Idaho State University AGN-201 Reactor Power Calibration Cadmium Ratio Improvement Using Monte Carlo Methods

By

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of the requirements for the degree of

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I would like to thank my Wife, whose continued love and support has held me accountable to finishing my thesis.

My thanks also goes out to my job at Puget Sound Naval Shipyard, who have been very supportive in giving me the time needed to finish this thesis.

A BRIEF BIOGRAPHY

I am Trevor Richard Boaz. I was born in raised in Port Orchard, Washington but came to Idaho State University back in the Fall of 2007. After (many) years, I completed my Bachelor of Science in Nuclear Engineering. During my time as an undergraduate, my interest in nuclear grew to include Monte Carlo Theory and model experiment validation. While completing my thesis I worked at Puget Sound Naval Shipyard and have since joined Knolls Atomic Power Laboratory in Niskayuna, New York.

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Thesis Abstract—Idaho State University (2019)

Using a computational model of the AGN-201 reactor, a correctly tailored power calibration using foil activation can be made, ensuring higher accuracy power levels in compliance with safety limits. This will be accomplished by reviewing documentation of AGN-201 models from Aerojet General Nucleonics, Monte Carlo N Particle (MCNP) models of the AGN-201, and foil activation methods for power calibration.

The focus of this calibration is through MCNP methods that have not been previously used for the AGN-201 reactor power calibration to relate reaction rate of fission in the reactor core to a power rate. Moreover, it will compare parameters calculated from laboratory to simulated data. Values that will be researched are: cadmium ratio, fraction of neutrons below cadmium cut off that are responsible for fission, and maximum to average flux ratio. MCNP analysis will also be included to determine if a method can relate MCNP6 models to a power output.

Keywords: cadmium ratio, power calibration, AGN-201 reactor, Monte Carlo methods, foil activation

INTRODUCTION

The cadmium separation method is a well-documented and standard method for power calibration of low power reactors [1]. The preferred method in high power reactors is to determine power using the calorimetric method. The calorimetric method utilizes coolant temperatures, and coolant flows. For reactors that do not have a steam plant connection, this method is not feasible. The AGN-201 utilizes a solid moderator with the fissile mixture homogenously dispersed throughout the core. AGN-201 Reactor has a limiting safety setting of up to 6 watts thermal [2]. Instead of measuring the heat from fission, the power level can be deduced from the mechanism driving the fission; neutrons [3].

At Idaho State University (ISU), the cadmium separation method for power calibration (also known as gold foil activation) is the standard method for calibrating the proportional counters attached to the reactor console. The method is performed yearly for reactor power validation and is also performed as part of reactor laboratory experiment courses. The cadmium separation method brings in multiple steps that can induce error in the calculation. The main issue stems from how data is collected to calculate the cadmium ratio.

Power calibration experiments on the AGN-201 reactor have resulted in changing cadmium ratio values, which has been shown to be due to the reactor flux profile compensating due to the cadmium insertion in the reactor. Meaning, cadmium is a neutron poison and has a sharp local negative effect on the reactor flux profile. The cadmium ratio

value is a material independent value that should not be affected by positioning in the reactor or be different between runs.

Later experiments revealed that the cadmium ratio would vary between experiments based on how the foil activations were compared. There have been two different approaches used to calculate a more accurate cadmium ratio. The first approach is to have only one reactor activity run. In this approach, gold foils are placed symmetrically across the glory hole, with one foil covered in a cadmium foil. The cadmium ratio is then calculated using the covered gold foil and its symmetrical gold foil across the core. This approach is not recommended. The flux profile adjusts to be higher on the side of the core away from the cadmium, to compensate, and the flux profile cannot be assumed to be symmetrical. Approach number two, is to have two separate reactor runs. On one of the reactor runs, all that is activated is gold foils at specified locations in the reactor core. During the second reactor run, a gold foil covered with cadmium can be placed at a same location as a gold foil on the previous run. Note, the reactor power and activation time of the foils must be the same. Using the cadmium covered gold foil and comparing it to the gold foil at the same reactor location results in a more accurate cadmium ratio

Other experimenters have utilized a modified cadmium ratio method, where if the cadmium ratio is known independently, that value can be used as a known constant [4] An accurate and verified cadmium ratio value can be used for all future experiments without the need to be recalculated every power run.

Anecdotal evidence has shown that no cadmium ratio value has been the same on power calibration between power calibration labs. Meaning, it is difficult to sustain a control over every variable that could affect the cadmium ratio.

Using a Monte Carlo N particle (MCNP) model, the values from the real world experiments can be evaluated against the simulated versions of the experiments. A reactor flux profile can be compared to data taken, gold foil activation rate can be compared, a cadmium ratio can be compared by several methods, and the overall power of the reactor and the concurrent activation of foils can be compared.

Literature Review

The power calibration background covered in depth in this thesis is largely two fold; lab results analysis, and MCNP comparison analysis. The laboratory procedure for power calibration has earlier roots than the MCNP comparison analysis.

The earliest record of the cadmium separation method for power calibrating the Aerojet-General Nucleaonics reactors was created in San Ramon, California in 1957 [1]. The method published allows for either indium or gold material to be used. In 1958, J. A.T Biel Hoag published "Nuclear Reactor Experiments", addressing that cadmium separation can be used to find the thermal activity of indium or gold foils [5].

A.T. Biel Hoag also describes a way to relate formation rate and decay rate of a radioisotope from time of quantification at a detector to the time of formation in the reactor. This is an important step and is still a major cornerstone of current power calibration at ISU AGN-201.

Cadmium separation analysis using MCNP has been performed using TRIGA reactors. [6] Tiyapun, et al., described how MCNP calculated flux values have to be scaled by various factors to equal the laboratory calculated flux. This provides a documented approach on adjusting MCNP values to equal experimental data. Los Alamos documents even address the issue of monitoring source convergence for power distribution using MCNP [7].

The power calibration method for "zero power" reactors is documented thoroughly, and multiple MCNP-based methods for power reactors have been analyzed. As of yet, no

research has been done on “zero power” reactors using MCNP methods for power calibration, but the theory basis should not differ greatly.

Power Calibration Theory

Idaho State University has an Aerojet General Nucleonics Type 201a (AGN-201) reactor that has limited safety setting of 6 watts [2]. The AGN was built commercially in the 1950’s for many applications. Currently at ISU, the reactor is used predominately for laboratory instruction and research purposes.

The purpose of this work is to develop and quantify parameters in the power calibration methods for the reactor. The end goal is to independently verify and improve terms used in calculating the AGN-201 reactor power level.

This will involve using experiment data compared to an adjusted Monte Carlo N Particle Code (MCNP) model of the AGN-201 reactor. For the purpose of the thesis, the adapted MCNP model of the AGN reactor will not be a benchmark model so much as a model of AGN-201 that has similar flux profile of the data. This is due to the fact that no benchmark of the AGN-201 exists. Creation of a benchmark for this thesis would be beyond the scope, time, and intentions of the thesis.

Foil Utilization

There are multiple ways to perform cadmium separation method. For the purpose this experiment, gold and cadmium were. Gold foils are ideal because gold only has one naturally occurring isotope.

Cadmium Ratio

The cadmium ratio is the activity of a saturated bare foil, divided by the activity of the foil completely covered in cadmium, and is represented in the equation below.

$$CR = \frac{A_{bare\ foil}}{A_{Covered\ Foil}} \quad [1]$$

Where

$$A_{bare\ foil} = Activity\ of\ a\ bare\ foil$$

$$A_{Covered\ Foil} = Activity\ of\ cadmium\ covered\ foil$$

Cadmium largely stops all neutrons below its cut off point at 0.4 electron volts [1]. As can been seen in the figure below, the covered foil responds mainly to the flux in the resonance and fast region. The bare foil responds to the epithermal region (sub-cadmium region), as well as to the resonance region and fast region (epi-cadmium). Figure Figure 1 shows the difference between indium and cadmium and explains the region of interest of the cadmium cross section [5].

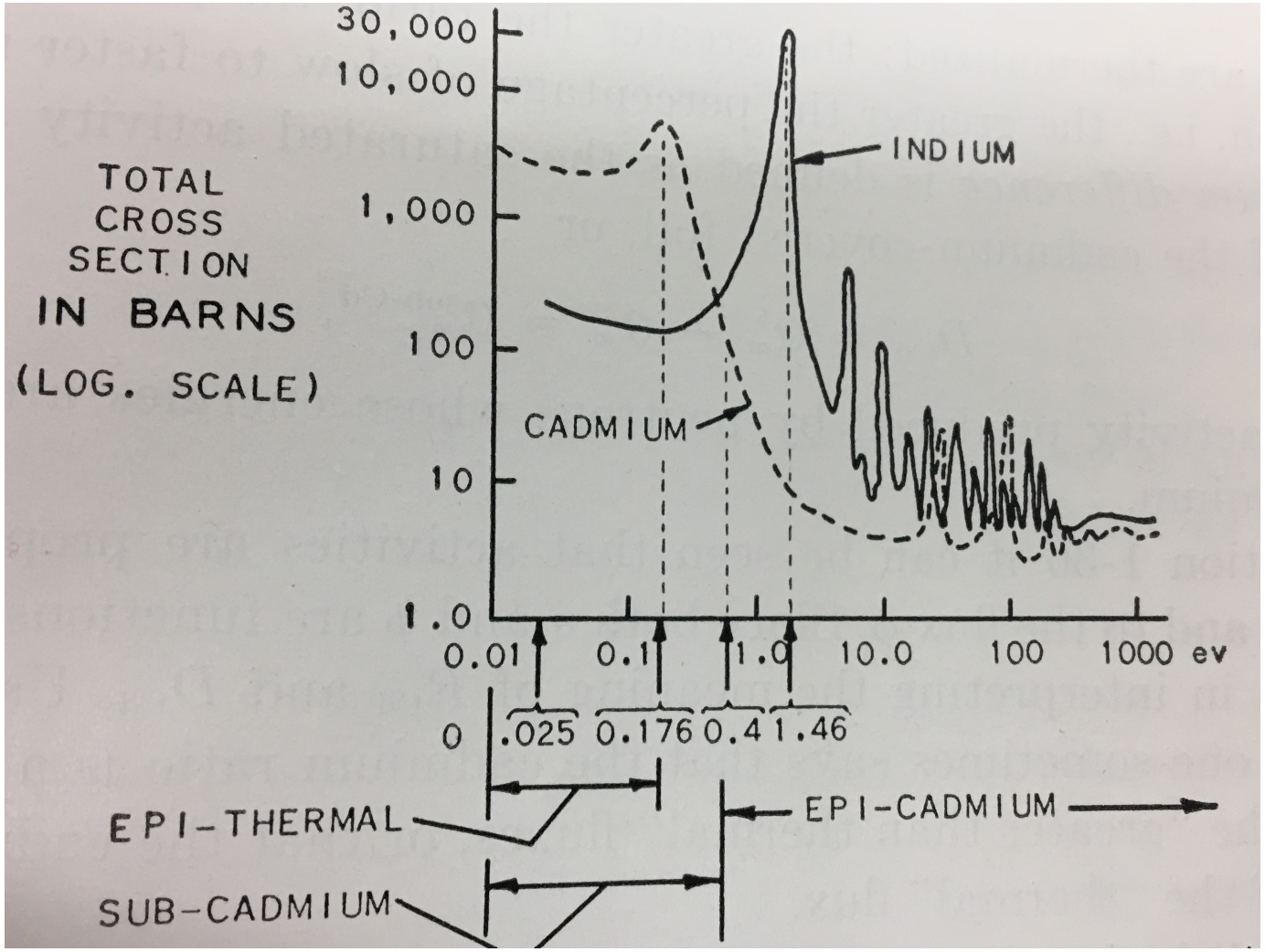

Figure 1: Cadmium Cut Off [5]

AGN-201 Description

The AGN 201 reactor is composed of four fuel control rods consisting of a homogeneous 20% enriched uranium and polyethylene blend inside the core. The core consists of a very small critical mass, so that without the rods inserted, the core is in a subcritical configuration. The fuel rods can be classified into three categories which affect the reactors operability; 2 safety control rods, 1 coarse control rod, and 1 fine control rod. Both the coarse control rod and the safety rods are lifted into the core by electromagnets attached to a reversible DC motor through lead screw assemblies. In the event of a SCRAM, the electromagnets are de-energized, and the rods are gravity and spring assisted in ejection to dashpots.

The whole reactor vessel is approximately 300 centimeters tall and consists of a tank cover, graphite reflector, fuel discs, a thermal fuse, and a core support assembly. The core of the AGN-201 consists of 665 grams of uranium-235 (^{235}U) at 20 wt.% enrichment. The control rods are made up of stacked fuel pellets in a 0.119-cm thick aluminum tube which is the fuel cladding. The aluminum tube is encased in another layer of aluminum, which is the main fission-product interface barrier for the fuel in the control rods. The safety and coarse rods consist of four stacked fuel pellets. Each of the fuel pellets are 4-cm thick and have a 4.6 cm diameter. The total fuel loading for each of the safety rods and the course control rod is 14.4g uranium-235. The fine control rod consists of four stack fuel pellets that are 4-cm thick with a 2.3-cm diameter with a total ^{235}U loading of 0.9 grams.

The core is a right circular cylinder consisting of nine fuel disks of varying height stacked vertically. ISU AGN-201 added a tenth special disk on the top of the other fuel

plates that is specified in other reports as the Reactivity-Adding Disk (RAD). The overall height has a lot of variations between AGN-201 documents and multiple thesis topics [8]. The general consensus is that the core diameter is 25.6 cm and has a core volume of 12.6 liters [11].

Variations of core height could potentially be blamed on the RAD. The RAD is a pure polyethylene disk with fuels pieces inserted in cut-outs throughout the disk. The RAD has a radius of 12.8-cm and a thickness of 1-cm. Core heights from varying thesis topics were compiled by Mackenzie Gorham and height values ranged from 23.6 cm to 25.4 cm. For the basis of this thesis the core height with the RAD is 24.7cm, based on Bower, M.W. thesis calculations. An additional fuel corner piece cut, or 1.0-cm thickness was added to the RAD in the late 1970's. In the 1980's three fuel pellets were planned to be place on the RAD, however, there is no formal documented proof of such change. However, a previous thesis did get vocal confirmation from a previous Reactor Supervisor who was around when the addition took place.

Out of the 9 fuel disks, the bottom four disks are 4.0-cm thick, the next three are 1.9-cm thick, and the last two are 1.0-cm thick for a total height of 23.7-cm. The bottom four fuel disks contain four vertical holes to allow insertion of the reactor rods and are spaced 10-cm from center of the rod hole to center of the closest rod hole.

To allow for reactor experiments, there is a 2.88-cm diameter cut out centered 12-cm from the reactor core bottom. A 0.33-cm thick aluminum tube with an inner diameter of 2.2-cm extends across the reactor core and is attached to the core tank. The purpose of this glory hole is to allow for the irradiation of samples in the core, as well as provide a fission product barrier.

The AGN-201 reactor contains a thermal fuse located just below the aluminum tube centerline in the reactor core. The thermal fuse is a cylindrical disk spanning 2.2-cm in diameter and 0.95-cm thick, and is attached to the core support assembly. In the event of the reactor going prompt critical, the thermal fuse will soften and drop the bottom half of the reactor. Figure 2 specifies each component of the AGN-201 reactor tank and reactor core [9].

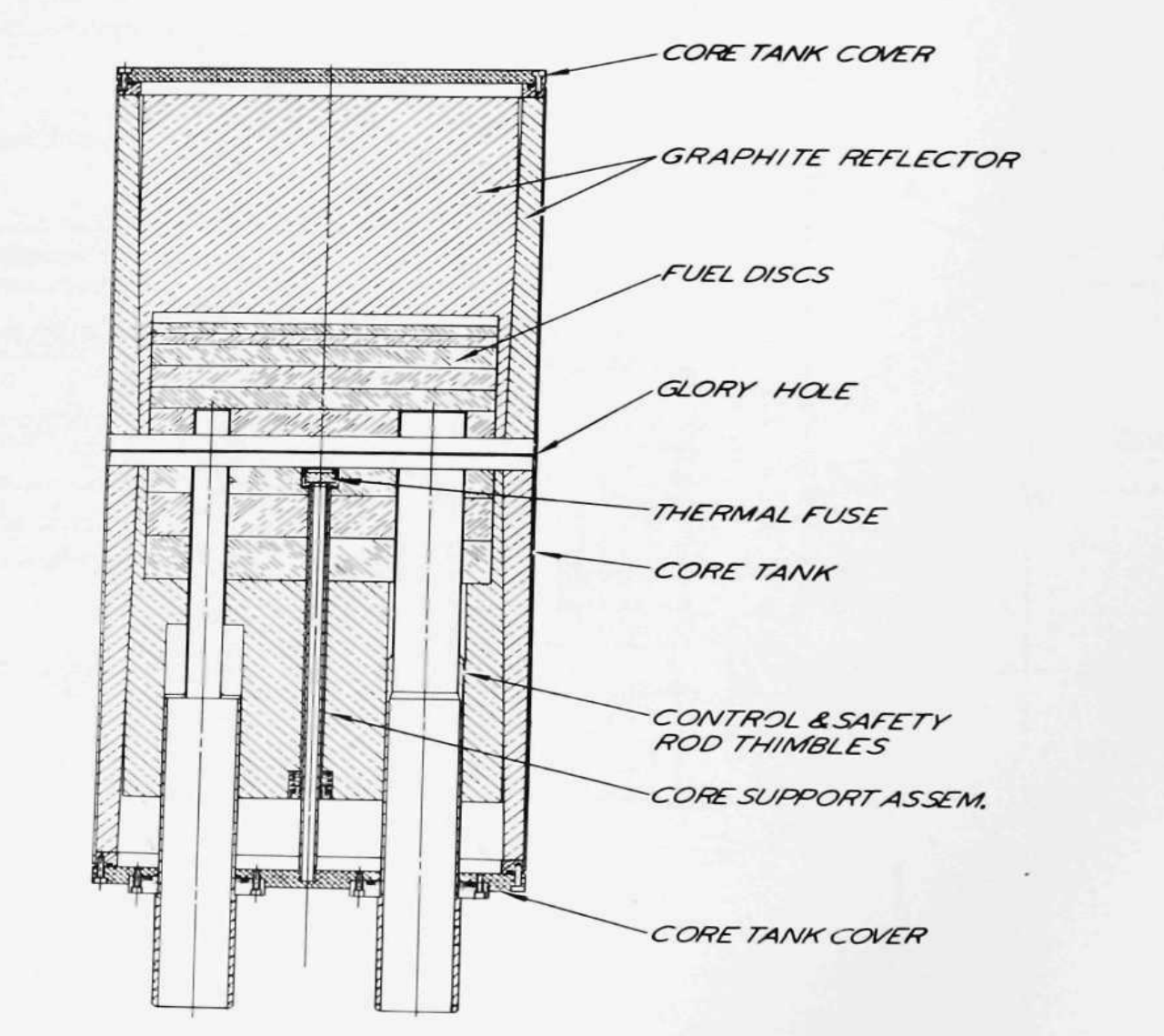

Figure 2: AGN-201 Assembly

MCNP Model

Initial model assessment was done using two of the AGN-201 Reactor models that were available, an unpublished model from University of New Mexico's AGN, and a model of Idaho State University's AGN-201. [8]

The AGN core is essentially a 665g U-235 cylinder core. For that geometry and amount of U-235 (at 20% enrichment) it would be impossible for the reactor to go critical. The water will limit leakage, which would affect the reactor flux profile, the required element needed for the reactor to go critical is polyethylene. Comparing run times of both the New Mexico model and Mackenzie's model shows a substantial difference in run times but with a roughly same value for K-effective. There are a few differences that can be noted between ISU's AGN and UNM AGN, namely that ISU has a Reactivity Adding Disc , while UNM has a fission plate[8]. Additionally, the AGN-201 had additional fuel fragments added on top of the core in the 1980's

The model used is not required to be a benchmark. Uncertainties from the model need to be minimized, but not eliminated for proper utilization.

Previous thesis have modeled the AGN-201 reactor, most recently Gorham made a proposed model for a benchmark in 2012[8]. However, as of 2018 it has not been recognized as a benchmark model. The University of New Mexico also had a thesis for creating an accurate MCNP model of the AGN-201, however it did not have the thermal fuse modeled. Both MCNP models were utilized in the creation of a more simplistic AGN-201 model.

Gorham's thesis was a proposal for a benchmark of the AGN and had extensive neutron spectroscopy to characterize the fuel impurities[8]. The material cards for the MCNP model were very detailed, but the extent of detail is not needed for proof of concept of this thesis. Material composition of major atom densities were selected for the reactor model, but material impurities were not modeled.

The majority of the inside of the AGN-201 reactor core tank cover was modeled, with the exception of the core tank cover, core support assembly and control safety rod thimbles. The reactor vessel and water tank which were not modeled outside the core tank for a several reasons. Extensive modeling of the reactor is outside the scope of the thesis, and the reactor tank provides neutron shielding but is not the moderator of the reactor.

The gloryhole of the reactor was modeled and centered at the origin. A centered gloryhole allows for easier modeling of materials in the core of the reactor, as well as making it more user friendly. The gloryhole extends across the X-plane, with the Z-plane used as height, and the Y-plane tangential to the X plane. The gloryhole consists of a modeled cylinder of air with a radius of 1.1 cm. The aluminum that is the gloryhole is 0.33 cm thick. A standard air composition was used based off of PNNL reference "Compendium of Material Composition Data for Radiation Transport Modeling". The value was input as density for ease of use, with a standard air density of 0.001205 grams per cubic centimeter. Standard air is assumed to be carbon (Zaid 6000), nitrogen (Zaid 7014), oxygen (Zaid 8016), and argon (Zaid 18000). The gloryhole atom density is listed in Table 1.

Gloryhole Composition (air and aluminum)	
Material ZAID	Mass percent (% of density)
6000	0.00124%
7014	75.527%
8016	23.178%
18000	1.283%

Table 1: Gloryhole Composition

The reactor's 9 fuel plates were modeled together at 23.7 cm diameter. The reactivity adding disc was not modeled. The reactor is modeled as if there is no burn-up and minimal impurities, so additional reactivity is not needed to make the reactor critical. The model is essentially modelled at beginning of core life. Below the gloryhole there is 12 cm of the core which represents 3 of the 4 cm fuel plates and the core extends 11.7 cm above the gloryhole. The material used in the core is listed in Table 2, and corresponds to 665 grams U-235 at 20 percent enrichment as uranium dioxide (UO_2) mixed homogenously in powder form in polyethylene (C_2H_2). The reactor core atom density is listed in Table 2.

Reactor Core Atom Density	
Material ZAID	Atom Density (Atom/barn*cm)
92235.70c	1.4221E-04
92238.70c	5.6957E-04
92234.70c	8.5492E-07
1001.70c	7.8546E-02
6000.70c	3.9277E-02

8016.70c	1.427693223911E-03

Table 2: Reactor Core Atom Density.

Graphite in the reactor tank extends above the core at 34.2 cm on the Z-plane, 34.5 cm below the Z-plane, and is a cylinder with a radius of 32.8 cm. The graphite is modeled around the reactor core. The final graphite atom density is noted in Table 3.

Graphite Cylinder	
Material ZAID	Atom Density (Atom/barn*cm)
6000.70c	7.76243731429E-02

Table 3: Graphite Cylinder Atom Density

The reactor core and moderator were modeled for cross sections at 70 degrees Fahrenheit. The 70 degrees better corresponds to a fully inserted coarse and fine control rod configuration of the reactor due to negative temperature coefficient of reactivity of the reactor. This in turn made modeling of the reactor control rods unnecessary, as well as allows for a symmetrical buckling across the core. By having a the nuclide cross sections at 70 degrees, the control rods can be assumed to be fully inserted into core, which allows for geometry simplification. With all control rods fully inserted in the core, the geometry is essentially a single cylinder.

Creation of the model proved be a significant investment of time. The focus was on a similar model to the actual AGN-201 reactor, but with simplifications. Another consideration, was the ease of use for adapting. The model would have to be adapted for multiple different variables and run multiple times. The last main consideration would be time to run , this interplays with simplification.

The general construction of the model began with the central irradiation facility which is furthermore referred to as the gloryhole. This is the access area for insertion of foils into the reactor core, which is central to modeling. Around the gloryhole the homogeneous reactor and moderator blend is modeled. Figure 3 is a top plane view of the reactor and glory hole, which was the first stage of the MCNP model construction

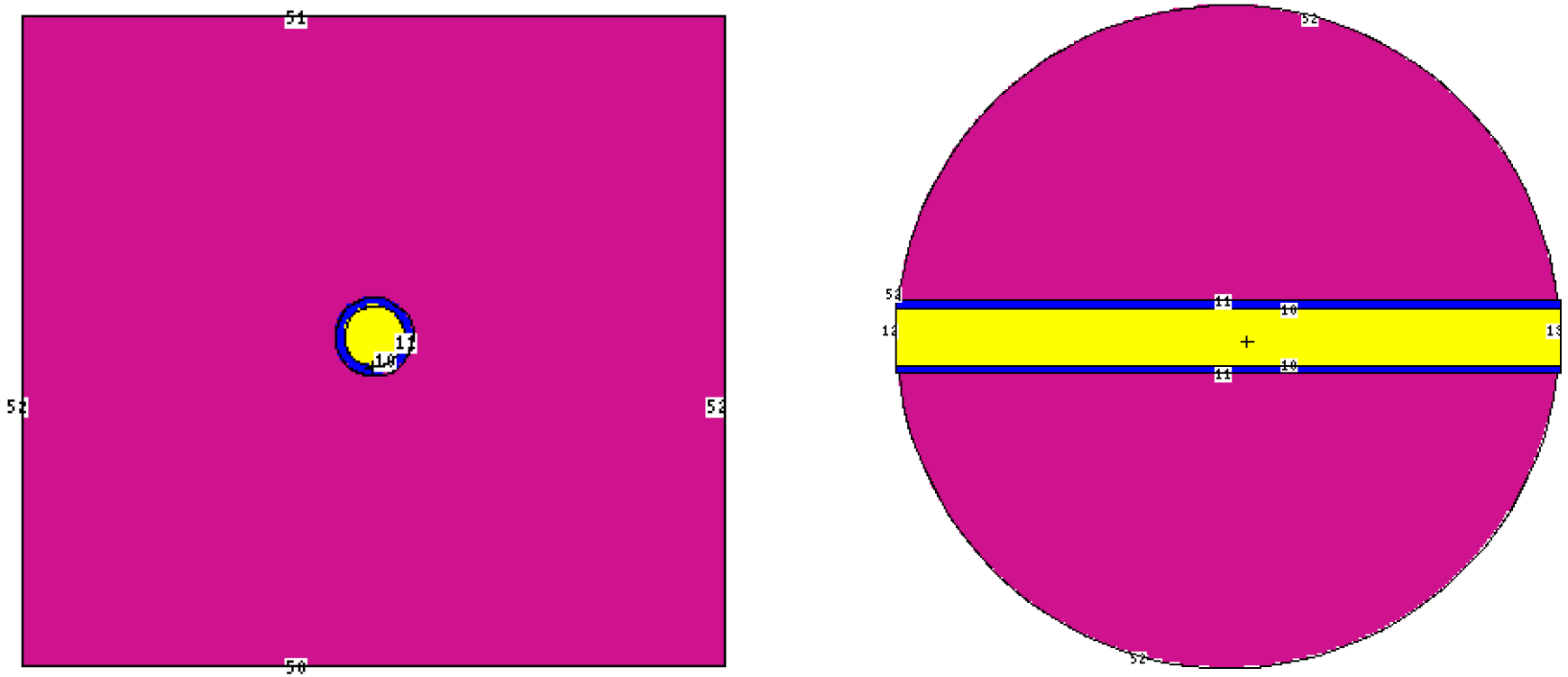

Figure 3: AGN-201 Reactor Core/Gloryhole

Figure 3 represents the core moderated, but the reflector is not modeled. The core is around with a 1.75 gram/cm^3 graphite reflector [1], which was the density given in the AGN Experiments Book as documentation. It should be noted that no precise measurement for other materials were given in the book, just round figures, so the value is not necessarily accurate. Initially, calculations were done to find criticality at this point, to see if there was a reasonable K value that would coincide with a lower graphite density. The graphite range with reasonable values averaged around 1.6 gram/cm^3, with the final graphite density being calculated after additional editing. In Figure 4, the additional graphite model is shown in top planar view and side plane view, both referenced at the center of the gloryhole. The figure helps give a frame of reference on how thick the graphite is around the core and serves as a cutaway image. Note, the graphite density was later adjusted when the simplified reactor was completely modeled, to have a K-effective in the appropriate critical range.

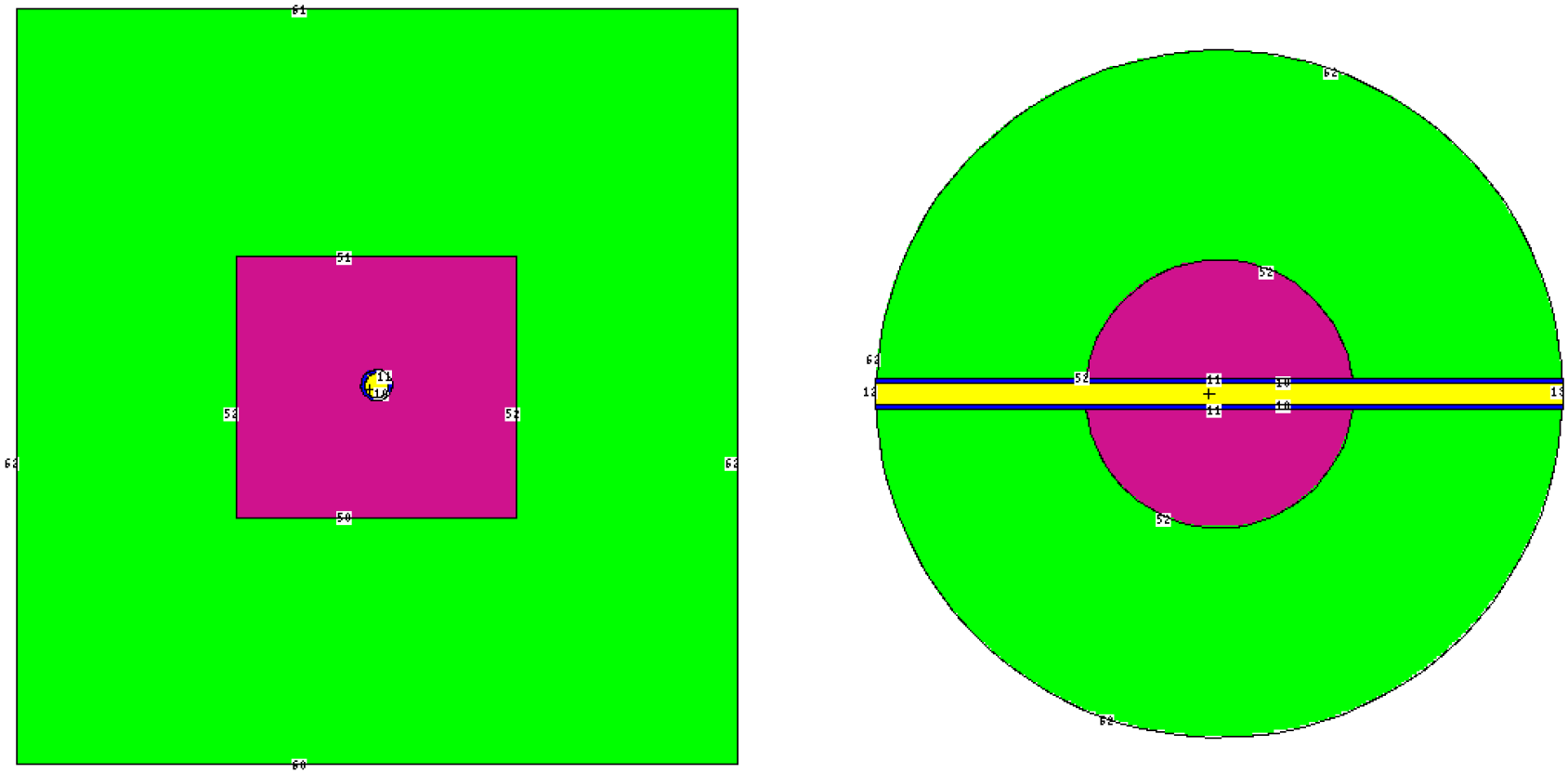

Figure 4: AGN-201 Reactor. Core and Graphite

The nature and proximity of the thermal fuse and its effects on the thermal neutron characteristics in the center of the core were a needed reactor characteristic that had to be modeled next. The fuse provides two features; a double fuel loading and a fuel gap below the reactor. The fuse double fuel loading acts as a thermal neutron sink, but also a source of fast neutrons [1]. The cutaway provides an air gap and a gap of additional fuel reactivity. The fuse and the air gap below the core was the first additional step in construction where surface errors were later realized and fixed. Although of no importance to calculations from MCNP, but from a model perspective, the order of cell cards of the air around the fuse and the fuse were switched with the placement of the reactor and graphite. This was done to serve as a simple fix of surface errors. The following figure shows the top plane view and side plane cutaway of the AGN-201 reactor with the addition of the fuse and the air gap.

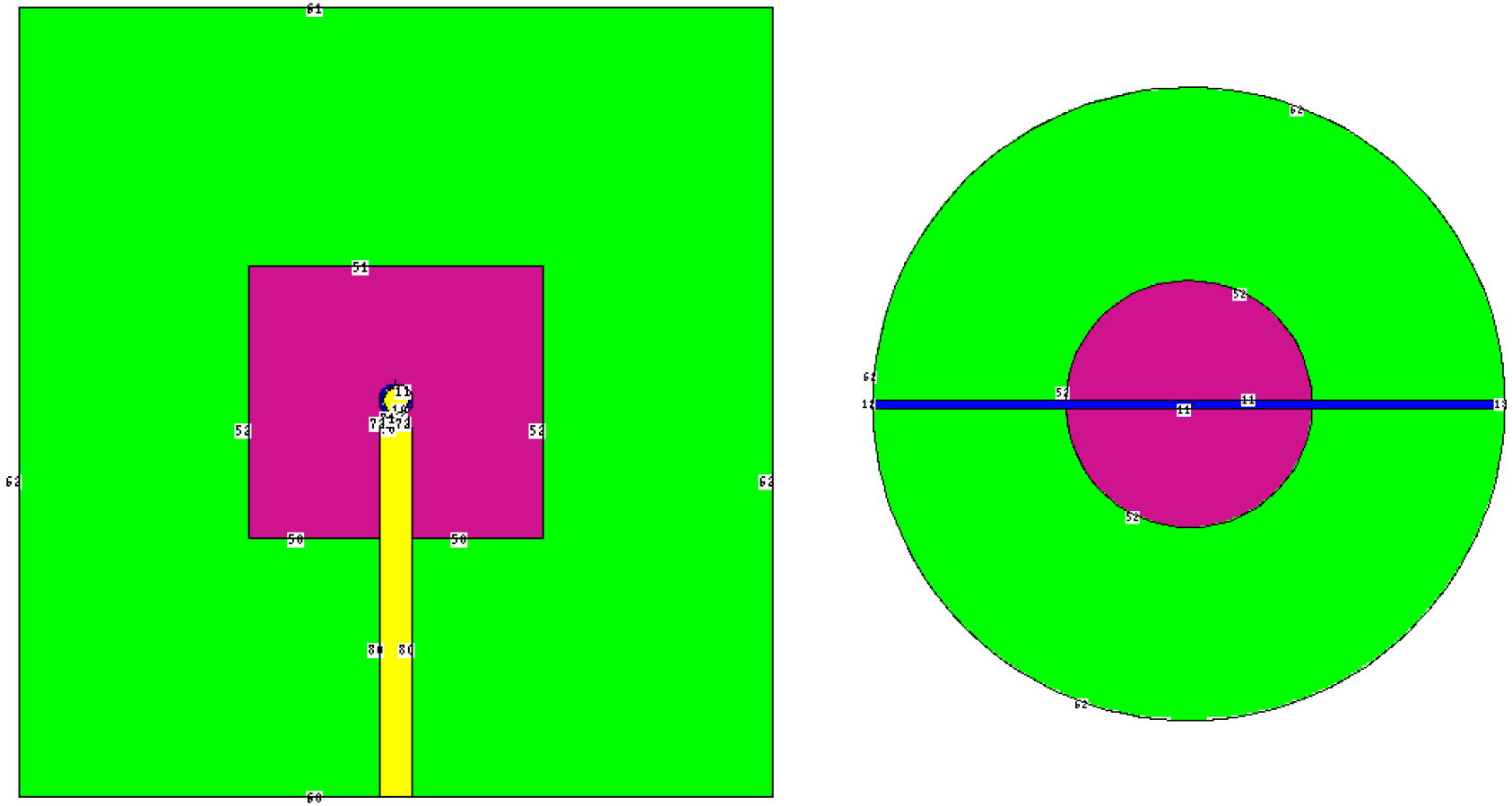

Figure 5: AGN-201 Reactor. Core/Graphite/Fuse

LABORATORY PROCEDURE

Power monitoring was based on previous power calibrations with the foil activation technique. Due to the variations of calculated powers at the same levels, the power level for the reactor runs were taken well below the safety limits [11]

Prior to inserting the gold foils into the reactor, the foils were weighed and taped to a thin aluminum rod at measured intervals. The center of the core from the entrance of the gloryhole is 53.5 inches; starting from the center each foil was then symmetrically placed. Aluminum has a low cross section of absorption and also a very short lived half -life. The low cross section lessens the effect the aluminum has on the flux, while the short lived half-life allows activity of the rod to quickly die away.

The reactor operator brought the power range to a previously specified level and kept the reactor at the power range required, while a certified observer slowly put the aluminum rod and the gold foils into the reactor. Due to the position of the gold foils in the reactor, the low k-excess of the reactor, and the parasitic absorption the gold foils cause; insertion of the aluminum rod has to be done at slow pace to avoid scramming the reactor. Initial insertion of the rod will result in an acute drop of power unless the reactor operator allows for more rod insertion.

The critical reactor temperature, rod positions, and the proportional counters current readings were recorded. These values allow for replicability of the laboratory, while the current reading gives a way to relate data and calibrate the channel ranges after power calibration analysis.

The foils were irradiated for a time (time is specified in calculations). Longer time in the reactor will irradiate the materials more, which will result in higher counts, and will also smooth out any rapid power level changes. After irradiation, the samples were counted in a High Purity Germanium Detector (HPGe). Count time in the HPGe detector depends on activity of the sample; with higher activity resulting in shorter time needed in the detector.

Following detector counting of the foils, the foils were removed, logged, and placed in the isotope vault.

Laboratory Values

The full table for the laboratory results can be seen in Appendix I. Figures Figure 6 Figure 7 below are the plotted values for all energies of the calculated flux, not only thermal. In the figures, the values for the cadmium covered foils are very obvious, and can show a graphical representation of the difference between thermal and fast neutron flux

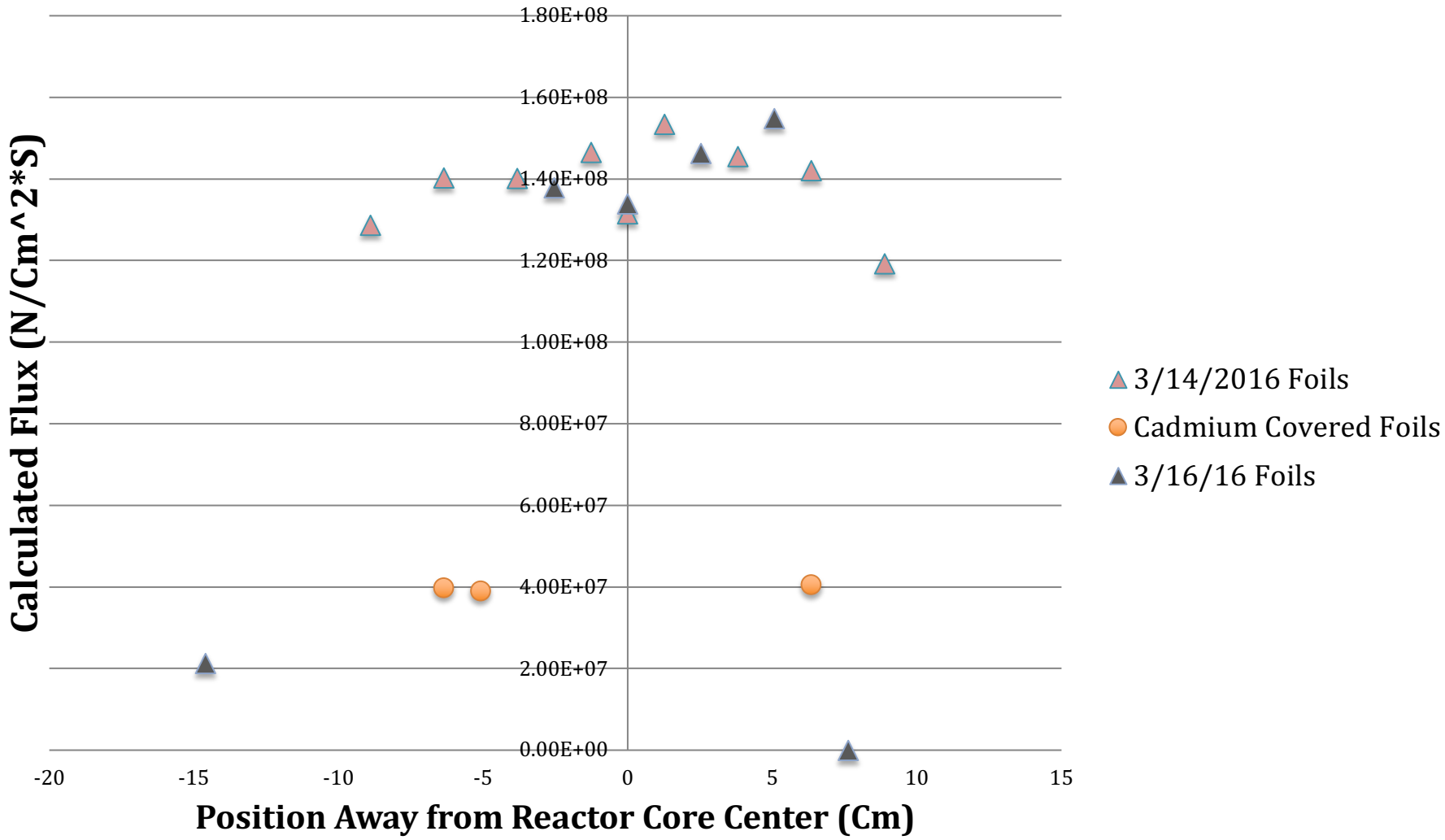

Figure 6: Calculated Flux at Foils from 3/14/2016 through 3/16/2016

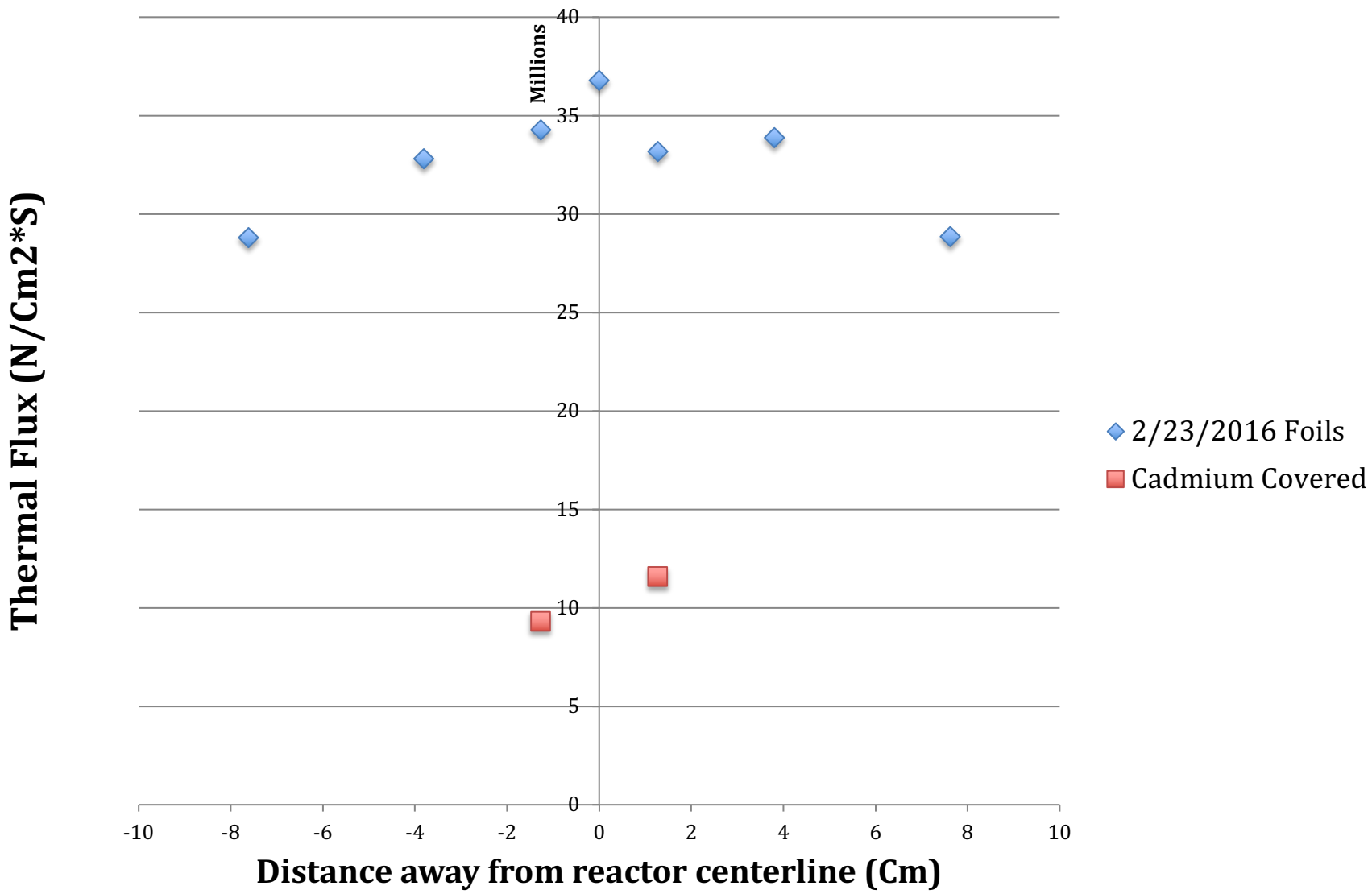

Figure 7 :Calculated Flux at Foils from 3/14/2016 through 3/16/2016

Cadmium Ratio

Analyzing the data from 3/14/2016 and 3/15/2016, two cadmium ratios were calculated resulting in a cadmium ratio of 3.51 and 3.52 respectively. Both values were calculated using two separate irradiations but are at the same locations in the reactor and the same power rating. The first irradiation involved no cadmium foils, and the second irradiation involved two cadmium covered foils.

Analyzing the data from 2/23/2016, two activation runs were performed. Both activation runs had cadmium foils. The cadmium ratios were calculated using the values between runs for foils at the same location in the core. The first activation had no cadmium covered foils, and the second activation had two cadmium covered foils. The calculated cadmium ratios are 3.74 and 2.92 respectively, which is wide-ranging considering the values should be the same.

The data shows that comparing a covered foil and an uncovered foil at the same symmetrical area from the center of the reactor and from the same activation results in a poor cadmium ratio value. The values using foils from the same reactor run have higher variability and more extreme values. It is likely that due to the insertion of neutron poisons at small points across the core result in an unsymmetrical flux profile, and the symmetrical points across the core experience a different flux.

The foil activation on 2/23/2016 with two cadmium foils resulted in the widest range of cadmium ratios than previous runs. It is possible that the cadmium has a higher effect for suppressing the flux across the core. Regardless of the effect, the result is that cadmium affects the flux profile drastically.

Future recommendations for getting a more accurate cadmium ratio include using smaller and thinner cadmium foils, as well as avoiding placing cadmium centerline in the core. Since reactivity can be a function of positioning, the negative effects of the cadmium on the flux profile could be minimized by placing the cadmium cover foil further away from the center of the core.

MCNP THEORY

F4 Tally

The principal tally used for computing the reaction rate of the foils in the reactor is the F4 tally in MCNP. The F4 Tally calculates fluence over the cell volume [12]. The F4 tally is calculated using a particle weight, energy, and track length inside a defined cell volume. This is the average interaction over the volume of the cell per source particle.

Modified F4 Tally

In MCNP6 you can modify the F4 tally in multiple ways. Namely, you can multiply the tally by a given value, which is a convenient way of doing unit conversion. This can also be used in application for reaction rate. Another method is to specify what energy ranges are of interest in the calculation. MCNP can energy bin values on specified ranges. Below is an example that is looking at two defined cells, and the total from both for fluence over the cells' volumes. The modifier multiplies each cell by a value of one, and the energy group of neutrons that is tracked is between 0 eV to 0.4 eV. The cell volume is also stated for each cell and the cell total, this is due to the fact that MCNP needs a volume but cannot calculate some due to geometry calculations. Figure 8 is an example of the MCNP6 input used.

```
c F4 tallies
F4:n 25 30  T                $Fissile Cells (Fuse, core, and both)
E4 4E-07 $.4 eV
SD4 3.61 12142.93 12146.54
```

Figure 8: F4 Modified Example

Shannon Entropy

A discussed issue that deals with MCNP power distribution is the convergence of the source. For K-effective calculations, plots of kcycle vs cycle are fine for judging convergence. However, for power distribution, local tally and heat distribution kcycle is not a good indicator [5]. K-effective can converge faster than the source shape of the model. Shannon Entropy is a valuable and effective means of checking characterizing convergence of a fission distribution.

One way to check the convergence is to use the runtape associated with the MCNP6 model run. There are two inputs that are required to plot the source convergence; on the command prompt "mcnp6 z r='runtapename'", then "kcode 6" [13] . Using the runtape to plote Shannon Entropy is an effective way to plot source convergence of the model. In Figure 9: Source Convergence For 500 and Figure 10 Source Convergence for 5000 Cycle, the Shannon entropy for the model showed quick convergence, but continuous oscillations.

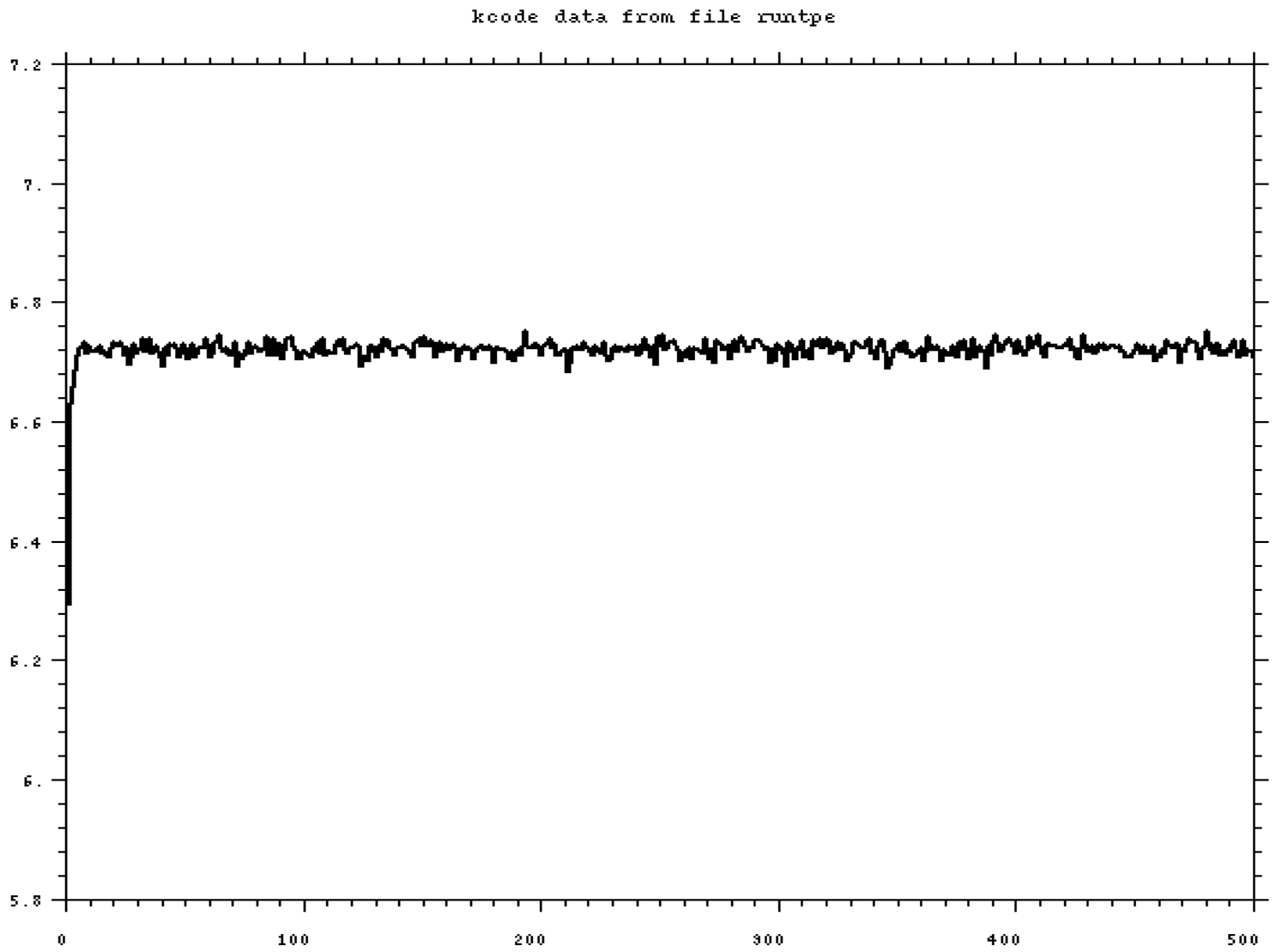

Figure 9: Source Convergence For 500

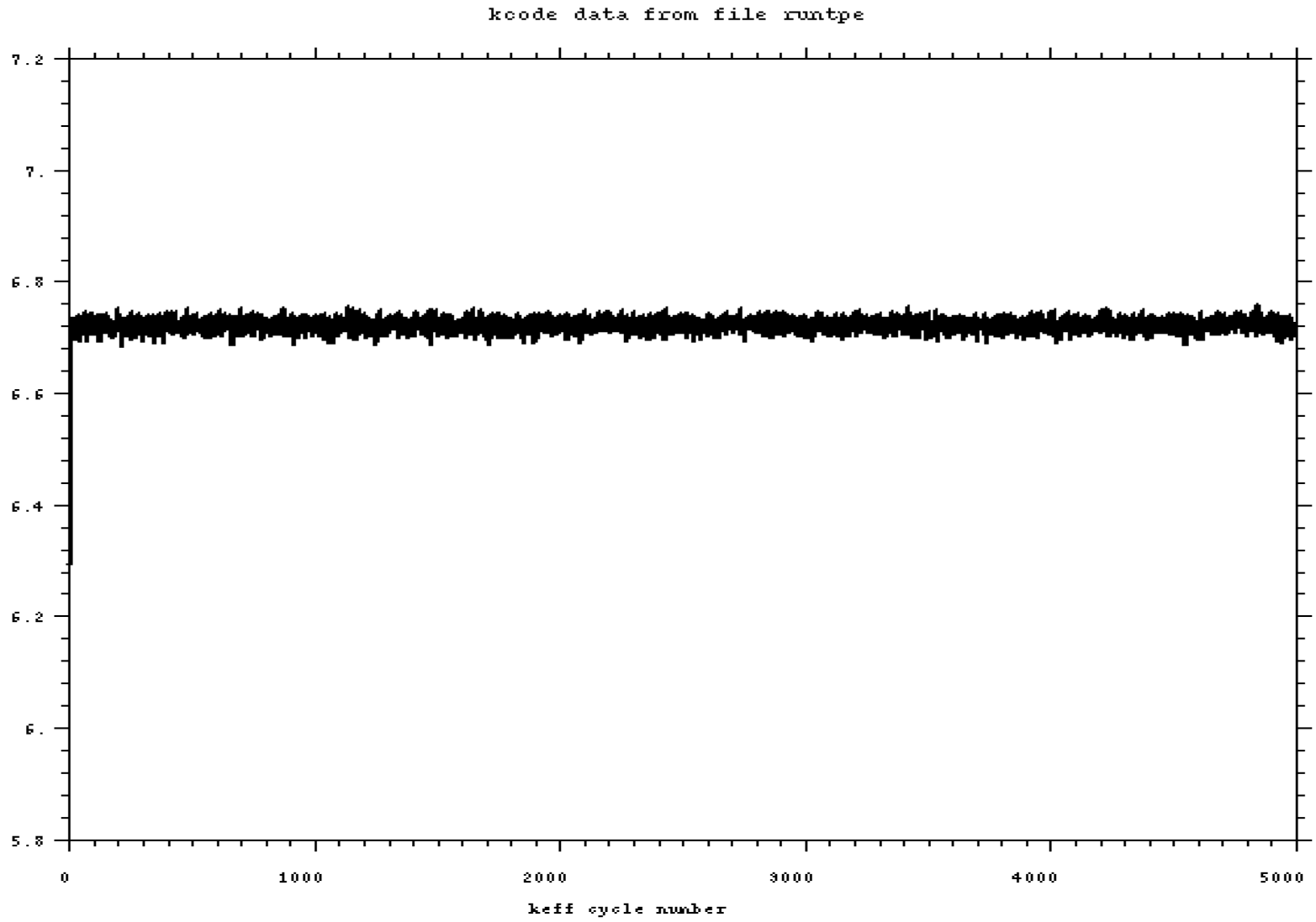

Figure 10 Source Convergence for 5000 Cycle

Modeling the Experiments

There has been multiple theses built around the AGN-201 reactor. University of New Mexico and Idaho State University both have a model. As of 2018, there is no accepted benchmark model for the AGN-201 reactor. Ideally, calculations would be done with a benchmark model for the most accurate results. Another option is to use a simplified model and compare results with experiment values to make sure the model values align close to experimental values.

Gold and Cadmium

Gold activation has many advantages for power calibration. Gold has only one isotope, which limits the complexity for energy peak counting. More isotopes equate to more energy peaks to count and different decay schemes to account for.

Gold has a relatively short half-life of 2.6952 days [14], which is advantageous since shorter half-life means quicker counts. Gold has a large thermal neutron cross-section which gives it a higher reaction rate and limits the amount of time needed to activate the foils. Gold only has one naturally occurring isotope; Au-197. Bombarding gold Au-197 yields Au-198 as shown in the equation below. Au-198 decays via beta decay into a mercury isotope (Hg-198).

$$^{197}_{79}Au + ^{1}_{0}n \rightarrow ^{198}_{79}Au$$

$$^{198}_{79}Au \xrightarrow{2.6952\ days} \gamma + ^{0}_{-1}e + ^{198}_{78}Hg$$

Gold has a downside of a large resonance at lower energies. Cadmium-113 is used to mitigate this issue. The contribution of thermal and resonance neutrons can be separated through the cadmium difference method. Cadmium has a large cross section for neutron energies below 0.4 eV, but a low cross section for higher neutron energies. Figure 11 compares the absorption cross section for both cadmium-113 and gold-97 [15].

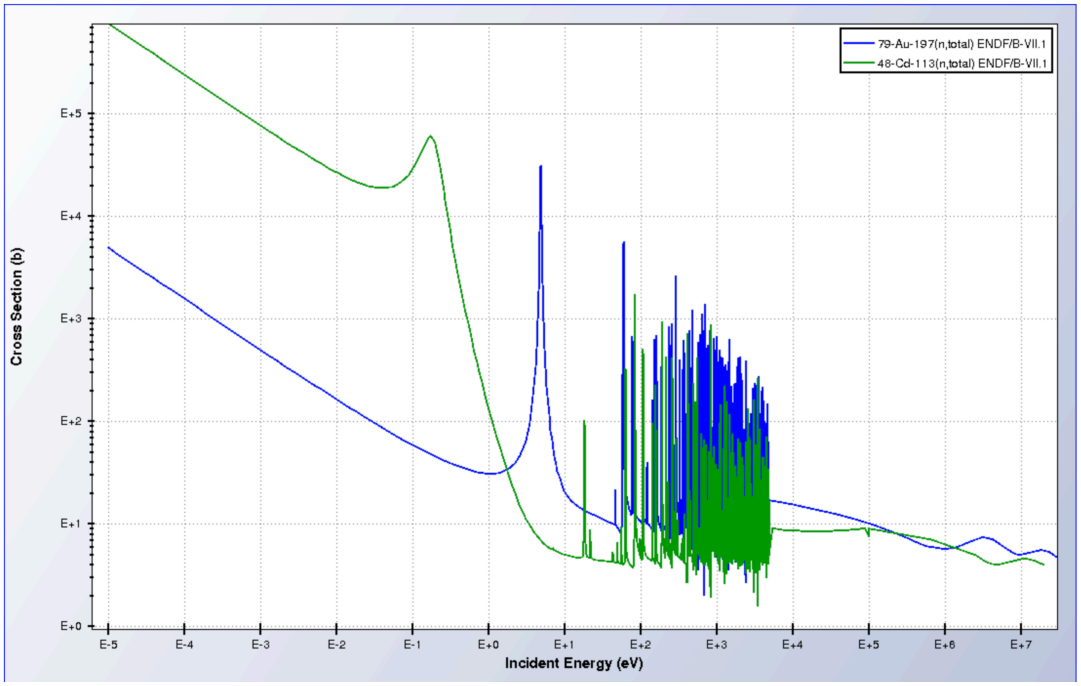

Figure 11: Cross sections of Cd-113 and Au-97

Covering a gold foil with cadmium will allow the foil to largely only be affected by resonance neutrons while an uncovered foil will respond to both thermal and resonant neutrons.

The difference between the covered and uncovered gold can be used to find the thermal neutron contribution, assuming all other factors such as location, symmetry, and

flux are kept the same. Dividing the activity of the uncovered foil by the cadmium covered foil yields the cadmium ratio.

Cadmium ratio is used to find the contribution of thermal neutrons to the activation of the foils. Specifically, cadmium-113 is the isotope of interest. Although , natural cadmium is used , the other nuclides besides cadmium-113 provide minimal absorption of thermal neutrons comparatively. Figure 12 and Figure 13 show the absorption cross section for all cadmium isotopes [15].

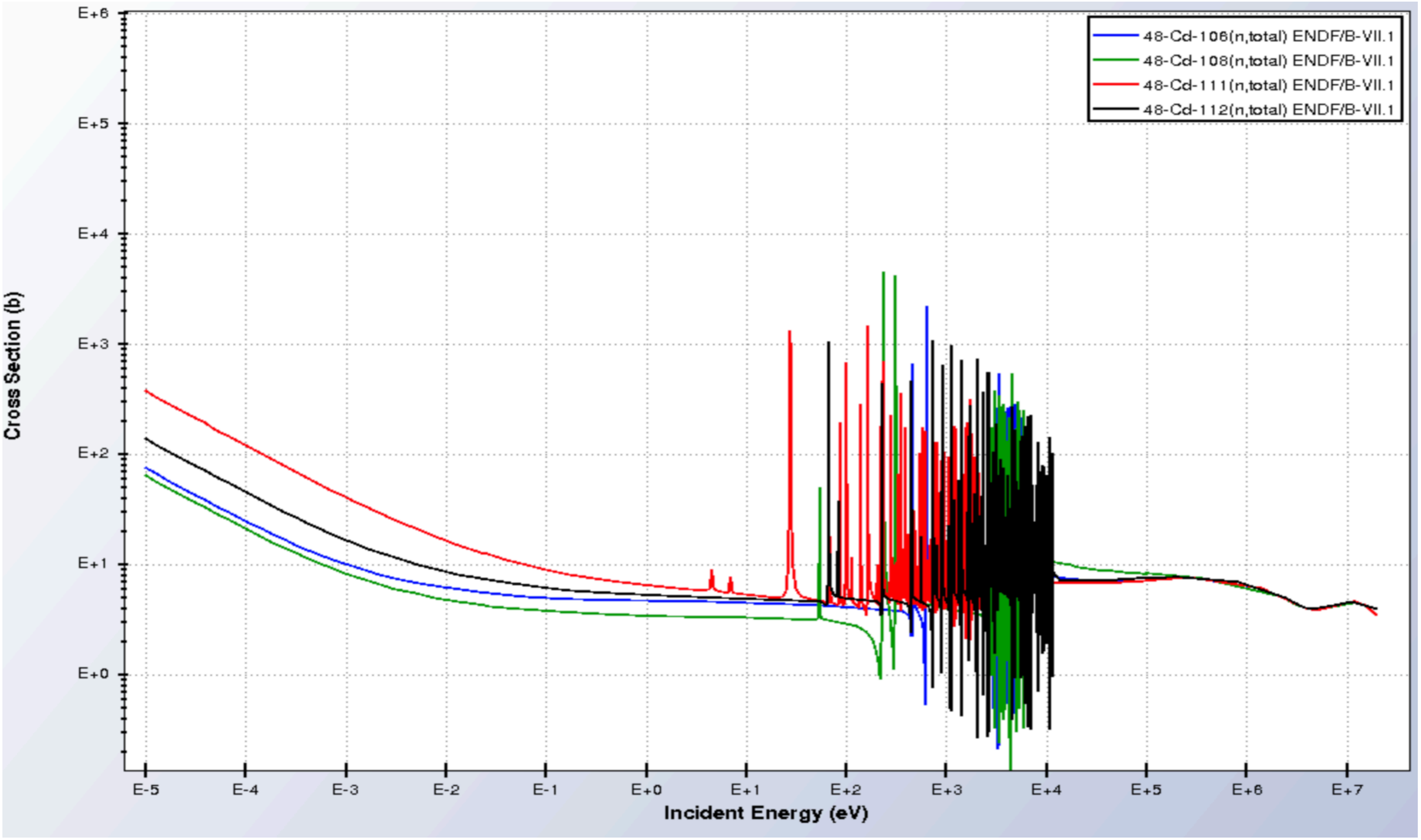

Figure 12: Cd-106, Cd-108, Cd-111, Cd-112 Cross sections

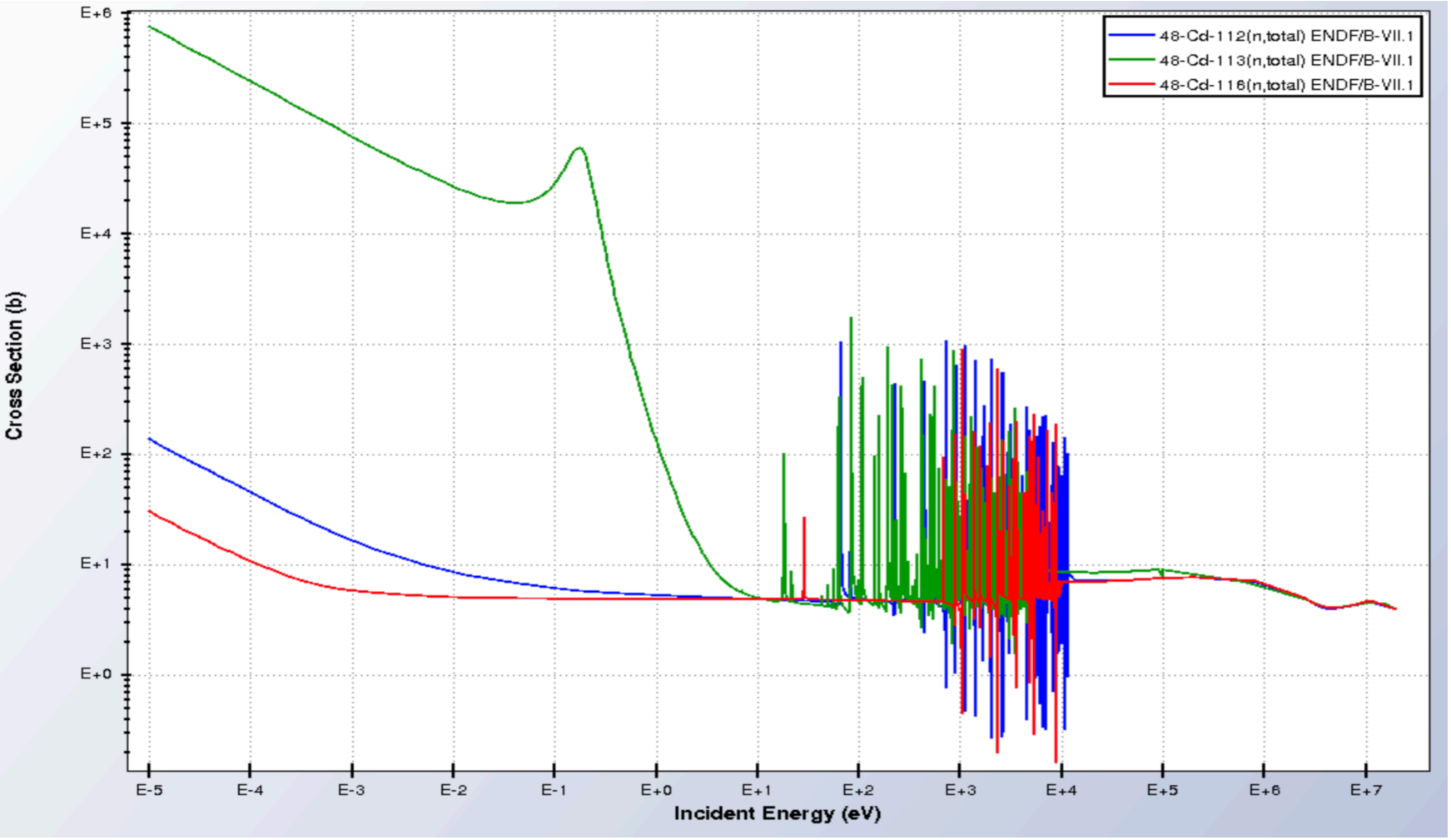

Figure 13: Cd-112, Cd-113, Cd-116 Cross sections

Foil Irradiation

The basis of foil irradiation is relating the thermal flux of the reactor to a reaction rate of the material being irradiated. Assuming that only thermal neutrons are contributing to the power of the reactor, the reaction rate of fission caused by the thermal-neutron flux can be related in the equation below [3]. Thermal flux is not the only contribution to the reactor power but is the major contributor. Adjusting the equation to address the effect of fast neutrons to the power will be addressed later.

$$RR_f \left(\frac{fissions}{second}\right) = \Sigma_f \int_{Volume} \phi \, dV \qquad [2]$$

where

ϕ = the thermal neutron flux

Σ_f = macroscopic fission cross section of the fuel

The flux profile of the reactor is dependent on both geometry of the system as well as the material of the system. If the average thermal-neutron flux is known, Equation (2) will be simplified to

$$RR_f = \Sigma_f \, \phi_{average} \, V_{core} \qquad [3]$$

where

$$\phi_{average} = \frac{\int_{Volume} \phi \, dV}{\int_{Volume} dV} \qquad [4]$$

The average thermal flux equals the flux over the core volume divided by the volume of the core. Using the following Equating 5 [16].

$$P = E_R \, \Sigma_f \, \phi_{average} \, V_{core} \qquad [5]$$

where

$$E_R = \text{recoverable energy per fission}$$

The equation we have so far is dependent on four variables: energy per fission, macroscopic cross section for the fuel, average thermal flux of the reactor, and the volume of the reactor core. All variables in the equation with the exception of the average thermal flux of the reactor can be calculated. Reactors that use U-235 for the fissile fuel can use an approximate conversion factor, where 1 watt = 3×10^{10} fissions/second [1].

Activity

Material decay and production of the gold in the reactor is equal to the equation below. The time rate of change of a nuclide is equal to the rate of production minus the rate of loss [16].

$$\frac{dN(t)}{dt} = \lambda N(t) + Production \qquad [6]$$

Where

$$\frac{dN(t)}{dt} = time\ rate\ of\ change\ of\ the\ nuclides\ atoms$$

$$\lambda = decay\ constant$$

Nuclide production is dependent on the flux, the macroscopic cross section of absorption of the nuclide, and the volume of the nuclide. Production is also equal to the reaction rate.

$$Production = \phi * \Sigma * Volume = Reaction\ Rate \quad [7]$$

Production is dependent on the energy of the neutrons and the corresponding energy dependent cross section. Gold production is thermal energy dominated.

$$Production = \phi_{Thermal} * \Sigma_{Thermal} * Volume = Reaction\ Rate \quad [8]$$

The reaction rate of nuclide production is equivalent to an activity of the foil while in the reactor, which is the continued production term of a nuclide while in a reactor.

$$Reaction\ Rate = A_{\infty} = \frac{mN_a\sigma\phi_{Thermal}}{M} \quad [9]$$

Where

$$A_{\infty} = Activity\ of\ the\ foil\ while\ in\ the\ reactor$$

$$m = mass\ of\ foil$$

$$M = atomic\ mass\ of\ nuclide$$

$$\sigma = microscopic\ cross\ section\ of\ absorption$$

$$\phi_{Thermal} = thermal\ flux\ of\ the\ reactor$$

The production and decay equation can be integrated over time, and results in the equation below

$$N(t) = n_0 e^{-\lambda t} + \frac{Production}{\lambda}(1 - e^{-\lambda t}) \quad [10]$$

Where,

$$N(t) = \text{number of atoms of the nuclide}$$

$$n_0 = amount\ of\ nuclide\ initial$$

$$t = time\ in\ reactor$$

Substitution and simplification can be done to make the equation more usable. Using un-irradiated gold foils, n_0 is zero. The production term is equal to reaction rate of activation, which is equal to A_∞. With these substitutions and then multiplying all sides of the equation by λ, the equation simplifies to terms of activity.

$$A(t) = A_\infty(1 - e^{-\lambda t}) \qquad [11]$$

Where,

$$A(t) = Activity\ of\ the\ nuclide$$

The equation above is only applicable while the nuclide is being irradiated in the reactor. After the material is pulled from the reactor, the activity of the nuclide follows the natural radioactive decay.

$$A(t) = A_0 e^{-\lambda t_1} \qquad [12]$$

Where,

$$A_0 = initial\ activity$$

$$t_1 = time\ after\ irradiation$$

The instant the nuclide is pulled from the reactor, the initial activity of the nuclide is equal to the activity in the reactor. That is; $A_0 = A_\infty$. The equation becomes the following

$$A(t) = A_{\infty} e^{-\lambda t_1} \quad [13]$$

Combing both equations yields an equation that deals with the activation and production of a nuclide in a reactor and its subsequent decay. The equation below does not take in affect the time and issues that come with counting the activity, those parameters will be accounted for in Equation 15.

$$A(t) = \phi_{Thermal} * \Sigma_{Thermal} * Volume(1 - e^{-\lambda t})\ (e^{-\lambda t_1}) \quad [14]$$

So far, the equation takes account the activity at a specified time, the thermal flux, the microscopic crossection of absorption, volume of the foils activated, as well as the time in reactor and time outside the reactor. The equation still has two unknowns and is not linearly independent. $A(t)$ and $\phi_{Thermal}$ are not known.

Activity can be solved using either a Geiger-Muller Tube (GM Tube), or High Purity Germanium Detector (HPGe). GM Tubes operate by detecting the charged particle emitted from the sample, which happens every time the nuclide decays. A HPGe responds to gamma rays emitted during nuclide decay which can happen when the atom is in a metastable state. The branching ratio is the probability that during radioactive decay the nuclide will release a gamma ray [17]. Detector efficiency is dependent on the HPGe detector as well as the energy peak that is of interest. The equation below takes account of what factors need to be accounted for in HPGe Detectors.

$$A(t) = \frac{Counts}{Branching\ Ratio * Detector\ Efficiency * Time} \quad [15]$$

Where,

$$A(t) = Activity\ of\ nuclide\ in\ detector$$

$$Time = Real\ time\ of\ counting\ in\ detector$$

For gold the branching is 0.9562 (95.62%) and detector efficiency is dependent on the detector as well as the energy of the gamma being counted.

The activity of the foils need to be adjust for thermal neutrons. Cadmium separation is the method of covering a foil with cadmium, activating the foil, and comparing the activity of the foil to an irradiated uncovered foil. Equation 1 equates the way to find the cadmium ratio, and Equation 16 equates the way to use the cadmium ratio to adjust foil activity for thermal neutrons [17].

$$A_{thermal} = A_{bare}(CR-1)/CR \qquad [16]$$

Detector efficiency is dependent on the detector used and also the energy peak being analyzed. Detector efficiency for High Purity Germanium detectors will be a very low value. The detector will have a calibration curve of energy vs efficiency, and a fitted equation. Antecedent evidence has shown that the equation used for the fitted energy vs efficiency curve needs to use all numbers provided regardless of significant figures. Premature rounding of numbers throws off the efficiency value drastically.

The flux of the reactor can be related to the activity of the irradiated foils. Using the thermal activity of the irradiated gold foil the flux of the reactor can be calculated using the following equation.

$$\varphi = \left[\frac{AW}{N_A m \sigma (1 - e^{-\lambda\tau})(e^{-\lambda t})} \right] \quad [17]$$

A is the thermal activity

W is the atomic mass of Au

M is the actual mass of the foil

N_A is the Avogadro number

σ is the microscopic cross-section

λ is the decay constant

t is the time elapsed after irradiation

τ is the irradiation time

The equation relates reaction rate to power production. The macroscopic cross section multiplied by the total thermal neutron flux of the core, over the entirety of the core is the reaction rate of fission production.

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MCNP IMPLEMENTATION

MCNP6 does not have a built in power option or power flux. MCNP6 runs cycles that are time independent. The closest option we have for a tally to relate to power production or power production over time is MCNP6's 'F4' Tally.

The F4 tally is useful for finding out a reaction rate. The F4 tally is the track-length flux averaged over a cell. F7 is a fission energy deposition averaged over a cell [18]. The F7 tally is essentially a modified F4 tally.

F4 Tally

As previously discussed, the F4 tally is the reaction rate tally, and is a way to find the average reaction rate of the cell. There are two areas of interest of the model that contain fissile material; the core and the fuse. The F4 tally is cell distinctive, meaning you have to specify which cell you are interested in for MCNP6 to tally it. There is a process for tallies where you can 'bin' multiple cells, which allows you to address multiple cells for the same tally. Along this process, it is possible to have MCNP6 take the total of all binned cells of the tally. The input looks similar to below, where a reaction rate tally looking at neutrons is binning two cells and the cells are totaled together.

F4:N Cell1 Cell2 T

Tallies can also be divided for energy groups of interest using energy cut-offs. The cadmium separation method is utilized to separate neutrons below 0.4eV. MCNP allows inputs that specify which tally it is editing. The input looks similar to below, where E4 is telling which tally it is editing, and E1 and E2 tell which energy intervals to tally between.

The standard nomenclature already includes 0eV as the starting point to energy bin, meaning the energy bins will go from 0 to E1, then E1 to E2. MCNP only accepts energy input in as MeV which has to be corrected to eV on the input.

E4 E1 E2

MCNP provides the relative error for each tally [12], the smaller the relative error the better the results of the tally. A smaller relative error means better statistical convergence of the model.

Cadmium Ratio

The cadmium ratio can be simulated in multiple ways, there are experiment validation and energy cut off comparison methods. The energy cut off comparison was created to simulate what the cadmium ratio is used for with minimal flux profile adjustment The idea being that the cadmium ratio is meant to separate completely the value of thermal neutrons from the value of total neutrons, without affecting the flux.

The energy cut-off method utilized the energy bin ability of MCNP6 for F4 tallies; that is separating what energy of neutrons are absorbed in the F4 volume. The cadmium ratio is found using two different energy bins, neutrons with energy below the cadmium cut off point and neutrons of all energy.

The input for MCNP6 for neutrons being absorbed in gold below 0.4eV and for neutrons being absorbed by all energies would look similar to below. Where F14 is an F4 tally for cells 1 and 2 which represent the gold foils, and E14 is for neutron energies of below 0.4 eV, F24 is an F4 tally for gold for all energy groups.

```
F14:n 1 2
E14 4E-07
F24:n 1 2
```

Running 15000 cycles and rejecting the first 5000 resulted in the Table 4. The MCNP values were calculated for two gold foils of the same mass at symmetrical core locations. The foil locations corresponded to 2.5 inches away from the core (either side). The cadmium ratios that resulted from Table 4 are 3.388 from the data at 2.5 inches from core and 3.368. Evaluations in other sections will use 3.378 for the MCNP cadmium ratio, which is the average of both the values. The relative error of the tallies are shown in Table 4.

Two cadmium ratio methods are compared to the MCNP value First method was calculated by using the data from Appendix I, February 23rd data. The cadmium ratio from February 23rd,2016 compared two power runs, one without any cadmium foils and one run with only two foils with both covered in cadmium. Method one calculates the cadmium ratio utilizing a run that only has minimal gold foils compared to a power run with two symmetrical cadmium foils. Figure 7 shown in previous sections, demonstrates the difference. Method two utilizes symmetrical foils from the same power run; one cadmium covered and one bare. Figure 6 from previous sections shows the data used for method two.

In order to reconcile the MCNP values and the calculated values into a term that relates the values ratio, the term calculated over evaluated is used. Calculated over evaluated (C/E) is the laboratory calculated cadmium ratio divided by the MCNP ratio. C/E gives the ratio of how close the MCNP value compares to the values calculated from laboratory experiments. Table 5 compares the range of values calculated by both methods and compares them to the MCNP values. Table 5 shows that method one has higher variance over values but is more overall conservative and method two is less conservative but values are in closer agreement.

Tally	MCNP Value (N/Cm^2 per particle)	Relative Error of Tally
Gold at 2.5 Inches from Center, Below .4eV	6.67E-04	.0146
Gold at -2.5 Inches from Center, Below .4eV	6.68E-04	.0145
Gold at 2.5 Inches from Center, All energies	2.26 E-03	.0157
Gold at -2.5 Inches from Center, All energies	2.25E-03	.0157

Table 4: F4 Values for .4 eV and for all energies

	Measure Method 1 Value	C/E Method 1	Measure Method 2 Value	C/E Method 2
Cadmium Ratio	2.92/3.74	.87/ 1.11	3.51/3.97	1.04/1.18

Table 5: Calculated over evaluated values

Max to Average

In the laboratory manual [17], there is a value that is used to relate the average flux of the reactor to the average flux of the reactor. AGN-201 reactor power calibration utilizes the maximum calculated flux of the gold foils to determine the average flux of the whole core using the aptly named "Max to Average" value. Previously, the laboratory value was calculated using Disnel, which is a multigroup code. MCNP6 is a continuous energy group code and could potentially increase the accuracy of the variable that is listed in the lab manual.

Using MCNP6, two values from the core were modelled; the average over the core and the fuse, as well as the highest value in the core. The average over the core was calculated using an F4 tally over the core and the fuse, while the maximum was found by taking an FMESH4 over the top portion of the core and then finding the maximum value in the MESHTAL given. The MCPLOT for the F4MESH Tally is shown in both Figure 14: FMESH4 Tally Above Gloryhole which shows a side view of the FMESH4 tally above the gloryhole and Figure 15: Top view of FMESH4 tally

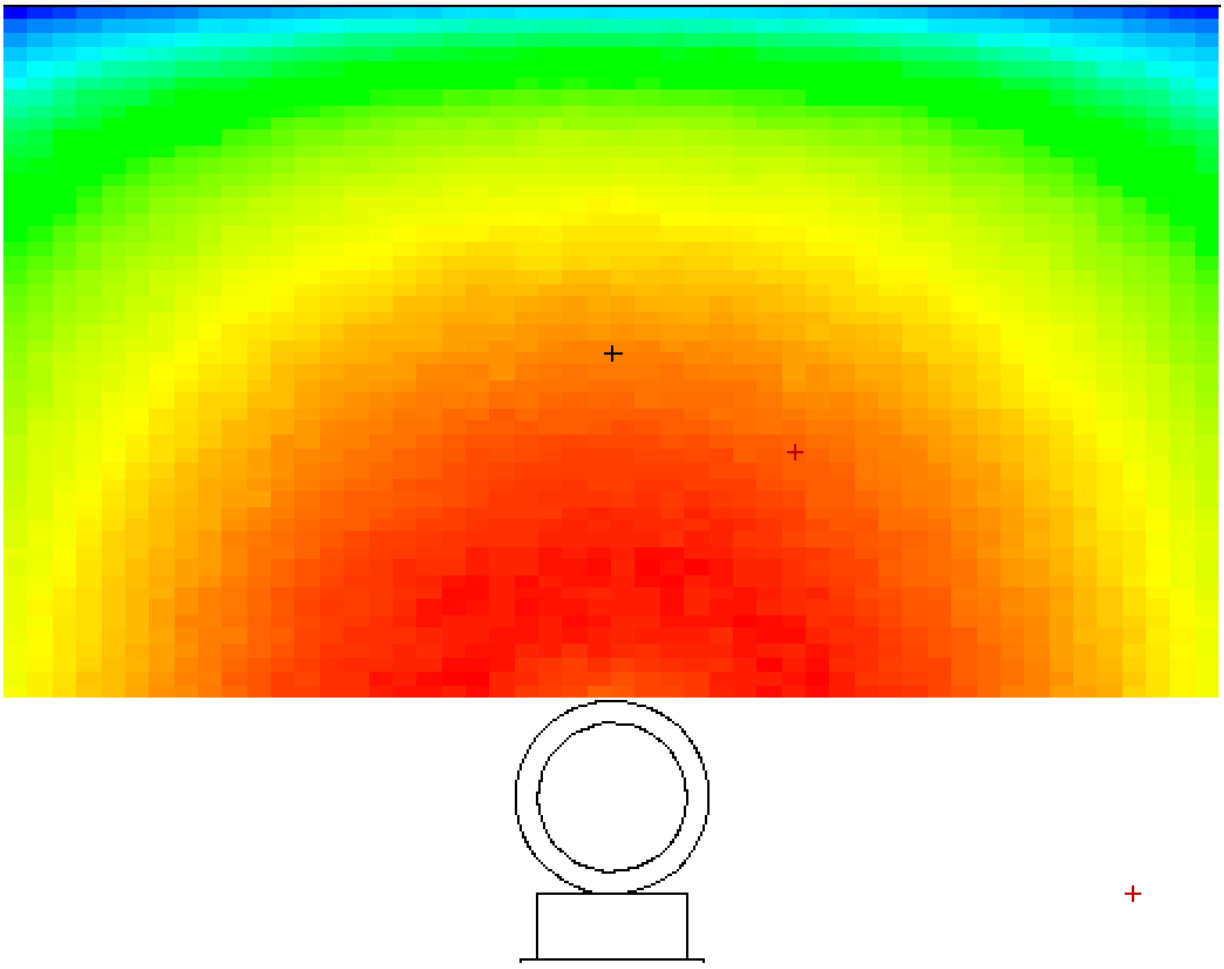

Figure 14: FMESH4 Tally Above Gloryhole

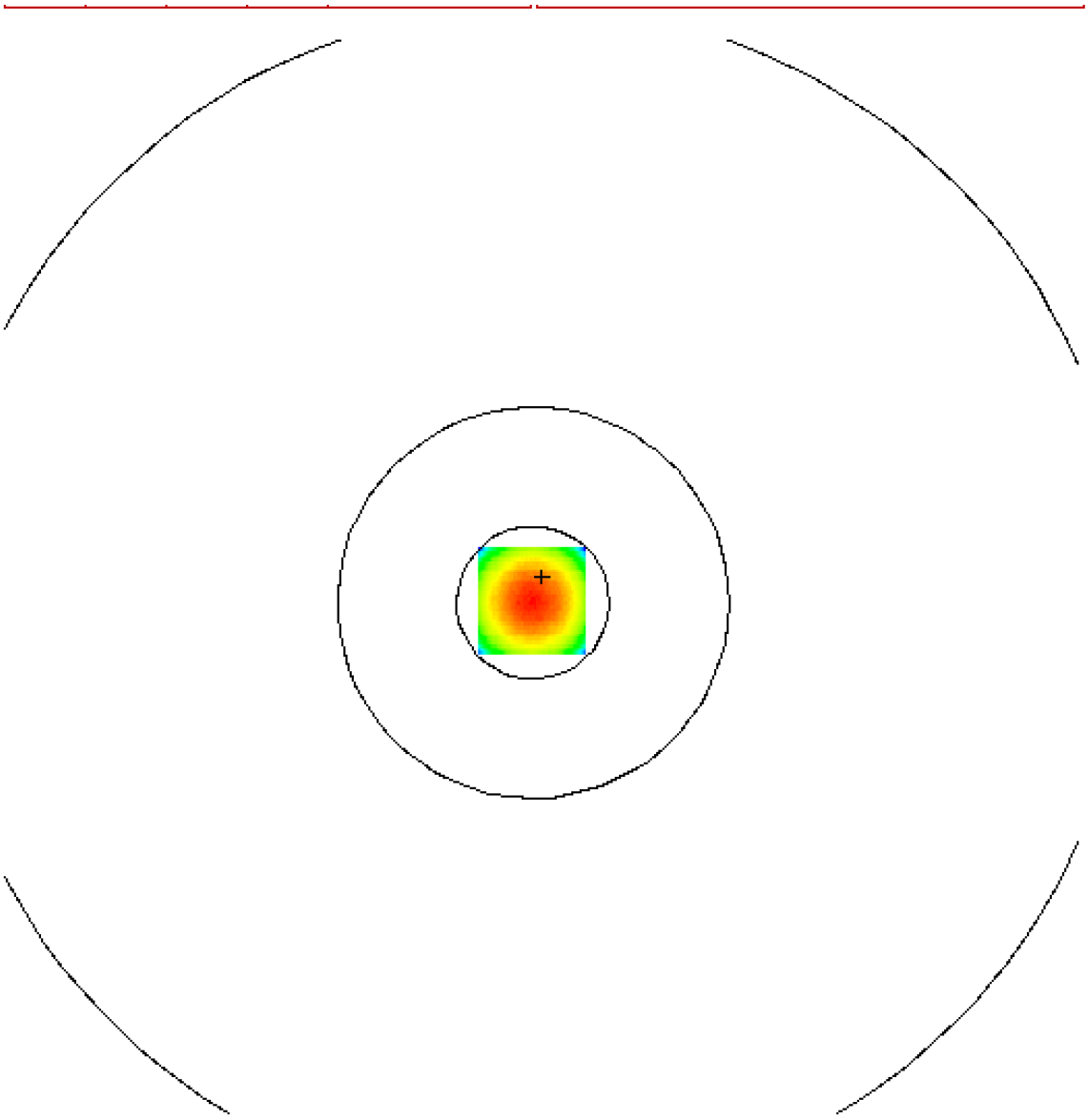

Figure 15: Top view of FMESH4 tally

The value given in the lab manual is 1.638 [17] and was a good starting point for where the calculated MCNP value should be. Multiple runs were performed to see the values change. Less total cycles resulted in a higher value max as can be seen in Table 6. The final result for the maximum to average value was 1.6598, with multiple outputs showing close agreement. Table 6 also shows the maximum relative error of the tallies used for calculation, since two different tallies were needed for each step the maximum error for each tally is the one used.

Total Cycles	Cycles Rejected	Maximum Flux Value (N/Cm^2 per particle)	Average Flux Value (N/Cm^2 per particle))	Maximum to Average Value	Relative Error of Tally
5000	2000	9.0668E-04	5.1756E-04	1.7518	.0047
20000	10000	8.7976E-04	5.1770E-04	1.6994	.0026
40000	20000	8.6694E-04	5.1772E-04	1.6745	.0015
100000	40000	8.5926E-04	5.1769E-04	**1.6598**	.0011

Table 6: Max to Average Values

Contribution of thermal neutrons to fission

Another value utilized in the laboratory manual for power calibration is contribution of thermal neutrons to fission. Ironically enough, this value is used to correct for the neutron energies that are above thermal. It is important in the cadmium separation method to account for the neutrons above the cadmium cut off energy, in order to fully calibrate the power level. Figure 16 is a table from a MCNP6 output that was created from running 100,000 cycles and rejecting 40,000 cycles with 10,000 neutrons per generation. Figure 16 shows percentages of fissions caused by neutrons of varying ranges. The 0.625eV and below range of neutrons are the cause of 95.74% of the fissions, and thus the calculation for

fission contribution from 0.4eV and below neutrons should be less than the value in Figure 16.

Energy	Cycles Run	Rejected	Value	Relative Error of Tally
Below 0.4 eV at Fuse	1000	500	8.33E-05	.0128
Below 0.4 eV at Core	1000	500	3.22E-05	.0005
All Energies at Fuse	1000	500	8.81E-05	.0122
All Energies at Core	1000	500	3.39E-05	.0005

Table 7: Absorption of neutrons, by energy in the core

```
 the final estimated combined collision/absorption/track-length keff = 1.00460 with an estimated standard deviation of 0.00003

 the estimated 68, 95, & 99 percent keff confidence intervals are 1.00457 to 1.00464, 1.00454 to 1.00467, and 1.00452 to 1.00469

 the final combined (col/abs/tl) prompt removal lifetime = 1.1630E-04 seconds with an estimated standard deviation of 1.0758E-08

 the average neutron energy causing fission = 1.2381E-02 mev
 the energy corresponding to the average neutron lethargy causing fission = 4.4582E-08 mev

 the percentages of fissions caused by neutrons in the thermal, intermediate, and fast neutron ranges are:
        (<0.625 ev):  95.74%         (0.625 ev - 100 kev):   3.73%          (>100 kev):   0.53%

 the average fission neutrons produced per neutron absorbed (capture + fission) in all cells with fission = 1.5150E+00
 the average fission neutrons produced per neutron absorbed (capture + fission) in all the geometry cells = 1.5039E+00

 the average number of neutrons produced per fission = 2.438
```

Figure 16: Fissions by Energy Range

The contribution of thermal neutrons to the flux is the addition of both the Fuse and Core and is 1.1554E-04 and the total for all energy absorbed in the core is 1.2195E-04. Absorption of thermal neutrons (0.4eV and below) over the absorption of all energy

neutrons yields the contribution of thermal neutrons and is 94.75%. The total flux is equivalent to the thermal flux divided by that value.

The example tally modifier used to calculate the thermal neutron contribution is in Figure 17: F4 Modifier Example for Fission Contribution The modifier used accounts for only absorptions leading to fission

```
c F4 tallies
F4:n 25                    $Fissile Cells (Fuse, core, and both)
FM4 -1 500 -6         $No modification
SD4 3.61    $Define volume for the bins
E4 4E-07 $ below 4 ev
c
F14:N 30
FM14 -1 100 -6
SD14 12142.93
E14 4E-07 $ below 4 ev
c
F24:n 25                    $Fissile Cells (Fuse, core, and both)
FM24 -1 500 -6         $No modification
SD24 3.61
c
F34:N 30
FM34 -1 100 -6
SD34 12142.93
```

Figure 17: F4 Modifier Example for Fission Contribution

MCNP Power Calculation

Multiple methods were attempted to find a method of relating power to reaction rate of the reactor core. To correctly adapt MCNP tallies to a power level, a variable called 'Neutrons Total' is introduced [19]. Neutrons Total relates neutrons to energy released per fission and a power rating, which is a value of neutrons per second that is required to meet a given power. The factor is needed because MCNP6 normalizes tally values per source particle. The equation below shows the relation.

$$Neutrons\ Total\ [\frac{neutrons}{second}] = \frac{P[watts]\ x\ \nu[\frac{n}{fission}]}{(1.6\ x\ 10^{-13}\left[watt\ \frac{s}{Mev}\right] x200[\frac{MeV}{fission}]} \quad [18]$$

Where

$$P = Power\ level$$

$$\nu = number\ of\ neutrons\ released\ per\ fission$$

The next step is to relate F4 tally to a flux. The next equation utilizes the neutron total, the F4 tally and corrects the value with the k_{eff}. The tally and k_{eff} are values that can be taken from MCNP. K_{eff} is need in Equation 19 to adjust for any difference from an assumed steady state condition of k=1.

$$\text{Flux [n/cm}^2\text{s]} = \text{F4 Tally [\#/cm}^2\text{]} \times \text{(Neutrons Total)} \times (1/\text{keff}) \quad [19]$$

The F4 Tally will be the most pivotal value for the equation. All references on this method don't specify a F4 modifier, but with following our normal format from previous equations the F4 tally will be for absorption below .4 eV. This value simulates the energy of neutrons below the cadmium cut off point. Table 8: Flux from F4 Calculation lists the values from MCNP as well as the values calculating flux from F4. Power is an assumed value, and for the equation is assumed to be 3 watts.

Variable Name	Value
P (watts)	3
ν (neutrons per fission)	2.438
Neutrons Total Value	2.2856*10^-15

F4 Tally (absorption of 0.4 eV and below)	5.18*10^-4
Volume for F4 Tally (cm^3)	12146.54
Keff	1.0041
Neutrons Total (Calculated Value)	2.2856*10^11
Flux Value .4eV and Below [n/cm^2s] (Calculated Value)	1.18*10^-13

Table 8: Flux from F4 Calculation

The flux value calculated does not mean much in itself, however using Equation 9 for reaction rate and multiplying that by energy per fission yields the power equation (adapted Equation 5). The flux value is the thermal absorption value, not absorption leading to fission.

$$Power = \frac{E_R m N_a \sigma \phi_{Thermal}}{M} \quad [20]$$

Equation 20 filled out is seen below. This equation should have the same power rating as the value put in Table 8: Flux from F4 Calculation , however the values differ because of the F4 Modifier as well as the cross section used.

$(200 * 10^6 eV) * \left(1.602 * 10^{-19} \frac{joules}{eV}\right) *$ (665 grams U-235)*($6.022*10^{23} \frac{atoms}{mol}$)*($580*10^{-24}$ cm^2) *($1.18*10^{-13} \frac{neutrons}{cm^2}$)/ ($235 \frac{grams}{mol}$)= 3.77 watts

The F4 modifier used was for absorbed neutrons below .4 eV, however, all energy neutrons need to be accounted for. The F4 tally also is for absorbed neutrons, not absorbed neutrons leading to fission. Adapting the value for absorption leading to fission, and also

correcting the value for neutrons of all energies. A one energy group correcting factor for thermal energy is used to correct for absorption leading to fission and the previous calculated value for thermal contribution to fission is used to correct for all energy ranges. The thermal cross section for absorption for thermal neutrons over the cross section for fission of thermal neutrons is used to correct the value for absorption leading to fission. This is for single energy groups, not continuous and is only used as a proof of concept.

$$\left(\frac{3.77\ Watts*585/698.9}{.95}\right)=3.33\text{watts}$$

MCNP6 can be used to relate power to flux values, but more work and tailoring will be need to be done to insure accuracy. Additionally, the source reference for which the equation was taken was not trying to relate a specific energy group to a power rating [19] Using a single energy group correcting factor, the power level is 11% off from what it should be.

CONCLUSION

Conclusion

The final values from this thesis work compared to the original lab values are contained in Table 9. The original values are either the values calculated from experimental data or given in the laboratory manual. The MCNP values are the calculated values from using the MCNP6 model. The overall net change to power is contained in the Table 9, and calculated by what the change of the original value was to the new power value from MCNP values.

Variable	Original Values	MCNP Calculated Values	% Change to Power
Cadmium Ratio	2.92, 3.03, 3.08, 3.51, 3.52, 3.97	3.39, 3.36	-5.7% to +7.2%
Max to Average	1.638	1.6598	-1.3%
Contribution of thermal neutrons to flux	96.1%	94.7459%.	+1.4%

Table 9: Original laboratory values compared to calculated

The overall effect on these new figures on power calibration is -5.592% to +7.33%, which means previous power calibrations using these methods have largely been conservative. The major cause for error in the power calibration stems from the cadmium ratio, which is the variable that is heavily affected by operation and human errors.
The new variable values will help improve the accuracy of future calibrations, but also provide independent verification that AGN-201 reactor past calculations have kept the reactors power range below safety limits. Overall, these values prove that we are way below safety limits and we have precision in our power ratings.

The calculated cadmium ratio proved to be a source of great variance and little information. There was no previous method created for calculating cadmium ratio in MCNP. The method contained in this thesis was adapted to relate the difference between neutrons of difference energies absorbed in gold for a symmetrical flux. Out of all the other variables calculated in the thesis cadmium ratio proved to be the hardest to develop a method to, as well as the most varied experimental value. The values contained in thesis will serve as a starting point on how laboratory methods setting up and calculating cadmium ratio should change. The different laboratory methods discussed in the thesis show that different ways of calculating the cadmium ratio affect the variance and accuracy. This suggests that there are multiple parameters that can affect the calculation.

Two other variables that are used in the laboratory procedure for calculating the power of the AGN-201 are the max to average ratio and the contribution of thermal neutrons to flux. Both these values are in close agreement to the laboratory manual values[17]. The fact that these values agree with other independent methods values gives the laboratory method more confident in accuracy.

MCNP can be used to relate F4 tallies to a given flux at a stated power level. The methods outline in this thesis give an inherent error of 11%. Further research can be used to improve the accuracy. There is not method out there based on power calibration using MCNP, only for looking at power peaks of reactor cores.

Additional Work

Values calculated from the MCNP method proved to be in high agreement to laboratory results, and thus parameters taken using MCNP analysis are likely of more accurate value than values that can be skewed from operator error. The model created to simulate the reactor is not a benchmark quality model, it was created to closely represent the reactor without exacting detail. All values would benefit validation with an AGN-201 reactor benchmark, but as of 2018, there is no benchmark mode available. Furthermore, values could be ran for more cycles while rejecting more cycles increasing the accuracy of the values.

Cadmium ratio improvement could be improved in laboratory settings in a number of ways. Multiple experiments varying the location of cadmium foils in the core could be completed to see if there is a trend of cadmium ratio changes depending on location.

The method of relating power level to a F4 tally can be improved by either utilizing MCNP cross sections for continuous energies, or by binning the F4 tally for different energy groups and then using multigroup cross sections. There is no method created so far that utilizes MCNP for power calibration.

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Appendix I

March 14/15/16, 2016 Lab Data.

Foils for Monday 3/14/2016	Wt (grams)	Position	Time In	Time Out	Irradiation time	Cadmium Wt	Count Start (Thursday 3/17/2016)	Time between Pull and count	Count Time (sec)	Count Total	Count/eff*branch*time	Mass Adjusted	Total Flux	Thermal Flux		
1	0.0156	Center	1:33pm	2:33	60 min		2:13pm	258000	302.8	2.64E+04	3.05E+03	3.05E+03	1.32E+08	9.42E+07	Mass Adjusted	
2	0.0138	"+0.5"	1:33pm	2:33	60 min		2:31pm	259080	302.6	2.40E+04	2.77E+03	3.13E+03	1.53E+08	1.10E+08	**Average Flux**	**6.99E+07**
3	0.014	"+1.5"	1:33pm	2:33	60 min		2:39PM	259560	302.6	2.34E+04	2.70E+03	3.01E+03	1.46E+08	1.04E+08	**Power**	**2.24E+00**
4	0.0136	"+2.5"	1:33pm	2:33	60 min		2:48PM	260100	302.3	2.15E+04	2.49E+03	2.85E+03	1.42E+08	1.02E+08		
5	0.0136	"+3.5"	1:33pm	2:33	60 min		2:55PM	260520	302	1.80E+04	2.08E+03	2.39E+03	1.19E+08	8.53E+07	Not Mass Adjusted	
6	0.0141	"-.5"	1:33pm	2:33	60 min		3:03PM	261000	302.6	2.38E+04	2.75E+03	3.04E+03	1.47E+08	1.05E+08	Average Flux	6.17E+07
7	0.0142	"-1.5"	1:33pm	2:33	60 min		3:24pm	262260	302.5	2.30E+04	2.66E+03	2.92E+03	1.40E+08	1.00E+08	Power	1.98E+00
8	0.0139	"-2.5"	1:33pm	2:33	60 min		3:31pm	262680	302.2	2.20E+04	2.55E+03	2.86E+03	1.40E+08	1.00E+08		
9	0.0135	"-3.5"	1:33pm	2:33	60 min		3:39Pm	263280	302.1	1.90E+04	2.20E+03	2.54E+03	1.29E+08	9.21E+07		

Foils for Tuesday 3/15/2016	Wt (grams)	Position	Time In	Time Out	Irradiation time	Cadmium Wt (grams)	Count Start	Count Time	Count Total	Count/eff*branch*time	Mass Adjusted	Total Flux	Thermal Flux
10	0.0092	"-2.5"	2:47	3:47	60 min	0.7034	3:47pm	172800	300.7	6.03E+03	7.01E+02	7.01E+02	3.98E+07	2.85E+07
11	0.0091	"+2.5"	2:47	3:47	60 min	0.6231	3:54pm	173220	300.7	5.99E+03	6.96E+02	7.04E+02	4.05E+07	2.90E+07

Cadmium Ratio Between 10 and 8)	3.52E+00
Cadmium Ratio Beween 11 and 4	3.51E+00

Foils for Wednesdays 3/16/16	Wt(grams)	Position	Time In	Time Out	Irradiation Time	Cadmium Wt (grams)	Count Start	Count Time	Count Total	Count/eff*branch*time	Mass Adjusted	Total Flux	Thermal Flux		
12	0.014	center	2:31	3:12	40 min		4:02Pm	89400	151.4	1.33E+04	3.07E+03	3.07E+03	1.34E+08	1.00E+08	**Average Flux**	**7.36E+07**
13	0.0135	"+1"	2:31	3:12	40 min		4:06pm	89640	302.9	2.70E+04	3.12E+03	3.23E+03	1.46E+08	1.09E+08	**power**	**2.36E+00**
14	0.0094	"+2"	2:31	3:12	40 min		4:14pm	90120	150.8	6.89E+03	1.60E+03	2.38E+03	1.55E+08	1.16E+08		
15	0.0137	"+3"	2:31	3:12	40 min		LEFT IN VAULT ACCIDENTLY						0.00E+00	0.00E+00		
16	0.0135	"-1"	2:31	3:12	40 min		4:18pm	90360	151.4	1.27E+04	2.93E+03	3.04E+03	1.38E+08	1.03E+08		
17	0.0141	"-2"	2:31	3:12	40 min	0.7034	4:23pm	90660	150.4	3.89E+03	9.04E+02	8.98E+02	3.90E+07	2.92E+07		
18	0.0138	"-5.75"	2:31	3:12	40 min	0.6231	4:27pm	90900	150.3	2.03E+03	4.72E+02	4.79E+02	2.13E+07	1.59E+07		

Cadmium Ratio Between 17 and 14	3.97E+00

Branching ratio	0.9558
Efficieinch	0.029924994
Half Life Au-198 Sec	233020.8
Decay Constant (S^-1)	2.97462E-06

February 23rd, 2016 Lab Data.

Foil	Locations (in inches)	Foil Weight (grams)	Time Pulled From Reactor	Time Detector Started Count	Difference of Times	Difference of time (s)	Counting Time	Counts		Corrected Activity	All Flux	Thermal
1	Center	0.0088	1:20	2:29	1:09	4140.0	301.3	5500.000	1170.31	1.17E+03	5.02E+07	3.68E+07
2	+.5'	0.0094	1:20	2:36	1:16	4560.0	301.2	5650.000	1203.24	1.20E+03	4.84E+07	3.32E+07
3	1.5'	0.0093	1:20	2:43	1:23	4980.0	301.1	5640.000	1199.91	1.20E+03	4.89E+07	3.39E+07
4	+3'	0.0093	1:20	2:49	1:29	5340.0	301.1	4800.000	1020.83	1.02E+03	4.17E+07	2.89E+07
5	-.5'	0.0093	1:20	2:56	1:36	5760.0	300.9	5690.000		1.21E+03	4.95E+07	3.43E+07
6	-1.5'	0.0095	1:20	3:03	1:43	6180.0	301.2	5680.000		1.20E+03	4.84E+07	3.28E+07
7	-3'	0.0093	1:20	3:09	1:49	6540.0	301.4	4776.092	1017.13	1.01E+03	4.16E+07	2.88E+07
8	-1.5'	0.0092	2:18	3:15	0:57	3420.0	300.6	1520.000		3.23E+02	1.33E+07	9.30E+06
9	+1.5'	0.0093	2:18	3:21	1:03	3780.0	300.7	1930.000		4.10E+02	1.67E+07	1.16E+07

Thermal Max **3.68E+07**
Thermal Av **2.34E+07**
Power **0.748151932**

From Genie 2000

Yield Percent of AU-198	0.9558
Detector Efficency for 411.8Kev	1.6384918
Au-198 Half life (sec)	233020.8
Decay Constatn S^-1	2.97462E-06
CD ratio between 8 and 6	**3.74E+00**
CD ratio between 9 and 3	2.92E+00

Appendix II: AGN201 Simplified Model

```
AGN201 Simplified Model
c  Trevor Boaz
c Cell Cards
c k=1.004
c Air in glory hole
10 300 -.001205 -10 12 -13                imp:n=1 $
c Aluminum around Glory Hole
20 200 -2.6989 10 -11 12 -13              imp:n=1 $
c Fuse
25 500 1.12828199E-01 70 -71 -72          imp:n=1 $
c Air around fuse
26 300 -.001205 60 -70 -80 #25            imp:n=1 $
c 26 0 60 -70 -80                         imp:n=1 $
c Reactor Core
30 100 1.199628696E-01 11 50 -51 -52  #26 #25       imp:n=1 $
c Graphite around core
40 400 7.9177862857143E-02  11 60 -61 -62 #30  #25 #26   imp:n=1 $
c BOUNDING
400 0 -500 #10 #20 #30 #40                imp:n=1 $
401 0  500                                imp:n=0 $

c Surface Cells
10 cx   1.1    $Inner glory hole dimensions diameter is 2.2cm
11 cx   1.43   $Outer glory hole dimension, thickness is .33cm
12 px -32.85   $Place holder of back edge of tube
13 px  32.85   $Place holder of front edge of tube
c Reactor Dimensions
50 pz -12      $Bottom of fuel plates
51 pz  11.7    $Top of fuel plates (NO RAD included 11.7+12=23.7)
52 cz  12.8    $Fuel radius
c Graphite Dimensions
60 pz -34.5    $Bottom of graphite
61 pz  34.2    $Top of graphite
62 cz  32.8    $Radius of graphite
c Fuse Dimensions
70 pz  -2.38  $just below glory hole 1.43+.95
71 pz  -1.43  $ top of fuse
72 cz   1.1   $diameter is 2.2
c Air around fuse
80 cz   1.36  $Widest dimension
c BOUNDING
```

```
500 so 100

c  Data Cards
kcode    10000 1.0   150   2000
ksrc     0 0 0
c ***************************************************
c Tally Cards
c Tally below is to find the peak flux using FMESH
c Reaction rate fmesh
c 3d Fmesh
c      up to the reactor core heigh of 11.7cm (top of core)
fmesh14:n geom=xyz origin= -9 -9 1.43 $ x, y, z
        imesh= 9      iints=50
        jmesh= 9      jints=50
        kmesh=11.7    kints=50
        Emesh= 4E-07
c F4 tallies
F4:n 25 30  T             $Fissile Cells (Fuse, core, and both)
c FM4 1.0  1.0 1.0          $No modification
E4 4E-07 $.4 eV
SD4 3.61 12142.93 12146.54  $Define volume for the bins
c
c F7:n 25 30
c FM7 1.0
c SD7 3.61 12142.93
c ****************************************
c Material Cards
c U02 in C2H4    atom density= 1.199628696E-01
m100 92235.70c   1.42205251457751E-04
    92238.70c   5.69573100E-04
    92234.70c   8.5492372154227E-07
    1001.70c    7.8545932730626E-02
    6000.70c    3.92771498281218E-02
    8016.70c    1.427693223911E-03
c
c Aluminum      rho= 2.6989 g/cc
m200 13027.70c   -1.0
c
c Air          rho=.001205
m300 6000       -.0000124 $ C
    7014       -.755268 $ N
    8016       -.231781 $ O
    18000       -.012827$ Argon
c Graphite      rho=1.58 g/cm3
m400 6000.70c    7.9177862857143E-02
c Fuse          atom density=1.12828199E-01
```

```
m500 92235.70c   2.63E-04
    92234.70c   1.58E-06
    92238.70c   1.05E-03
    92236.70c   2.76E-06
    8016.70c    2.64E-03
    8017.70c    1.00E-06
    6000.70c    3.63E-02
    1001.70c    7.26E-02
    1002.70c    8.35E-06
c Gold          rho=19.32 g/cm3
c m600 79197     -1
c Cadmium       rho=8.650 g/cm3
c m700 48000     -1
c S-Alpha/Beta Treatments
mt100   poly.10t o2/u.10t
mt200   al27.12t
mt400   grph.10t
mt500   poly.10t o2/u.10t
```