm}^2)\ x \\ & \textit{Dose Function}\ (\text{rem/hr})/(\text{n/cm}^2 - \text{s}) \end{aligned} \tag{1}$$

$$\begin{aligned} \textit{Dose}\ (\text{mrem}) = \quad & \\ & \textit{Modified Tally Ouput}\ (\text{rem/hr})(\text{s/SP})\ x \\ & \textit{Neutron Rate}\ (\text{n/s})\ x \\ & \textit{Time}\ (\text{hr})\ x \\ & 1000\ \text{mrem/rem} \end{aligned} \tag{2}$$

In Table 7, the modified tally output values, as calculated by the DF card in MCNP output file, can be seen in the third column. The neutron rate and time used in the calculation are shown in the next two columns. The last column shows the results of the final calculated dose in mrem.

Table 7: MCNP Tally Output & Post Processing

Dosimeter Number	Distance from Source (cm)	Modified Tally Output at 1,000,000 particles (rem/hr)*(s/sp)	Neutron Rate (n/s)	Time (hr)	MNCP Modeled Dose (mrem)
1	40.01	9.90E-09	5.00E+07	4	1979
2	50.29	6.72E-09	5.00E+07	4	1343
3	72.91	2.97E-09	5.00E+07	4	594
4	50.29	6.73E-09	5.00E+07	4	1345
5	72.91	2.96E-09	5.00E+07	4	592
6	50.29	6.71E-09	5.00E+07	4	1341
7	72.91	2.94E-09	5.00E+07	4	588
8	50.29	6.86E-09	5.00E+07	4	1371
9	72.91	3.60E-09	5.00E+07	4	720
10	114.36	1.60E-11	5.00E+07	4	3
11	91.52	2.94E-11	5.00E+07	4	6
12	83.91	4.19E-11	5.00E+07	4	8
13	128.32	1.16E-10	5.00E+07	4	23
14	128.32	1.26E-10	5.00E+07	4	25
15	128.32	1.29E-10	5.00E+07	4	26
16	99.81	5.08E-10	5.00E+07	4	102
17	128.32	9.30E-10	5.00E+07	4	186
18	99.81	5.06E-10	5.00E+07	4	101
19	99.81	1.79E-09	5.00E+07	4	359
20	99.81	4.91E-10	5.00E+07	4	98

Due to the configuration of the neutron generator, which was set in pulsed mode with a 50% duty cycle, the neutron rate was estimated as 5×10^7 n/s. The irradiation time for the experiment was four hours. Multiplying the MCNP tally output by the neutron rate and time gives final units of mrem. The predicted output ranges from 1979 mrem at the unshielded dosimeter located directly above the neutron generator to 3 mrem at a dosimeter located on the outside of the concrete structure. The doses at the locations representative of the 17 dosimeters on the string above the neutron generator were plotted against the point detector coordinates to gain an idea of symmetry about x=0 and y=0 within the dose maps in Figures 28 and 29.

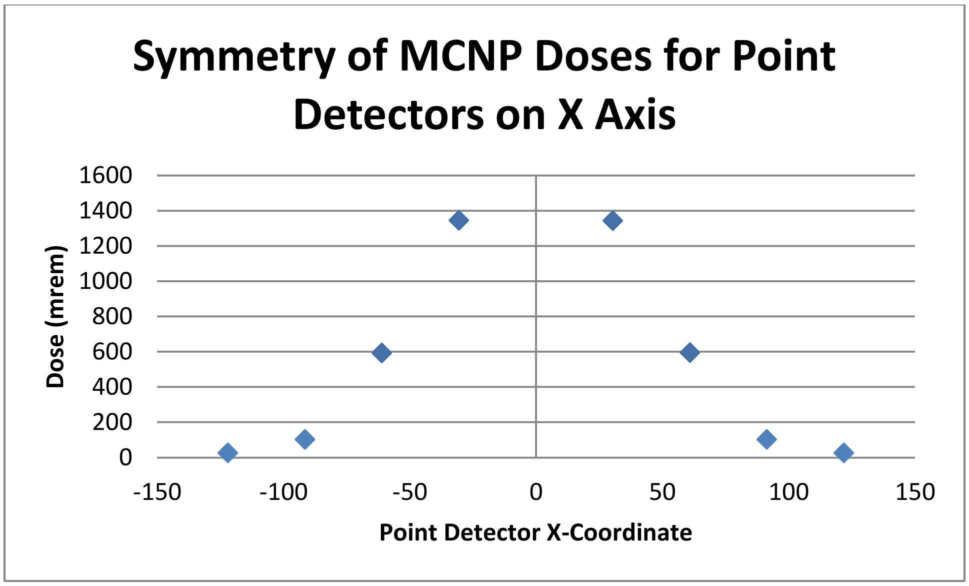

Figure 28: Plot of Symmetry Present in Dose Maps – X Axis

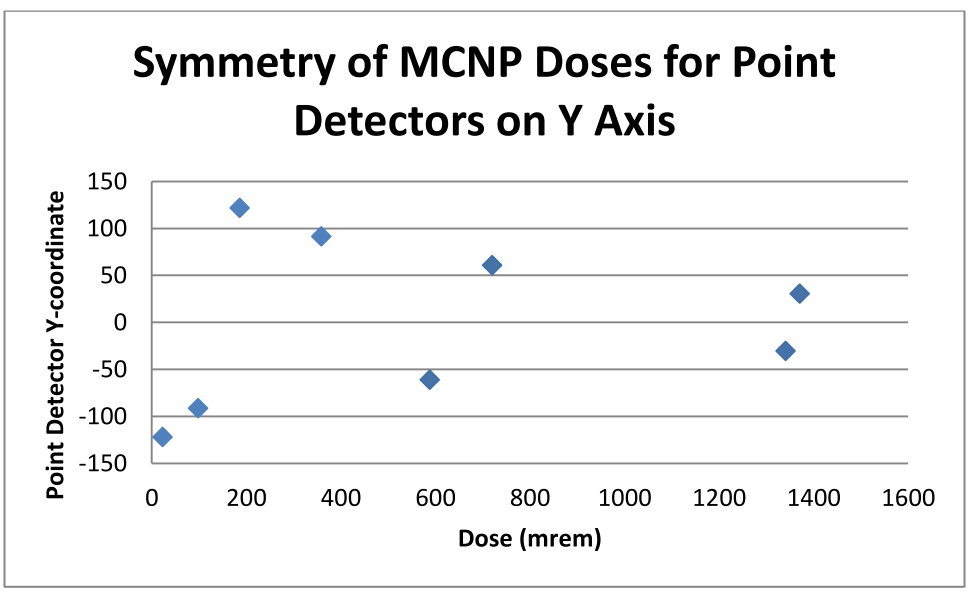

Figure 29: Plot of Symmetry Present in Dose Map – Y Axis

Looking at the above plots, it can be seen that significant symmetry exists in the MCNP predicted doses, which is expected because the neutron generator shielding facility is symmetric in its configuration.

EXPERIMENTAL RESULTS

Landauer provided a Radiation Dosimetry Report outlining the thermal and fast neutron deep, eye, and shallow dose equivalents for each of the 20 dosimeters. Deep dose equivalent (DDE) applies to the external whole body exposure at a tissue depth of 1 cm, eye dose equivalent (LDE) applies to external exposure of the lens at a tissue depth of 0.3 cm, and the shallow dose equivalent (SDE) applies to the external exposure of the skin or extremity at a tissue depth of 0.0007 cm [16]. According to a Landauer representative, a calibration factor is used depending on the energy of the neutrons. For 14 MeV neutrons, the calibration factor used is 1.7. To compute the corrected total neutron dose, the listed fast neutron dose (NF) is multiplied by 1.7 and then added to the thermal neutron dose (NT). For example, performing the calculation on dosimeter 1, the DDE NF is (2030 x 1.7) + the DD NT of 90 = 3541 mrem. This calculation was performed for the remaining 19 dosimeters, and the results are shown in Table 8.

Table 8: Post Processing for Dosimeter output

Dosimeter Number	DDE NF	DDE NF * 1.7	DDE NT	Dosimeter Experimental Dose (mrem)
1	2030	3451	90	3541
2	580	986	30	1016
3	70	119	0	119
4	420	714	20	734
5	290	493	10	503
6	650	1105	30	1135
7	220	374	10	384
8	530	901	30	931
9	250	425	20	445
10	<20	<20	<20	<20
11	<20	<20	<20	<20
12	<20	<20	<20	<20
13	<20	<20	<20	<20
14	<20	<20	<20	<20
15	<20	<20	<20	<20
16	30	51	0	51
17	<20	<20	<20	<20
18	<20	<20	<20	<20
19	100	170	0	170
20	<17	<18	<19	<20

Results below the minimum measureable dose equivalent were indicated with an "M" on the dosimeter report, and shown as less than 20 mrem in Figure 31 because the minimum measureable dose equivalent on this type of dosimeter was 20 mrem. The output ranges from 3541 mrem at the unshielded dosimeter above the neutron generator to less than 20 mrem at the dosimeters located at the edge of the concrete structure. The doses at the locations representative of the 17 dosimeters on the string above the neutron generator were plotted against the point detector coordinates to gain an idea of symmetry about x=0 and y=0 within the dose maps in Figures 30 and 31.

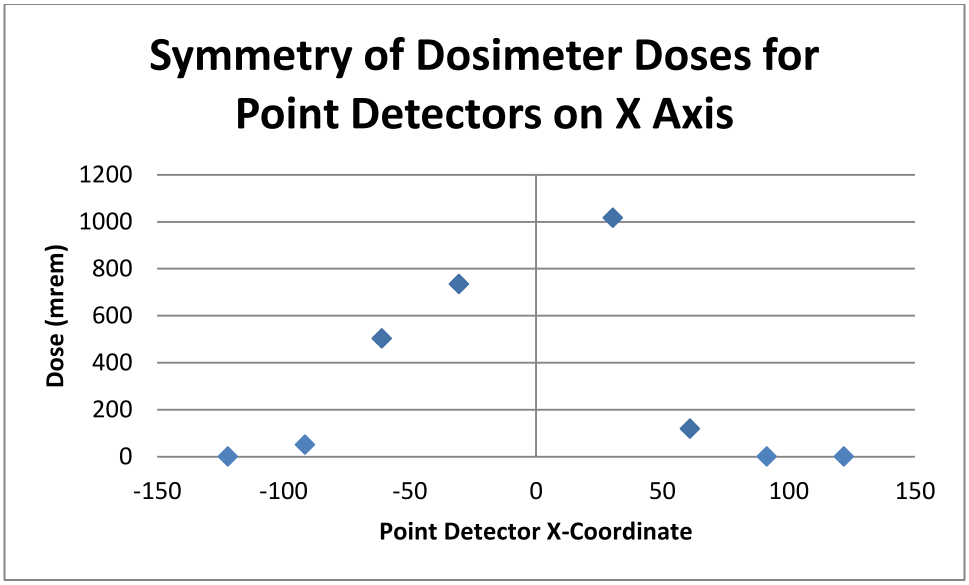

Figure 30: Plot of Symmetry Present in Dose Maps – X Axis

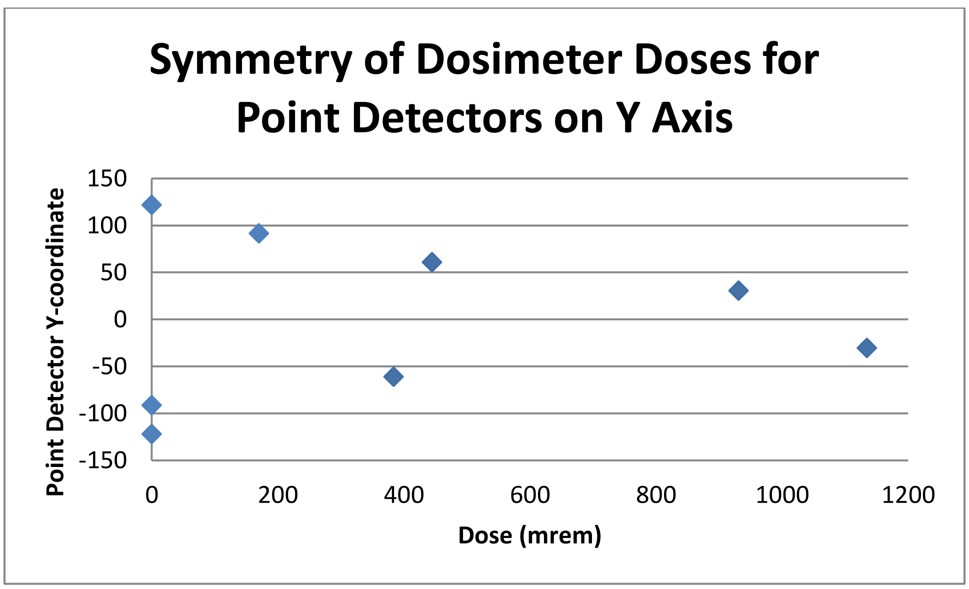

Figure 31: Plot of Symmetry Present in Dose Maps – Y Axis

Just like the MCNP predicted doses, the dosimeter symmetry plots show significant symmetry exists in the dosimeter predicted doses, which is expected because the neutron generator shielding facility is symmetric in its configuration

PRESCILA HAND HELD RESULTS

The PRESCILA probe was used to measure the neutron dose rate at the various locations of the dosimeters. This dose rate was output in mrem/hr and, in order to predict the dose to a person standing in the various dosimeter locations for the entire experiment, the dose rate was multiplied by the experimental fours that the neutron generator was run. Table 9 shows the calculations performed.

Table 9: PRESCILA Post Processing.

Dosimeter Number	PROBE Measured Dose Rate (mrem/hr)	PROBE Measured Dose (mrem)
1	790	3160
2	565	2260
3	290	1160
4	495	1980
5	350	1400
6	620	2480
7	335	1340
8	490	1960
9	290	1160
10	5.5	22
11	6	24
12	22	88
13	15	60
14	13	52
15	18	72
16	60	240
17	95	380
18	32	128
19	195	780
20	66	264

The predicted dose in mrem is shown in the last column.

COMPARISON

In order to understand if the MCNP model predicts doses aligned with experimental dosimeter values and measurements by the PRESCILA probe, the MCNP doses were plotted against the dosimeter and PRESCILA doses in Figure 32.

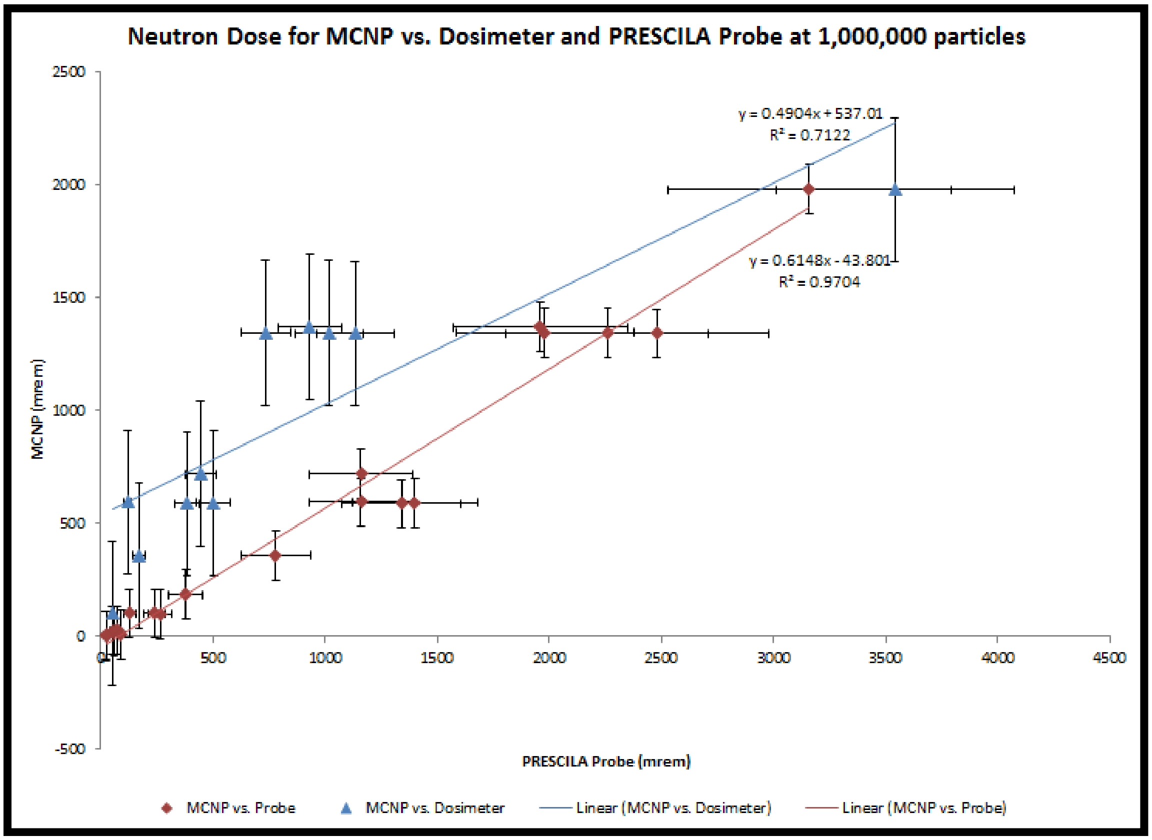

Figure 32: MCNP versus PRESCILA probe and Dosimeter data

When plotting the MCNP output versus the PRESCILA output, the trend line follows a linear trend indicating that the difference between the two is a consistent factor. When plotting the MCNP output versus the experimental dosimeter data, the trend line does not fit a linear trend as well as before. There is some disagreement; however, based on assumptions used to calculate the dose on the dosimeters, it is far more expected to see a visible deviance. The error in the MCNP values is the standard error found through a regression analysis. The error in the PRESCILA was estimated at the ANSI calibration standard of 20%, and the error in the dosimeters was estimated at 15% based on the provided photon accuracy in the calibration sheet [9].

The source was modeled as an isotropic disc source and the dose reduction away from the source is expected to follow somewhat closely to the inverse square law. The doses predicted by each of the modes (Dosimeter, MCNP, and PRESCILA) were plotted in Figure 33 against the distance from the source.

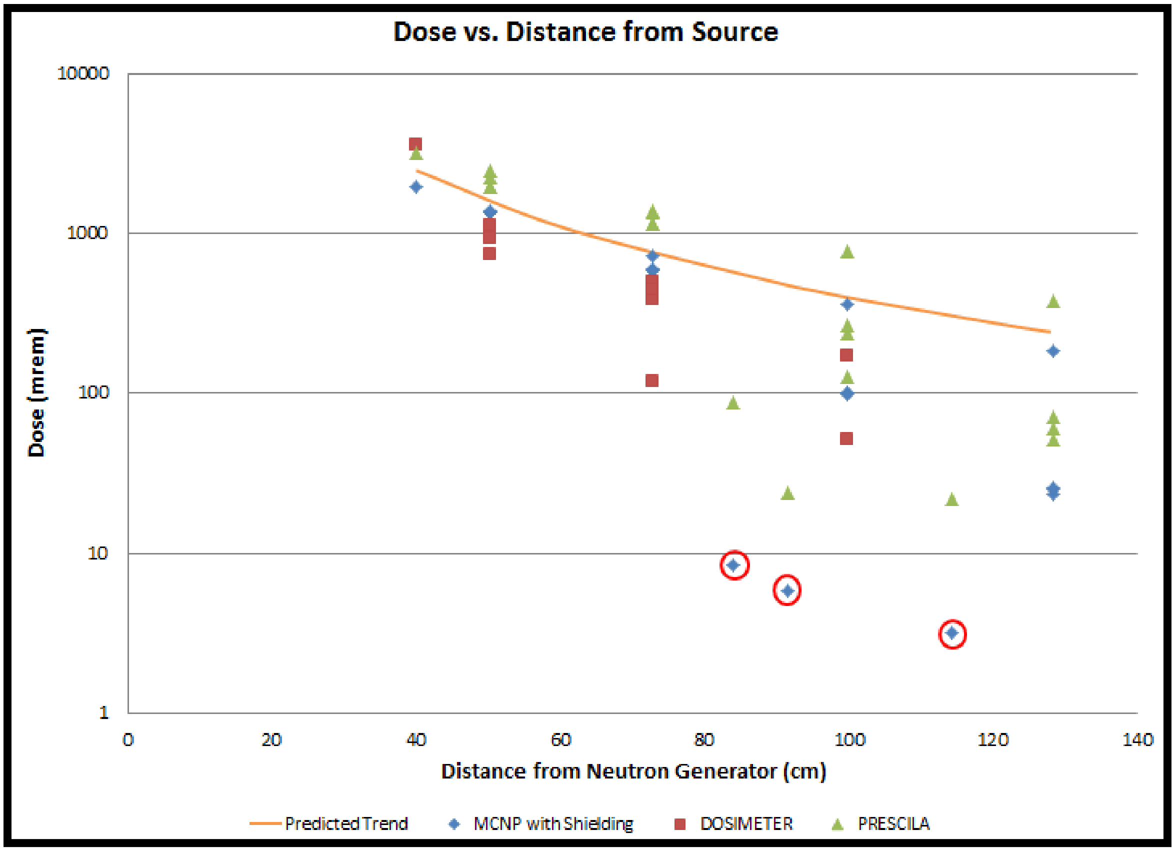

Figure 33: Doses versus Distance.

A predicted trend line following a $\frac{1}{4\pi r^2}$ reduction was added as an orange line to the graph. The one tracking most closely to a line is the MCNP predicted doses. There are three points from the MCNP data that fall well outside of the predicted trend line; these are circled above in red. These three represent the dose values from the dosimeters that were taped onto the side of the concrete blocks. It is expected due to shielding that the dose from these would be well below the predicted line. It is not expected that the results for MCNP, PRESCILA, and the dosimeters track perfectly to the trend due to the shape of the target in the neutron generator and potential scattering effects from the shielding configuration.

In order to understand how the results deviate from an unshielded configuration, all shielding was removed and the MCNP model re-run. The doses were calculated for the same positions and plotted on Figure 34.

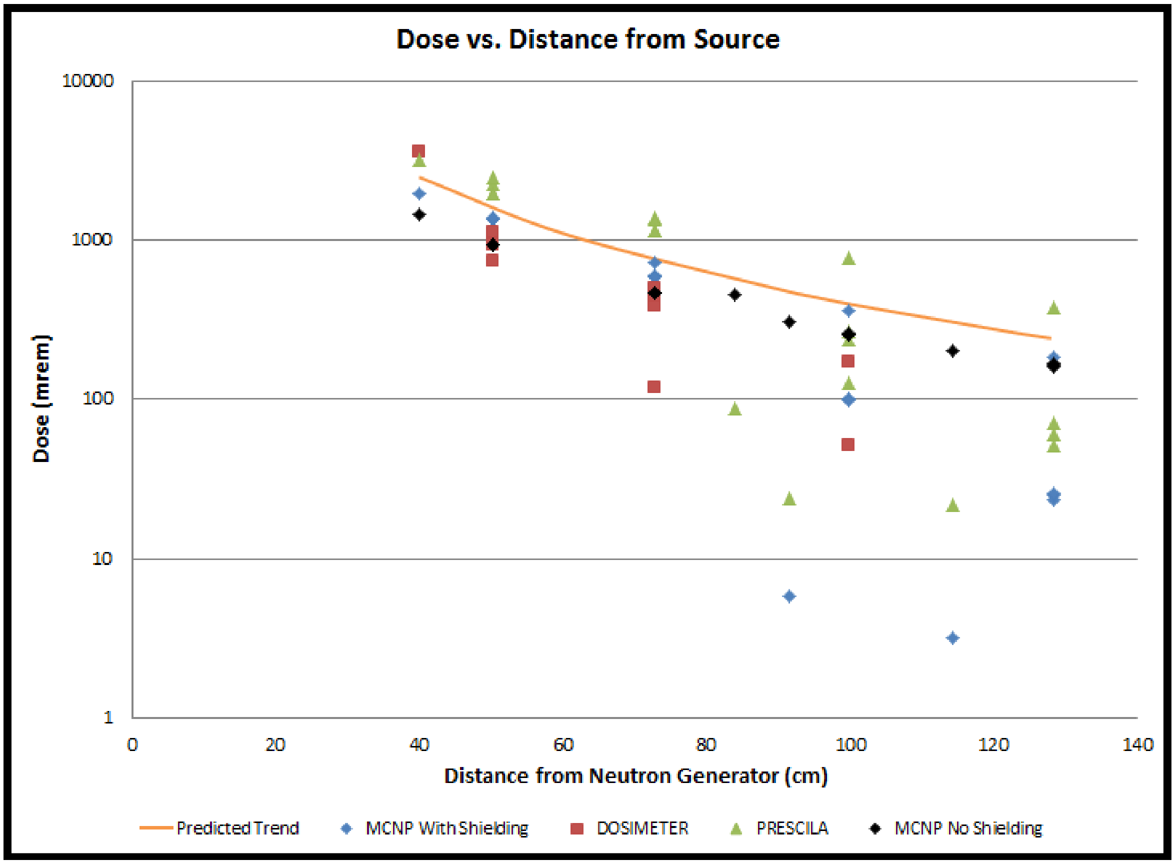

Figure 34: Dose versus Distance with MCNP No Shielding

The black points represent the doses at the various locations for an unshielded neutron generator. The dose values are lower in the unshielded setup at the selected locations because there is less reflection from the concrete. It can also be seen that the unshielded values track much closer to a $\frac{1}{4\pi r^2}$ trend.

Chapter 6: Conclusions

Figure 32 showed that the different methods varied by some consistent factor. Noting that in Landauer's analysis of the dosimeters, a conservative assumption was applied that all neutrons hitting the dosimeter are at 14 MeV. This is not an accurate representation of what is actually occurring because the neutrons are undergoing many different interactions within the shielding material, and the energy is reduced as a consequence. Figure 35 shows four different MCNP point source plots where the flux was plotted as a function of energy.

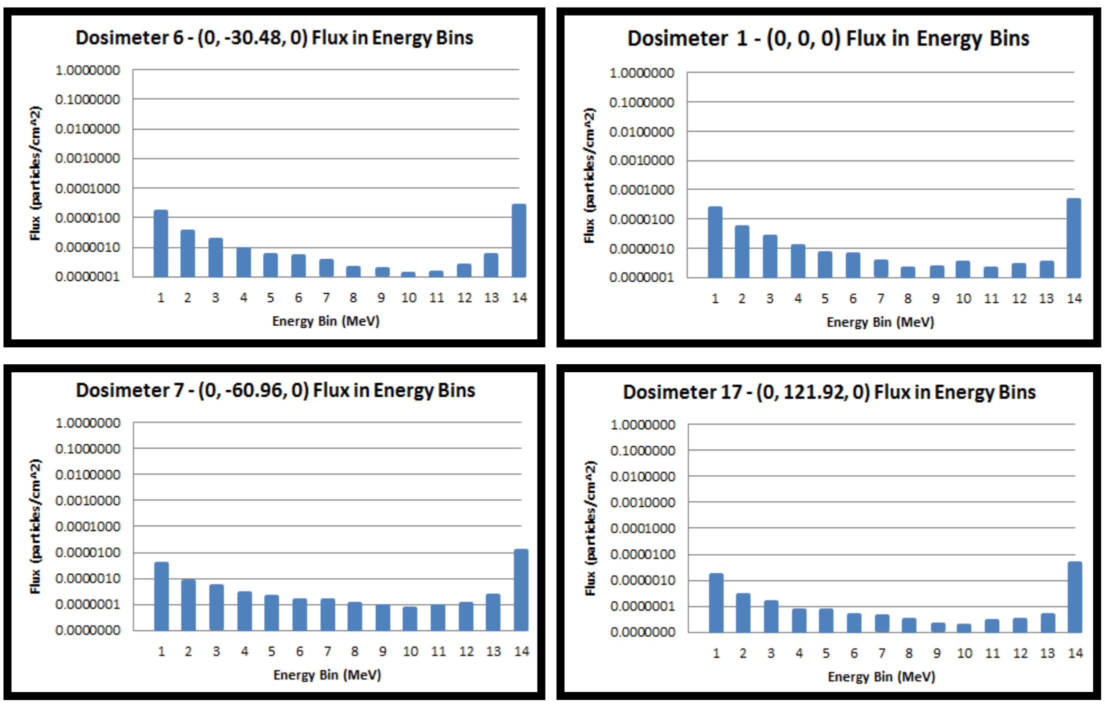

Figure 35: Flux contribution for Dosimeters 1, 6, 7, and 17

These plots show that a considerable amount of the neutrons undergo interactions that reduce their energy. At lower energies, the cross sections are higher, thus increasing the reaction rates. Since the Landauer dose calculation is not accounting for these lower energy particles, their reported value, after normalizing to their 1.7 calibration factor, is higher than the MCNP value. When calculating doses received by radiation workers, it is always best to err on the side of conservatism; however, it is important to understand why the doses are higher than MCNP predicts.

Figure 34 shows that there a significant effects from shielding that reduce the doses at locations on the concrete shield beyond a simple $\frac{1}{4\pi r^2}$ trend. At locations above the neutron generator where shielding does not completely cover the neutron generator, there is reflection occurring that causes the doses to be higher than in the unshielded configuration. The doses at the locations above the neutron generator still follow closely to a $\frac{1}{4\pi r^2}$ trend.

To put the neutron doses from the neutron generator into perspective, the annual whole body radiation limit to radiation workers is 5000 mrem [17]. The highest dose in four hours predicted by MCNP model was 1979 mrem, from the PRESCILA was 2260 mrem, and from the dosimeters was 3541 mrem. If an experimenter stood 15.75 inches above the neutron generator with the neutron generator operating in pulse mode for four hours, he/she would not receive the yearly limit of whole body dose. In order to exceed the limit, an experimenter would have to be 15.75 inches above the operational neutron generator for 5.7 hours (using the most conservative measurement – dosimeters).

Merely four feet from the neutron generator the highest predicted dose from the PRESCILA probe was 380 mrem. Based on those numbers, an experimenter standing four feet from the operational neutron generator would have to stand there for 53 hours to exceed the 5,000 mrem/yr limit. As previously discussed, there are multiple layers of safeguards in place to alert experimenters of an operational neutron generator as well as prevent non-experimenters from entering the facility and room. A situation where an experimenter would accidentally be exposed to his/her maximum yearly limit is highly unlikely given the predicted dose rates and safeguards.

Appendix A

Appendix A includes the MCNP code used to model the shielded configuration of the neutron generator.

```
Neutron Generator - Isotropic 14MeV neutron source in shielding
c
c ******************** File Description ******************************
c
c Description: Place dosimeters around a 14 MeV cylindrical neutron source
c            surrounded by concrete, steel, and borated poly layers
c
c Author: Kristen McConnell
c
c Modifications: 9/08/2013 - Created concrete geometry, outside wall
c              9/12/2013 - Added outer brick geometry
c              9/22/2013 - Added Source (Alex Fay's)
c              9/25/2013 - Modified Alex Fay's source
c              10/7/2013 - Added iron block beneath neutron source
c              10/9/2013 - Changed the way the problem boundary was set up to reflect a more realistic wall
c              10/9/2013 - Change the problem boundary to concrete & make a new problem boundary
c              10/13/2013 - Added Energy Bins
c              10/13/2013 - Added Weight Windows
c              10/30/2013 - Changed leaded concrete density to 4.11
c
c ***  Cell Cards ***
c
c # mat#   mat_rho    surface_relations              importances  comments
1   400   -2.300000  -10 12 13 14 15 16 17 18 19 26 28 imp:n=1      $concrete cube of bricks
2   100   -0.001205   (10 -11 20 21 22 23 24 25):-28   imp:n=1      $air
3   000          27                          imp:n=0      $problem boundary (outside concrete wall)
4   200   -7.874000   -12:-13:-14:-15:-26 28          imp:n=1      $iron sheets
5   300   -1.000000   -16:-17:-18:-19                imp:n=1      $borated poly sheets
6   600   -4.110000   -20:-21:-22:-23:-24:-25         imp:n=1      $leaded concrete
7   400   -2.300000   -27 11                        imp:n=1      $concrete surrounding walls
```

```
c *** Surface Cards ***
c
10  BOX -88.9 -83.82 -154.305 177.8 0 0 0 198.12 0 0 0 114.3        $ concrete cube (bricks)
11 BOX -180.34 -297.18 -154.305 914.4 0 0 0 914.4 0 0 0 914.4       $ inside concrete wall
12 BOX -27.78125 -22.86 -53.6575 55.5625 0 0 0 7.62 0 0 0 13.6525   $ iron sheet
13 BoX -22.70125 -15.24 -53.6575 7.62 0 0 0 60.96 0 0 0 13.6525     $ iron sheet
14 BOX 15.08125 -15.24 -53.6575 7.62 0 0 0 60.96 0 0 0 13.6525      $ iron sheet
15 BOX -27.78125 45.72 -53.6575 55.5625 0 0 0 5.08 0 0 0 13.6525    $ iron sheet
16 BOX -27.78125 -27.94 -53.6575 55.5625 0 0 0 5.08 0 0 0 13.6525   $ poly sheet
17 BOX -27.78125 -15.24 -53.6575 5.08 0 0 0 60.96 0 0 0 13.6525     $ poly sheet
18 BOX 22.70125 -15.24 -53.6575 5.08 0 0 0 60.96 0 0 0 13.6525      $ poly sheet
19 BOX -27.78125 50.8 -53.6575 55.5625 0 0 0 5.08 0 0 0 13.6525     $ poly sheet
20 BOX -27.78125 -57.15 -40.005 71.755 0 0 0 29.21 0 0 0 14.605     $ leaded concrete
21 BOX -41.75125 -44.7675 -40.005 13.97 0 0 0 29.5275 0 0 0 14.605  $ leaded concrete
22 BOX -57.6263 -15.24 -40.005 29.845 0 0 0 71.12 0 0 0 14.605      $ leaded concrete
23 BOX -57.9438 55.88 -40.005 85.725 0 0 0 29.21 0 0 0 14.605       $ leaded concrete
24 BOX 27.7813 -27.94 -40.005 13.97 0 0 0 88.9 0 0 0 14.605         $ leaded concrete
25 BOX 41.7513 -11.43 -40.005 13.97 0 0 0 58.42 0 0 0 14.605        $ leaded concrete
26 BOX -15.08125 -15.24 -61.2775 30 0 0 0 60.96 0 0 0 7.62          $ iron sheet (underneath)
27 BOX -271.78 -388.62 -245.745 1097.28 0 0 0 1097.28 0 0 0 1097.28 $ outside concrete wall
28 BOX -15.07 -15.24 -53.6575 30 0 0 0 60.96 0 0 0 13.6525          $ air around neutron gen

c *** Data Cards ***
c
c  ----- Materials
c
c     -- AIR
c
M100  6000    0.000124   $ carbon
      7014    0.755526   $ nitrogen
      8016    0.231781   $ oxygen
      18000   0.012827   $ argon
c
c     -- IRON
c
M200  26000  1.000000   $ this one should be obvious
c
c     -- BORATED POLYETHYLENE, 10 wt% B4C
c
M300  1001   0.627759   $ hydrogen (in the poly)
      5011   0.046690   $ boron (in the B4C)
      6000   0.325552   $ carbon (both poly and B4C)
c
```

```
c
c     -- CONCRETE, ORDINARY (NIST)
c
M400  1001   0.305330   $ hydrogen
      6000   0.002880   $ carbon
      8016   0.500407   $ oxygen
      11023  0.009212   $ sodium
      12000  0.000725   $ magnesium
      13027  0.010298   $ aluminum
      14000  0.151042   $ silicon
      19000  0.003578   $ potassium
      20000  0.014924   $ calcium
      26000  0.001605   $ iron
c
c     -- LEADED CONCRETE (20% lead by mass)
c
M600  1001   -0.0177391   $ hydrogen
      6000   -0.0020870   $ carbon
      8016   -0.4598261   $ oxygen
      11023  -0.0121739   $ sodium
      12000  -0.0010435   $ magnesium
      13027  -0.0160000   $ aluminum
      14000  -0.2438261   $ silicon
      19000  -0.0080000   $ potassium
      20000  -0.0344348   $ calcium
      26000  -0.0052174   $ iron
      82000  -0.1996522   $ lead
c  ----- Physics Options (With Importances if Necessary)
c
MODE N
PHYS:N 15.0 0
c
NPS  10000      $-- particle histories
c
c
SDEF  POS=0 0 -41.005 AXS=0 1 0 EXT=0 RAD=d1 PAR=1 ERG=14.0
      ARA=3.14159
```

```
SI1 0 1
SP1 -21 1
c
F15:N  0 0 0 0.5    &  $Dosimeter 1
-121.92 0 0 0.5     &  $Dosimeter 15
-91.44 0 0 0.5      &  $Dosimeter 16
-60.96 0 0 0.5      &  $Dosimeter 5
-30.48 0 0 0.5      &  $Dosimeter 4
30.48 0 0 0.5       &  $Dosimeter 2
60.96 0 0 0.5       &  $Dosimeter 3
91.44 0 0 0.5       &  $Dosimeter 18
121.92 0 0 0.5      &  $Dosimeter 14
0 30.48 0 0.5       &  $Dosimeter 8
0 60.96 0 0.5       &  $Dosimeter 9
0 91.44 0 0.5       &  $Dosimeter 19
0 121.92 0 0.5      &  $Dosimeter 17
0 -30.48 0 0.5      &  $Dosimeter 6
0 -60.96 0 0.5      &  $Dosimeter 7
0 -91.44 0 0.5      &  $Dosimeter 20
0 -121.92 0 0.5     &  $Dosimeter 13
91.44 0 -43.815 0.5 &  $Dosimeter 11
0 114.3 -43.815 0.5 &  $Dosimeter 10
0 -83.82 -43.815 0.5   $Dosimeter 12
DF15 ic=10 iu=1 LOG
c WWG 15 2
WWP:N 5 3 5 0 0 0
wwn1:n    1.9967E+00  5.0000E-01 -1.0000E+00  1.1385E+00  4.2574E+00
          2.1246E+00  1.1786E+02
print110
print30
```

Appendix B

Appendix B contains the radiation dosimetry report provided by Landauer after analyzing the 20 dosimeters.

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Page	1 of 3
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Copy / Version	1 / 1

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Radiation Dosimetry Report

Account : 6470

**No NVLAP accreditation is available from NVLAP for thermal neutron or X type dosimeters. When exposure results are reported for thermal neutrons or X type dosimeters, this report contains data that are not covered by the NVLAP accreditation.

Copy 1 : Original sent to UNIVERSITY OF TEXAS, AT AUSTIN, ENV HLTH & SFTY, P O BOX 7729, AUSTIN, TX 7

Participant Number	Name		Dosimeter	Use	Rad. Type	Rad. Quality	Dose Equivalent (mrem) for Periods Shown Below DDE-Deep Dose Equivalent LDE-Lens Dose Equivalent SDE-Shallow Dose Equivalent												Inception Date	Serial Number
	ID Number	Birth Date					Period Shown Below			Quarter to Date			Year to Date			Lifetime to Date				
							DDE	LDE	SDE	DDE	LDE	SDE	DDE	LDE	SDE	DDE	LDE	SDE		
For Monitoring Period:							2013-08-23 to 2013-11-22			QUARTER 3			2013			LIFETIME				
00000	CONTROL		Ta	CNTRL																7731995B
	Control Dose Used		Ta				7	7	7											
00004	PHYSICIAN 1		Ta	AREA	PN		2253	2253	2248										2013/08	7731996B
					P	M	133	133	128											
					N	T	90	90	90											
					N	F	2030	2030	2030											
00006	PHYSICIAN 2		Ta	AREA	PN		702	703	703										2013/08	7731997B
					P	M	92	93	93											
					N	T	30	30	30											
					N	F	580	580	580											
00007	PHYSICIAN 3		Ta	AREA	PN		87	87	86										2013/08	7731998B
					P		17	17	16											
					N	T	M	M	M											
					N	F	70	70	70											
00008	PHYSICIAN 4		Ta	AREA	PN		523	523	523										2013/08	7731999B
					P	H	83	83	83											
					N	T	20	20	20											
					N	F	420	420	420											
00009	PHYSICIAN 5		Ta	AREA	PN		338	339	339										2013/08	7732000B
					P		38	39	39											
					N	T	10	10	10											
					N	F	290	290	290											
00010	PHYSICIAN 6		Ta	AREA	PN		771	772	772										2013/08	7732001B
					P	M	91	92	92											
					N	T	30	30	30											
					N	F	650	650	650											

This report must not be used to claim product certification, approval, or endorsement by NVLAP, NIST, or any agency of the federal government.

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Page	2 of 3
Analytical Work Order / QC Release	1323911433 / CHA
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Radiation Dosimetry Report

Account : 6470

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Participant Number	Name		Dosimeter	Use	Rad. Type	Rad. Quality	Dose Equivalent (mrem) for Periods Shown Below DDE-Deep Dose Equivalent LDE-Lens Dose Equivalent SDE-Shallow Dose Equivalent												Inception Date	Serial Number
							Period Shown Below			Quarter to Date			Year to Date			Lifetime to Date				
	ID Number	Birth Date					DDE	LDE	SDE	DDE	LDE	SDE	DDE	LDE	SDE	DDE	LDE	SDE		
For Monitoring Period:							2013-08-23 to 2013-11-22			QUARTER 3			2013			LIFETIME				
00011	PHYSICIAN 7		Ta	AREA	PN P N N	 T F	262 32 10 220	263 33 10 220	263 33 10 220										2013/08	7732002B
00012	PHYSICIAN 8		Ta	AREA	PN P N N	 H T F	673 113 30 530	673 113 30 530	667 107 30 530										2013/08	7732003B
00013	PHYSICIAN 9		Ta	AREA	PN P N N	 T F	315 45 20 250	315 45 20 250	315 45 20 250										2013/08	7732004B
00015	PHYSICIAN 10		Ta	AREA			M	M	M										2013/08	7732005B
00016	PHYSICIAN 11		Ta	AREA			M	M	M										2013/08	7732006B
00017	PHYSICIAN 12		Ta	AREA	P P N N	 T F	2 2 M M	2 2 M M	1 1 M M										2013/08	7732007B
00019	PHYSICIAN 13		Ta	AREA			M	M	M										2013/08	7732008B
00020	PHYSICIAN 14		Ta	AREA			M	M	M										2013/08	7732009B
00021	PHYSICIAN 15		Ta	AREA			M	M	M										2013/08	7732010B
00022	PHYSICIAN 16		Ta	AREA	PN P N N	 T F	32 2 M 30	32 2 M 30	31 1 M 30										2013/08	7732011B

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Page	3 of 3
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Copy 1 : Original sent to UNIVERSITY OF TEXAS, AT AUSTIN, ENV HLTH & SFTY, P O BOX 7729, AUSTIN, TX 7

Participant Number	Name		Dosimeter	Use	Rad. Type	Rad. Quality	Dose Equivalent (mrem) for Periods Shown Below DDE-Deep Dose Equivalent LDE-Lens Dose Equivalent SDE-Shallow Dose Equivalent												Inception Date	Serial Number
							Period Shown Below			Quarter to Date			Year to Date			Lifetime to Date				
	ID Number	Birth Date					DDE	LDE	SDE	DDE	LDE	SDE	DDE	LDE	SDE	DDE	LDE	SDE		
For Monitoring Period:							2013-08-23 to 2013-11-22			QUARTER 3			2013			LIFETIME				
00023	PHYSICIAN 17		Ta	AREA	P P N N	 T F	2 2 M M	2 2 M M	2 2 M M										2013/08	7732012B
00024	PHYSICIAN 18		Ta	AREA			M	M	M										2013/08	7732013B
00025	PHYSICIAN 19		Ta	AREA	PN P N N	 T F	121 21 M 100	121 21 M 100	119 19 M 100										2013/08	7732014B
00026	PHYSICIAN 20		Ta	AREA	P P N N	 T F	2 2 M M	2 2 M M	2 2 M M										2013/08	7732015B

Radiation Dosimetry Report

Annual Radiation Exposure Limits (mrem) :

Whole body, blood forming organs, gonads	5,000
Lens of Eye	15,000
Extremities and Skin	50,000
Fetal (Gestation period)	500
General Public	100

Based on the US NRC Regulations, Title 10, Part 20, Code of Federal Regulations and adopted by many states. Certain state and other regulatory agencies may adhere to different limits.
Control Dosimeter: A control dosimeter is included with each shipment of dosimeters for monitoring radiation exposure received during transit. At the customer's facility, store the control in a radiation free area during the wear period.
Minimal Dose Equivalent Reported: Dose equivalents below the minimum measurable quantity for the current monitoring period are recorded as "M." The minimal reporting levels vary by the dosimeter type and radiation quality. "SL" is an elective option for the minimal dose equivalent reported where exposures less than 10 mrem report as "SL" (excludes fetal dosimeters), and/or exposures at or more than 10 mrem begin reporting at 10 mrem and report in increments of 10 mrem.

Dosimeter Type	M (DDE,LDE,SDE)	M (SDE Only)	SL (DDE,LDE,SDE)
Luxel+®	1	-	10
InLight®	5	-	10
Whole Body Beta	-	10	10
Ring	-	30	-
Neutrak® Neutron Fast	20	-	-
Neutrak® Neutron Thermal/Fast	10	-	-

Special Calculations: Special dose calculations can be applied to radiation workers who wear lead aprons.
EDE 1 - two dosimeters: one worn at the waist level under lead apron and one worn at the collar level outside lead apron. 1.5 (Waist DDE) + 0.04 (Collar DDE) = Assigned Deep Dose Equivalent.
EDE 2 - one dosimeter: one worn at the collar level outside lead apron. 0.3 (Collar DDE) = Assigned Deep Dose Equivalent.
EDE 122 - one dosimeter: one worn at the collar level outside lead apron. Collar DDE / 5.6 = Assigned Deep Dose Equivalent.
Calc3 - Lens of Eye dosimeter. 0.5 (Lens of Eye LDE) = Assigned Lens of Eye Dose Equivalent.
Lens.175 - Lens of Eye dosimeter. 0.175 (Lens of Eye LDE) = Assigned Lens of Eye Dose Equivalent.
EDE1-NTC EDE1 without Thyroid Collar assigned deep dose equivalent =
0.06 × (collar dose - waist dose) + waist dose
EDE1-TC EDE1 with Thyroid Collar assigned deep dose equivalent =
0.02 × (collar dose - waist dose) + waist dose
The "ASSIGNED" line follows all of the original whole body dosimeter doses with the EDE 1 or EDE 2 calculation results or Landauer's standard Dose Assessment Protocol (deep and shallow whole body dose from the highest reading whole body dosimeter, lens dose from dosimeter closest to the eye).

Ring Dosimeter Reading: Ring dosimeter readings report as a shallow dose.

Fetal Dosimeter: A declared pregnant worker will possess a fetal exposure on an extra page of the report based upon the whole body dosimeter worn closest to the fetus. The fetal dose is reported for the current wear period, plus the estimated dose from conception to declaration (if provided by customer), and the total dose from declaration to present.

Use	Description	Use	Description
AREA	Area Monitor	OEXTRM	Other Extremity
CHEST	Chest	OWHBDY	Other Whole Body
CNTRL	Control	RANKLE	Right Ankle
COLLAR	Collar	RFINGR	Right Hand Ring
EYE	Eye	RUARM	Right Upper Arm
FETAL	Fetal	RULEG	Right Upper Leg
LANKLE	Left Ankle	RWRIST	Right Wrist
LFINGR	Left Hand Ring	SPECL	Special Purpose
LUARM	Left Upper Arm	UPBACK	Upper Back
LULEG	Left Upper Leg	WAIST	Waist
LWBACK	Lower Back	WHBODY	Whole Body
LWRIST	Left Wrist		

Code	Radiation Quality Description (Type and/or Energy)
B	beta
BH	beta high energy, e.g. Strontium, Phosphorus
BL	beta low energy e.g. Thallium, Krypton
BS	Strontium beta
BT	Thallium beta
BU	Uranium beta
BN	beta, neutron mixture
NF	neutron fast
NT	neutron thermal
P	photon (x or gamma ray)
PB	photon, beta mixture
PBN	photon, beta, neutron mixture
PH	photon high energy greater than 200 keV
PL	photon low energy less than 40 keV
PM	photon medium energy 40 keV to 200 keV
PN	photon, neutron mixture

First Line Explanation
Participant Number: Unique number assigned by Landauer.
Name: Participant to whom the dosimeter is assigned.
Dosimeter: Badge type according to radiation monitoring needs.

Dosimeter	Code	Type of Radiation Monitored				
		Photons			Neutrons	
		X	Gamma	Beta	Fast	Fast/ Thermal
InLight Basic	B1P	Yes	Yes	Yes		
InLight Basic	B4P	Yes	Yes	Yes		
InLight Basic	B2C	Yes	Yes	Yes		
InLight Basic	B4C	Yes	Yes	Yes		
InLight LDR	L2P	Yes	Yes	Yes		
InLight LDR	L2J	Yes	Yes	Yes		
InLight LDR	L2T	Yes	Yes	Yes		Yes
InLight LDR	L2D	Yes	Yes	Yes	Yes	Yes
InLight LDR	L4P	Yes	Yes	Yes		
Luxel+	Pa	Yes	Yes	Yes		
Luxel+	Ja	Yes	Yes	Yes	Yes	
Luxel+	Ta	Yes	Yes	Yes		Yes
Luxel+ Escort	Pa	Yes	Yes			
Neutrak	N				Yes	
Neutrak	E					Yes
Ring, Single TLD	U	Yes	Yes	Yes		

Deep, Eye and Shallow Dose Equivalents: Deep dose equivalent (DDE) applies to external whole body exposure at a tissue depth of 1 cm (1000 mg/cm^2).
Eye dose equivalent (LDE) applies to external exposure of the lens at a tissue depth of 0.3 cm (300 mg/cm^2).
Shallow dose equivalent (SDE) applies to the external exposure of the skin or extremity at a tissue depth of 0.007 cm (7 mg/cm^2) averaged over an area 1 cm^2.

Deep, eye and shallow dose equivalents report for the time frame indicated by "For Monitoring Period." These doses represent the dose received only for the account/subaccount specified. Individual radiation component results and combined totals report in separate lines.
Quarterly accumulated results reflect total dose received within a calendar 3-months time frame and the customer defined start day. (Note: Quarterly accumulated columns are eliminated for bimonthly service or display "Not applicable.") Year to date accumulation totals dose received from the beginning of the current year to report date. Lifetime accumulation totals all dose received from inception date of dosimeter service to report date, and could include earlier dose history if supplied by customer. Reported quarterly, annual and lifetime dose accumulations represent the doses totaling from all account/subaccount dosimeters to be reported at the customer level.

Inception Date: The date Landauer began keeping dosimeter records for a given dosimeter for a badging participant on the current customer.

Serial Number: Dosimeter serial number.

Second Line Explanation
Participant's personal information consisting of ID number and birth date. This information can be suppressed on "Duplicate and Original Reports" for privacy and/or posting needs.

Notes: Text messages explaining any abnormalities or comments. The notes with message appears on a separate line below all dosimeter exposure information.

U.S. Patents
6,316,702; 6,127,685; 5,892,234

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