URANIUM-232 BERYLLIDE NEUTRON SOURCE

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By

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The discovery of nuclear reactions need not bring about the destruction of mankind any more than the discovery of matches.

Albert Einstein

To Alice.

And the sincere hope that she might one day open this thesis and read this dedication

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It wasn't easy leaving my homeland of Philadelphia: the land of cheesesteaks, soft pretzels, the E-A-G-L-E-S Eagles!, well laid out streets and drivers who know how to use a turn signal - none of which can be found in Atlanta. It was much harder to enroll at the Georgia Institute of Technology, one of the finest engineering schools in the world. The fact that I am typing this now is proof that I have survived my foray into the Southland and even more heroically, Georgia Tech. This could have never been possible without the support, guidance and patience of dozens of individuals. I will try to list them all here. And while there are many listed on these two pages, let it be known that I am not simply naming anyone who said "hi" to me in the hallway; these individuals all earned this acknowledgement!

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A	Activity
a	A Constant, Ratio of Masses in a Collision
B	Fuel Burn-up Factor
χ	Concentration of an Isotope in ppm
-dE/dx	Stopping Power
E_α	Alpha Particle Energy
E_{ex}	Energy Required to Excited a Nucleus to a Higher Energy Level
E_n	Neutron Energy
$\varepsilon(E)$	Stopping Cross Section
f^a_{kl}	Fraction of a emissions with a given alpha particle energy
G	Number of Energy Groups
I	Number of Isotopes of an Element
J	Number of Elemental Constituents
K	Number of Alpha Emission Energy Groups
λ	Decay Constant
m	Mass
mml	Measured mass of a given molecule
mml'	Theoretical mass of a given molecule with varying isotopic composition from the measured mass mml
N	Total Atom Density
N_i	Atom Density of Isotope i
n	A Constant, number of beryllium atoms in a reaction

ν	Number of neutrons produced per ppm of a given isotope in a mixture
Q	Reaction Energy
P	Probability
R	Ratio (target/source)
σ_i	(α,n) Cross Section
S	Stopping Power
t	time
$^{dirty}U$	Dirty Uranium, ^{233}U with 300 ppm ^{232}U Contamination
X	An Actinide
Y	Neutron Yield

SUMMARY

In this thesis the history and technical details of isotopic (α,n) neutron sources are discussed. The use of ^{232}U as part of dirty uranium was proposed as a new isotope for use in beryllium (α,n) neutron sources. A $^{232}UBe_{13}$ neutron source was designed and modeled and the fluence and flux distributions were calculated. The ^{232}U decay chain emits six high energy alpha particles in quick succession and is ideal for use in a beryllium (α,n) neutron source. ^{232}U is an undesirable byproduct in the production of ^{233}U in the thorium fuel cycle; its concentrations can vary from 5-3000 ppm in bred ^{233}U. A 1.1018-cm diameter by 1.1018-cm tall cylinder of $^{233}UBe_{13}$ with 300ppm ^{232}U at 0.74 GBq (20 mCi) was modeled and found to have a peak yield of $3.5*10^5$ n/s after 10.17 years. At this peak yield, the $^{232}UBe_{13}$ source has better neutron production efficiency per initial alpha emission activity than other beryllide neutron sources.

CHAPTER 1

INTRODUCTION

1.1 Historical Review of Isotopic (α,n) Neutron Sources

The history of neutron sources is the history of modern nuclear engineering. The neutron was first discovered by Chadwick in 1932 when he bombarded a beryllium foil with alpha particles. In the years after his discovery other researchers produced neutrons by bombarding light elements with alpha particles from alpha emitting nuclides (isotopic (α,n) neutron sources) for their own experiments, until fission reactors were developed. Even these fission reactors required a neutron source to begin their chain reactions. Without these kinds of neutron sources there would be no nuclear power today, and nuclear applications would likely be limited to a mere scientific curiosity.

The Manhattan Project of World War II and the Cold War that immediately followed were a golden age for nuclear science. Researchers all over the world were performing a multitude of experiments to better understand the nucleus of the atom. Many of these experiments required an efficient source of neutrons. There was a major effort to understand isotopic neutron sources and the way to make them more efficient, cost effective and safe.

The first isotopic neutron sources utilized the alpha emissions of radium-226 (^{226}Ra) and its progeny incident on a beryllium mixture to produce neutrons. ^{226}Ra was first used because it was well studied as a radioactive source and it was relatively plentiful compared with other high energy alpha emitting isotopes. Polonium-210 (^{210}Po, RaF), which itself is a decay product of ^{226}Ra, was also used as an isotope in early

neutron sources. Another early radioactive material used in isotopic neutron sources was actinium-227 (^{227}Ac), but because of its relative scarcity, this source was rarely used.

It was found early on that beryllium had the best neutron yields of the light elements. Therefore nearly all isotopic neutron sources after the 1950's were a combination of an alpha emitter and beryllium. However some isotopic neutron sources used fluorine, boron or lithium instead of beryllium. As a matter of nomenclature, isotopic neutron sources have usually taken the form <alpha emitter (or dominant radionuclide) chemical symbol> - <light element chemical symbol>. Usually the exact isotopes of the nuclides are omitted from this notation. Ex: Ra-Be, Pu-Be, Ra-F.

As time progressed transuranic elements were created and extracted, mostly from the waste streams of the nuclear weapon programs. These isotopes, for example plutonium-239 (^{239}Pu) and americium-241 (^{241}Am), were also found to have desirable traits for an isotopic neutron source. The low gamma emissions, high abundances and half lives of the new artificial alpha emitters ultimately made them more desirable for use as neutron sources and the older Ra-Be and Ac-Be sources were phased out.

These neutron sources had been manufactured in large numbers and have been used reliably by researchers, oil expatiators, the nuclear power industry, the military and others since their creation. However political and security concerns in recent years have led to a global desire to retire isotopic neutron sources. Replacing these neutron sources are d-t neutron sources and accelerator based neutron sources. Both of these sources have their advantages over isotopic neutron sources, but they also have disadvantages such as very fast neutron energies and the devices themselves can often require a very large volume and energy to operate.

One of the goals of this thesis is to outline the science behind isotopic neutron sources and how they are manufactured. Presently most of this knowledge is spread throughout decades of journals. It is the hope of the author that this thesis may be used as an easy reference for those engineers in the future who wish to understand how to produce these neutron sources. Another goal of this paper is to outline the use of uranium-232 (^{232}U), an isotope with superior qualities for use in future isotopic neutron sources.

1.2 Historical Review of Uranium-232

Uranium-232 was first discovered by Gossman in the Manhattan Project (Newton, 1949). This isotope had no immediate engineering applications and it existed mostly as a scientific curiosity. It was briefly considered for use in the space program, but the hard gamma emissions of its progeny made its use in space based applications or most other applications very limited. ^{232}U was the second isotope found to undergo cluster (exotic) decay (Bonetti et al., 1990) and was studied extensively for it decay properties.

There has also been extensive study into the creation of ^{232}U, specifically how it is inadvertently created when breeding uranium-233 (^{233}U) in the thorium fuel cycle. Because of the hard gamma emissions of the ^{232}U progeny, many engineers have proposed methods for breeding ^{233}U with as little ^{232}U production as possible. One of the principal reasons that the thorium fuel cycle is not in use today is because of the ^{232}U content (IAEA TECDOC -1450, 2005).

CHAPTER II

(α,n) Reactions

2.1 (α,n) Reaction Theory

Perhaps one of the most well studied alpha (α) particle nuclear reactions is the reaction in which an α particle undergoes fusion with a light nuclei and the compound nucleus ejects a neutron when returning to a ground state.

2.1.1 (α,n) Targets

Several light nuclei are prone to undergoing the (α,n) reaction. The nuclei are referred to as targets and are listed in Table 1, along with the reaction energy (Q value), threshold energy for the reaction and relative neutron yield and neutron energy for a 5.5 MeV α particle.

Table 1: Traits of Common (α,n) Targets

Target	Q[1] (MeV)	Threshold Energy[1] (MeV)	Mean E_n [2] for E_α= 5.5 MeV	Yield[2] (n/10^6 α) E_α= 5.5 MeV
^{6}Li	-3.975	6.620	NA	NA
^{7}Li	-2.79	4.382	0.5883	3.156
^{9}Be	5.702	exothermic	5.005	80.073
^{10}B	1.06	exothermic	2.243	5.72
^{11}B	0.157	exothermic	2.993	23.724
^{13}C	2.215	exothermic	4.72	9.904
^{17}O	0.587	exothermic	2.523	0.152
^{18}O	-0.697	0.852	2.374	0.333
^{19}F	-1.95	2.361	1.304	0.106

[1]Values in MeV Source: Shultis and Faw, 2000
[2]E_n in MeV, derived from SOURCES-4c

As shown in Table 1, the greatest neutron yield is achieved when beryllium is used as a target material in the (α,n) reaction. Therefore, beryllium has usually been the material of choice when producing a neutron source. The full reaction is:

$$^{9}_{4}Be(\alpha,n)^{12}_{6}C^{*} \qquad (1)$$

Theoretically any nuclide above helium is prone to undergo the (α,n) reaction, but only those isotopes which can undergo an exothermic or low energy threshold (α,n) reaction are considered. Even under the best circumstances the likelihood of an (α,n) reaction is quite small. For instance the (α,n) cross section for an 8 MeV alpha particle incident on beryllium is on the order of 0.6 barns. Typically the probability for most target nuclei is on the order of a few reactions per million alpha particles. The (α,n) cross section generally increases with alpha particle energy, as shown on Figure 1.

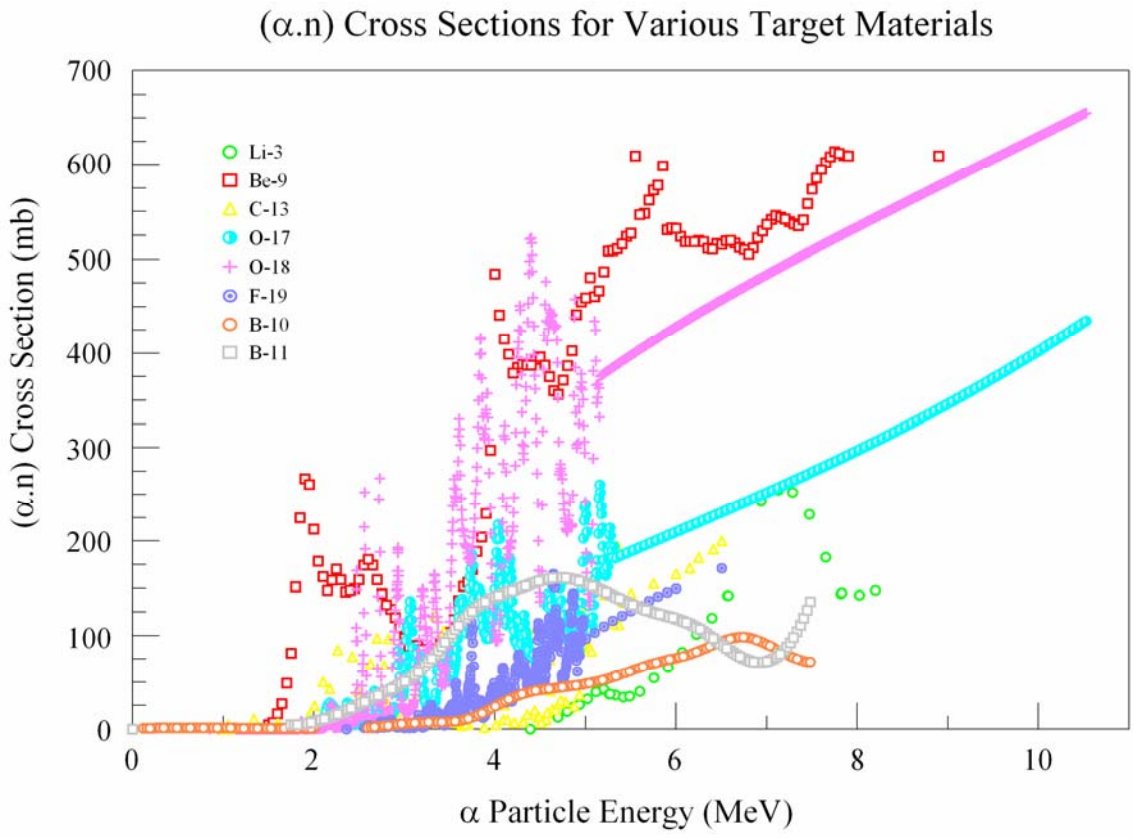

Figure 1: (α,n) cross sections, source: SOURCES-4c (tape3), 2002

When an alpha particle fuses with a target nucleus, the compound nucleus is in an energetic state. If this added energy is enough to overcome the threshold energy requirement for the (α,n) reaction, then a neutron will be emitted. Occasionally the α particle will bring in enough energy (often through its kinetic energy) that the compound nucleus will be left in an excited state after the neutron is emitted. Neutron emissions are characterized by the energy state in which they leave the nucleus (Van der Zwan, 1968). These neutron groups are identified by n_0, n_1, n_2, etc. corresponding to the energy state of the nucleus.

2.1.2 (α,n) Radionuclides

Alpha emitting radionuclides or "emitters" or "source" are a necessary component of (α,n) neutron sources. Alpha particles are a natural decay mode for many nuclei with an atomic number greater than 82 (lead). Lanthanides are also prone to undergoing α decay but these nuclides typically have a very long half-life and low decay energy. As a rule of thumb, the decay energy of an isotope is inversely proportional to its half life. Ergo isotopes with very long half lives such as Th-232 and many lanthanides have low decay energies. Since the probability of an (α,n) reaction increases with α energy and the number of α emissions of an isotope is inversely proportional to its half-life, the ideal α emitters would have a relatively short half life. It should be noted though, that isotopes with too short a half life will become too impractical to produce and use effectively as an (α,n) neutron source. Therefore, a reasonable half-life for α emitters, for use in a neutron source, should be on the order of 1 to 25,000 years. Table 2 list many common (save 4n series) radionuclides used in (α,n) neutron sources today, along with their half-lives,

common decay energies and thick target yield with beryllium (in $n/10^6\ \alpha$). Neutron spectra of some targets can be found in Appendix C.

Table 2: (α,n) Source Properties

Source	$t_{1/2}$[1]	Most Common E_α (MeV)[2]	Yield ($n/10^6\alpha$)[3]
^{239}Pu-Be	24110 yr	5.156	57.2
^{210}Po-Be	138.38 d	5.304	63.7
^{241}Am-Be	432.2 yr	5.486	71.5
^{238}Pu-Be	87.7 yr	5.499	72.1
^{238}Cm-Be	18.1 yr	5.805	105.3
^{244}Cm-Be	162.8 d	6.113	88.1
^{226}Ra-Be	1600 yr	4.784, 5.490, 6.002, 7.687, 5.304	521.6
^{227}Ac-Be	21.8 yr	6.038, 5.716, 6.819, 7.386, 6.623	736.8
4n Series			
^{228}Ra-Be	5.75 yr	5.4223, 5.686, 6.288, 6.779, 6.051 (36%), 8.784 (64%)	707.2
^{228}Th-Be	1.911 yr	as ^{228}Ra	707.2
^{232}U-Be	68.9 yr	5.320 + ^{228}Ra	755.5

[1] Source: Tuli, 2005
[2] in MeV, Source: Stabin and de Luz, 2002
[3] in $n/10^6\ \alpha$, From eq. (14)

Alpha decay is rarely the only decay method of a given isotope. Minor gamma (γ) or Auger electron emissions can accompany or follow an α emission. Furthermore the progeny of an α emitting nuclide is usually itself radioactive. Thus other emissions must be considered when handling or preparing the radionuclides with respect to radiation safety and other forms of radiation damage.

Additional α emissions can also contribute to the neutron yield of a neutron source. For many of the α emitters list on Table 2, the half life of the immediate progeny is longer than that of the parent nuclide. Therefore these extra alpha emissions can

usually be neglected from neutron yield calculations. However for ^{226}Ra, ^{227}Ac and 4n Series, the progeny's alpha emissions cannot be ignored. These isotopes are near the end of the uranium (4n+2), actinium (4n+3) and thorium (4n) decay chains, respectively; therefore as the long-lived ^{226}Ra, ^{227}Ac and ^{232}U (and ^{228}Ra/Th) decay, the activity of their progeny will rapidly build up and release additional α and beta (β) particles. The additional α emissions of the progeny will increase the total (α,n) output of the source by an order of magnitude. ^{228}Th and ^{227}Ac sources will come into secular equilibrium within a year of source assembly. ^{226}Ra will come into secular equilibrium with all of it progeny through ^{210}Pb (22.3 yr half-life) within one year. Only one α emission of ^{210}Po remains after ^{210}Pb. ^{228}Ra and ^{232}U sources take 5 and 10.17 years, respectively, for their α emitting progeny to come into equilibrium. The downside to using these kinds of neutron sources is the strong gamma emissions that usually accompany these decay chains.

2.2 (α,n) Yield

2.2.1 Solving for the Initial (α,n) Yield

The (α,n) neutron yield and energy flux for a given source arrangement may be solved using complicated transport equations and solved through Monte Carlo Methods. Those equations can be simplified to algebraic and light calculus expressions. The algorithm for finding the exact neutron yield is explained in great detail in Wilson et. al (1999), but the important equations of that document are presented here.

The probability of an alpha particle with energy E_α, undergoing an (α,n) reaction with target nucleus *I*, before it comes to rest, can be expressed by:

$$P_i(E_\alpha) = \int_0^{E_\alpha} \frac{N_i \sigma_i(E)}{-\left(\frac{dE}{dx}\right)} dE \tag{2}$$

Where N_i is the atom density of the target, $\sigma_i(E)$ is the (α,n) cross section of the target at energy E, $-dE/dx$ is the materials stopping power.

Since all (α,n) problems involve more than one material (emitters, target, impurities, progeny, additional targets and emitters), it is more accurate to describe the neutron output in terms of the stopping cross sections $\varepsilon(E)$:

$$P_i(E_\alpha) = \frac{N_i}{N} \int_0^{E_\alpha} \frac{\sigma_{ig}(E)}{\varepsilon(E)} dE \tag{3}$$

Where N is the total atom density of the system and $\varepsilon(E)$ is defined as:

$$\varepsilon(E) \cong \frac{1}{\sum_{j=1}^{J} N_j} \sum_{j=1}^{J} N_j \frac{-1}{N} \frac{dE}{dx} \tag{4}$$

and J is the number of elemental constituents.

It may be necessary to also separate the cross section σ_i into G cross sections each representing the (α,n) cross section of a neutron leaving the nucleus in energy level g:

$$P_i(E_\alpha) = \frac{N_i}{N} \sum_{g=0}^{G} \int_0^{E_\alpha} \frac{\sigma_{ig}(E)}{\varepsilon(E)} dE \tag{5}$$

Where σ_{ig} is the cross section of an (α,n) reaction leaving the nucleus in energy level g.

An expression can be derived for the total (α,n) yield of an isotope k which has L unique alpha energy emissions. Where f^{α}_{kl} is the fraction of all emissions of k with alpha emission of energy level l. The total neutron yield of the system per unit volume will be:

$$Y = \sum_{k=1}^{K} \lambda_k \sum_{l=1}^{L} f^{\alpha}_{kl} \sum_{i=1}^{I} P_i(E_l) \tag{6}$$

Where K is the number of alpha emitting isotopes and λ_k is the decay constant of isotope k.

2.2.2 Estimating the (α,n) Yield of an Aged Source

The above equations are very helpful in determining the initial (α,n) production of a neutron source. The neutron yield depends on the present activity of the alpha emitter(s). The activity of an isotope at a given time elapsed t is:

$$A(t) = A_0 * \exp(-\lambda_d * t) \tag{7}$$

Where A_0 is the initial activity and λ_d is the decay constant of the decay driving isotope.

The decay driving isotope is the usually the isotope with the highest activity in the system and is likely controlling the activity of its progeny. It need not be an α emitter itself, as in ^{227}Ac-Be and ^{228}Ra-Be sources. For a source with K decay driving isotopes and a time, t, after secular equilibrium is reached, the neutron yield can be estimated by:

$$Y(t) = \sum_{k=1}^{K} Y_{k,\max} * \exp(-\lambda_{d,k} * t) \tag{8}$$

Where $Y_{k,max}$ is the maximum number of (α,n) neutrons produced by alpha emissions of isotope k and its progeny at the peak (α,n) emission rate of the source, $\lambda_{d,k}$ is the decay constant of the decay driving isotope.

2.3 (α,n) Neutron Energy Distribution

In an (α,n) reaction, an alpha particle will collide with a target nucleus and form a compound nucleus. Due to conservation of momentum the compound nucleus will be forced in a given direction and speed. But since the neutron is released in such a short period of time, velocity of the compound nucleus may be neglected. Therefore

momentum must be conserved for the before system (alpha and target) and after system (neutron and product (recoil) nucleus). Because of the conservation of momentum, the energy of the neutron is dependent on the direction from which it is emitted, relative to the incoming α particle. The neutron will have the most energy when it is released in the same direction of the incoming alpha particle. For simplicity, the emission of a neutron is assumed to be isotropic and therefore the neutron will be emitted in all directions equally. The range of energies of a neutron emitted from energy level *m* in the laboratory frame of reference is described by:

$$E_{n(m)}^{\pm} = \left(\sqrt{E_\alpha a_1} \left(\frac{1}{1+a_2} \right) \pm \sqrt{Q_m \frac{1}{1+a_3} + E_\alpha \frac{a_2}{1+a_2} \frac{1}{1+a_3}} \right)^2 \tag{9}$$

Where:

$$a_1 = \frac{m_n}{m_a} \tag{10}$$

$$a_2 = \frac{m_t}{m_a} \tag{11}$$

$$a_3 = \frac{m_n}{m_r} \tag{12}$$

$$Q_m = Q - E_{ex} \tag{13}$$

and E_{ex} is the excitation level of the recoil (compound) nucleus, m_n is the mass of the nucleus, m_r is the mass of the recoil (compound) nucleus, m_α is the mass of the alpha particle and m_t is the mass of the target nuclei.

The minimum and maximum neutron energies can be found with equation (9).

SOURCES-4c is a computer code which calculates models (α,n) reactions. To simplify the calculations, SOURCES-4c assumes that the neutron emissions are isotropic and uniformly disperses the range of neutron energies into *G* evenly spaced energy groups (Wilson et. al, 1999).

The emissions of neutrons in an (α,n) reactions are not isotropic. For better accuracy (Van der Zwan, 1968), each of the *G* energy groups can be weighted based on the probability of a neutron being emitted at a given angle for a given neutron energy. Van der Zwan (1968) published Figure 2, showing the neutron angular cross-sections (distribution) for neutron energy groups 0-2 at various α energies.

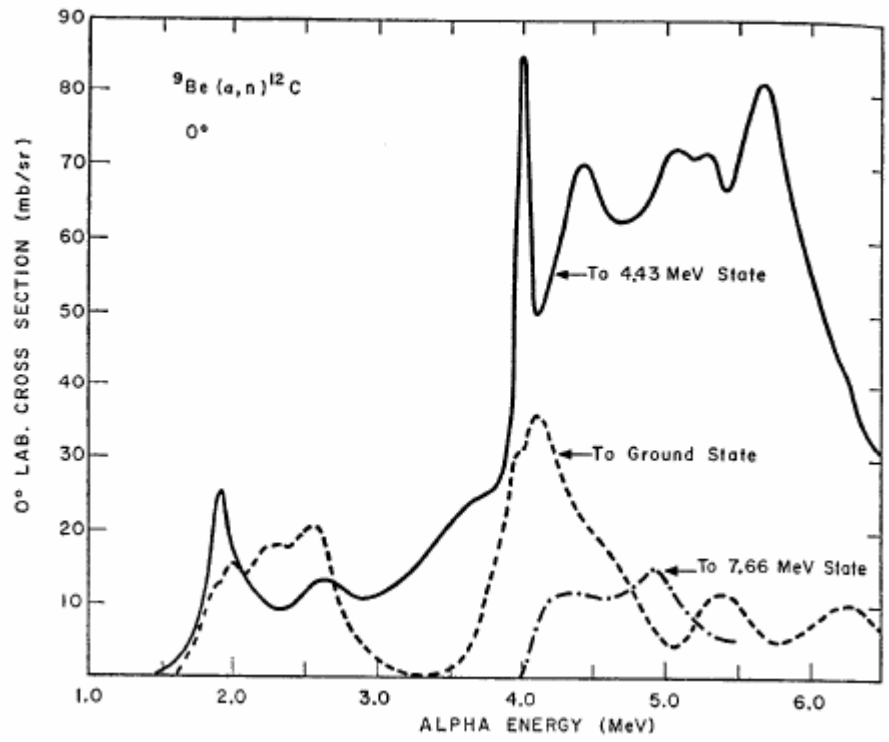

Figure 2: Angular cross sections for $^{9}Be(\alpha,n)^{12}C$ for n_0, n_1, n_2 neutron groups (experimental)

2.4 Thick Target Yields

The simplest and most efficient method of producing (α,n) neutrons is by projecting a monoenergetic beam of alpha particles upon a thick target material. Accelerators are large, bulky and expensive and are therefore not a practical means of producing neutrons for simple applications. But this method of producing (α,n) reactions is an excellent way of determining (α,n) cross sections, as the beam energy is tightly controlled and there are rarely contaminants to interfere with the reaction. Ergo the thick target yield is also considered to be the theoretical maximum neutron yield. Since the Be(α,n) reaction is the most important to nuclear engineers, its thick target yields and cross sections are well studied.

An empirical formula was developed by Anderson and Hertz (1971) to find the theoretical maximum number of neutrons that could be produced by alpha particles incident upon beryllium:

$$\begin{aligned} Y_{TT} &= 0.080 * E_{\alpha}^{4.05} \qquad (4.1 < E_{\alpha} \leq 5.7) \\ Y_{TT} &= 0.800 * E_{\alpha}^{2.75} \qquad (5.7 < E_{\alpha} \leq 10.0) \end{aligned} \tag{14}$$

Where Y_{TT} is the number of neutrons produced per 10^6 alpha particles and E_{α} is the energy in MeV of the incident alpha particle. Equation (14) was used to find the theoretical maximum neutron yield for Table 2.

Similar approximations have been made and reworked by Geiger and Van der Zwan (1975):

$$Y_{TT} = 0.1444 * E_{\alpha}^{3.65} \tag{15}$$

And by Shultis and Faw (2000):

$$\begin{aligned} Y_{TT} &= 0.115 * E_{\alpha}^{3.82} \qquad (3.5 < E_{\alpha} \leq 6.5) \\ Y_{TT} &= E_{\alpha}^{2.64} \qquad (6.5 < E_{\alpha} \leq 10.0) \end{aligned} \tag{16}$$

All three reaction yields very similar neutron yields outputs in the range of alpha energies of most concern to nuclear engineers, 5.14-6.10 MeV, the alpha energies available from ^{241}Am through ^{242}Cm. The yields for equations (14), (15) and (16) diverge from each other above 6.5 MeV. The equation by Anderson and Hertz best matches the experimental neutron output (Geiger and Van der Zwan, 1975) for higher energies.

Using the data provided by Bair and del Campo (1979), several new empirical equations have been created to describe the thick target (α,n) neutron yield of beryllium and several other targets materials. Equations (17) – (22) are plotted against the source data in figure 3.

Beryllium:

$$\begin{aligned} Y_{TT} &= 0.1776 * E_{\alpha}^{3.5} \qquad (3.0 < E_{\alpha} \leq 6.5) \\ Y_{TT} &= .8392 * E_{\alpha}^{2.6796} \qquad (6.5 < E_{\alpha} \leq 9.0) \end{aligned} \tag{17}$$

Fluorine:

$$\begin{aligned} Y_{TT} &= 4 * 10^{-5} * E_{\alpha}^{7.1634} \qquad (3.5 < E_{\alpha} \leq 5.5) \\ Y_{TT} &= .0037 * E_{\alpha}^{4.5095} \qquad (5.5 < E_{\alpha} \leq 8.0) \end{aligned} \tag{18}$$

Natural Boron:

$$Y_{TT} = 9.0179 E_{\alpha} - 29.234 \qquad (3.5 < E_{\alpha} \leq 7.5) \tag{19}$$

Boron-10:

$$\begin{aligned} Y_{TT} &= 1 * 10^{-4} * E_{\alpha}^{6.4345} \quad (3.5 < E_{\alpha} \leq 5.5) \\ Y_{TT} &= .007 * E_{\alpha}^{3.9779} \qquad (5.5 < E_{\alpha} \leq 7.5) \end{aligned} \tag{20}$$

Boron-11:

$$Y_{TT} = 9.0901 * E_{\alpha} - 31.163 \qquad (3.5 < E_{\alpha} \leq 7.5) \tag{21}$$

Natural Lithium:

$$\begin{aligned} Y_{TT} &= 2.122 * E_{\alpha}^{2} - 19.07 * E_{\alpha} + 42.286 \qquad (5.5 < E_{\alpha} \le 6.0) \\ Y_{TT} &= 22.859 * E_{\alpha} - 135.5 \qquad (6.0 < E_{\alpha} \le 9.0) \end{aligned} \tag{22}$$

West and Sherwood performed an experiment in 1982, similar to Bair and del Campo (1979) and obtained the graph in Figure 4 as their results for the thick target (α,n) yields in several materials.

The relation between the thick target yield and the theoretical yield of a homogenous (α,n) system can be found by:

$$Y_{source} = Y_{TT} * \frac{R * S_{target}}{R * S_{target} + \sum_{i=1}^{I} f_i S_i} \tag{23}$$

Where R is the ratio of target to non-targets and S_{target} is the stopping power of the target and S_{emmit} is the stopping power of the emitter and f_i is the fraction of element i with stopping power S_i.

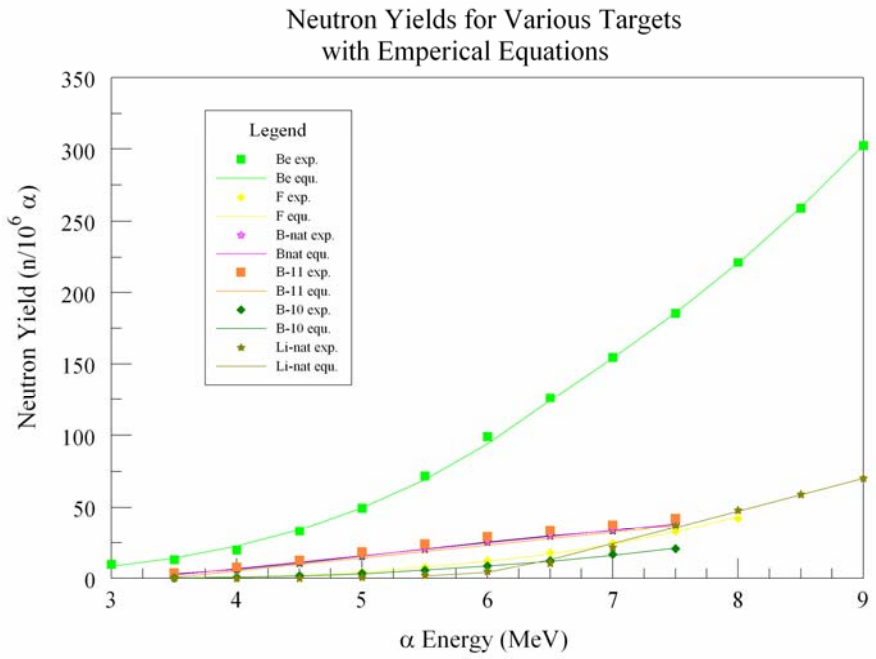

Figure 3: Neutron yields for various targets with plots of equations (17)-(22)

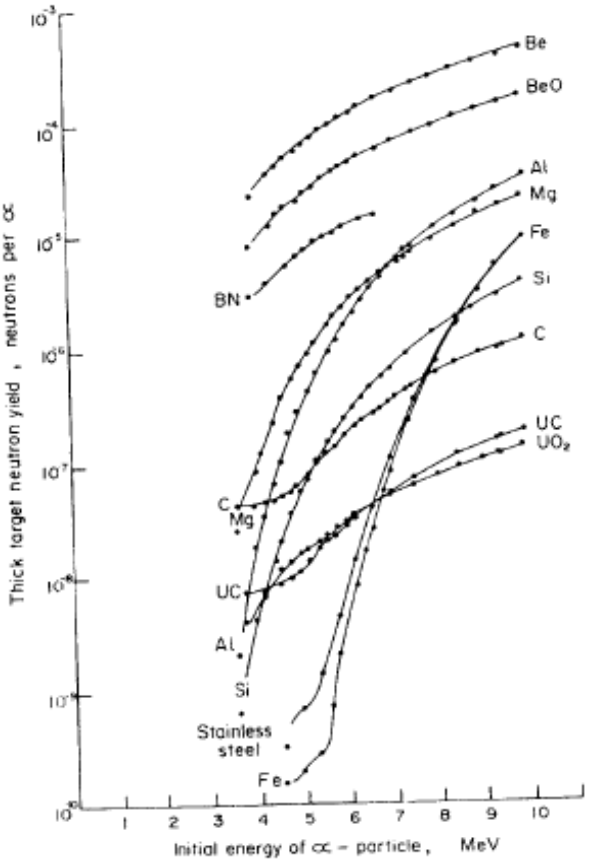

Figure 4: Thick target neutron yields for various materials. (Source: West and Sherwood, 1982, experimental)

Equation (23) is a variation of the equation by Runnalls and Boucher (1957) generalized to systems with more than a single element of target and single isotope emitter. Most (α,n) neutron systems are not entirely homogenous (Van der Zwan, 1968) and, therefore, equation (23) alone is not valid. This is because emitters will tend to form clusters of their own in whatever compound they occupy. Therefore the neutron yield must be calculated within the clusters with equation (23) and then a thick target yield equation may be employed (or another variation on equation (23)) using the α flux

entering that system (target matrix). A further explanation on clusters and the material science of neutron sources can be found in section 2.8.

2.5 Multiplication Reactions

There are other reactions that may occur within an (α,n) neutron source. These reactions are split into two categories: neutron multiplication reactions and photonuclear reactions. Magnitudes of the neutron multiplying reactions are dependent on the source geometry. To maximize these reactions, the volume-to-surface area ratio should be as great as possible to facilitate more reactions for a given volume before the product particles can escape. Therefore both the volume and shape of the source play an important role in determining the magnitude of the multiplication and photonuclear reactions.

2.5.1 Neutron Multiplying Reactions

Within (α,n) sources, neutrons may also be created by multiplication reactions. Beryllium has a high (n,2n) cross section, as do many actinides currently used in beryllium (α,n) neutron sources. Some alpha emitters are also fissionable so they may undergo (n,f) reactions.

2.5.2 Photonuclear Reactions

Some isotopes are sensitive to the photonuclear (γ,n) reaction. At very high energies a photofission (γ,f) may occur in fissionable isotopes, but these reactions are quite rare and are not thoroughly discussed in this thesis. For photonuclear reactions to

occur, a nucleus must capture a gamma ray of energy greater than the separation energy of one of its neutrons. The two most common isotopes that are susceptible to photonuclear reactions are deuterium and beryllium at energies above 2.225 MeV and 1.666 MeV (Profio, 1979) respectively. Some isotopic neutron sources are entirely dependent upon the (γ,n) reaction and are called photoneutron sources. In Table 3 (Cierjacks, 1983) some common photoneutron sources and their properties are listed. The yields of Table 3 assumes a spherical (γ,n) source 2.38 cm in diameter, and a 3.2 mm thick blanket of target (Be or D) material.

2.6 Breakaway Neutrons

Alpha particles may induce the release of a neutron without fusing with a target material. Beryllium is well-known for undergoing the breakaway reaction:

$$ {}^{9}_{4}Be(\alpha,\alpha')^{9}_{4}Be^{*} \rightarrow {}^{8}_{4}Be + {}^{1}_{0}n \qquad (24) $$

Unlike the reaction shown in equation (1) which is exothermic, the reaction shown in equation (24) is endothermic and has a threshold energy of 4.3 MeV. This becomes increasingly important as the α energy increases.

At present a major hurdle in creating a reliable (α,n) computer code is finding the proper fraction of neutron producing reactions (Shores et al., 2003).

Table 3: (γ,n) Source Properties, Source: Cierjacks, 1983					
γ Source	$t_{1/2}$	E_γ (MeV)	Target	E_n (keV)	Yield (n/10^{10} Bq)
^{24}Na	15 hr	2.7541	Be	967	340000
		2.7541	D	263	330000
^{28}Al	2.24 m	1.7787	Be	101	32600
^{38}Cl	37.3 m	2.1676	Be	446	43100
^{56}Mn	2.58 hr	1.8107	Be	129	91500
		2.1131	Be	98	91500
		2.9598	Be	1149	91500
		2.9598	D	365	162
^{72}Ga	14.1 m	1.8611	Be	174	64900
		2.2016	Be	476	64900
		2.5077	Be	748	64900
		2.5077	D	140	25100
^{76}As	26.3 hr	1.7877	Be	109	3050
		2.0963	Be	383	3050
^{88}Y	107 d	1.8361	Be	152	229000
		2.734	Be	949	229000
		2.734	D	253	160
^{116m}In	54.1m	2.1121	Be	397	15600
^{124}Sb	60.2 d	1.691	Be	23	210000
^{140}La	40.3 hr	2.5217	Be	760	10200
		2.5217	D	147	6600
^{144}Pr	17.3 m	2.1856	Be	462	690

2.7 Contaminants

2.7.1 Chemical Contamination

Chemical contaminants can have two serious implications on a neutron source. First, they destroy the homogeneity of the source and can reduce the number of (α,n) reactions by increasing the stopping power of the system. This reduces the possibility of an α particle colliding with its intended target. Additionally some contaminant materials can themselves be targets. On Table 4, taken from West and Sherwood (1982), α particle interactions with several target materials and the γ emissions they often released in these

reactions are listed. These γ emissions add to the radiation dose of a neutron source. The γ emissions can be used as a method for determining what chemical contaminants exist in a material.

2.7.2 Isotopic Contamination

Isotopic separation can never economically produce a pure sample of a given isotope in large quantities, but it can produce some nearly pure samples. For example, a ^{239}Pu-Be source may contain Pu-239 (91.6%), Pu-240 (7.7%), Pu-241 (0.7%) and trace amounts of Pu-238 and Pu-242. Pu-241 with a half-life of 14.4 years will beta decay to Am-241 which itself is an emitter of choice in many (α,n) sources.

The decay of Pu-241 into Am-241 cannot be ignored because it will increase the neutron yield over time. For instance, Anderson (1967) claims that for the above isotopic composition the neutron yield will increase the first year by bout 2%. After 69.5 years the neutron yield will peak at about 33% more than the original neutron yield. A more thorough discussion of this phenomenon can be read in Anderson's paper.

The emissions of radioactive contaminants and their progeny should always be considered when they are present in appreciable quantities within an (α,n) neutron source.

2.8 Material Science of (α,n) Sources

As discussed in section 2.4, the (α,n) reaction is highest when the medium is primarily composed of the target isotope. To most efficiently utilize the α emissions, the

Table 4: α induced reaction in light elements Source: West and Sherwood, 1982		
Element	E_γ (MeV)	Origin Reaction
Be	4.439	$^{9}Be(\alpha,n\gamma)^{12}C$
B	0.718	$^{10}B(\alpha,\alpha'\gamma)^{10}B$
	1.632	$^{11}B(\alpha,n\gamma)^{14}N$
	2.313	$^{11}B(\alpha,n\gamma)^{14}N$
C	6.130	$^{13}C(\alpha,n\gamma)^{16}O$
O	0.351	$^{18}O(\alpha,n\gamma)^{21}Ne$
	1.395	$^{18}O(\alpha,n\gamma)^{21}Ne$
	1.982	$^{18}O(\alpha,\alpha'\gamma)^{18}O$
F	0.583	$^{19}F(\alpha,n\gamma)^{22}Na$
	0.891	$^{19}F(\alpha,n\gamma)^{22}Na$
	1.236	$^{19}F(\alpha,\alpha'\gamma)^{19}F$
	1.275	$^{19}F(\alpha,p\gamma)^{22}Ne$
	1.528	$^{19}F(\alpha,n\gamma)^{22}Na$
	1.555	$^{19}F(\alpha,p\gamma)^{22}Ne$
	2.081	
Mg	0.844	$^{24}Mg(\alpha,p\gamma)^{27}Al$
	1.014	$^{24}Mg(\alpha,p\gamma)^{27}Al$
	1.273	$^{26}Mg(\alpha,n\gamma)^{29}Si$
	1.369	$^{24}Mg(\alpha,\alpha'\gamma)^{24}Mg$
	1.779	$^{25}Mg(\alpha,n\gamma)^{28}Si$
Al	0.709	$^{27}Al(\alpha,n\gamma)^{30}P$
	0.844	$^{27}Al(\alpha,\alpha'\gamma)^{27}Al$
	1.014	$^{27}Al(\alpha,\alpha'\gamma)^{27}Al$
	1.263	$^{27}Al(\alpha,p\gamma)^{30}Si$
	1.264	$^{27}Al(\alpha,n\gamma)^{30}P$
	1.454	$^{27}Al(\alpha,n\gamma)^{30}P$
	2.236	$^{27}Al(\alpha,p\gamma)^{30}Si$
Si	1.266	$^{28}Si(\alpha,p\gamma)^{31}S$
	1.779	$^{28}Si(\alpha,\alpha'\gamma)^{28}Si$
	2.230	$^{29}Si(\alpha,n\gamma)^{32}S$
	2.234	$^{28}Si(\alpha,p\gamma)^{31}S$

ratio of target atoms to alpha emitter atoms should be as high as possible, typically in excess of 200:1 (Wauchope and Baird, 1959), with the α emitters distributed as homogenously as possible within the target mixture. Aside from the homogeneity of the source, the frequency of reactions shown in equations (1) and (23) are independent of the macroscopic source geometry.

In the case of the common target beryllium, homogeneity may be difficult to achieve. The radionuclide and target will tend to form segregated clusters of atoms within the mixture. This severely limits the collisions of α particles with beryllium if the α emitter cluster is wider than the range of the alpha particle. Runnalls and Boucher (1956) report that the range of a typical α particle is about 20 μm. They also explain that actinides will form an XBe_{13} intermetallic compound (crystal), where X is the chemical symbol for the given actinide.

2.8.1 X Beryllide

The XBe_{13} crystal is referred to as X beryllide. This crystal also forms its own clusters that can be quite large when they are in a large matrix of beryllium. This effectively reduces neutron production since the alpha emitters will be locally surrounded by more α emitters and less beryllium nuclei. The formations of XBe_{13} crystals are a cause for concern since most alpha emitters of value in the construction of beryllium (α,n) neutron sources are actinides.

The positive side to the formation of the intermetallic compound XBe_{13} is that its density is much greater than the theoretical density of the emitter and beryllium combined (when not in crystal form). In the case of UBe_{13}, McElfresh et al. (1990) reported the density to be 4.359 g/cm^3. The theoretical density of the mixture, however, was calculated to be 3.070 g/cm^3. Wauchope and Baird (1959) explain that the greater density of the XBe_{13} allows for the best neutron economy with regards to volume. For this reason, many modern neutron sources have a 13:1 ratio of beryllium to α emitter and are comprised almost entirely of XBe_{13} crystals.

CHAPTER 3

ISOTOPIC NEUTRON SOURCE CONSTRUCTION

3.1 Source Preparation

The most important aspect of creating an isotopic neutron source is the metallurgical and/or chemical process of creating the source material. The preparation of the source will vary depending on the target material and radionuclide. Older, and less efficient, neutron sources will have fine (≤200 mesh) particles of actinide metal or oxide compressed with fine particles of the target nuclide (or some compound containing the target nuclei). Michaud and Boucher (1960) found that 200 mesh particles were the most economical for achieving a high neutron yield.

Presently most (α,n) sources are made of the intermetallic compound: XBe_{13}. Radium sources made of $RaCO_3$ in a fine beryllium powder (IAEA, 2003).

To produce the most amount of (α,n) reactions, neutron sources should be as dense and homogenous as possible. Therefore ideal neutron sources are made entirely of a ceramic, crystal or metallic compound. Most (α,n) sources use an actinide and beryllium, so barring any major nuclear or metallurgical discovery, it is doubtful that new (α,n) sources will stop using the XBe_{13} composition. Therefore the procedures for creating an X-Be compound is summarized here. The process of creating an X-Be compounds are explained in more detail by Tate and Coffinberry (1958), Runnalls and Boucher (1956), and Michaud and Boucher (1960).

Tate and Coffinberry (1958) created a Pu-Be mixture by melting the metals of the two elements together under inert atmospheric pressure and thoroughly mixing the two. While this is the simplest option, several chemical and safety concerns must be addressed

when using this method. First, plutonium metal is usually prepared through fluorination which is costly. Second, preparing and handling plutonium metal requires extensive safety precautions.

Runnalls and Boucher (1956) created X-Be alloys by fluorination reduction at 1400 K using the reaction (in vacuum):

$$2XF_{3(s)} + (3+n)Be_{(s)} \rightarrow 2X_{(l)} + nBe_{(l)} + 3BeF_{2(s)} \tag{25}$$

Where *n* is a positive number and *X* represents an actinide element.

Runnalls and Boucher (1956) write that the beryllium fluoride product were easily distilled away at these temperatures leaving the X and Be metals. The X-Be alloys were created by melting the metals together at about 1650 K.

Michaud and Boucher (1960) created a X-Be compound by reducing X-oxide with beryllium in vacuum at 1725 K. The reaction follows:

$$\begin{aligned} XO_2 + (n)Be &\rightarrow 2BeO + (n-2)Be && (n>15) \\ XO_2 + (n)Be &\rightarrow XBe_{13} + 2BeO + (n-15)Be && (n \geq 15) \end{aligned} \tag{26}$$

Assuming the beryllium-to-actinide ratio was greater than 13:1, this resulted in an XBe_{13} embedded in a porous beryllium matrix. Sintering reduced the porosity of the beryllium and removed the BeO from the mixture. Michaud and Boucher (1960) wrote that Sintering was best achieved from 1673-1723 K to sinter mixtures with a beryllium to plutonium (actinide) ratio of 15:1. Ratios greater than 30:1 had to be sintered at a temperature no greater than 1473 K, lest the neutron output would drop. This is because the beryllium matrix would begin to melt around the solid $PuBe_{13}$ (XBe_{13}) clusters. These clusters would migrate near each other and form larger clusters. These larger clusters

would limit the effect beryllium-to-plutonium ratio of the system and thus reduce the neutron output.

Michaud and Boucher (1960) reported that compressing the neutron source would have little effect on the volume of the source beyond sintering.

Wauchope and Baird (1959) performed experiments using a method similar to Michaud and Boucher (1960). They reported two concerns when preparing X-Be compounds in this manner, the first being the tendency for the mixture to violently outgas at 393 K and 693 K. They recommended holding the temperature of the furnace at these temperatures until the pressure inside the furnace returned to atmospheric. They also found that the mixtures were "very porous and mushroom shaped at the top". To remedy this outcome they recommended placing a weight atop the container in which the alloy is being prepared. Michaud and Boucher found that compound made with a 13:1 (exact) beryllium-to-actinide ratio did not sinter well at 1673 K. They believe this is due to the high melting point of the XBe_{13} compound. They found that adding a "small excess of beryllium" aided the sintering of the neutron source. Wauchope and Baird used 200 mesh beryllium powder to replicate this procedure.

It is ideal to create a neutron source with a beryllium-to-actinide ratio of 13:1 because it is the most volumetrically efficient. Wauchope and Baird (1959) explain that the ≥200:1 ratio causes the overall volume of the source to be quite large for many α emitters and sometimes impractical for many applications. If the application of the neutron source allows for a physically larger neutron source, then the engineer may create a mixture with a ratio in excess of 13:1, perhaps even as large a 300:1, as this allows a better neutron production per alpha particle.

Special care must be taken to ensure that the XBe_{13} clusters are as fine and homogenously spread throughout the mixture as possible. As mentioned in section 2.8 the range of an α in beryllium is about 20 μm; therefore the XBe_{13} clusters should be much smaller than this, and the distance between the XBe_{13} clusters should be about 20 μm. If the distance is any greater then this extra barrier would be a waste of beryllium and volume.

It is also wise to make the source as dense as possible as this will increase the atom density of the source and all its constituents proportionately. If the system is truly homogenous then equation (5) implies that the similar atom density increase in all materials will have no effect of the neutron output of the system. Rarely are neutron sources completely homogenous and therefore equation (2) can be applied in certain local areas (fractures, contaminants and other irregularities) of the source where the increase in atom density need not be proportional.

While a theoretical density can be achieved, most neutron sources will only achieve about 60% theoretical density. This is because of the economic and safety concerns associated with sintering a dangerous radioactive material.

Ansell and Hall (1971, a) write that beryllium powder will corrode stainless steel at temperatures above 873 K. Therefore beryllium powder should be avoided when preparing neutron sources with an inner cladding of stainless steel. Many early neutron sources used Monel-K for this reason, since many sources were used in fission reactors where temperatures could be very high.

3.2 Source Geometry and Cladding

3.2.1 Cylindrical Sources

As mentioned in section 2.5, in order to maximize the secondary reactions of the (α,n) source and create more neutrons, the source should have as large a volume-to-surface-area ratio as possible. The best geometry to use is a sphere. However spherical geometry is not always easy to produce or use in the lab or industry and therefore a cylinder is the next best choice. The source cylinder should have dimensions such that the diameter is equal to the height.

The large volume-to-surface-area ratio of this cylinder will maximize multiplication neutrons reactions. But it will also increase the number of elastic and inelastic collisions that occur within the source. Therefore the energy spectrum of the source will be different when viewed from the radial sides of the source than the top of bottom of the source. For larger sources Kumar and Nagarajan (1977) state that the flux may vary up to 20%.

It is also necessary to contain the radioactive material so that it does not escape. The method of containment is to doubly encase the source in a strong and chemically resistant metal. Wauchope and Baird (1959) surrounded their experimental sources in two layers of aluminum. The MDS Nordian catalogue (IAEA-TECDOC-1357, 2003) features a Ra-Be source double encased in Monel-K which was silver soldered together. The Numec Pu-Be source (Profio, 1976) has the source surrounded by tantalum then an outer shell of stainless steel. Lorch (1973) had two layers of stainless steel. All of the stainless steel shells were carefully welded shut.

It should be noted that there must be some empty space between the source material and first encasing shell. This allows for the buildup of helium gas from α decay and prevents any warping or damage to the material. Ansell and Hall (1971, I) write that the gap between the source and the cladding should be sufficient to account for all of the helium (and fission gases) that will develop over the (indefinite) life of the source. This protective measure may not be applicable for some source loadings and geometries. Therefore the cladding must be strong enough to withstand the internal pressures that will build up for a wide range of temperatures.

Many cylindrical neutron source designs featured an optional hook or ring. If the application demands such an accessory, the design engineer must be certain that the hook or other attachment does not penetrate the outer encasement, for radiation leakage concerns.

3.2.2 "Switchable" Neutron Source

Ansell and Hall (1971, I) provide a design for an isotropic neutron source, that may be turned "on" and "off". Two half shells of an alpha-emitting foil and beryllium can be brought into contact with each other by means of a rotating spindle as shown in figure 5. More information about this kind of source can be found in their paper.

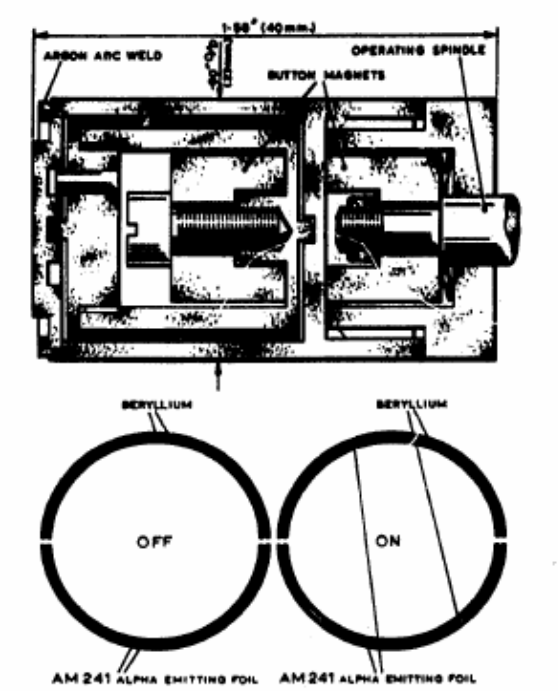

Figure 5: "Switchable"(α,n) source design, source: Ansell and Hall, 1971 a

3.3 Photonuclear Neutron Source Creation

Photonuclear neutron sources are significantly easier to produce than the (α,n) counterparts. According to Cierjacks (1983) these sources are typically spherical. These sources have a center that is made up of the γ emitter (which is in a chemically stable state). For the sources listed in Table 3, these compounds are: NaF, Al, CCl_4, MnO_2, Ga_2O_3, As_2O_3, Y_2O_3, In, Sb, La_2O_3 and Pr_2O_3.

This material is then encased in a single layer of aluminum. However it is not inconceivable that other materials such as stainless steel can also be used. The cladding should be thick enough to ensure that the radioactive material will not escape passively nor in an accident, but not too thick that it would greatly reduce the energy of the γ rays. The cladding is then surrounded by a spherical blanket containing deuterium or beryllium. While it is not included in Cierjacks' description it would not be a bad idea to

also have a thin cladding surrounding the blanket with beryllium. This is to help reduce the number of toxic beryllium particles contaminate the nearby environment.

The half-life of most (γ,n) sources are quite short. Therefore a (γ,n) source is prepared with a stable isotope of the source element. The entire neutron source is bombarded by a strong neutron field. The stable isotopes will capture some of those neutrons to form the unstable γ source. As these atoms decay, they will emit the γ ray that may undergo the (γ,n) reaction.

CHAPTER 4

URANIUM-232

The Brookhaven National Nuclear Database (Tuli, 2005) reports that ^{232}U has a 68.9 ±.4 year half-life and decays into ^{228}Th, where it joins the thorium (4n) decay chain. After ^{228}Th decays, the progeny (^{224}Ra) will quickly decay to a stable ^{208}Pb. The full decay scheme of ^{232}U and its primary emissions can be found in Figure 6. The ^{232}U decay chain requires 10.17 years to come into secular equilibrium. Other characteristics of ^{232}U can be found in Appendix C.

Uranium-232 was first created during the Manhattan project. It underwent more extensive study in the late 1950's and early 1960's when it was a candidate for use in space-based electricity generation systems (Rohrman, 1961). ^{232}U was not selected for use in the space program largely because of the strong gamma emissions of it progeny, particularly ^{208}Tl which emits a 2.615 MeV gamma ray.

Currently there is a small push (Kang and Hipple, 2001) to mix ^{232}U with uranium nuclear weapons in all of the established nuclear weapon nations. They claim that because of the strong gamma field associated with the progeny of short lived ^{232}U, extensive shielding is required to shield against the radiation. This discourages the procurement of uranium nuclear weapons by independent groups because they will have to invest more resources into shielding against the radiation. Ideally, intelligence agencies should be able to detect when a rouge group or nation has ^{232}U tainted material because the radiation is very easy to detect and the thick shielding required to hide the radiation is also quite noticeable as it requires a lot of dense material or a very deep hole or cave. Therefore it follows that the traffic of ^{232}U tainted nuclear weapons is very easy to detect.

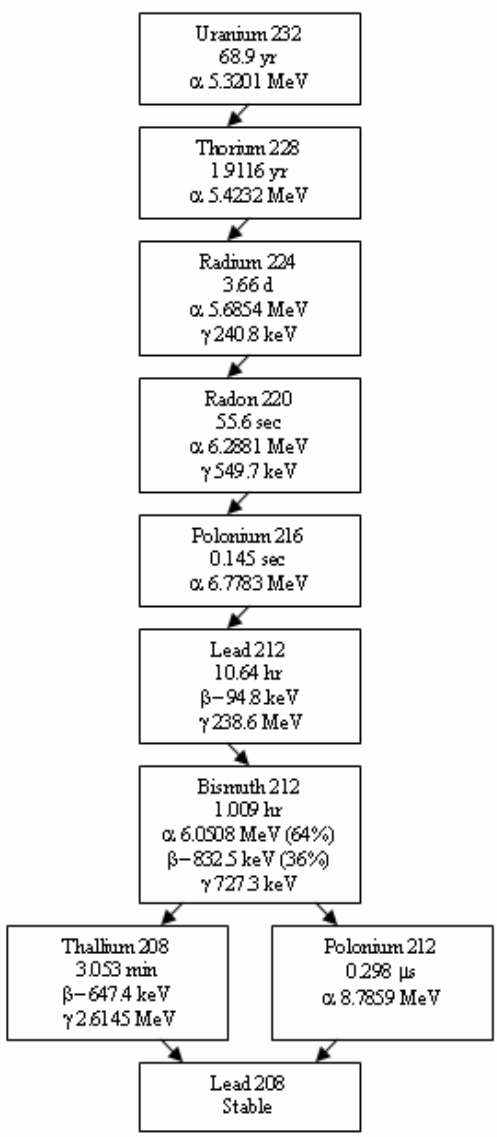

Figure 6: Uranium-232 decay chain with most common decay energies

There are no other engineering or practical applications of ^{232}U at this time. However the ^{232}U decay chain emits six high-energy α emissions in quick succession, making it an ideal candidate for a beryllium neutron source if the strong gamma field can be tolerated and the user is willing to wait 3-5 years for the neutron source to mature to a very high neutron output. Using equation (3), the theoretical peak neutron output was found to be 782.1 n/10^6 α after summing over all six α emissions, following the example

of Geiger and Van der Zwan (1975). The theoretical (α,n) neutron yields of other common neutron sources mentioned in the above paper were recalculated using decay data compiled by Stabin and da Luz (2002) from Brookhaven National Laboratory and are listed on Table 1. It is clear from the table that the ^{232}U decay chain has the highest neutron output.

4.1 Historical Review of ^{232}U and Thorium (4n) Decay Chain (α,n) Sources

There has been no effort to use ^{232}U as a driver for an (α,n) neutron source in the past. Nor has there been any papers written explicitly on the topic. There are very few mentions of ^{232}U or the Thorium (4n) decay chain as an (α,n) neutron producer.

The first mention was made by Arnold at the Second United Nations Atoms for Peace Conference in 1958. He presented a similar paper with ANS Conference (Nuclear Safety) in 1964. Unpublished results of his calculations were printed in Benedict, Pigford and Levi's Nuclear Chemical Engineering (1981). Arnold discussed the potential of ^{232}U/^{228}Th to produce neutrons in an (α,n) reaction, but discussed it only in a waste management context. Arnold showed preliminary results of the neutron output of several isotopes found in nuclear waste when there are contaminant light elements, such as beryllium and fluorine.

Ansell and Hall (1971, II) included ^{232}U in their table of potential (α,n) sources. This table showed that ^{232}U had the greatest potential for producing (α,n) neutrons. They also discussed how limited ^{228}Th (α,n) sources have been produced. ^{228}Th-Be sources were produced by irradiating ^{226}Ra with neutrons until it forms ^{228}Th. ^{228}Th is the first

progeny of ^{232}U, and therefore a ^{228}Th and ^{232}U will have similar neutron yields and spectrums.

^{228}Th-Be (Ansell and Hall, 1971, II) and ^{228}Ra-Be (Ansell and Hall, 1971, I) sources were described as being high intensity (α,n) sources which are best used in research and development reactors where their gamma emissions can be tolerated. Ansell and Hall concluded their paper (1971, II) by predicting that a demand for high intensity isotopic neutron sources would lead to increased development of ^{244}Cm-Be, ^{227}Ac-Be and ^{232}U-Be neutron sources. This thesis is the first work to develop a ^{232}U-Be(α,n) source.

In 1978, Kumar and Nagarajan published the neutron spectrum of a ^{228}Ra-Be source. ^{228}Ra beta decays into ^{228}Ac with beta decays into ^{228}Th. Therefore both a ^{228}Ra-Be and ^{232}U-Be source would utilize most of the same alpha emissions, but a ^{232}U-Be source would yield additional neutrons due to the α emission of ^{232}U.

4.2 Uranium-232 Production

^{232}U was first created in 1943 by Gofman (and Seaborg) (Seaborg, Kathren and Gough, 1994) via the reaction (Newton, 1949):

$$^{232}_{90}Th(d,2n)^{232}_{91}Pa(\beta^-)^{232}_{92}U \tag{27}$$

Today, commercial high purity U-232 is produced in the following reaction (Guglielmetti et al., 2000) :

$$^{232}_{90}Th(p,n)^{232}_{91}Pa(\beta^-)^{232}_{92}U \tag{28}$$

Other modern techniques for creating ^{232}U use the following reactions (Mann and Schenter, 1977):

$$^{230}_{90}Th(n,\gamma)^{231}_{90}Th(\beta^-)^{231}_{91}Pa(n,\gamma)^{232}_{91}Pa(\beta^-)^{232}_{92}U \tag{29}$$

$$^{233}_{92}U(n_{fast},2n)^{232}_{92}U \tag{30}$$

The threshold energy for the reaction in equation (30) is about 6 MeV.

The reaction in equation (29) requires that ^{230}Th (ionium, Io) be isotopically separated from the much more prevalent ^{232}Th. Ionium is a decay product in the ^{238}U (4n+2) decay chain and is very prevalent in nature. However natural uranium is almost always found with natural thorium (^{232}Th). According to Rohrman (1961) ionium makes up anywhere from 2%-40% of thorium, depending largely on where it is found. Ionium is usually less than 5% of thorium. This implies that isotopic separation, which is always difficult and expensive, will be needed to produce ^{232}U without significant isotopic impurities from the neutron captures of ^{232}Th.

Kim et al. (1972) explain that isotopic separation is not necessary. Natural thorium is only found in two long lived forms, ^{232}Th and ^{230}Th (Io). Consider the following reactions:

$$^{230}_{90}Th(n,\gamma)^{231}_{90}Th(\beta^-,t_{1/2}=1.063d)^{231}_{91}Pa(\beta^-,t_{1/2}=32{,}000yr) \tag{31}$$

$$^{232}_{90}Th(n,\gamma)^{233}_{90}Th(\beta^-,t_{1/2}=22.3m)^{231}_{91}Pa(\beta^-,t_{1/2}=27d) \tag{32}$$

After both ^{230}Th and ^{232}Th absorb a neutron, they will beta decay into ^{231}Pa and ^{233}Pa, with half-lives of 32,000 yr and 27 d respectively. Therefore after a cool down period of 10 months all of the remaining protactinium will be ^{231}Pa. The protactinium can then be chemically separated to again undergo neutron activation and decay into ^{232}U:

$$^{231}_{91}Pa(n,\gamma)^{232}_{91}Pa(\beta^-)^{232}_{92}U \tag{33}$$

Using the above procedure is more efficient than isotopically enriching the ionium in the thorium.

The production scheme of ^{232}U is shown on Figure 7 (Kim et al.). That scheme mentions a cool down period after each neutron irradiation. Because of the 27 day half-life of ^{233}Pa, the first cool down should last 270 days. After 10 half lives a radioisotope is considered to have completely decayed. Since the half-life of ^{231}Pa is 32,800 years, ^{231}Pa will be the only remaining protactinium isotope after 270 days. The second cool down period is to allow the ^{232}Pa created toward the end of the irradiation to decay to ^{232}U.

The scheme also recommends the time that each material should be irradiated with neutrons. This is found by deriving when the rate of isotope to be created will equal 80% of destruction (by decay, neutron activation, fission, etc) of the same isotope.

The production scheme will require two kinds of chemical separations as shown on Figures 8 and 9 (Kim et al., 1972). Each flowchart describes the procedure for the target material after the first and second kind of irradiation cycle respectively.

According to Kim et al. (1972), the first chemical separation is largely to remove ^{231}Pa. However uranium and fission products will also have to be removed. Fission products will be made from the fission of uranium, as well as fast fissions of thorium. The uranium extracted during this cycle will contain about 15% ^{232}U. This is unsatisfactory for applications requiring pure ^{232}U, but it can be used later on for other applications. Lastly thorium will also have to be extracted so that it may be subjected to several more cycles in the reactor.

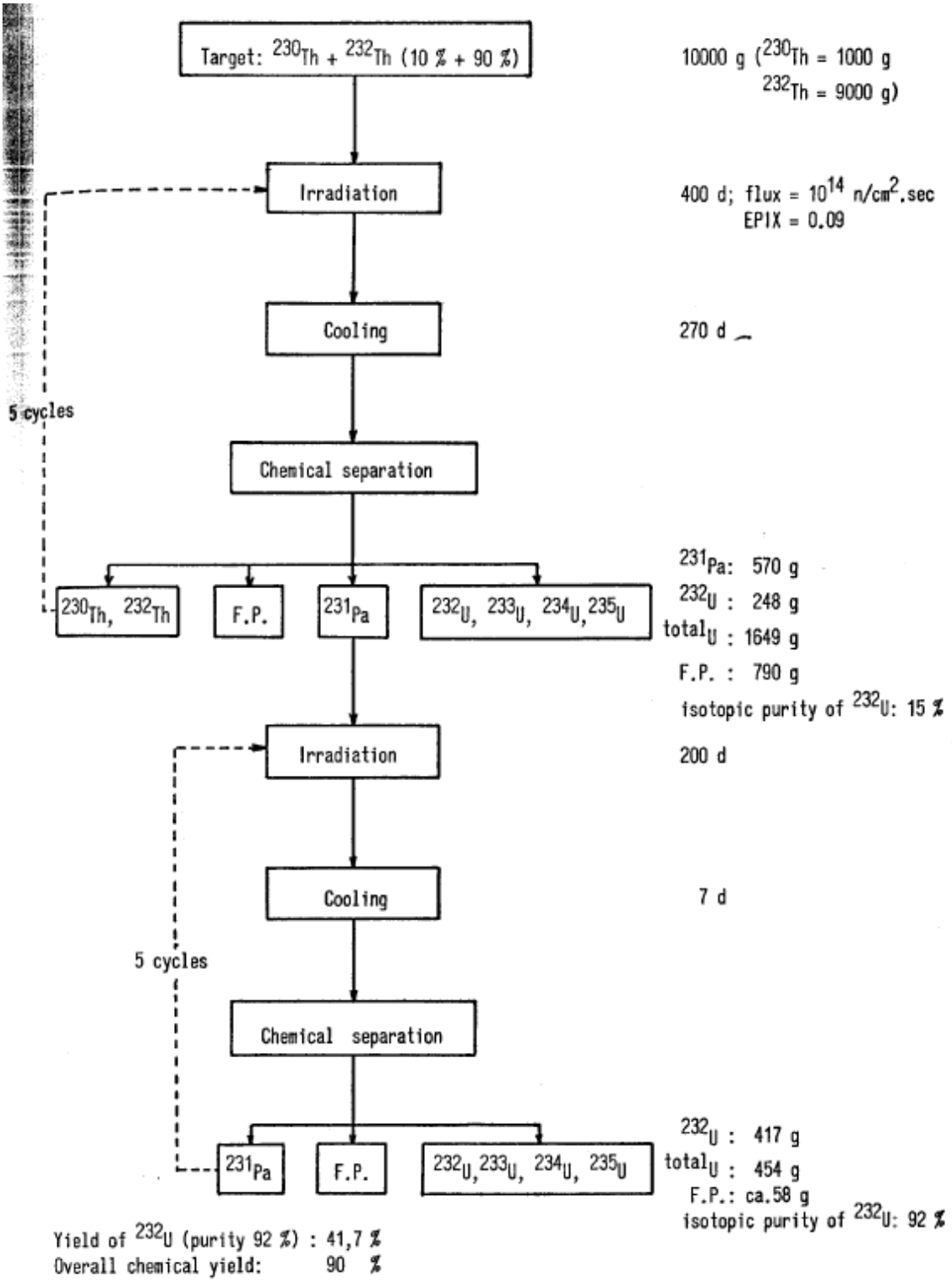

Figure 7: ^{232}U production flowchart, Source: J. I. Kim et al. 1972

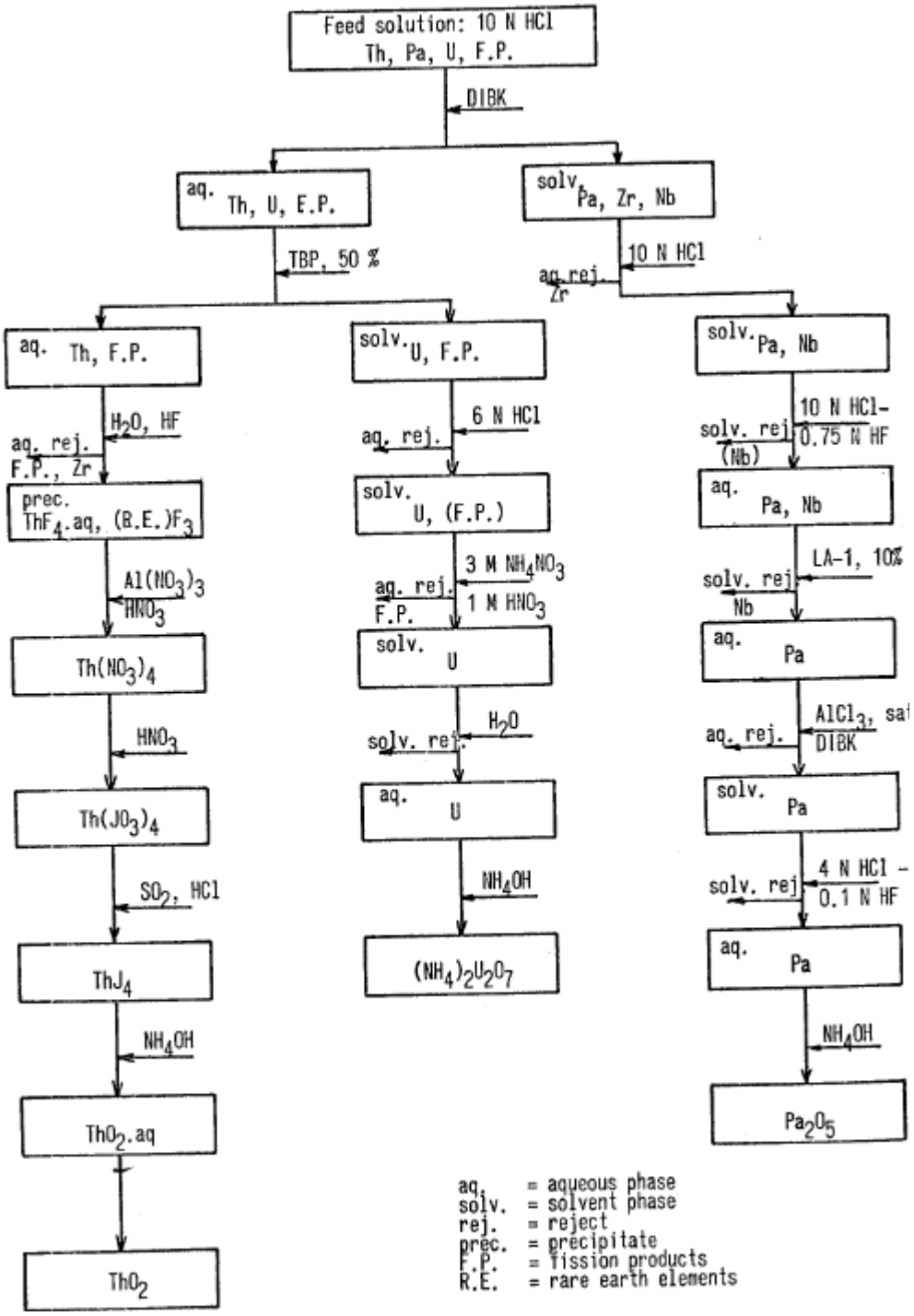

Figure 8: Chemical removal of ^{231}Pa and ^{232}U flowchart , source: Kim et al. 1972

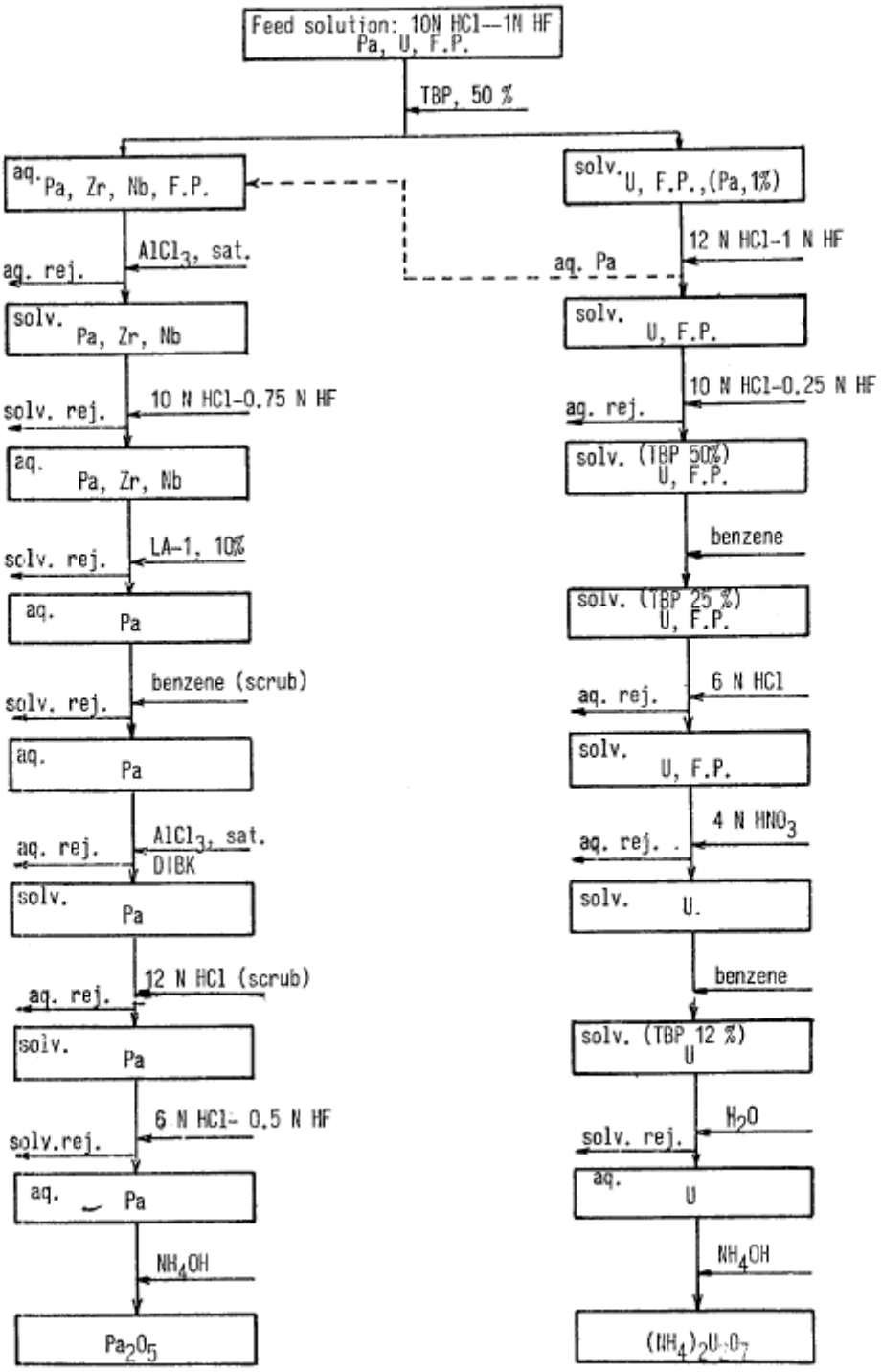

Figure 9: Chemical recovery of ^{232}U flowchart ,source: Kim et al. 1972

Kim et al. (1972) explain that the second chemical separation is to remove ^{232}U, now enriched to a high and very satisfactory 92%. Fission products and higher actinides will also have to be removed leaving only ^{231}Pa. This material will also be placed back in the neutron flux.

The uranium will exit the process in the form of $(NH_4)_2U_2O_7$. Considering the quantities of uranium to be extracted starting with 10 kg of thorium, this mixture and amount is safe from a criticality viewpoint. ^{232}U has a critical radius of about 3.31 cm under normal pure-metal density.

4.3 Dirty Uranium

The reactions described in reactions (29) and (30) are the most important reactions when creating ^{232}U, although they are unwanted reactions during the breeding of ^{233}U in the thorium fuel cycle. According to the IAEA (2005) the ^{232}U yields in bred ^{233}U can vary from 5-3000 ppm depending on the method by which ^{233}U is bred. ^{233}U with a high ^{232}U contamination is said to be “dirty uranium”. The presence, short half-life and the strong gamma emissions of ^{232}U and its progeny make the handling of ^{233}U dangerous. Adequate shielding must be in place at any facility that handles bred ^{233}U. Therefore there has been a great deal of effort to reduce the production of ^{232}U when breeding ^{233}U.

The creation of ^{232}U when breeding ^{233}U is inevitable. But there exists methods to reduce the ^{232}U yield. Most often, this requires recycling thorium from previous efforts to breed ^{233}U. The reason why recycling thorium will reduce the ^{232}U yield in subsequent breeding attempts is because most of the ionium will have been converted in the first

cycle of breeding. Therefore reaction in equation (30) will become the dominant producer of ^{232}U after most or all of the ionium has been consumed.

In addition to ^{232}U production and related safety concerns, a report from Hanford Atomic Products Operation (Hanford Laboratories) (HW-78100, 1963) found that is was more economical to recycle thorium when breeding ^{233}U than it was to create fresh thorium.

Mann and Schenter (1977) found that relative impurities of ^{232}U in bred ^{233}U increase with neutron irradiation time. The impurity is also dependent on the neutron energy, geometry and target chemical and isotopic composition. More complete results of their investigation and that of other studies can be found in their paper and the IAEA report IAEA-TECDOC-1450. Generally, ^{232}U production is reduced with $E_n<6$ MeV (reaction (30)) and when pure ^{232}Th is used. Therefore a ^{232}Th breeding blanket on the outskirts of a reactor is generally a superior method for breeding ^{233}U with as little ^{232}U impurity as possible, because the flux will be less energetic on the edges of the reactor than when the thorium is intimately mixed with the fuel.

CHAPTER 5

$^{232}UBe_{13}$ NEUTRON SOURCE

5.1 Description of Source

A neutron source comprised of $^{232}UBe_{13}$ was designed and its performance simulated in the present work. Very little ^{232}U exists in a pure form and what does exists is too expensive to produce an appreciably sized neutron source. However significant amounts of ^{232}U are inadvertently created during the breeding of ^{233}U can be used to construct a source. Therefore a source consisting dirty of $^{233}UBe_{13}$ with a contamination of 300 ppm ^{232}U, was modeled. Hereafter this source will be referred to as $^{dirty}UBe_{13}$.

The source was modeled as a cylinder with its diameter equal to its height. This was done to maximize the multiplication reactions that occur within the cylindrical source, as these dimensions have the greatest volume-to-surface area ratio. Ideally a spherical geometry would be selected to maximize the multiplication reactions, but a cylinder was chosen for ease of potential manufacturing (Wauchope and Baird, 1959). The model source has a diameter and height of 1.1018 cm

The source model contains 1.043 GBq (0.0282 Ci, 3.004 g) of ^{233}U, 0.747 GBq (0.0202 Ci, 0.9019 mg) of ^{232}U and 1.511 g of beryllium at construction. Anderson (1967) indicates that the $PuBe_{13}$ compound may not form successfully for sources containing under a gram of plutonium. It was assumed that this condition also applies to all actinides when forming an XBe_{13} compound,, so the source was designed to have over one gram of uranium. The density of the source was set to the theoretical density of 4.2975 g/cm^3.

The source dimensions were selected so that this could be made into a demonstration source. It was small enough that very little material was needed, but large enough that the UBe_{13} could be sintered properly.

There exists a 0.491 cm helium collection gap on all sides of the UBe_{13} source to contain the helium particle that will accumulate due to α decay. The source would be doubly encapsulated in stainless steel (modeled as iron) to prevent release of material. The diagram of the sample source can be found in Fig. 9.

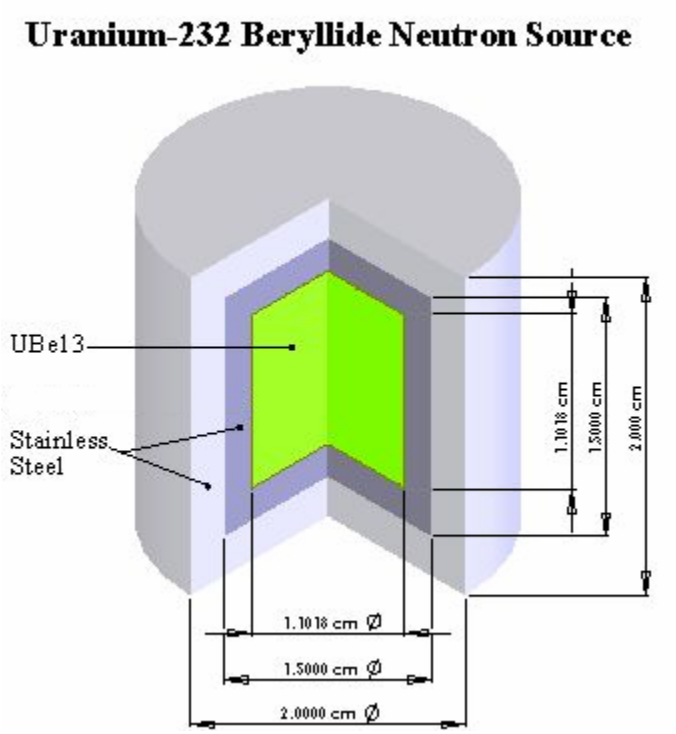

Figure 10: Design of example $^{dirty}UBe_{13}$ neutron source

The source design was evaluated to determine if the steel encapsulation would safely contain the accumulation of helium gas over the lifetime of the neutron source. It was predicted that the source would satisfy these requirements. The internal helium pressure will be 749 kPa (7.4 atm) over the lifetime of the ^{232}U (~700 yrs). Assuming a

conservative yield strength of 250 MPa, the containment is expected to fail after about 40,000 years.

Heat generation from radioactive decay and reactions within the source is expected to be negligible. Therefore the source can passively maintain a safe temperature with its environment.

5.2 Multiplication Blanket

Simulations with a 6 cm diameter sphere surrounding the neutron source and cladding for the purpose of moderating the neutrons and creating additional neutrons by multiplication reactions were performed. The neutron spectrum was assumed to have come from a source aged 10.17 years or more. Several materials were selected for the sphere: Be, $Be_{12}W$, BeHD, W and Pb. The beryllium compounds were selected because of the large (n,2n) and (γ,n) probability of beryllium. Deuterium can undergo a (γ,n) photonuclear reaction (Profio, 1979) at the source gamma-ray energies, which could possibly reduce the number of hard gamma rays leaving the source. Lead and tungsten were selected because they have reasonable (n,2n) probabilities at fast neutron energies and because they can be used to shield against the strong gamma emissions.

5.3 Dirty Uranium Fuel

A sample source of dirty uranium fuel UO_2 was also modeled. It was assumed that the fuel was 10.17 years or older so that U-232 was in equilibrium with its daughters.

5.4 Computer Codes

Two computer codes were used to determine the total neutron yield and spectrum of the sample neutron source. The first code SOURCES-4c was used to find the (α,n) reactions. MCNP5 was used to determine the secondary reactions.

SOURCES-4c and its predecessor versions, is a statistical code that find the (α,n), (α,α'n), spontaneous fission and delayed neutron yields for a neutron source system. SOURCES-4c uses the equations outlined in section 2.2.1 and 2.3 of this thesis to solve for the (α,n) neutron yield and spectrum of a homogenous system. The code is also capable of performing thick and thin target problem and layered systems, (Wilson et. al, 1999).

SOURCES-4c is presently considered the dominant (α,n) reaction code. ORIGEN (SCALE5) is also capable of performing (α,n) reactions, because the SOURCES code has been integrated into the ORGIEN code. The source code for SOURCES-4c is available and can be easily modified and recompiled. The source file for ORIGEN is not available. Since several data cards and lines of code had to changed to evaluate this problem, SOURCES-4c was selected.

MCNP5 is a deterministic transport code that is used to determine many neutron, electron and gamma reactions. It is very reliable and its solutions are universally considered correct within the calculated error. MCNP5 was selected to evaluate this problem because of its ease and widespread usage.

5.5 Evaluation

5.5.1 (α,n) Yields

The neutron yield is source was modeled using the SOURCES-4c computer code for source ages of 0, 1, 3, 5, 7.5, and 10.17 years. The composition of the source varied at these times due to the buildup of the ^{232}U and ^{233}U decay chains. All of the ^{232}U and ^{233}U progeny were included as they built in with time. The SOURCES-4c code returns the neutron output (in n/cm^3), assuming an infinite homogenous medium. It computed the neutron energy distribution from 0 to 13.5 MeV in 54 evenly spaced energy groups (width 250 keV).

The current release of SOURCES-4c was incapable of modeling this source, as it will cease functioning for alpha particle energies over 7.9 MeV since exact cross section data are only available up to that energy. The code and its data library were modified to allow energies up to 10 MeV by assuming a constant value of the cross section beyond the last available data point, as previously performed by Shores et al. (2003). This modification can be found in Appendix B. The extension of the data is required as ^{232}U decay product ^{212}Po emits an 8.7844 MeV alpha particle. This is a middling assumption since the total cross section for the Be(α,n) reaction measured by Gibbons and Macklin (1964) remains nearly constant from 7.9-8.5 MeV, but sharply increases from 8.5-8.8 MeV. Therefore, the neutron yields may be underestimated for ^{212}Po. The data library was also modified to include the latest half-life data from Brookhaven National Lab (Tuli, 2005) for the isotopes found in the source. These modifications can be found in Appendix B.

All input files used with SOURCES-4c can be found in Appendix A.1.

5.5.2 Multiplication Neutrons and Dose Calculations

The neutron energy spectrum and yield found by the SOURCE-4c code were used in an MCNP5 model of the source. The leakage neutron spectrum from the source was computed with the MCNP5 code. This calculated spectrum included any multiplication due to secondary nuclear reactions (n,2n), (n,f) and (n, γ) as well as energy-degrading interactions in the source for all SOURCES-4c evaluation at the various source ages. A separate MCNP5 run was used to calculate the leakage spectrum due to photonuclear neutron yield. The 2.615 MeV gamma rays from the decay of ^{208}Tl and all other gamma emissions of the radionuclides were included in these calculations. The addition of the weaker gamma rays into the input decks served to find the gamma dose rate (calculated on a separate spreadsheet). This step was also done for all SOURCES-4c evaluated ages of the source.

All MNCP input decks for multiplication and photonuclear reactions can be found in Appendix A.2.

A separate run to calculate the photonuclear yield of the 4.438 MeV gamma ray emitted from the excited ^{12}C nucleus after reaction (1) (Profio, 1979). But because of the relative scarcity of the 4.438 MeV gamma rays and the rare improbable nature of the photonuclear reaction, the 4.438 MeV (γ,n) reactions were omitted from the results and future MCNP simulations.

5.5.3 Multiplication Blanket

An MCNP model was used also used to evaluate the multiplication and photonuclear reactions along with the spectra of the neutrons and gamma particles exiting the multiplication blanket describe in section 5.2. MCNP models were made for a 10.17 year or older neutron source, surrounded by the blankets described in that section. These MCNP input decks can be found in Appendix A.2.

5.5.4 Dirty Uranium Fuel

A SOURCE-4c input file was also set up for UO_2 fuel. This input deck was made to evaluate the (α,n) yield of dirty uranium fuel. No calculations were done for multiplication neutrons since the geometry of fuel pellets vary too greatly. So only the (α,n) production traits of spent dirty uranium fuel was evaluated. Further evaluation of spent dirty uranium fuel will have to be evaluated on a case-by-case basis. The SOURCES-4c input file for the dirty uranium fuel can be found in Appendix A.1.

CHAPTER 6

RESULTS

6.1 Neutron Source

6.1.1 Source Neutron Output

The source magnitude was found to vary with time starting at $6.7*10^4$ neutrons/sec and peaking 10.17 years later at about $3.5*10^5$ neutrons/sec. The specific (α,n) neutron yields from ^{232}U and its progeny were found to be 43 and 560 $n/10^6$ α at birth and after 10.17 years, respectively. After 10.17 years the neutron yield decreases in direct proportion to the ^{232}U activity. The graph of neutron emission rate verses time can be found in Fig. 11. At the peak neutron emission, 90.08% of all the (α,n) reactions were from the ^{232}U decay chain's alpha emissions; the other 9.92% of the (α,n) reactions being from the ^{233}U decay chains. The ^{232}U decay chain (α,n) reactions account for 86.42% of the total neutron output. A break down of how each isotope contributed to the production of neutrons at 10.17 years can be found in Table 5 along with the contributions of the multiplication and photonuclear reactions. The probability of a photonuclear reaction is greater for the 4.438 MeV gamma from the $^{12}C^*$ atom than from ^{208}Tl, but the limited number of $^{12}C^*$ gamma rays per unit time results in a smaller number of neutrons produced from the higher energy gamma rays.

Including only the (α,n) reactions from ^{232}U and its progeny, the source had a peak output of 561.1 $n/10^6$ α particles or 74.3% of the value predicted by equation (14). The shortcoming of reaching the predicted value is due to the low beryllium-to-alpha-emitter atomic ratio; specifically because the alpha particles will have a smaller chance of interacting with a beryllium nucleus if there are more non-beryllium nuclei in its path.

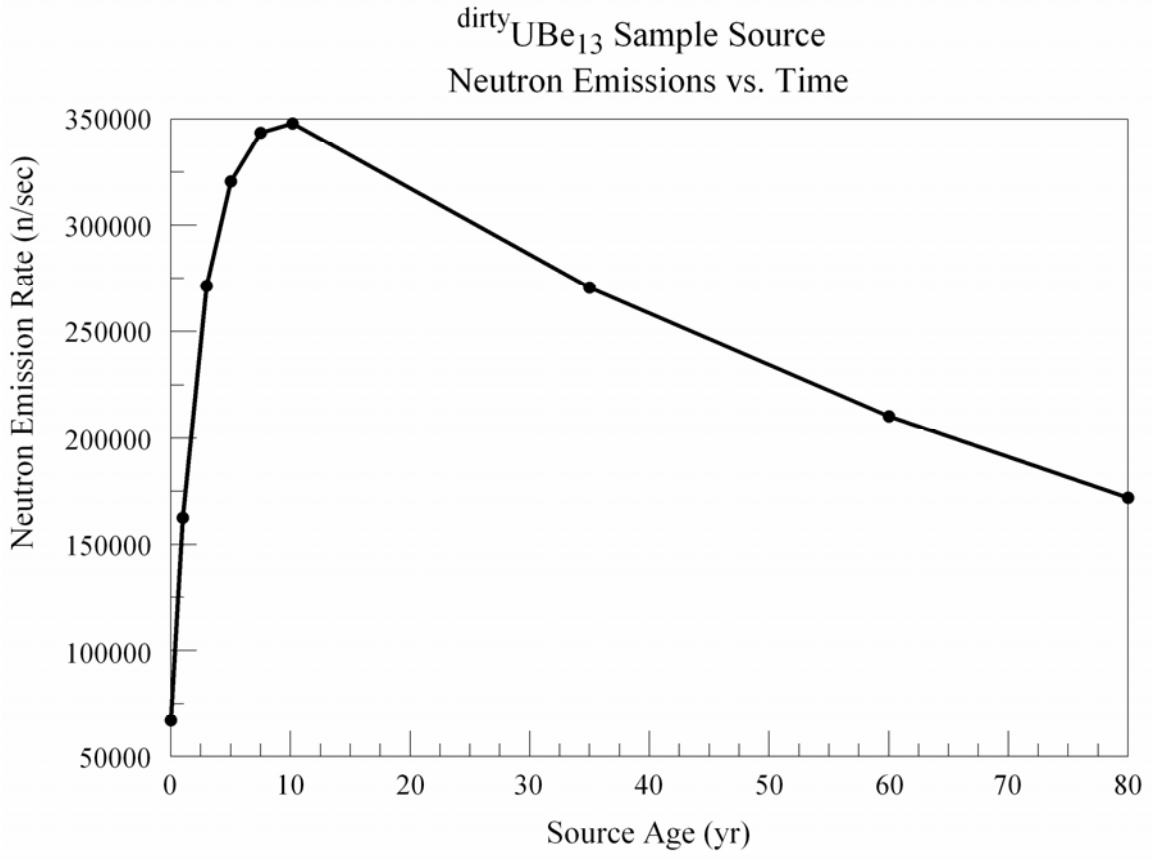

Figure 11: Neutron emissions vs. time for $^{dirty}UBe_{13}$ example source (calculated)

Ergo the alpha particle is more likely to be slowed down by the Columbic force of the heavier atoms before interacting with a beryllium nucleus. Equation (23) can be used to predict the neutron output given the chemistry of the system. These results were 116% of the value predicted by equation (23). This is within the error of the SOURCES-4c code of ±18% (See below).

Table 5: Neutron Yields for sample $^{dirty}UBe_{13}$ Source (calculated)			
	Yield (n/sec)	Fraction of (α,n)	Fraction of Total
(α,n) Reactions			
Isotope			
^{212}Bi	16374	4.91%	4.66%
^{213}Bi	1	0.00%	0.00%
^{212}Po	77327	23.17%	22.03%
^{213}Po	138	0.04%	0.04%
^{216}Po	59544	17.84%	16.96%
^{217}At	105	0.03%	0.03%
^{220}Rn	50699	15.19%	14.44%
^{221}Fr	50	0.02%	0.01%
^{224}Ra	36881	11.05%	10.51%
^{225}Ac	6	0.00%	0.00%
^{228}Th	30891	9.25%	8.80%
^{229}Th	33	0.01%	0.01%
^{232}U	28967	8.68%	8.25%
^{233}U	32775	9.82%	9.34%
^{232}U Chain	300683	90.08%	85.66%
^{233}U Chain	33108	9.92%	9.43%
Total (α,n)	333791		95.09%
Secondary Reactions			
(n,2n)	1516		0.43%
(n,f)	11738		3.34%
Photonuclear (γ,n)	3968		1.13%
Total	351010		100.00%

The source spectrum, normalized to a neutron emission of unity, is plotted in Fig. 12 for varying times. At the loading of the neutron source, only ^{232}U and ^{233}U atoms are emitting alpha particles and, therefore, the 0 year spectrum is largely determined by the neutrons from ^{232}U and ^{233}U (α,n) reactions. As time progresses the contributions of other progeny's alpha emissions build in and have an impact of the neutron energy spectrum.

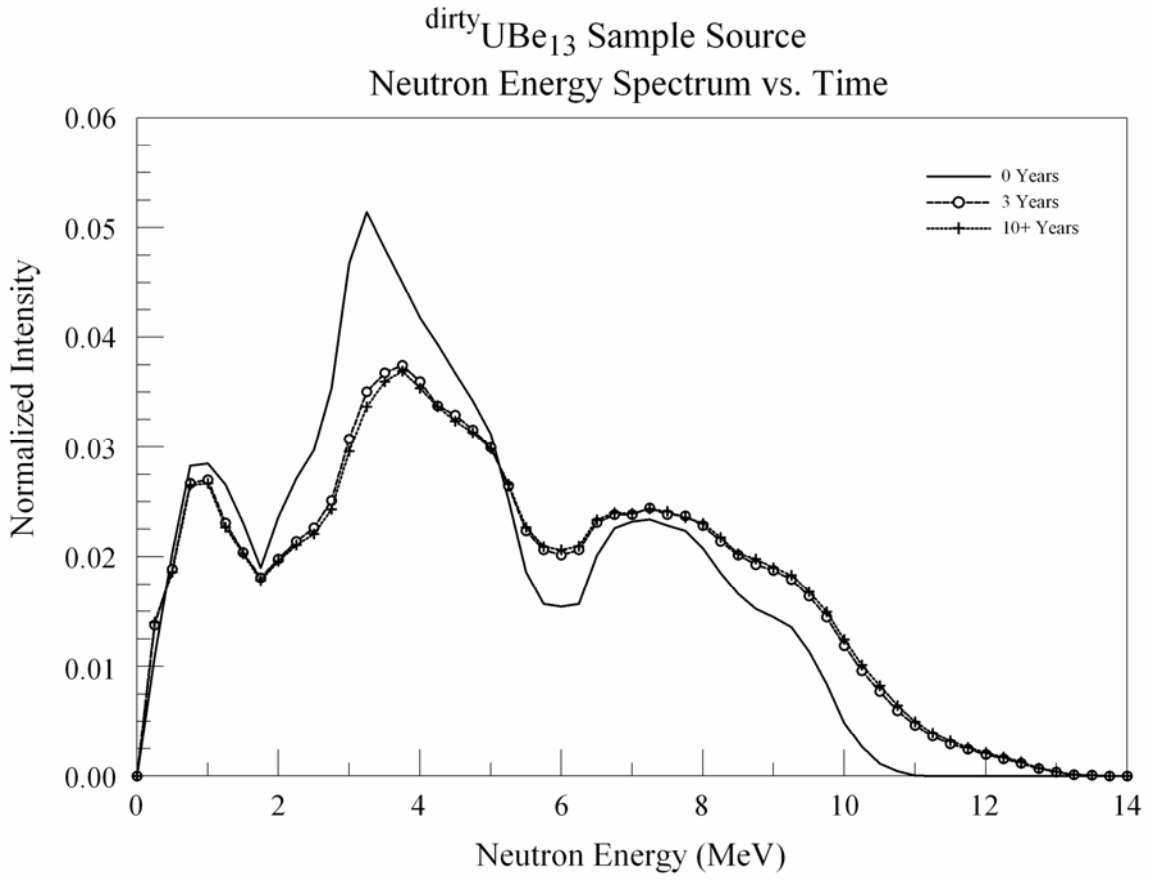

Figure 12: Neutron energy spectrum for $^{dirty}UBe_{13}$ example source at varying times (calculated)

Peaks were predicted in the spectrum at 1.5, 3.75, 4.25, 6.75, 9.25 and a broad peak at 7.5 MeV. By comparison Ansell and Hall measured peaks in a ^{228}Th (α,n) source at 3.5, 5.5, 6.25, 8.5 and 9.25 MeV. Figure 13 shows the spectrum of the two neutron sources at peak neutron production. The graph shows a rough correlation between the two spectra. The discrepancies are likely due to SOURCE-4c assuming the neutron emissions to be isotropic and it's 18% overall error.

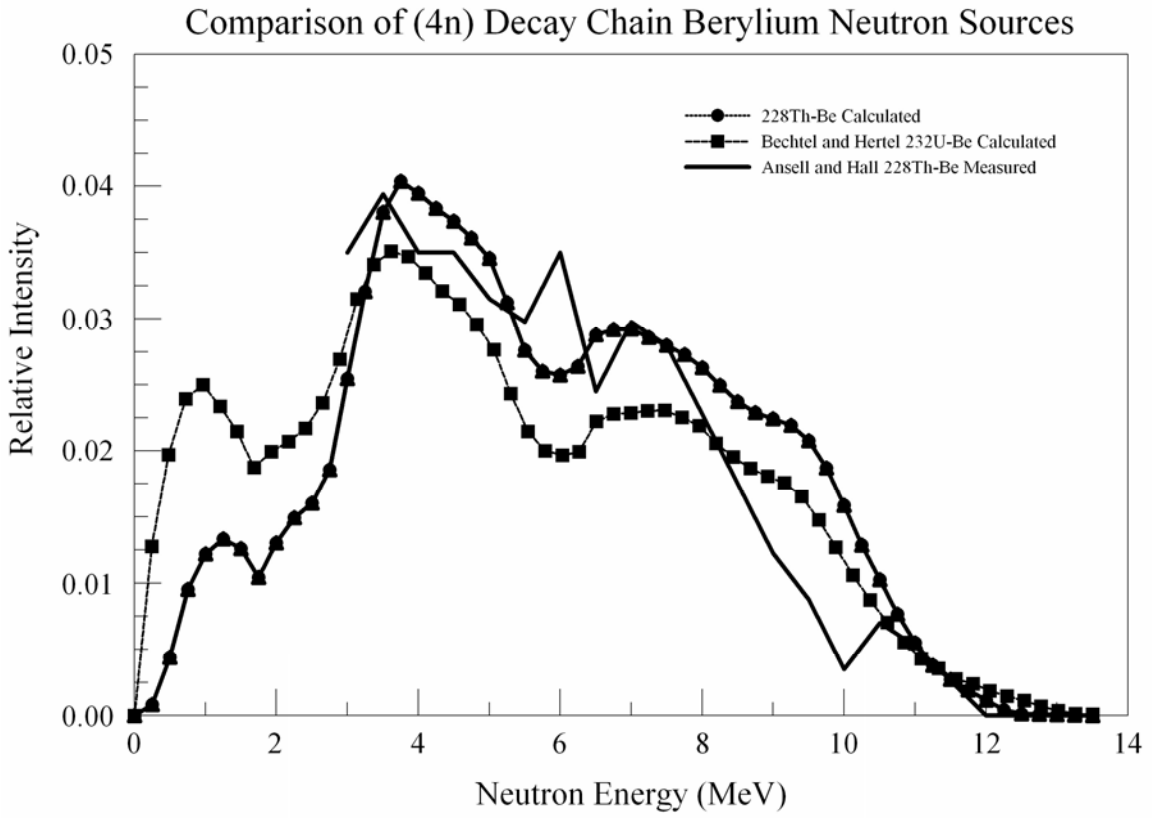

Figure 13: Comparison of calculated $^{dirty}UBe_{13}$, $^{228}ThBe13$ and measured ^{228}Th-Be source spectra

Compared with measured results of Pu-Be neutron sources, Charlton et al. (1997) found that the spectrum derived by SOURCES-3A was correct within 18% for all energies. This benchmark was used for later versions of the SOURCES code. This rather large error, occurs mostly at lower energies where the spectrum is underestimated by the code. It was postulated in that paper that the large error was the result of the experimenters (who measured the flux spectrum of the neutron source) neglecting to report the contaminants in their sample. If these contaminants were neglected in the SOURCES input file, then it follows that the contribution of the contaminants to the neutron spectrum will also be omitted. Shores et al. (2003) show better agreement by modifying the fraction of breakup reactions (reaction in equation (24). However a final

version of the modified code was never released and the authors made no attempt to make such modifications. The final error from the SOURCES-4c, MCNP5 and spreadsheet calculations is about 20% for all energies and yields.

6.1.2 Source Dose Rate

Much like the neutron emission, the ambient dose equivalent rate from the source will vary with time and follows the total activity of the source. The peak total ambient dose rate (ICRP, 1996; ICRU, 1998) was calculated to be 0.105 mSv/hr at 1 meter; the neutron contribution accounted for 0.004 mSv/hr. The graph of ambient dose rate at one meter from the source as a function of time can be found in Fig. 14. Most (96.2%) of the dose rate comes from the intense gamma field surrounding the source. Because of the high dose rate, a ^{232}U driven neutron source should only be used where there is adequate shielding.

6.2 Multiplication Blanket

As expected, it was found that the overwhelming factor influencing neutron multiplication was the (n,2n) reaction, accounting for at least 94.5% of all multiplication neutrons produced in the various blanket materials. The photonuclear reactions were most prevalent in the beryllium materials. The multiplication factors, fraction (γ,n) neutrons, and total relative dose can be found in Table 6 for these materials. The normalized neutron energy spectrum can be found in Figure 15.

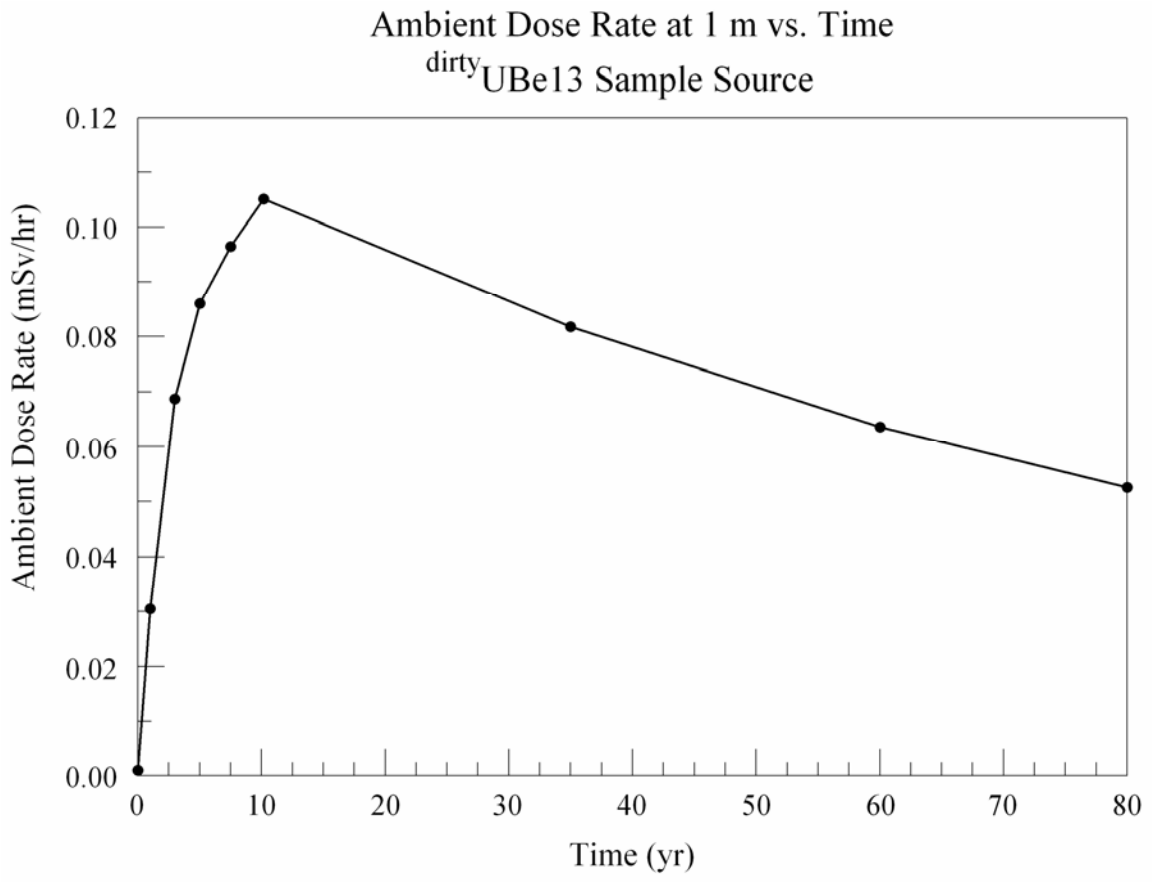

Figure 14: Ambient dose rate at 1 m for $^{dirty}UBe_{13}$ example source (calculated)

Table 6: Neutron Multiplication for 6 cm diameter spherical blanket (calculated)			
Material	Multiplication Factor	Fraction Yield (γ,n)	Relative Total Dose
Bare	1.0000	1.1%	100%
Be	1.0940	5.5%	93%
BeHD	1.0303	4.5%	76%
$Be_{12}W$	1.0657	3.7%	98%
W	1.0261	1.1%	23%
Pb	1.0109	1.1%	37%

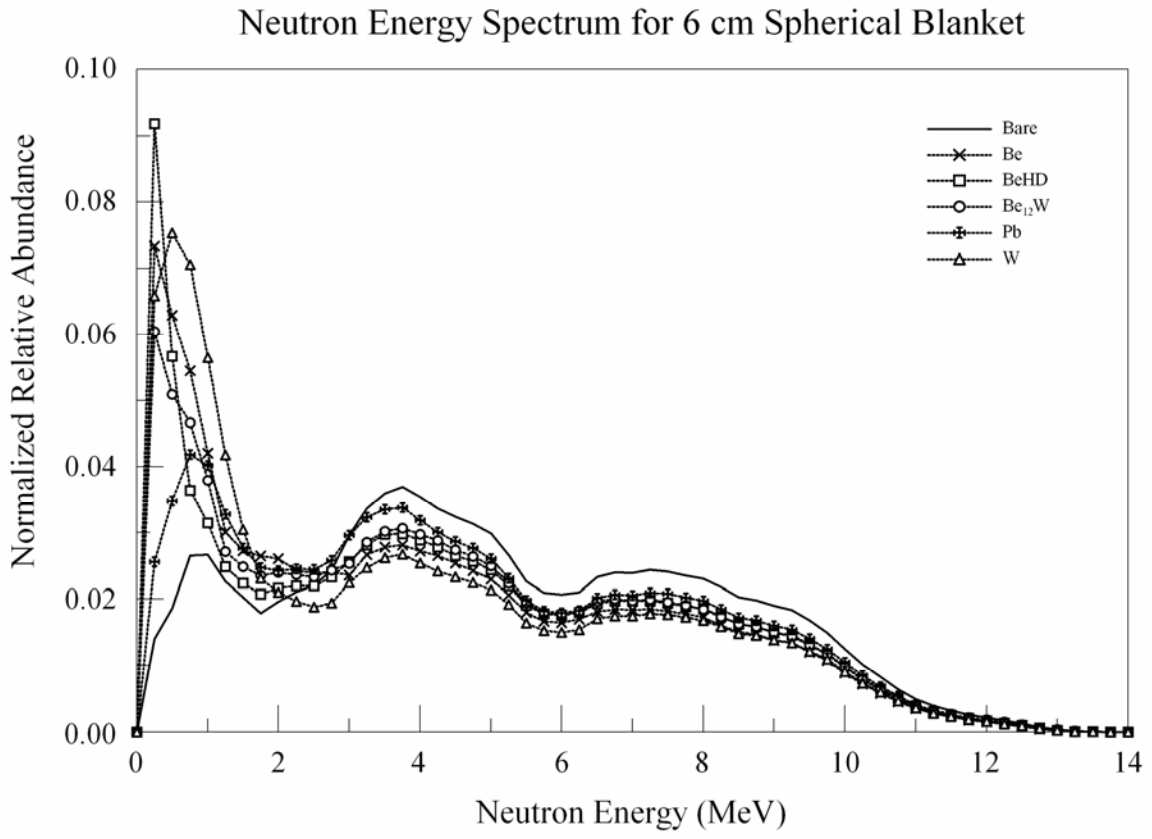

Figure 15: Neutron energy spectrum for $^{dirty}UBe_{13}$ example source surrounded in a 6 cm spherical blanket (calculated)

6.3 Dirty Uranium Fuel

The homogenous (α,n) production rate of 10.17 year or older UO_2 was found to be 725 n/sec*cc.

CHAPTER VII

DISCUSSION

A $^{232}UBe_{13}$ driven neutron source would make an excellent neutron source because of its superior neutron production capability. It will produce more neutrons than any other (α,n) or (γ,n) neutron source per unit activity and per unit gamma dose respectively. Its theoretical maximum from equation (14), 755.1 n/10^6 α is over a factor of 10 higher than many beryllium (α,n) neutron sources in use today (~57-100 n/10^6 α), and its simulated output (561.1 n/10^6 α) is still greater than the theoretical maximum for every other beryllium (α,n) neutron source, save ^{227}Ac-Be (736.8 n/10^6 α) sources. However if the ^{227}Ac-Be source were to have the same 13:1 beryllium-to-alpha-emitter ratio as the $^{232}UBe_{13}$ source, then the $^{232}UBe_{13}$ source would have a superior output.

The density was assumed to be theoretical. In practice though, densities are typically around 60% of theoretical. It was found that the (α,n) neutron yield decreased by 17.3% (464 n/10^6 α). This is still a very respectable output that exceeds most other (α,n) sources.

7.1 Applications in the Thorium Fuel Cycle

A ^{232}U driven neutron source will be very economical if and when the thorium fuel cycle is utilized, since only a few grams of bred ^{233}U will be required to produce a strong neutron source – especially when ^{233}U will be produced by the ton. A ^{232}U driven neutron source could be best used in a thorium fuel cycle reactor. This is because the reactor and associated facilities will already be adequately shielded against the hard gamma emissions. Potentially fuel rods of $^{dirty}UBe_2$ could be made to add to the fuel rods

in place in a thorium cycle reactor. These rods could be used as control rods in such a reactor.

7.2 Generalizing Results for Dirty Uranium

Since the concentrations of ^{232}U will vary in dirty uranium, it is useful to have an approximation for finding the peak (α,n) neutron yield per unit volume for varying concentrations of ^{232}U. Assuming a source was constructed with pure sample of a given compound (e.g. UBe_{13} or UO_2), the peak (α,n) fluence is:

$$Y_{peak} \approx (\chi\upsilon)_{^{232}U} + (\chi\upsilon)_{^{233}U} \tag{34}$$

Where Y is the peak emission rate in n/cm^3-sec, χ is the concentration of each respective isotope in ppm and ν is a constant that can be found in table 7 given in terms of n/cm^3-ppm-sec.

Table 7 lists the constants for unit volume neutron production for UBe_{13} and UO_2 at theoretical density, since these are the most likely compounds for producing neutrons with dirty uranium. Note that these constants assume theoretical density and have an error of ±18%. Equation (34) only estimates the (α,n) yields. Multiplication reactions cannot be estimated easily per unit volume, and are not included in that equation.

Table 7: ν values for Various Compounds with Dirty Uranium (n/ppm*sec*cc) (calculated)

Compound	$\nu_{U\text{-}232}$	$\nu_{U\text{-}232}$
UBe_{13}	954.09	.0315
UO_2	2.163	$6.56*10^{-5}$

Equation (34) can be easily modified to account for burn-up factors, *B*, in the case of spent fuel:

$$Y_{peak} \approx \left(B\chi\upsilon\right)_{^{232}U} + \left(B\chi\upsilon\right)_{^{233}U} \tag{35}$$

Similarly the specific volume neutron production in spent fuel waste products can be estimated over time by applying the decay equation after the ^{232}U is in equilibrium with its progeny:

$$Y(t) \approx \left(B\chi\upsilon e^{-\lambda t}\right)_{^{232}U} + \left(B\chi\upsilon\right)_{^{233}U} \tag{36}$$

Where λ is the decay constant for ^{232}U and t is the time elapsed after ^{232}U-progeny equilibrium is achieved.

CHAPTER VIII

CONCLUSIONS

The physics and material science of the (α,n) reaction and neutron source were researched and discussed in this thesis. A new emitter was chosen, uranium-232, for what promises to be an inexpensive, easy-to-manufacture new choice of isotope for use in (α,n) neutron sources.

Theoretical calculations suggest that a $^{232}UBe_{13}$ driven neutron source should have a greater neutron output per alpha particle than other beryllium neutron sources. The calculated neutron yield was found to be greater than the theoretical yield of any other (α,n) neutron source except ^{227}Ac. But with similar geometries, compositions and activities, a $^{232}UBe_{13}$ source should be superior to other (α,n) neutron sources. The disadvantages fo the neutron source are that it takes about 5 years to achieve a very high neutron output and that the gamma field surrounding the source is very high. However if the gamma emissions can be tolerated and the source can be allowed to age, then this source should outperform any other isotopic neutron source aside from Cf-252.

APPENDIX

APPENDIX A

SOURCES-4c and MCNP5 INPUT CARDS

APPENDIX A.1

SOURCES-4c Input Cards

$^{dirty}UBe_{13}$ at 0 years Input Card (tape1)

```
233UBe13 300ppm 232 @ 0.0 yr
1  2  1
12  0
  004 .928571429
  081 0.00e-00
  082 0.00e-00
  083 0.00e-00
  084 0.00e-00
  085 0.00e-00
  086 0.00e-00
  087 0.00e-00
  088 0.00e-00
  089 0.00e-00
  090 0.00e-00
  092 .07142648
 54  13.5 0.0
14
  0832120  0.0000e00
  0832130  0.0000e00
  0842120  0.0000e00
  0842130  0.0000e00
  0842160  0.0000e00
  0852170  0.0000e00
  0862200  0.0000e00
  0872210  0.0000e00
  0882240  0.0000e00
  0892250  0.0000e00
  0902280  0.0000e00
  0902290  0.0000e00
  0922320  2.2015e18
  0922330  7.3360e21
1  4000
  0040090  .928571429
```

$^{dirty}UBe_{13}$ at 1 year Input Card (tape1)

```
233UBe13 300ppm 232 @ 1.0 yr
1  2  1
12  0
  004 .928571429
  081 1.96e-13
  082 3.36e-08
  083 2.38e-11
  084 4.32e-16
  085 3.91e-20
  086 1.66e-13
  087 2.25e-16
  088 9.45e-10
  089 1.05e-12
  090 4.91e-07
  092 .07142648
 54  13.5 0.0
14
  0832120  1.1130e12
  0832130  3.4047e08
  0842120  5.8681e01
  0842130  4.4451e-1
  0842160  4.4417e07
  0852170  4.0203e03
  0862200  1.7032e10
  0872210  2.3151e07
  0882240  9.6867e13
  0892250  1.0754e11
  0902280  1.8477e16
  0902290  3.1942e16
  0922320  2.1794e18
  0922330  7.3360e21
1  4000
  0040090  .928571429
```

$^{dirty}UBe_{13}$ at 3 years Input Card (tape1)

```
233UBe13 300ppm 232 @ 3.0 yr
1  2  1
12  0
  004 .928571429
  081 4.23e-13
  082 2.47e-07
  083 1.50e-10
  084 9.31e-16
  085 1.26e-19
  086 3.57e-13
  087 7.25e-16
  088 2.04e-09
  089 3.37e-12
  090 1.32e-06
  092 .07142648
 54  13.5 0.0
14
  0832120  2.3966e12
  0832130  1.0951e09
  0842120  1.2635e02
  0842130  1.4297e00
  0842160  9.5639e07
  0852170  1.2931e04
  0862200  3.6673e10
  0872210  7.4464e07
  0882240  2.0858e14
  0892250  3.4590e11
  0902280  3.9786e16
  0902290  9.5889e16
  0922320  2.1360e18
  0922330  7.3359e21
1  4000
  0040090  .928571429
```

$^{dirty}UBe_{13}$ at 5 years Input Card (tape1)

```
233UBe13 300ppm 232 @ 5.0 yr
1  2  1
12  0
  004 .928571429
  081 5.26e-13
  082 5.67e-07
  083 3.86e-10
  084 1.16e-16
  085 2.13e-19
  086 4.44e-13
  087 1.22e-15
  088 2.53e-09
  089 5.69e-12
  090 2.04e-06
  092 .07142648
 54  13.5 0.0
14
  0832120  2.9812e12
  0832130  1.8499e09
  0842120  1.5717e02
  0842130  2.4152e00
  0842160  1.1897e08
  0852170  2.1844e04
  0862200  4.5617e10
  0872210  1.2579e08
  0882240  2.5945e14
  0892250  5.8430e11
  0902280  4.9489e16
  0902290  1.5977e17
  0922320  2.0935e18
  0922330  7.3359e21
1  4000
  0040090  .928571429
```

$^{dirty}UBe_{13}$ at 7.5 years Input Card (tape1)

```
233UBe13 300ppm 232 @ 7.5 yr
1  2  1
12  0
  004 .928571429
  081 5.75e-13
  082 1.03e-06
  083 8.41e-10
  084 1.27e-15
  085 3.21e-19
  086 4.86e-13
  087 1.85e-15
  088 2.77e-09
  089 8.59e-12
  090 2.86e-06
  092 .07142648
 54  13.5 0.0
14
  0832120  3.2599e12
  0832130  2.7932e09
  0842120  1.7186e02
  0842130  3.6468e00
  0842160  1.3009e08
  0852170  3.2983e04
  0862200  4.9882e10
  0872210  1.8993e08
  0882240  2.8370e14
  0892250  8.8227e11
  0902280  5.4116e16
  0902290  2.3965e17
  0922320  2.0415e18
  0922330  7.3358e21
1  4000
  0040090  .928571429
```

$^{dirty}UBe_{13}$ at 10.17 years Input Card (tape1)

```
233UBe13 300ppm 232 @ 10.17 yr
1  2  1
12  0
  004 .928571429
  081 5.86e-13
  082 1.54e-06
  083 3.24e-11
  084 1.29e-15
  085 4.38e-19
  086 4.95e-13
  087 2.52e-15
  088 2.83e-09
  089 1.17e-11
  090 3.71e-06
  092 .07142648
 54  13.5 0.0
14
  0832120  3.3215e12
  0832130  3.8096e09
  0842120  1.7511e02
  0842130  4.9737e00
  0842160  1.3255e08
  0852170  4.4984e04
  0862200  5.0825e10
  0872210  2.5904e08
  0882240  2.8907e14
  0892250  1.2033e11
  0902280  5.5140e16
  0902290  3.2578e17
  0922320  1.9874e18
  0922330  7.3357e21
1  4000
  0040090  .928571429
```

$^{dirty}UBe_{13}$ (60% theoretical density) at 10.17 years Input Card (tape1)

```
233UBe13 300ppm 232 @ 10.17 yr 60% theoretical density
1  2  1
12  0
  004 .928571429
  081 5.86e-13
  082 1.54e-06
  083 3.24e-11
  084 1.29e-15
  085 4.38e-19
  086 4.95e-13
  087 2.52e-15
  088 2.83e-09
  089 1.17e-11
  090 3.71e-06
  092 .07142648
 54  13.5 0.0
14
  0832120  1.9929e12
  0832130  2.2857609
  0842120  1.0266e02
  0842130  2.9842e00
  0842160  0.7953e08
  0852170  2.6990e04
  0862200  3.0495e10
  0872210  1.5542e08
  0882240  7.7344e14
  0892250  0.7220e11
  0902280  3.3084e16
  0902290  1.9547e17
  0922320  1.1924e18
  0922330  4.4014e21
1  4000
                          0040090  .928571429
```

$^{dirty}UO_2$ at 10.17 years Input Card (tape1)

```
dirtyUO2 300ppm 232 @ 10.17 yr
1  2  1
12  0
  008 .66666667
  081 2.74e-12
  082 7.21e-06
  083 1.51e-10
  084 6.02e-15
  085 2.04e-18
  086 2.31e-12
  087 1.18e-14
  088 1.32e-08
  089 5.47e-11
  090 1.73e-05
  092 .33333333
 54  13.5 0.0
14
  0832120  1.1054e13
  0832130  1.2678e10
  0842120  5.8275e02
  0842130  1.6552e01
  0842160  4.4110e08
  0852170  1.4970e05
  0862200  1.6914e11
  0872210  8.6205e08
  0882240  9.6197e14
  0892250  4.0044e12
  0902280  1.8350e17
  0902290  1.0842e18
  0922320  6.6136e18
  0922330  2.4412e22
2  1000
  0080170  .0002533
  0080180  .00205
```

$^{228}ThBe_{13}$ Neutron Source in Secular Equilibrium (tape1)

```
228ThBe13 in secular equilibrium
1  2  1
8  0
  004 .928571429
  081 7.76e-08
  082 4.51e-05
  083 4.28e-06
  084 1.71e-10
  086 6.55e-08
  088 3.72e-04
  090 7.10e-02
 54  13.5 0.0
6
  0832120  4.2292e17
  0842120  2.2296e07
  0842160  1.6877e13
  0862200  6.4714e15
  0882240  3.6806e19
  0902280  7.0208e21
1  4000
  0040090  .928571429
```

APPENDIX A.2

MCNP5 Input Decks

$^{dirty}UBe_{13}$ Source at 0 years Neutron Multiplication Input Deck

```
c 233UBe13 300ppm U-232 Cylinder 0 years
c cell cards
1     1 -4.2674 -1              $source proper
2     2 -7.874  -2  1            $ encapsulation
3     0         -3  2           $  air
4     0          3             $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 1.5
4     SO 30
c end of cube surfaces

IMP:N 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3
SI1 0 53I 13.5
SP1 0.00E+00 2.87E-04 6.36E-03 1.60E-02 1.85E-02 1.84E-02 1.60E-02 1.12E-02
    1.67E-02 2.12E-02 2.55E-02 3.29E-02 4.95E-02 5.59E-02 5.28E-02 5.00E-02
    4.63E-02 4.33E-02 4.11E-02 3.86E-02 3.56E-02 2.86E-02 2.09E-02 1.74E-02
    1.72E-02 1.71E-02 2.28E-02 2.61E-02 2.65E-02 2.65E-02 2.64E-02 2.57E-02
    2.39E-02 2.15E-02 1.92E-02 1.78E-02 1.68E-02 1.58E-02 1.34E-02 9.84E-03
    5.82E-03 3.15E-03 1.39E-03 5.47E-04 5.80E-05 1.18E-12 9.46E-13 7.60E-13
    6.11E-13 4.90E-13 3.93E-13 3.15E-13 2.52E-13 2.01E-13 1.62E-13
SI2 .5509
SI3 .5509
c
F1:N 3
E0 0 55I 14
FC1 number passing through 1.5 cm radius sphere
c
M1 4009 .928547421 92232.69c 2.142e-5 92233.69c 0.0711 90228.92C 5.37e-7
    82208.60c 3.0104e-4
M2 26056 1
NPS 1000000
```

$^{dirty}UBe_{13}$ Source at 1 year Neutron Multiplication Input Deck

```
c 233UBe13 300ppm U-232 Cylinder 1 year
c cell cards
1     1 -4.2674 -1              $source proper
2     2 -7.874  -2  1            $ encapsulation
3     0         -3  2           $  air
4     0          3             $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 1.5
4     SO 30
c end of cube surfaces

IMP:N 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3
SI1 0 53I 13.5
SP1 0.00E+00 2.87E-04 6.36E-03 1.60E-02 1.85E-02 1.84E-02 1.60E-02 1.12E-02
    1.67E-02 2.12E-02 2.55E-02 3.29E-02 4.95E-02 5.59E-02 5.28E-02 5.00E-02
    4.63E-02 4.33E-02 4.11E-02 3.86E-02 3.56E-02 2.86E-02 2.09E-02 1.74E-02
    1.72E-02 1.71E-02 2.28E-02 2.61E-02 2.65E-02 2.65E-02 2.64E-02 2.57E-02
    2.39E-02 2.15E-02 1.92E-02 1.78E-02 1.68E-02 1.58E-02 1.34E-02 9.84E-03
    5.82E-03 3.15E-03 1.39E-03 5.47E-04 5.80E-05 1.18E-12 9.46E-13 7.60E-13
    6.11E-13 4.90E-13 3.93E-13 3.15E-13 2.52E-13 2.01E-13 1.62E-13
SI2 .5509
SI3 .5509
c
F1:N 3
E0 0 55I 14
FC1 number passing through 1.5 cm radius sphere
c
M1 4009 .928547421 92232.69c 2.142e-5 92233.69c 0.0711 90228.92C 5.37e-7
    82208.60c 3.0104e-4
M2 26056 1
NPS 1000000
```

$^{dirty}UBe_{13}$ Source at 3 years Neutron Multiplication Input Deck

```
c 233UBe13 300ppm U-232 Cylinder 3 years
c cell cards
1     1 -4.2674 -1              $source proper
2     2 -7.874  -2  1            $ encapsulation
3     0         -3  2           $  air
4     0          3             $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 1.5
4     SO 30
c end of cube surfaces

IMP:N 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3
SI1 0 53I 13.5
SP1 0.00E+00 6.53E-04 4.66E-03 1.08E-02 1.35E-02 1.43E-02 1.32E-02 1.05E-02
    1.37E-02 1.63E-02 1.82E-02 2.18E-02 3.07E-02 3.68E-02 4.00E-02 4.09E-02
    3.94E-02 3.78E-02 3.68E-02 3.56E-02 3.38E-02 2.97E-02 2.52E-02 2.31E-02
    2.29E-02 2.33E-02 2.64E-02 2.75E-02 2.77E-02 2.80E-02 2.81E-02 2.75E-02
    2.64E-02 2.49E-02 2.35E-02 2.25E-02 2.18E-02 2.10E-02 1.95E-02 1.71E-02
    1.41E-02 1.14E-02 9.10E-03 7.08E-03 5.43E-03 4.21E-03 3.47E-03 2.89E-03
    2.35E-03 1.85E-03 1.37E-03 9.09E-04 4.51E-04 1.42E-04 1.02E-04
SI2 .5509
SI3 .5509
c
F1:N 3
E0 0 55I 14
FC1 number passing through 1.5 cm radius sphere
c
M1 4009 .928547421 92232.69c 2.142e-5 92233.69c 0.0711 90228.92C 5.37e-7
    82208.60c 3.0104e-4
M2 26056 1
NPS 1000000
```

$^{dirty}UBe_{13}$ Source at 5 years Neutron Multiplication Input Deck

```
c 233UBe13 300ppm U-232 Cylinder 5 years
c cell cards
1     1 -4.2674 -1              $source proper
2     2 -7.874  -2  1            $ encapsulation
3     0         -3  2           $  air
4     0          3             $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 1.5
4     SO 30
c end of cube surfaces

IMP:N 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3
SI1 0 53I 13.5
SP1 0.00E+00 6.72E-04 4.57E-03 1.05E-02 1.33E-02 1.41E-02 1.31E-02 1.05E-02
    1.36E-02 1.61E-02 1.79E-02 2.13E-02 2.97E-02 3.57E-02 3.93E-02 4.04E-02
    3.90E-02 3.75E-02 3.66E-02 3.54E-02 3.37E-02 2.97E-02 2.54E-02 2.34E-02
    2.31E-02 2.36E-02 2.66E-02 2.76E-02 2.77E-02 2.81E-02 2.82E-02 2.76E-02
    2.65E-02 2.51E-02 2.37E-02 2.27E-02 2.20E-02 2.13E-02 1.98E-02 1.75E-02
    1.46E-02 1.18E-02 9.50E-03 7.42E-03 5.72E-03 4.44E-03 3.65E-03 3.04E-03
    2.48E-03 1.95E-03 1.45E-03 9.57E-04 4.75E-04 1.50E-04 1.08E-04
SI2 .5509
SI3 .5509
c
F1:N 3
E0 0 55I 14
FC1 number passing through 1.5 cm radius sphere
c
M1 4009 .928547421 92232.69c 2.142e-5 92233.69c 0.0711 90228.92C 5.37e-7
    82208.60c 3.0104e-4
M2 26056 1
NPS 1000000
```

$^{dirty}UBe_{13}$ Source at 7.5 years Neutron Multiplication Input Deck

```
c 233UBe13 300ppm U-232 Cylinder 7.5 years
c cell cards
1     1 -4.2674 -1              $source proper
2     2 -7.874  -2  1            $ encapsulation
3     0         -3  2           $  air
4     0          3             $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 1.5
4     SO 30
c end of cube surfaces

IMP:N 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3
SI1 0 53I 13.5
SP1 0.00E+00 6.80E-04 4.53E-03 1.04E-02 1.31E-02 1.40E-02 1.30E-02 1.04E-02
    1.35E-02 1.60E-02 1.77E-02 2.10E-02 2.93E-02 3.53E-02 3.91E-02 4.03E-02
    3.89E-02 3.74E-02 3.65E-02 3.53E-02 3.36E-02 2.98E-02 2.55E-02 2.35E-02
    2.33E-02 2.38E-02 2.67E-02 2.76E-02 2.78E-02 2.81E-02 2.83E-02 2.76E-02
    2.66E-02 2.52E-02 2.38E-02 2.28E-02 2.21E-02 2.14E-02 1.99E-02 1.77E-02
    1.47E-02 1.20E-02 9.67E-03 7.56E-03 5.83E-03 4.53E-03 3.73E-03 3.11E-03
    2.53E-03 1.99E-03 1.48E-03 9.77E-04 4.85E-04 1.53E-04 1.10E-04
SI2 .5509
SI3 .5509
c
F1:N 3
E0 0 55I 14
FC1 number passing through 1.5 cm radius sphere
c
M1 4009 .928547421 92232.69c 2.142e-5 92233.69c 0.0711 90228.92C 5.37e-7
    82208.60c 3.0104e-4
M2 26056 1
NPS 1000000
```

$^{dirty}UBe_{13}$ Source at 10.17 years Neutron Multiplication Input Deck

```
c 233UBe13 300ppm U-232 Cylinder 10.17 years
c cell cards
1     1 -4.2674 -1              $source proper
2     2 -7.874  -2  1            $ encapsulation
3     0         -3  2           $  air
4     0          3             $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 1.5
4     SO 30
c end of cube surfaces

IMP:N 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3
SI1 0 53I 13.5
SP1 0.00E+00 6.83E-04 4.53E-03 1.03E-02 1.31E-02 1.40E-02 1.30E-02 1.04E-02
    1.35E-02 1.59E-02 1.77E-02 2.10E-02 2.92E-02 3.52E-02 3.90E-02 4.02E-02
    3.88E-02 3.74E-02 3.64E-02 3.53E-02 3.36E-02 2.98E-02 2.55E-02 2.36E-02
    2.33E-02 2.38E-02 2.67E-02 2.76E-02 2.78E-02 2.81E-02 2.83E-02 2.76E-02
    2.66E-02 2.52E-02 2.38E-02 2.29E-02 2.22E-02 2.14E-02 1.99E-02 1.77E-02
    1.48E-02 1.20E-02 9.72E-03 7.60E-03 5.87E-03 4.56E-03 3.75E-03 3.12E-03
    2.54E-03 2.00E-03 1.49E-03 9.82E-04 4.87E-04 1.54E-04 1.11E-04
SI2 .5509
SI3 .5509
c
F1:N 3
E0 0 55I 14
FC1 number passing through 1.5 cm radius sphere
c
M1 4009 .928547421 92232.69c 2.142e-5 92233.69c 0.0711 90228.92C 5.37e-7
    82208.60c 3.0104e-4
M2 26056 1
NPS 1000000
```

$^{dirty}UBe_{13}$ Source at 1 year Photon Interactions Input Deck

```
c 233UBe13 300ppm U-232 Cylinder 1 year
c cell cards
1     1 -4.2674 -1            $source proper
2     2 -7.874  -2  1          $ encapsulation
3     0         -3  2         $ blanket
4     0         -4  3         $air
5     0          4            $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3 PAR=2
SI1  L  0.0106 0.0728042 0.0749694 0.0849 0.2114 0.23336 0.25261 &
    0.277358 0.48595 0.51077 0.583191 0.5877 0.6501 0.7052 &
    0.72204 0.7487 0.76313 0.8212 0.860564 0.8833 0.9276 &
    0.9827 1.0939 1.1608 1.1852 1.2828 2.614533 0.0108 &
    0.07481 0.07711 0.0873 0.115183 0.17668 0.238632 0.300087 &
    0.4152 0.0103 0.0111 0.039857 0.0708319 0.0728715 0.076862 &
    0.07929 0.0826 0.0898 0.2882 0.32803 0.4337 0.45298 &
    0.473 0.72733 0.78537 0.893408 0.95212 1.0736 1.07862 &
    1.5127 1.6205 1.6797 1.806 0.54973 0.0117 0.08107 &
    0.08378 0.0949 0.240986 0.0123 0.084373 0.08847 0.131613 &
    0.16641 0.20593 0.215983 0.013 0.05778 0.12908
SP1 0.00397519 0.002768926 0.004674275 0.00206984 0.000243994 &
    0.000420822 0.000945821 0.008649466 6.85378E-05 0.030979068 &
    0.115828818 5.48302E-05 4.93472E-05 3.01566E-05 0.000275522 &
    5.89425E-05 0.002481067 5.48302E-05 0.01702478 4.24934E-05 &
    0.000179569 0.000278263 0.000548302 1.50783E-05 2.33028E-05 &
    7.12793E-05 0.13592409 0.057495567 0.039599596 0.066633935 &
    0.029737773 0.002254131 0.000197998 0.164871394 0.012489103 &
    0.000544494 0.030461228 0.000224652 0.004036113 0.000219702 &
    0.000372389 0.00014926 0.000250163 0.000164491 0.000112326 &
    0.001283179 0.000475957 6.47301E-05 0.001382178 0.000190383 &
    0.02505436 0.004196034 0.001439293 0.000647301 6.09225E-05 &
    0.002147517 0.001104219 0.005673404 0.000220844 0.000342689 &
    0.000434072 0.001561138 0.000498803 0.000826261 0.000376958 &
    0.015611379 0.037374806 0.004700749 4.54663E-05 0.000502826 &
```

```
     0.000399178 7.55202E-05 0.00097868 0.151397751 0.002522034 &
     0.000860444
SI2 .5509
SI3 .5509
c
PHYS:P J J J -1 J
Mode N P
F1:N 4
E1 0 55I 14
F11:P 4
E11 0 26I 2.7
FC1 number neutrons passing through 3.5 cm radius sphere
FC11 number gamma passing through 3.5 cm radius sphere
c
M1 4009.26u .928571429 92233.27u .07142648 82208.24u 3.52e-6
M2 26056 1
NPS 5e7
```

$^{dirty}UBe_{13}$ Source at 3 years Photon Interactions Input Deck

```
c 233UBe13 300ppm U-232 Cylinder 3 years
c cell cards
1     1 -4.2674 -1            $source proper
2     2 -7.874  -2  1          $ encapsulation
3     0         -3  2         $ blanket
4     0         -4  3         $air
5     0          4            $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3 PAR=2
SI1  L  0.0106 0.0728042 0.0749694 0.0849 0.2114 0.23336 0.25261 &
     0.277358 0.48595 0.51077 0.583191 0.5877 0.6501 0.7052 &
     0.72204 0.7487 0.76313 0.8212 0.860564 0.8833 0.9276 &
     0.9827 1.0939 1.1608 1.1852 1.2828 2.614533 0.0108 &
     0.07481 0.07711 0.0873 0.115183 0.17668 0.238632 0.300087 &
     0.4152 0.0103 0.0111 0.039857 0.0708319 0.0728715 0.076862 &
     0.07929 0.0826 0.0898 0.2882 0.32803 0.4337 0.45298 &
     0.473 0.72733 0.78537 0.893408 0.95212 1.0736 1.07862 &
     1.5127 1.6205 1.6797 1.806 0.54973 0.0117 0.08107 &
     0.08378 0.0949 0.240986 0.0123 0.084373 0.08847 0.131613 &
     0.16641 0.20593 0.215983 0.013 0.05778 0.12908
SP1 0.004360916 0.003037603 0.005127836 0.002270684 0.00026767 &
     0.000461656 0.001037597 0.009488751 7.51882E-05 0.033985068 &
     0.127068065 6.01506E-05 5.41355E-05 3.30828E-05 0.000302257 &
     6.46619E-05 0.002721813 6.01506E-05 0.01867675 4.66167E-05 &
     0.000196993 0.000305264 0.000601506 1.65414E-05 2.5564E-05 &
     7.81957E-05 0.149113247 0.063074549 0.043442074 0.073099643 &
     0.032623326 0.002472856 0.00021721 0.180869402 0.013700962 &
     0.000597329 0.03341698 0.00024645 0.00442775 0.00024102 &
     0.000408523 0.000163743 0.000274437 0.000180452 0.000123225 &
     0.00140769 0.00052214 7.10111E-05 0.001516295 0.000208856 &
     0.027485466 0.004603189 0.001578952 0.000710111 6.6834E-05 &
     0.002355897 0.001211366 0.006223912 0.000242273 0.000375941 &
     0.000476192 0.00171262 0.000547203 0.000906436 0.000413535 &
     0.017126202 0.039516477 0.004970114 4.80716E-05 0.000531639 &
```

```
     0.000422052 7.98477E-05 0.001034761 0.07288857 0.001214202 &
     0.00041425
SI2 .5509
SI3 .5509
c
PHYS:P J J J -1 J
Mode N P
F1:N 4
E1 0 55I 14
F11:P 4
E11 0 26I 2.7
FC1 number neutrons passing through 3.5 cm radius sphere
FC11 number gamma passing through 3.5 cm radius sphere
c
M1 4009.26u .928571429 92233.27u .07142648 82208.24u 3.52e-6
M2 26056 1
NPS 5e7
```

$^{dirty}UBe_{13}$ Source at 5 years Photon Interactions Input Deck

```
c 233UBe13 300ppm U-232 Cylinder 5 Years
c cell cards
1     1 -4.2674 -1            $source proper
2     2 -7.874  -2  1          $ encapsulation
3     0         -3  2         $ blanket
4     0         -4  3         $air
5     0          4            $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3 PAR=2
SI1  L  0.0106 0.0728042 0.0749694 0.0849 0.2114 0.23336 0.25261 &
    0.277358 0.48595 0.51077 0.583191 0.5877 0.6501 0.7052 &
    0.72204 0.7487 0.76313 0.8212 0.860564 0.8833 0.9276 &
    0.9827 1.0939 1.1608 1.1852 1.2828 2.614533 0.0108 &
    0.07481 0.07711 0.0873 0.115183 0.17668 0.238632 0.300087 &
    0.4152 0.0103 0.0111 0.039857 0.0708319 0.0728715 0.076862 &
    0.07929 0.0826 0.0898 0.2882 0.32803 0.4337 0.45298 &
    0.473 0.72733 0.78537 0.893408 0.95212 1.0736 1.07862 &
    1.5127 1.6205 1.6797 1.806 0.54973 0.0117 0.08107 &
    0.08378 0.0949 0.240986 0.0123 0.084373 0.08847 0.131613 &
    0.16641 0.20593 0.215983 0.013 0.05778 0.12908
SP1 0.004418567 0.00307776 0.005195625 0.002300702 0.000271209 &
    0.000467759 0.001051314 0.009614192 7.61822E-05 0.034434347 &
    0.12874789 6.09457E-05 5.48512E-05 3.35202E-05 0.000306252 &
    6.55167E-05 0.002757795 6.09457E-05 0.018923654 4.7233E-05 &
    0.000199597 0.0003093 0.000609457 1.67601E-05 2.59019E-05 &
    7.92295E-05 0.151084507 0.063908387 0.044016373 0.074066012 &
    0.033054603 0.002505547 0.000220082 0.183260475 0.013882087 &
    0.000605225 0.033858748 0.000249708 0.004486284 0.000244206 &
    0.000413923 0.000165908 0.000278065 0.000182837 0.000124854 &
    0.0014263 0.000529043 7.19498E-05 0.001536341 0.000211617 &
    0.02784882 0.004664043 0.001599826 0.000719498 6.77175E-05 &
    0.002387042 0.00122738 0.006306192 0.000245476 0.000380911 &
    0.000482487 0.001735261 0.000554437 0.000918419 0.000419002 &
    0.017352609 0.04107787 0.005166495 4.9971E-05 0.000552646 &
```

```
     0.000438729 8.30027E-05 0.001075647 0.059722628 0.000994879 &
     0.000339424
SI2 .5509
SI3 .5509
c
PHYS:P J J J -1 J
Mode N P
F1:N 4
E1 0 55I 14
F11:P 4
E11 0 26I 2.7
FC1 number neutrons passing through 3.5 cm radius sphere
FC11 number gamma passing through 3.5 cm radius sphere
c
M1 4009.26u .928571429 92233.27u .07142648 82208.24u 3.52e-6
M2 26056 1
NPS 5e7
```

$^{dirty}UBe_{13}$ Source at 7.5 years Photon Interactions Input Deck

```
c 233UBe13 300ppm U-232 Cylinder 7.5 Years
c cell cards
1     1 -4.2674 -1            $source proper
2     2 -7.874  -2  1          $ encapsulation
3     0         -3  2         $ blanket
4     0         -4  3         $air
5     0          4            $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3 PAR=2
SI1  L  0.0106 0.0728042 0.0749694 0.0849 0.2114 0.23336 0.25261 &
     0.277358 0.48595 0.51077 0.583191 0.5877 0.6501 0.7052 &
     0.72204 0.7487 0.76313 0.8212 0.860564 0.8833 0.9276 &
     0.9827 1.0939 1.1608 1.1852 1.2828 2.614533 0.0108 &
     0.07481 0.07711 0.0873 0.115183 0.17668 0.238632 0.300087 &
     0.4152 0.0103 0.0111 0.039857 0.0708319 0.0728715 0.076862 &
     0.07929 0.0826 0.0898 0.2882 0.32803 0.4337 0.45298 &
     0.473 0.72733 0.78537 0.893408 0.95212 1.0736 1.07862 &
     1.5127 1.6205 1.6797 1.806 0.54973 0.0117 0.08107 &
     0.08378 0.0949 0.240986 0.0123 0.084373 0.08847 0.131613 &
     0.16641 0.20593 0.215983 0.013 0.05778 0.12908
SP1 0.004448246 0.003098433 0.005230524 0.002316156 0.00027303 &
     0.000470901 0.001058376 0.00967877 7.66939E-05 0.034665641 &
     0.129612685 6.13551E-05 5.52196E-05 3.37453E-05 0.000308309 &
     6.59568E-05 0.002776319 6.13551E-05 0.019050764 4.75502E-05 &
     0.000200938 0.000311377 0.000613551 1.68727E-05 2.60759E-05 &
     7.97617E-05 0.152099335 0.064337657 0.044312029 0.07456351 &
     0.033276629 0.002522377 0.00022156 0.184491428 0.013975332 &
     0.00060929 0.034086176 0.000251386 0.004516418 0.000245847 &
     0.000416704 0.000167022 0.000279933 0.000184065 0.000125693 &
     0.00143588 0.000532597 7.24331E-05 0.00154666 0.000213039 &
     0.02803588 0.004695371 0.001610572 0.000724331 6.81724E-05 &
     0.002403075 0.001235624 0.00634855 0.000247125 0.000383469 &
     0.000485728 0.001746917 0.000558161 0.000924588 0.000421816 &
     0.017469165 0.041329489 0.005198142 5.02771E-05 0.000556031 &
```

```
     0.000441416 8.35111E-05 0.001082236 0.053581644 0.000892581 &
     0.000304522
SI2 .5509
SI3 .5509
c
PHYS:P J J J -1 J
Mode N P
F1:N 4
E1 0 55I 14
F11:P 4
E11 0 26I 2.7
FC1 number neutrons passing through 3.5 cm radius sphere
FC11 number gamma passing through 3.5 cm radius sphere
c
M1 4009.26u .928571429 92233.27u .07142648 82208.24u 3.52e-6
M2 26056 1
NPS 5e7
```

$^{dirty}UBe_{13}$ Source at 10.17 years Photon Interactions Input Deck

```
c 233UBe13 300ppm U-232 Cylinder 10.17 years/vacuum
c cell cards
1     1 -4.2674 -1            $source proper
2     2 -7.874  -2  1          $ encapsulation
3     0         -3  2         $ blanket
4     0         -4  3         $air
5     0          4            $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3 PAR=2
SI1  L  0.0106 0.0728042 0.0749694 0.0849 0.2114 0.23336 0.25261 &
    0.277358 0.48595 0.51077 0.583191 0.5877 0.6501 0.7052 &
    0.72204 0.7487 0.76313 0.8212 0.860564 0.8833 0.9276 &
    0.9827 1.0939 1.1608 1.1852 1.2828 2.614533 0.0108 &
    0.07481 0.07711 0.0873 0.115183 0.17668 0.238632 0.300087 &
    0.4152 0.0103 0.0111 0.039857 0.0708319 0.0728715 0.076862 &
    0.07929 0.0826 0.0898 0.2882 0.32803 0.4337 0.45298 &
    0.473 0.72733 0.78537 0.893408 0.95212 1.0736 1.07862 &
    1.5127 1.6205 1.6797 1.806 0.54973 0.0117 0.08107 &
    0.08378 0.0949 0.240986 0.0123 0.084373 0.08847 0.131613 &
    0.16641 0.20593 0.215983 0.013 0.05778 0.12908
SP1 0.004459427 0.003106221 0.00524367 0.002321977 0.000273717 &
    0.000472084 0.001061036 &
    0.009703097 7.68867E-05 0.034752772 0.129938462 6.15093E-05 &
    5.53584E-05 3.38301E-05 &
    0.000309084 6.61225E-05 0.002783297 6.15093E-05 0.019098647 &
    4.76697E-05 0.000201443 &
    0.00031216 0.000615093 1.69151E-05 2.61415E-05 7.99621E-05 &
    0.152481632 0.064499368 &
    0.044423406 0.074750924 0.033360269 0.002528717 0.000222117 &
    0.184955142 0.014010459 &
    0.000610822 0.034171851 0.000252017 0.00452777 0.000246464 &
    0.000417751 0.000167442 &
    0.000280636 0.000184528 0.000126009 0.001439489 0.000533935 &
    7.26152E-05 0.001550548 &
```

```
     0.000213574 0.028106347 0.004707172 0.00161462 0.000726152 &
     6.83437E-05 0.002409115 &
     0.00123873 0.006364507 0.000247746 0.000384433 0.000486949 &
     0.001751307 0.000559564 &
     0.000926911 0.000422877 0.017513074 0.041433369 0.005211207 &
     5.04035E-05 0.000557428 &
     0.000442525 8.3721E-05 0.001084956 0.051257776 0.000853869 &
     0.000291315
SI2 .5509
SI3 .5509
c
PHYS:P J J J -1 J
Mode N P
F1:N 4
E1 0 55I 14
F11:P 4
E11 0 26I 2.7
FC1 number neutrons passing through 3.5 cm radius sphere
FC11 number gamma passing through 3.5 cm radius sphere
c
M1 4009.26u .928571429 92233.27u .07142648 82208.24u 3.52e-6
M2 26056 1
NPS 5e7
```

4.438 γ in $^{dirty}UBe_{13}$ Source at 10.17 years Photon Interactions Input Deck

```
c 233UBe13 300ppm U-232 Cylinder 10.17 years/vacuum 4.438 gammas
c cell cards
1     1 -4.2674 -1              $source proper
2     2 -7.874  -2  1           $ encapsulation
3     0         -3  2           $ blanket
4     0         -4  3           $air
5     0          4              $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0     0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=4.438 EXT=D3 PAR=2
SI2 .5509
SI3 .5509
c
PHYS:P J J J -1 J
Mode N P
F1:N 4
E1 0 55I 14
F11:P 4
E11 0 26I 2.7
FC1 number neutrons passing through 3.5 cm radius sphere
FC11 number gamma passing through 3.5 cm radius sphere
c
M1 4009.26u .928571429 92233.27u .07142648 82208.24u 3.52e-6
M2 26056 1
NPS 5e7
```

Beryllium Blanket Neutron Multiplication Input Deck

```
c 233UBe13 300ppm U-232 Cylinder Be Blanket
c cell cards
1     1 -4.2674 -1              $source proper
2     2 -7.874  -2  1             $ encapsulation
3     3 -1.848  -3  2             $blanket
4     0         -4  3           $  air
5     0          4              $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3
SI1 0 53I 13.5
SP1 0.00E+00 6.83E-04 4.53E-03 1.03E-02 1.31E-02 1.40E-02 1.30E-02 1.04E-02
    1.35E-02 1.59E-02 1.77E-02 2.10E-02 2.92E-02 3.52E-02 3.90E-02 4.02E-02
    3.88E-02 3.74E-02 3.64E-02 3.53E-02 3.36E-02 2.98E-02 2.55E-02 2.36E-02
    2.33E-02 2.38E-02 2.67E-02 2.76E-02 2.78E-02 2.81E-02 2.83E-02 2.76E-02
    2.66E-02 2.52E-02 2.38E-02 2.29E-02 2.22E-02 2.14E-02 1.99E-02 1.77E-02
    1.48E-02 1.20E-02 9.72E-03 7.60E-03 5.87E-03 4.56E-03 3.75E-03 3.12E-03
    2.54E-03 2.00E-03 1.49E-03 9.82E-04 4.87E-04 1.54E-04 1.11E-04
SI2 .5509
SI3 .5509
c
F1:N 3
E0 0 55I 14
FC1 number passing through 1.5 cm radius sphere
c
M1 4009 .928547421 92232.69c 2.142e-5 92233.69c 0.0711 90228.92C 5.37e-7
    82208.60c 3.0104e-4
M2 26056 1
M3 4009 1
NPS 1000000
```

$Be_{12}W$ Blanket Neutron Multiplication Input Deck

```
c 233UBe13 300ppm U-232 Cylinder Be12W Blanket
c cell cards
1     1 -4.2674 -1            $source proper
2     2 -7.874  -2  1          $ encapsulation
3     3 -3.2    -3  2          $blanket
4     0         -4  3         $  air
5     0          4            $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3
SI1 0 53I 13.5
SP1 0.00E+00 6.83E-04 4.53E-03 1.03E-02 1.31E-02 1.40E-02 1.30E-02 1.04E-02
    1.35E-02 1.59E-02 1.77E-02 2.10E-02 2.92E-02 3.52E-02 3.90E-02 4.02E-02
    3.88E-02 3.74E-02 3.64E-02 3.53E-02 3.36E-02 2.98E-02 2.55E-02 2.36E-02
    2.33E-02 2.38E-02 2.67E-02 2.76E-02 2.78E-02 2.81E-02 2.83E-02 2.76E-02
    2.66E-02 2.52E-02 2.38E-02 2.29E-02 2.22E-02 2.14E-02 1.99E-02 1.77E-02
    1.48E-02 1.20E-02 9.72E-03 7.60E-03 5.87E-03 4.56E-03 3.75E-03 3.12E-03
    2.54E-03 2.00E-03 1.49E-03 9.82E-04 4.87E-04 1.54E-04 1.11E-04
SI2 .5509
SI3 .5509
c
F1:N 3
E0 0 55I 14
FC1 number passing through 1.5 cm radius sphere
c
M1 4009 .928547421 92232.69c 2.142e-5 92233.69c 0.0711 90228.92C 5.37e-7
    82208.60c 3.0104e-4
M2 26056 1
M3 74000.21c .076923 4009.24c .9230769
NPS 1000000
```

BeHD Blanket Neutron Multiplication Input Deck

```
c 233UBe13 300ppm U-232 Cylinder BeHD Blanket
c cell cards
1     1 -4.2674 -1             $source proper
2     2 -7.874  -2  1           $ encapsulation
3     3 -.694   -3  2           $blanket
4     0         -4  3          $  air
5     0          4             $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3
SI1 0 53I 13.5
SP1 0.00E+00 6.83E-04 4.53E-03 1.03E-02 1.31E-02 1.40E-02 1.30E-02 1.04E-02
    1.35E-02 1.59E-02 1.77E-02 2.10E-02 2.92E-02 3.52E-02 3.90E-02 4.02E-02
    3.88E-02 3.74E-02 3.64E-02 3.53E-02 3.36E-02 2.98E-02 2.55E-02 2.36E-02
    2.33E-02 2.38E-02 2.67E-02 2.76E-02 2.78E-02 2.81E-02 2.83E-02 2.76E-02
    2.66E-02 2.52E-02 2.38E-02 2.29E-02 2.22E-02 2.14E-02 1.99E-02 1.77E-02
    1.48E-02 1.20E-02 9.72E-03 7.60E-03 5.87E-03 4.56E-03 3.75E-03 3.12E-03
    2.54E-03 2.00E-03 1.49E-03 9.82E-04 4.87E-04 1.54E-04 1.11E-04
SI2 .5509
SI3 .5509
c
F1:N 3
E0 0 55I 14
FC1 number passing through 1.5 cm radius sphere
c
M1 4009 .928547421 92232.69c 2.142e-5 92233.69c 0.0711 90228.92C 5.37e-7
    82208.60c 3.0104e-4
M2 26056 1
M3 4009.24c .333 1002.24c .333 1001.24c .333
NPS 1000000
```

Tungsten Blanket Neutron Multiplication Input Deck

```
c 233UBe13 300ppm U-232 Cylinder W Blanket
c cell cards
1     1 -4.2674 -1             $source proper
2     2 -7.874  -2  1           $ encapsulation
3     3 -19.3   -3  2           $blanket
4     0         -4  3           $  air
5     0          4              $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3
SI1 0 53I 13.5
SP1 0.00E+00 6.83E-04 4.53E-03 1.03E-02 1.31E-02 1.40E-02 1.30E-02 1.04E-02
    1.35E-02 1.59E-02 1.77E-02 2.10E-02 2.92E-02 3.52E-02 3.90E-02 4.02E-02
    3.88E-02 3.74E-02 3.64E-02 3.53E-02 3.36E-02 2.98E-02 2.55E-02 2.36E-02
    2.33E-02 2.38E-02 2.67E-02 2.76E-02 2.78E-02 2.81E-02 2.83E-02 2.76E-02
    2.66E-02 2.52E-02 2.38E-02 2.29E-02 2.22E-02 2.14E-02 1.99E-02 1.77E-02
    1.48E-02 1.20E-02 9.72E-03 7.60E-03 5.87E-03 4.56E-03 3.75E-03 3.12E-03
    2.54E-03 2.00E-03 1.49E-03 9.82E-04 4.87E-04 1.54E-04 1.11E-04
SI2 .5509
SI3 .5509
c
F1:N 3
E0 0 55I 14
FC1 number passing through 1.5 cm radius sphere
c
M1 4009 .928547421 92232.69c 2.142e-5 92233.69c 0.0711 90228.92C 5.37e-7
    82208.60c 3.0104e-4
M2 26056 1
M3 74000.21c 1
NPS 1000000
```

Lead Blanket Neutron Multiplication Input Deck

```
c 233UBe13 300ppm U-232 Cylinder Pb Blanket
c cell cards
1     1 -4.2674 -1              $source proper
2     2 -7.874  -2  1           $ encapsulation
3     3 -11.35  -3  2           $blanket
4     0         -4  3           $  air
5     0          4              $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0     0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3
SI1 0 53I 13.5
SP1 0.00E+00 6.83E-04 4.53E-03 1.03E-02 1.31E-02 1.40E-02 1.30E-02 1.04E-02
    1.35E-02 1.59E-02 1.77E-02 2.10E-02 2.92E-02 3.52E-02 3.90E-02 4.02E-02
    3.88E-02 3.74E-02 3.64E-02 3.53E-02 3.36E-02 2.98E-02 2.55E-02 2.36E-02
    2.33E-02 2.38E-02 2.67E-02 2.76E-02 2.78E-02 2.81E-02 2.83E-02 2.76E-02
    2.66E-02 2.52E-02 2.38E-02 2.29E-02 2.22E-02 2.14E-02 1.99E-02 1.77E-02
    1.48E-02 1.20E-02 9.72E-03 7.60E-03 5.87E-03 4.56E-03 3.75E-03 3.12E-03
    2.54E-03 2.00E-03 1.49E-03 9.82E-04 4.87E-04 1.54E-04 1.11E-04
SI2 .5509
SI3 .5509
c
F1:N 3
E0 0 55I 14
FC1 number passing through 1.5 cm radius sphere
c
M1 4009 .928547421 92232.69c 2.142e-5 92233.69c 0.0711 90228.92C 5.37e-7
    82208.60c 3.0104e-4
M2 26056 1
M3 82000.42c 1
NPS 1000000
```

Beryllium Blanket Photon Interactions Input Deck

```
c 233UBe13 300ppm U-232 Cylinder Be Blanket
c cell cards
1     1 -4.2674 -1              $source proper
2     2 -7.874  -2  1             $ encapsulation
3     3 -1.848  -3  2             $ blanket
4     0         -4  3           $air
5     0          4              $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3 PAR=2
SI1  L  0.0106 0.0728042 0.0749694 0.0849 0.2114 0.23336 0.25261 &
    0.277358 0.48595 0.51077 0.583191 0.5877 0.6501 0.7052 &
    0.72204 0.7487 0.76313 0.8212 0.860564 0.8833 0.9276 &
    0.9827 1.0939 1.1608 1.1852 1.2828 2.614533 0.0108 &
    0.07481 0.07711 0.0873 0.115183 0.17668 0.238632 0.300087 &
    0.4152 0.0103 0.0111 0.039857 0.0708319 0.0728715 0.076862 &
    0.07929 0.0826 0.0898 0.2882 0.32803 0.4337 0.45298 &
    0.473 0.72733 0.78537 0.893408 0.95212 1.0736 1.07862 &
    1.5127 1.6205 1.6797 1.806 0.54973 0.0117 0.08107 &
    0.08378 0.0949 0.240986 0.0123 0.084373 0.08847 0.131613 &
    0.16641 0.20593 0.215983 0.013 0.05778 0.12908
SP1 0.004459427 0.003106221 0.00524367 0.002321977 0.000273717 &
    0.000472084 0.001061036 &
    0.009703097 7.68867E-05 0.034752772 0.129938462 6.15093E-05 &
    5.53584E-05 3.38301E-05 &
    0.000309084 6.61225E-05 0.002783297 6.15093E-05 0.019098647 &
    4.76697E-05 0.000201443 &
    0.00031216 0.000615093 1.69151E-05 2.61415E-05 7.99621E-05 &
    0.152481632 0.064499368 &
    0.044423406 0.074750924 0.033360269 0.002528717 0.000222117 &
    0.184955142 0.014010459 &
    0.000610822 0.034171851 0.000252017 0.00452777 0.000246464 &
    0.000417751 0.000167442 &
    0.000280636 0.000184528 0.000126009 0.001439489 0.000533935 &
    7.26152E-05 0.001550548 &
```

```
     0.000213574 0.028106347 0.004707172 0.00161462 0.000726152 &
     6.83437E-05 0.002409115 &
     0.00123873 0.006364507 0.000247746 0.000384433 0.000486949 &
     0.001751307 0.000559564 &
     0.000926911 0.000422877 0.017513074 0.041433369 0.005211207 &
     5.04035E-05 0.000557428 &
     0.000442525 8.3721E-05 0.001084956 0.051257776 0.000853869 &
     0.000291315
SI2 .5509
SI3 .5509
c
PHYS:P J J J -1 J
Mode N P
F1:N 4
E1 0 55I 14
F11:P 4
E11 0 26I 2.7
FC1 number neutrons passing through 3.5 cm radius sphere
FC11 number gamma passing through 3.5 cm radius sphere
c
M1 4009.26u .928571429 92233.27u .07142648 82208.24u 3.52e-6
M2 26056.26u 1
M3 4009.26u 1
NPS 5e7
```

$Be_{12}W$ Blanket Photon Interactions Input Deck

```
c 233UBe13 300ppm U-232 Cylinder Be12W Blanket
c cell cards
1     1 -4.2674 -1            $source proper
2     2 -7.874  -2  1          $ encapsulation
3     3 -3.2    -3  2          $ blanket
4     0         -4  3          $air
5     0          4             $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3 PAR=2
SI1  L  0.0106 0.0728042 0.0749694 0.0849 0.2114 0.23336 0.25261 &
    0.277358 0.48595 0.51077 0.583191 0.5877 0.6501 0.7052 &
    0.72204 0.7487 0.76313 0.8212 0.860564 0.8833 0.9276 &
    0.9827 1.0939 1.1608 1.1852 1.2828 2.614533 0.0108 &
    0.07481 0.07711 0.0873 0.115183 0.17668 0.238632 0.300087 &
    0.4152 0.0103 0.0111 0.039857 0.0708319 0.0728715 0.076862 &
    0.07929 0.0826 0.0898 0.2882 0.32803 0.4337 0.45298 &
    0.473 0.72733 0.78537 0.893408 0.95212 1.0736 1.07862 &
    1.5127 1.6205 1.6797 1.806 0.54973 0.0117 0.08107 &
    0.08378 0.0949 0.240986 0.0123 0.084373 0.08847 0.131613 &
    0.16641 0.20593 0.215983 0.013 0.05778 0.12908
SP1 0.004459427 0.003106221 0.00524367 0.002321977 0.000273717 &
    0.000472084 0.001061036 &
    0.009703097 7.68867E-05 0.034752772 0.129938462 6.15093E-05 &
    5.53584E-05 3.38301E-05 &
    0.000309084 6.61225E-05 0.002783297 6.15093E-05 0.019098647 &
    4.76697E-05 0.000201443 &
    0.00031216 0.000615093 1.69151E-05 2.61415E-05 7.99621E-05 &
    0.152481632 0.064499368 &
    0.044423406 0.074750924 0.033360269 0.002528717 0.000222117 &
    0.184955142 0.014010459 &
    0.000610822 0.034171851 0.000252017 0.00452777 0.000246464 &
    0.000417751 0.000167442 &
    0.000280636 0.000184528 0.000126009 0.001439489 0.000533935 &
    7.26152E-05 0.001550548 &
```

```
    0.000213574 0.028106347 0.004707172 0.00161462 0.000726152 &
    6.83437E-05 0.002409115 &
    0.00123873 0.006364507 0.000247746 0.000384433 0.000486949 &
    0.001751307 0.000559564 &
    0.000926911 0.000422877 0.017513074 0.041433369 0.005211207 &
    5.04035E-05 0.000557428 &
    0.000442525 8.3721E-05 0.001084956 0.051257776 0.000853869 &
    0.000291315
SI2 .5509
SI3 .5509
c
PHYS:P J J J -1 J
Mode N P
F1:N 4
E1 0 55I 14
F11:P 4
E11 0 26I 2.7
FC1 number neutrons passing through 3.5 cm radius sphere
FC11 number gamma passing through 3.5 cm radius sphere
c
M1 4009.26u .928571429 92233.27u .07142648 82208.24u 3.52e-6
M2 26056.26u 1
M3 74182.26u .02115 74183.26u .01177 74184.26u
    .0235384 74186.26u .02185 4009.26u .9230769
NPS 5e7
```

BeHD Blanket Photon Interactions Input Deck

```
c 233UBe13 300ppm U-232 Cylinder BeHD Blanket
c cell cards
1     1 -4.2674 -1             $source proper
2     2 -7.874  -2  1           $ encapsulation
3     3 -.423   -3  2           $ blanket
4     0         -4  3           $air
5     0          4              $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3 PAR=2
SI1  L  0.0106 0.0728042 0.0749694 0.0849 0.2114 0.23336 0.25261 &
    0.277358 0.48595 0.51077 0.583191 0.5877 0.6501 0.7052 &
    0.72204 0.7487 0.76313 0.8212 0.860564 0.8833 0.9276 &
    0.9827 1.0939 1.1608 1.1852 1.2828 2.614533 0.0108 &
    0.07481 0.07711 0.0873 0.115183 0.17668 0.238632 0.300087 &
    0.4152 0.0103 0.0111 0.039857 0.0708319 0.0728715 0.076862 &
    0.07929 0.0826 0.0898 0.2882 0.32803 0.4337 0.45298 &
    0.473 0.72733 0.78537 0.893408 0.95212 1.0736 1.07862 &
    1.5127 1.6205 1.6797 1.806 0.54973 0.0117 0.08107 &
    0.08378 0.0949 0.240986 0.0123 0.084373 0.08847 0.131613 &
    0.16641 0.20593 0.215983 0.013 0.05778 0.12908
SP1 0.004459427 0.003106221 0.00524367 0.002321977 0.000273717 &
    0.000472084 0.001061036 &
    0.009703097 7.68867E-05 0.034752772 0.129938462 6.15093E-05 &
    5.53584E-05 3.38301E-05 &
    0.000309084 6.61225E-05 0.002783297 6.15093E-05 0.019098647 &
    4.76697E-05 0.000201443 &
    0.00031216 0.000615093 1.69151E-05 2.61415E-05 7.99621E-05 &
    0.152481632 0.064499368 &
    0.044423406 0.074750924 0.033360269 0.002528717 0.000222117 &
    0.184955142 0.014010459 &
    0.000610822 0.034171851 0.000252017 0.00452777 0.000246464 &
    0.000417751 0.000167442 &
    0.000280636 0.000184528 0.000126009 0.001439489 0.000533935 &
    7.26152E-05 0.001550548 &
```

```
     0.000213574 0.028106347 0.004707172 0.00161462 0.000726152 &
     6.83437E-05 0.002409115 &
     0.00123873 0.006364507 0.000247746 0.000384433 0.000486949 &
     0.001751307 0.000559564 &
     0.000926911 0.000422877 0.017513074 0.041433369 0.005211207 &
     5.04035E-05 0.000557428 &
     0.000442525 8.3721E-05 0.001084956 0.051257776 0.000853869 &
     0.000291315
SI2 .5509
SI3 .5509
c
PHYS:P J J J -1 J
Mode N P
F1:N 4
E1 0 55I 14
F11:P 4
E11 0 26I 2.7
FC1 number neutrons passing through 3.5 cm radius sphere
FC11 number gamma passing through 3.5 cm radius sphere
c
M1 4009.26u .928571429 92233.27u .07142648 82208.24u 3.52e-6
M2 26056.26u 1
M3 4009.26u .5 1002.24u .5
NPS 5e7
```

Tungsten Blanket Photon Interactions Input Deck

```
c 233UBe13 300ppm U-232 Cylinder W Blanket
c cell cards
1     1 -4.2674 -1             $source proper
2     2 -7.874  -2  1           $ encapsulation
3     3 -19.3   -3  2           $ blanket
4     0         -4  3           $air
5     0          4              $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3 PAR=2
SI1  L  0.0106 0.0728042 0.0749694 0.0849 0.2114 0.23336 0.25261 &
    0.277358 0.48595 0.51077 0.583191 0.5877 0.6501 0.7052 &
    0.72204 0.7487 0.76313 0.8212 0.860564 0.8833 0.9276 &
    0.9827 1.0939 1.1608 1.1852 1.2828 2.614533 0.0108 &
    0.07481 0.07711 0.0873 0.115183 0.17668 0.238632 0.300087 &
    0.4152 0.0103 0.0111 0.039857 0.0708319 0.0728715 0.076862 &
    0.07929 0.0826 0.0898 0.2882 0.32803 0.4337 0.45298 &
    0.473 0.72733 0.78537 0.893408 0.95212 1.0736 1.07862 &
    1.5127 1.6205 1.6797 1.806 0.54973 0.0117 0.08107 &
    0.08378 0.0949 0.240986 0.0123 0.084373 0.08847 0.131613 &
    0.16641 0.20593 0.215983 0.013 0.05778 0.12908
SP1 0.004459427 0.003106221 0.00524367 0.002321977 0.000273717 &
    0.000472084 0.001061036 &
    0.009703097 7.68867E-05 0.034752772 0.129938462 6.15093E-05 &
    5.53584E-05 3.38301E-05 &
    0.000309084 6.61225E-05 0.002783297 6.15093E-05 0.019098647 &
    4.76697E-05 0.000201443 &
    0.00031216 0.000615093 1.69151E-05 2.61415E-05 7.99621E-05 &
    0.152481632 0.064499368 &
    0.044423406 0.074750924 0.033360269 0.002528717 0.000222117 &
    0.184955142 0.014010459 &
    0.000610822 0.034171851 0.000252017 0.00452777 0.000246464 &
    0.000417751 0.000167442 &
    0.000280636 0.000184528 0.000126009 0.001439489 0.000533935 &
    7.26152E-05 0.001550548 &
```

```
    0.000213574 0.028106347 0.004707172 0.00161462 0.000726152 &
    6.83437E-05 0.002409115 &
    0.00123873 0.006364507 0.000247746 0.000384433 0.000486949 &
    0.001751307 0.000559564 &
    0.000926911 0.000422877 0.017513074 0.041433369 0.005211207 &
    5.04035E-05 0.000557428 &
    0.000442525 8.3721E-05 0.001084956 0.051257776 0.000853869 &
    0.000291315
SI2 .5509
SI3 .5509
c
PHYS:P J J J -1 J
Mode N P
F1:N 4
E1 0 55I 14
F11:P 4
E11 0 26I 2.7
FC1 number neutrons passing through 3.5 cm radius sphere
FC11 number gamma passing through 3.5 cm radius sphere
c
M1 4009.26u .928571429 92233.27u .07142648 82208.24u 3.52e-6
M2 26056.26u 1
M3 74182.26u .275 74183.26u .153 74184.26u
    .306 74186.26u .284
NPS 5e7
```

Lead Blanket Photon Interactions Input Deck

```
c 233UBe13 300ppm U-232 Cylinder Pb Blanket
c cell cards
1     1 -4.2674 -1              $source proper
2     2 -7.874  -2  1           $ encapsulation
3     3 -11.35  -3  2           $ blanket
4     0         -4  3           $air
5     0          4              $outside universe
c end of cell card

c Beginning of Surfaces for UBe cylinder
1     RCC  0.  0.  -0.5509  0.  0.  1.1018   0.5509
2     RCC  0.  0.  -1.0    0.  0.  2.000  1.0
3     SO 3.0
4     SO 3.5
5     SO 30
c end of cube surfaces

IMP:N 1 1 1 1 0
SDEF pos=0 0 .5509 RAD=D2 CEL=1 ERG=D1 EXT=D3 PAR=2
SI1  L  0.0106 0.0728042 0.0749694 0.0849 0.2114 0.23336 0.25261 &
    0.277358 0.48595 0.51077 0.583191 0.5877 0.6501 0.7052 &
    0.72204 0.7487 0.76313 0.8212 0.860564 0.8833 0.9276 &
    0.9827 1.0939 1.1608 1.1852 1.2828 2.614533 0.0108 &
    0.07481 0.07711 0.0873 0.115183 0.17668 0.238632 0.300087 &
    0.4152 0.0103 0.0111 0.039857 0.0708319 0.0728715 0.076862 &
    0.07929 0.0826 0.0898 0.2882 0.32803 0.4337 0.45298 &
    0.473 0.72733 0.78537 0.893408 0.95212 1.0736 1.07862 &
    1.5127 1.6205 1.6797 1.806 0.54973 0.0117 0.08107 &
    0.08378 0.0949 0.240986 0.0123 0.084373 0.08847 0.131613 &
    0.16641 0.20593 0.215983 0.013 0.05778 0.12908
SP1 0.004459427 0.003106221 0.00524367 0.002321977 0.000273717 &
    0.000472084 0.001061036 &
    0.009703097 7.68867E-05 0.034752772 0.129938462 6.15093E-05 &
    5.53584E-05 3.38301E-05 &
    0.000309084 6.61225E-05 0.002783297 6.15093E-05 0.019098647 &
    4.76697E-05 0.000201443 &
    0.00031216 0.000615093 1.69151E-05 2.61415E-05 7.99621E-05 &
    0.152481632 0.064499368 &
    0.044423406 0.074750924 0.033360269 0.002528717 0.000222117 &
    0.184955142 0.014010459 &
    0.000610822 0.034171851 0.000252017 0.00452777 0.000246464 &
    0.000417751 0.000167442 &
    0.000280636 0.000184528 0.000126009 0.001439489 0.000533935 &
    7.26152E-05 0.001550548 &
```

```
     0.000213574 0.028106347 0.004707172 0.00161462 0.000726152 &
     6.83437E-05 0.002409115 &
     0.00123873 0.006364507 0.000247746 0.000384433 0.000486949 &
     0.001751307 0.000559564 &
     0.000926911 0.000422877 0.017513074 0.041433369 0.005211207 &
     5.04035E-05 0.000557428 &
     0.000442525 8.3721E-05 0.001084956 0.051257776 0.000853869 &
     0.000291315
SI2 .5509
SI3 .5509
c
PHYS:P J J J -1 J
Mode N P
F1:N 4
E1 0 55I 14
F11:P 4
E11 0 26I 2.7
FC1 number neutrons passing through 3.5 cm radius sphere
FC11 number gamma passing through 3.5 cm radius sphere
c
M1 4009.26u .928571429 92233.27u .07142648 82208.24u 3.52e-6
M2 26056.26u 1
M3 82206.24u .241 82207.24u .221 82208.24u .524
NPS 5e7
```

APPENDIX A.3

MCNP5 Criticality Input Decks

Critical Radius of ^{232}U Input Deck

```
c 232U Critical Radius
1     1 -18.47298 -1 imp:n=1
2     0 1           imp:N=0

1     SO 3.31

M1 92232 1
KSRC 0 0 0
KCODE 1e5 10 50
PRINT
```

Critical Radius of $^{232}UBe_{13}$ Input Deck

```
c 232UBe13 Critical Radius
1     1 -4.2674 -1 imp:n=1
2     0 1          imp:N=0

1     SO 10.774

M1 92232 0.0714285714 4009 0.9285714286
KSRC 0 0 0
KCODE 1e5 10 50
PRINT
```

Critical Radius of $^{dirty}UBe_{13}$ Input Deck

```
c dirtyUBe13 Critical Radius
1     1 -4.2674 -1 imp:n=1
2     0 1          imp:N=0

1     SO 11.355

M1 92232 2.1429e-5 92233 0.071407143 4009 0.9285714286
KSRC 0 0 0
KCODE 1e5 10 50
PRINT
```

APPENDIX B

CHANGES TO SOURCES-4c

Changes to SOURCES-4c Source Code

The lines below were changed in the code. These lines were changed to allow the code to evaluate alpha particle energies up to 10.0 MeV. The previous limit was 6.5 MeV.

Line #	Replaced Code
989	if (eamax.le.10.0) go to 860
2162	if(eala(mm).gt.10.0)eala(mm)=10.0
2595	if (eamax.le.10.0) go to 860
3857	if(eala(mm,k).gt.10.0)eala(mm,k)=10.0
3921	if(eala(mm,k).gt.10.0)eala(mm,k)=10.0
4357	if (eamax.le.10.0) go to 860

Changes to tape3

The following lines replaced the default lines in tape3. These lines added an additional data point to the data files for beryllium. These new data points allowed the code to evaluate alpha energies up to 10.0 MeV.

Line #	Replaced Text
59	00040090 130 be(nat)ge76 131 points ge76 tab.4, be(alpha,n) xec-sec,adj*.885
92	7.8500e+006.0985e+027.9000e+006.0941e+0210.000e+006.0941e+02
93	00040090 130 be(nat)ge76 131 points ge76 tab.4, be(alpha,n) xec-sec
126	7.8500e+006.8910e+027.9000e+006.8860e+0210.000e+006.8860e+02

Changes to tape4

The following lines were added or replaced existing lines in tape4. These lines added an additional data point to the data files for beryllium. These new data points allowed the code to evaluate alpha energies up to 10.0 MeV.

The following line replaced the default line in tape4.

Line #	Replaced Text
33	00040090 5131 Geiger and Van der Zwan *

A new line was added between 165 and 166 of this code.

Add New Line	Added Text
166	10.0 .054 .184 .055 .076 .631

Changes to tape5

The following lines replaced the default lines in tape5. These lines were changed to reflect updated decay data.

This line was changed to allow the code to utilize these very energy alpha particles, with the given alpha energy limit.

Line #	Replaced Text
2328	10.000 2.3714e-05 10.000 1.0420e-05 10.000 1.1030e-04

This line was changed to match the current half life of ^{232}U (Tuli, 2005).

Line #	Replaced Text
2476	3.19007e-10 0.00000e+0 1.71000e+00 8.92204e-01 3.72278e+00

APPENDIX C

PHYSICAL CONSTANTS

Properties of Uranium-232

Table 8: Properties of ^{232}U		
Property	Value	Source
Mass	232.03713	Chart of the Nuclides, 2002
Density	18.52 g/cc	Tuli, 2005
Melting Point	1408 K	Tuli, 2005
Boiling Point	4404 K	Tuli, 2005
Oxidation	+3, +4, +5, +6	Tuli, 2005
Spin	0	Tuli, 2005
Half-Life	68.9 yr	Tuli, 2005
$t_{1/2\ SF}$	$2.55*10^{15}$ yr	Calculated[1] from Bonetti et al. 2000
$t_{1/2\ Ne}$	$2.746*10^{12}$ yr	Calculated[1] from Bonetti et al. 1990
λ	$3.1878*10^{-10}$ 1/s	Calculated[1] from Tuli
λ_{SF}	$8.6*10^{-24}$ 1/s	Bonetti et al. 2000
λ_{Ne}	$8.0*10^{-22}$ 1/s	Bonetti et al. 1990
α Emissions	5.139 MeV (.3%) 5.26336 (31.55%) 5.32012 (68.15%)	Stabin and de Luz, 2002
Critical Radius	3.31 cm	Calculation with MCNP5

[1] Calculated with equation (37)

$$\lambda = \frac{\ln(2)}{t_{1/2}} \tag{37}$$

Properties of UBe_{13}

Table 9: UBe_{13} Properties		
Property	Value	Source
Melting Point	2250 K	Predel and Maeling, 1998
Mass		
$^{nat}UBe_{13}$	355.187	Calculated[1]
$^{232}UBe_{13}$	349.1954	Calculated[1]
$^{dirty}UBe_{13}$	350.1976	Calculated[1]
Density		
$^{nat}UBe_{13}$	4.3590	McElfresh et al. 1990
$^{232}UBe_{13}$	4.2852	Calculated[1] from McElfresh et al.
$^{dirty}UBe_{13}$	4.29750486	Calculated[1,2] from McElfresh et al.
Critical Radius		
$^{232}UBe_{13}$	10.774	Calculated with MCNP5
$^{dirty}UBe_{13}$	11.355	Calculated with MCNP5

[1] Isotopic mass data from Chart of the Nuclides, 2002

[2] $^{dirty}U$ defined as ^{233}U with 300 ppm ^{232}U

Densities of Selected Nuclear Materials

Table 10: Densities of Nuclear Materials		
Compound	Density (g/cc)	Source
BeH_2	0.63	Smith et al. 1988
BeD_2	0.75	Calculated[1] from Smith et al.
BeHD	0.69	Calculated[1] from Smith et al.
$Be_{12}W$	3.2	Yamada et al. 2003
$^{227}AcBe_{13}$	4.2237	Calculated[1] from McElfresh et al.
$^{227}ThBe_{13}$	4.2237	Calculated[1] from McElfresh et al.
$^{228}ThBe_{13}$	4.2360	Calculated[1] from McElfresh et al.
$^{nat}UBe13$	4.3590	McElfresh et al. 1990
$^{232}UBe_{13}$	4.2852	Calculated[1] from McElfresh et al.
$^{dirty}UBe_{13}$	4.29750486	Calculated[1,2] from McElfresh et al.
$^{233}UBe_{13}$	4.29750479	Calculated[1] from McElfresh et al.
$^{238}PuBe_{13}$	4.3590	Calculated[1] from McElfresh et al.
$^{239}PuBe_{13}$	4.3713	Calculated[1] from McElfresh et al.
$^{241}AmBe_{13}$	4.3959	Calculated[1] from McElfresh et al.
$^{242}CmBe_{13}$	4.4085	Calculated[1] from McElfresh et al.
$^{244}CmBe_{13}$	4.4330	Calculated[1] from McElfresh et al.
$^{nat}UO_2$	10.96	Alfa Aesar 2004 a
$^{dirty}UO_2$	10.72	Calculated[1,2] from Alfa Aesar 2004 a
$RaCO_3$	8.3735	Calculated from Alfa Aesar 2004 b
NaF	2.558	Alfa Aesar 2004 c
CCl_4	1.5900	Fisher Scientific 2004
MnO_2	5.445	Alfa Aesar 2004 d
GaO_3	6.44	Alfa Aesar 2004 e
As_2O_3	3.74	Alfa Aesar 2004 f
Y_2O_3	5.01	Alfa Aesar 2003
La_2O_3	6.51	Strem Chemicals 2001
Pr_2O_3	7.07	Alfa Aesar 2004 g

[1] Isotopic mass data from Chart of the Nuclides, 2002, calculated with equation (38)
[2] $^{dirty}U$ defined as ^{233}U with 300 ppm ^{232}U

$$\rho' = \rho * \frac{mml}{mml'} \tag{38}$$

Where ρ is the known density of a compound with mass *mml*, ρ' is the theoretical density of a similar compound with different isotopic (or elemental) composition and mass *mml'*.

Neutron Spectrum of Various Target Materials

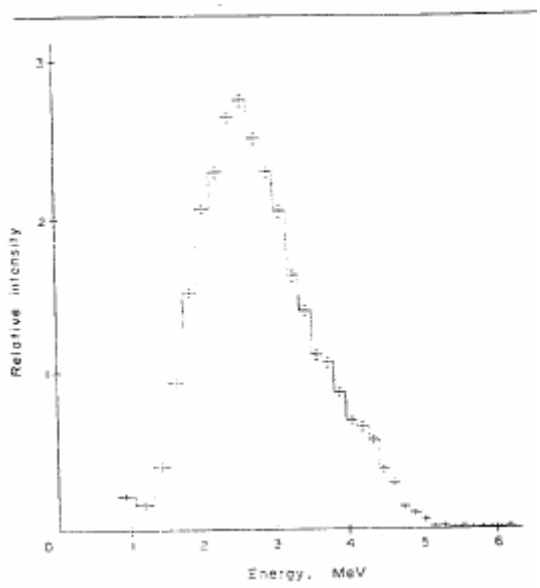

Figure 16: ^{241}Am-B neutron spectrum, Source: Lorch, 1973

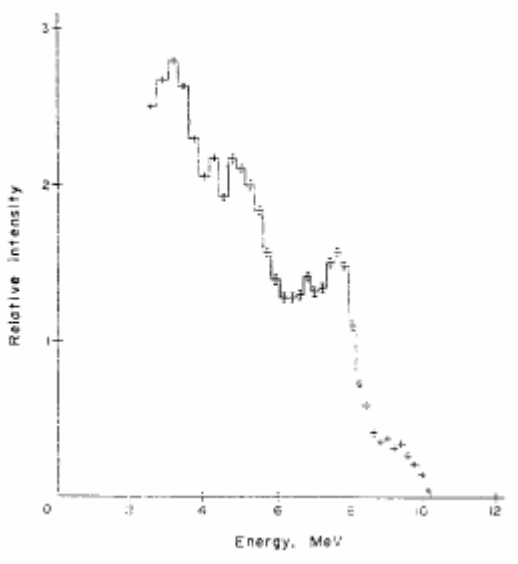

Figure 17: ^{241}Am-Be neutron spectrum, Source: Lorch, 1973

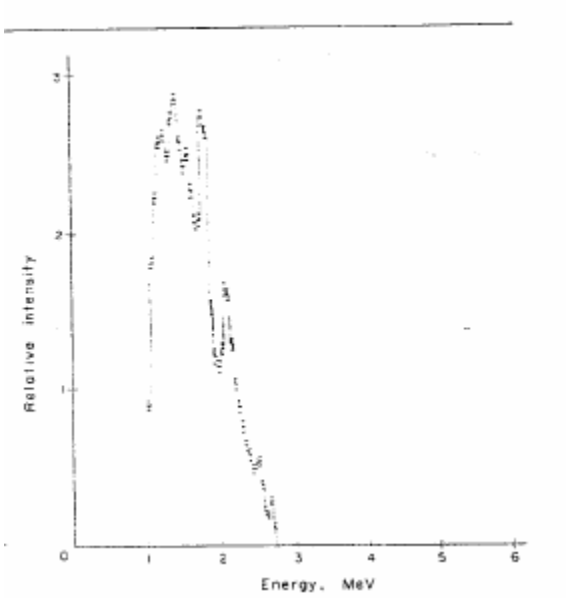

Figure 18: ^{241}Am-F neutron spectrum, Source: Lorch, 1973

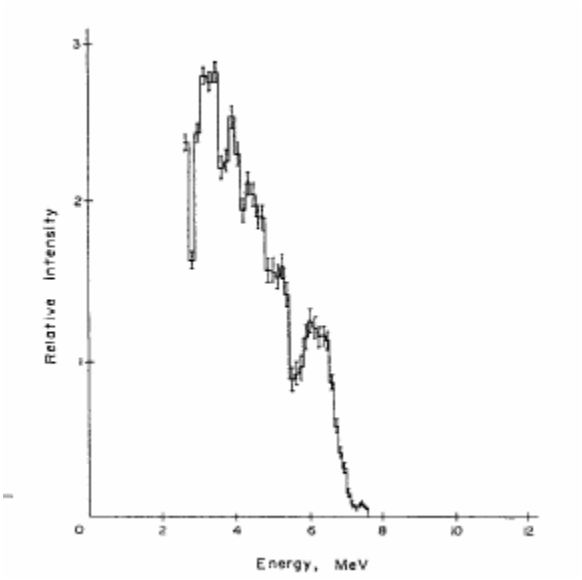

Figure 19: ^{239}Pu-^{13}C neutron spectrum, Source: Lorch, 1973

^{232}U Power Curve

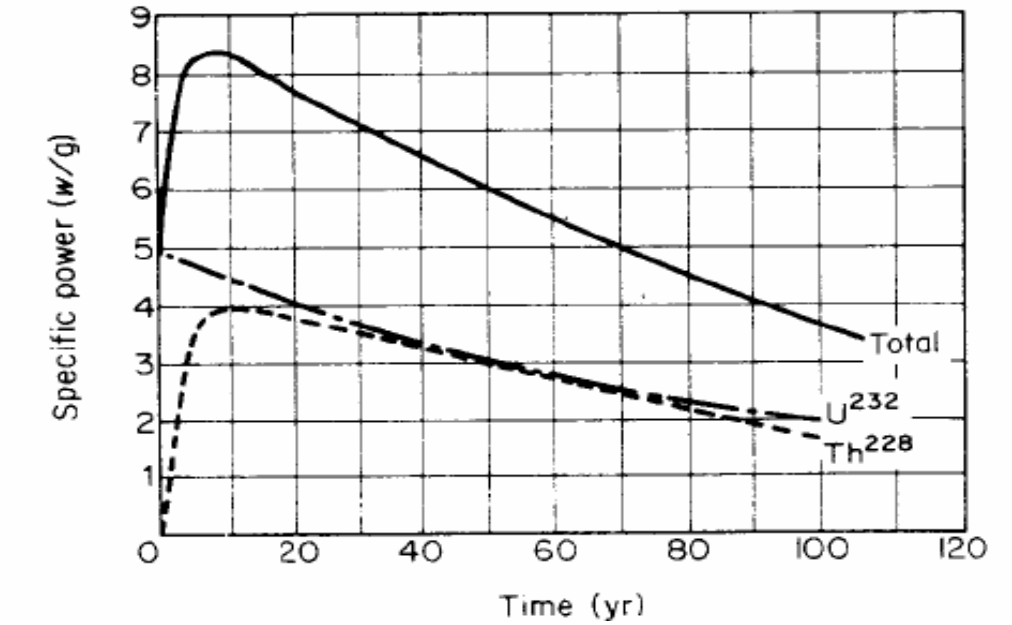

Figure 20: ^{232}U power curve, source: Corliss and Harvey, 1964

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Alfa Aesar (2004) c. Sodium Fluoride. MSDS 12964. Ward Hill, MA.

Alfa Aesar (2004) d. Manganese (II) Oxide. MSDS 11870. Ward Hill, MA.

Alfa Aesar (2004) e. Gallium (III) Oxide. MSDS 10508. Ward Hill, MA.

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