Nuclear Training Course 427
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Nuclear Theory

NUCLEAR TRAINING DEPARTMENT

COURSE 427

NUCLEAR THEORY

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INDEX.427

Nuclear Training Course

COURSE 427

4 - Level
2 - Science Fundamentals
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INDEX

(Mechanical Maintainers do not require the following modules.)

ATOMIC STRUCTURE

OBJECTIVES

At the conclusion of this lesson the trainee will be able to:

1. Name the fundamental atomic particles and state their mass and electric charge.
2. Describe the Bohr model picture of the atom.
3. Understand and use the ${}^{A}_{Z}X$ notation for nuclides.
4. Define and describe isotopes.

ATOMIC STRUCTURE

For nearly 2500 years it has been believed that the vast array of substances which surround us are made of tiny particles called atoms. In the early nineteenth century John Dalton, an English Chemist, formalized this belief in his Atomic Theory which contains six basic points:

1. All matter is composed of particles called atoms;

2. These particles are far too small to be observed with the naked eye;

3. Different elements are made of atoms with different masses;

4. All atoms of the same element are identical;

5. Atoms combine in simple ratios to form new substances but the atoms themselves remain unchanged;

6. Atoms cannot be divided, created, or destroyed.

Much of this theory has proved correct. We now know that atoms are also made of more fundamental particles; the proton, the neutron and the electron.

FUNDAMENTAL PARTICLES

Proton

A proton is a very small particle. Its diameter is one hundred thousandth (1/100,000 = 10^{-5}) of the diameter of a hydrogen atom. A hydrogen atom's diameter is between one and two billionths of a meter (about 1.6×10^{-9}m).

The proton carries a single unit positive charge (+1e) and has a mass of about one mass unit (1u). This mass unit is very small; 1u = 1.66×10^{-27}kg. The proton mass is 1.0073u and most of the time we round this off to 1u.

Neutron

A neutron is a neutral (uncharged) particle the same size as the proton. It has a mass of 1.0087u, about 2.5 electron masses heavier than the proton. We generally approximate its mass as 1u.

Electron

An electron is the smallest of the three fundamental particles, having a mass of only 0.00054u, about 1/1840 of the mass of a nucleon. (Nucleon is a name for either of the two heavier fundamental particles).

The electron carries a unit negative charge (-1e). The unit charge is so small it takes a flow of 6.24 trillion charges a second to measure one microampere.

We shall now see how these fundamental particles are put together to form atoms.

STRUCTURE OF ATOMS

Niels Bohr, a Danish Physicist, received the Nobel Prize in 1922 for his theory about the structure of atoms. The picture which the Bohr model presents is of a positively charged nucleus, containing protons and neutrons, surrounded by negatively charged electrons in orbit. The atom, according to this model, looks like a very small solar system.

A neutral (i.e., uncharged) atom must have the same number of electrons in orbit as there are protons in the nucleus. The first three elements, hydrogen, helium and lithium are shown in Figure 1.1.

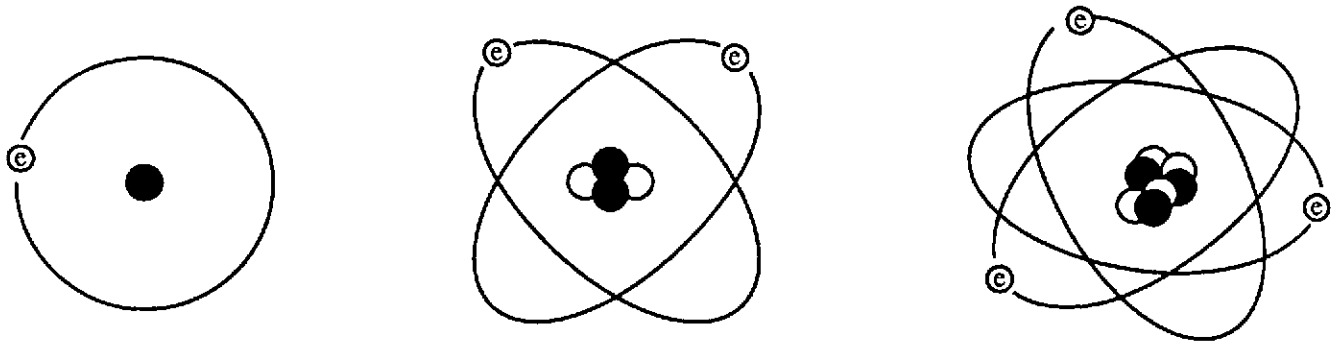


Figure 1.1

Over 110 elements are known, each having a characteristic number of protons in the nucleus. Ninety of these elements exist naturally in the world around us and most of these (81) are made of stable atoms. Unstable atoms (i.e., radioactive atoms) also occur naturally, and the "manmade" atoms are unstable.

ATOMIC NOTATION

Each element can be identified and represented by its chemical symbol, its atomic number (number of protons), and its atomic mass number (equal to the number of nucleons) as follows:

$^{A}_{Z}X$ Where: Z = Atomic Number
X = Chemical Symbol
A = Atomic Mass Number

For example, the three elements of Figure 1.1 are:

Hydrogen $^{1}_{1}H$ (1 proton)

Helium $^{4}_{2}He$ (2 protons, 2 neutrons)

Lithium $^{7}_{3}Li$ (3 protons, 4 neutrons)

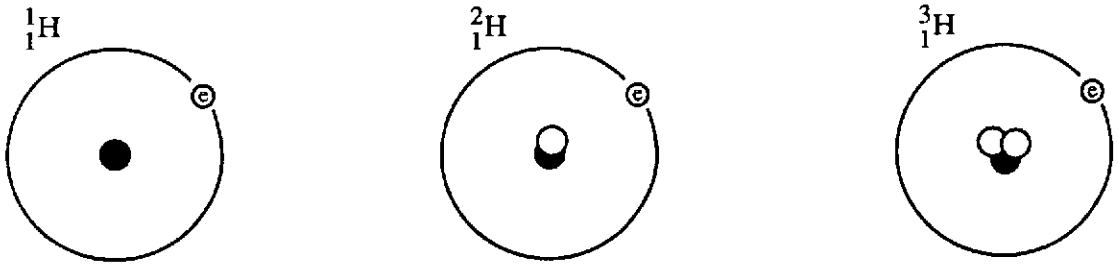
Since the number of protons uniquely determines the chemical symbol we often write these in a simpler way; for example, $^{4}_{2}He$ becomes He-4 or helium-4.

When the numbers of nucleons in the nucleus of an atom are shown (i.e., both Z and A) then the atom is sometimes called a nuclide.

ISOTOPES

The lithium atom in Figure 1.1 has 3 protons and 4 neutrons in its nucleus. Only 92.5% of naturally occurring lithium atoms are like this. The other 7.5% of lithium atoms have three protons and three neutrons. We call these different kinds of lithium isotopes of lithium. Isotopes of an element are atoms which have the same number of protons with a varying number of neutrons. All isotopes of a given element have similar chemical and physical properties but show very large variations in nuclear properties.

The isotopes of two elements (hydrogen and uranium) are of particular significance in this course. Hydrogen has three isotopes shown in Figure 1.2. The first two occur naturally, although deuterium is only 0.015% abundant (about one atom in every 7000). Unfortunately we need deuterium in the form of heavy water (D_2O) to make the CANDU reactor work. This requires an expensive separation process (discussed in detail in another course). The third isotope, tritium, is produced in our reactors; it is radioactive and can be a serious health hazard.



Hydrogen

Deuterium
or
Heavy Hydrogen

Tritium
or
Radioactive
Heavy Hydrogen

Figure 1.2

Uranium used for CANDU fuel has two isotopes, $^{238}_{92}U$ and $^{235}_{92}U$. U-235 is 0.7% abundant and will fission (split, releasing energy) when struck by a low energy (slow speed) neutron. It is said to be fissile. U-235 is the only naturally occurring fissile material. U-238 is 99.3% abundant and is not fissile. Nevertheless, it strongly affects the behavior of nuclear fuel, as we will see later.

ASSIGNMENT

1. List the mass, charge, and location of each of the fundamental particles.

2. Define an isotope.

J.E. Crist

RADIOACTIVITY - SPONTANEOUS NUCLEAR PROCESSES

OBJECTIVES

At the conclusion of this lesson the trainee will be able to:

1. For α, β and γ decays
   a) Write a typical equation for the production of each type of radiation.
   b) List the physical properties of each type of radiation.
   c) Discuss how each type of radiation interacts with matter.
2. State how to shield against alphas and betas.
3. State how to shield against γ rays and calculate γ ray shielding in terms of $\frac{1}{2}$ value layers.

RADIOACTIVITY - SPONTANEOUS NUCLEAR PROCESSES

The property we know as radioactivity was first observed in 1896 by Becquerel. He was carrying out experiments with fluorescent salts (which contained uranium) and found that some photographic plates had been exposed despite being well wrapped against light.

Later research showed that the "rays" that he had observed were of three distinct types. We now call these alpha particles (α), beta particles (β) and gamma rays (γ). We also know of several other types of rays or emissions but these three are the commonest.

TYPES OF EMISSIONS

All natural nuclides of atomic number greater than 82 are unstable (i.e., radioactive) and eventually decay (or disintegrate) by emitting an alpha particle or a beta particle. The new nuclides formed (daughter nuclides) also decay until a stable nuclide of atomic number 82 or less is formed. Several naturally occurring radioactive nuclides with lower mass number are also known and many other manmade radioactive nuclides have been found.

Alpha Emissions

The alpha particle is emitted, typically, from a heavy nuclide such as U238. This is expressed as:

$$^{238}_{92}U \longrightarrow \alpha + {}^{234}_{90}Th$$

Examination of the alpha particle shows it is a helium-4 nucleus so you can write:

$$^{238}_{92}U \longrightarrow {}^{4}_{2}He + {}^{234}_{90}Th$$

(parent) (α) (daughter)

OR $$^{A}_{Z}X \longrightarrow \alpha + {}^{A-4}_{Z-2}X$$

These equations represent a parent nucleus emitting a fast moving helium-4 nucleus (α particle) and resulting in a new daughter nucleus.

The alpha particle does not have any electrons (remember it is a helium nucleus) and therefore will have a charge of +2e, (usually given simply as +2). The mass of the alpha is 4.0015u and its speed when first emitted is typically a few percent of the speed of light.

Beta Emissions

Beta particles are emitted by neutron-rich nuclides, i.e. a nuclide with too many neutrons. This is a typical example:

$$^{90}_{38}Sr \longrightarrow \beta + ^{90}_{39}Y$$

You may put the mass number and charge number onto the β symbol if desired, giving:

$$^{90}_{38}Sr \longrightarrow {}^{0}_{-1}\beta + ^{90}_{39}Y$$

OR $$^{A}_{Z}X \longrightarrow \beta + {}_{Z+1}^{A}X$$

As noted from the above expressions the daughter nuclide from beta decay appears one position higher in the table of the elements. A neutron in the nucleus has changed into a proton so the atomic number goes up one.

The beta particle <u>is</u> a fast moving electron. It has the same mass as any other electron, 0.00054u, and it has the same charge, -1. The speeds of beta particles range from about 90 to 99% of the speed of light. They are very fast!

Gamma Emissions

After an alpha or beta emission the residual nucleus will usually be in an excited state. Excited states must not be confused with the concept of unstable nuclides. Both stable and unstable nuclides can be in an excited state. The mode of de-excitation could be by emission of a suitable particle (α, β, neutron, proton) but in most cases the de-excitation takes place by the emission of one or more gamma photons. The name photon is used to emphasize that gamma radiation has particle-like properties. A typical example is written:

$$^{60}_{27}Co \longrightarrow \beta + ^{60}_{28}Ni^{*} \ (\beta \text{ emission})$$

Then $$^{60}_{28}Ni^{*} \longrightarrow \gamma + ^{60}_{28}Ni \ (\gamma \text{ emission})$$

The cobalt-60 emits a beta leaving the daughter nickel-60 nucleus in an excited state (indicated by the asterisk). Almost immediately the excited nickel-60 emits γ-rays until it is de-excited. The duration of the excited state is very short, usually less than 10^{-9}s so we usually write the beta and gamma decays as though they are a single event.

$$^{60}_{27}\text{Co} \longrightarrow \beta + \gamma + {}^{60}_{28}\text{Ni}$$

The generalized gamma decay can be written:

$$^{A}_{Z}\text{X}^{*} \longrightarrow \gamma + {}^{A}_{Z}\text{X}$$

As you can see there is no change in Z or A because the gamma ray has no charge and no mass so it cannot affect the charge and mass of the nuclide.

Gamma rays are electro-magnetic radiations like light rays, radio waves and x-rays. A gamma photon has more energy than most x-ray photons which in turn have more energy than ultra violet photons and so on, down to the longest wave length radio waves. Figure 2.1 shows the electromagnetic spectrum. Note that long waves imply low frequencies, low photon energies, and wave like properties. High energy γ-rays are more particle-like in their interactions. The gamma ray speed is the same as that of light in a vacuum.

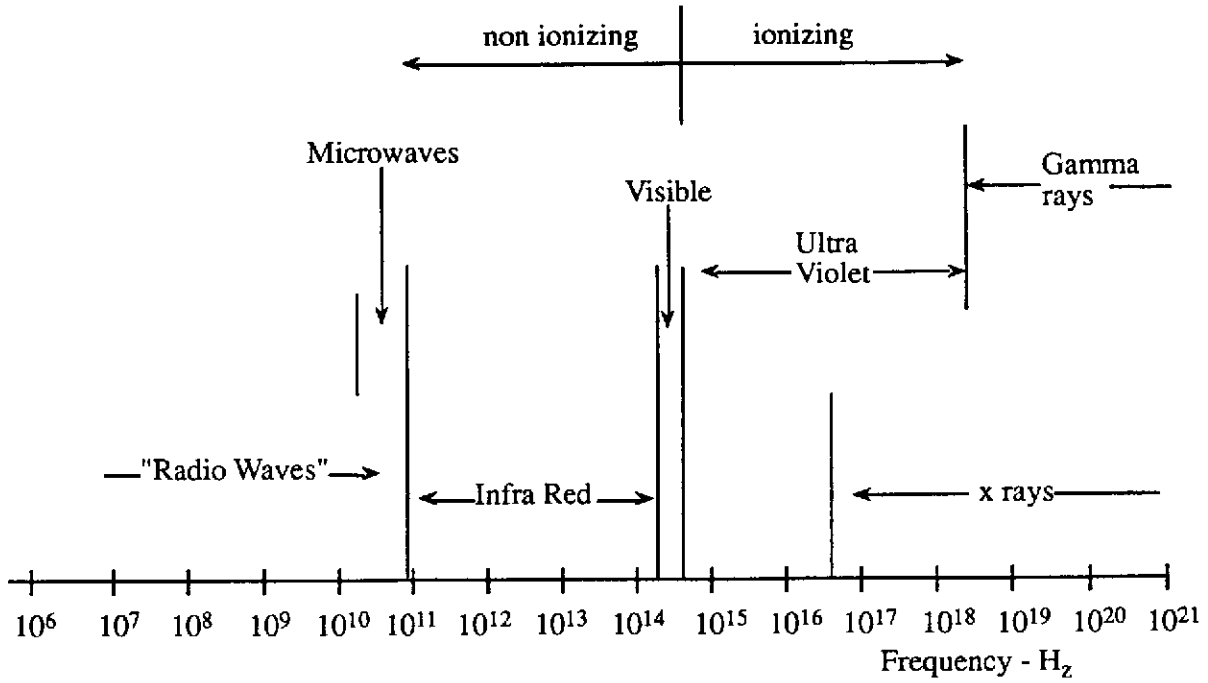


Figure 2.1: Electro Magnetic Spectrum

INTERACTION OF PARTICLES OR RAYS WITH MATTER

Alpha and beta particles are classed as ionizing particles. This is because they carry electric charge which causes the atoms they approach to separate into ions. Each separation creates an ion-pair.

Ionization by Alpha Particles

Alpha particles with their charge of +2 and their mass of 4u create intense ionization. In dry air the alpha causes about 50 000 ion-pairs per centimeter of its path, giving up about 33 eV per pair produced. A 4 MeV alpha travels about 2.5 cm before all its energy is used up. It slows down and stops and becomes a normal helium atom by adopting two electrons from neighbouring atoms.

In liquids or solids the ion-pairs per centimeter is much greater and the distance travelled by the alpha is much less. In general the range (straight line distance) of alpha particles in solid materials is less than 0.1 mm (about the thickness of a sheet of paper).

Ionization by Beta Particles

Beta particles have a charge of -1, a mass of 0.0005u and are travelling very fast (90-99%c). They cause less intense ionization than alpha particles, typically 100 - 300 ion-pairs per centimeter of path in dry air. Because of their small mass the beta particles are deflected easily and do not travel in a straight line. In air their total length of path would be typically 20 m. Beta particles are more penetrating than alphas and will penetrate a sheet of paper. Generally a mm or so of a dense material will be sufficient to stop them.

Gamma Ray Interaction with Atoms

Gamma rays behave differently from alpha and beta particles. First, they have no charge and no mass. Secondly, they do not lose their energies in small, scattered amounts, but give it away in larger chunks when they undergo a reaction. Three of the possible reactions between gamma rays and atoms are described below.

1. The photoelectric effect.

   This gamma ray interaction can take place for gamma rays of low energy. An incident gamma ray reacts with an electron in an atomic orbit. The gamma photon gives all of its energy to an orbiting electron. The gamma ray ceases to exist and the electron is ejected from the atom and behaves like a beta particle. In many materials, the photoelectric effect is not important for photon energies above 0.1 MeV.

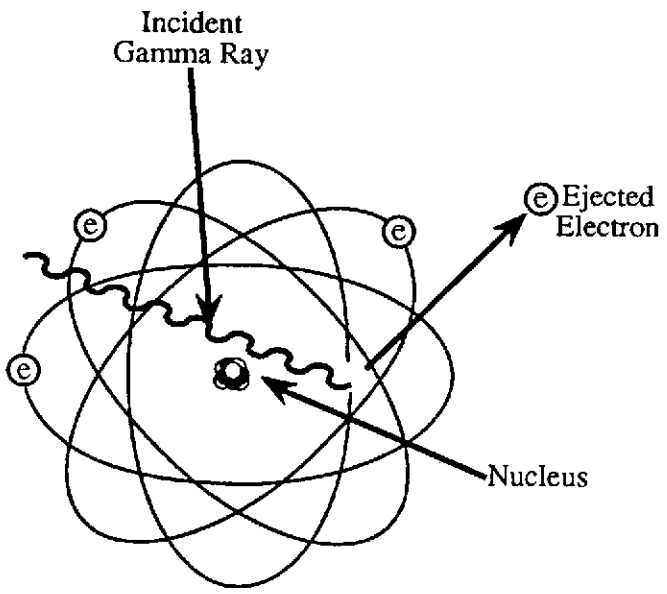


Figure 2.2: Photo Electric Effect

2. The Compton Effect

This gamma ray interaction occurs mainly for gamma photons with energies from about 0.1 to 10 MeV. The incident gamma ray is "scattered" by hitting an electron. The electron in turn is given some of the gamma ray energy and ejected from the atom. This electron is usually more energetic than the photoelectron and will cause ionization exactly like a beta particle.

The scattered gamma ray is really a different gamma ray as the original photon is absorbed and a new one emitted at a lower energy.

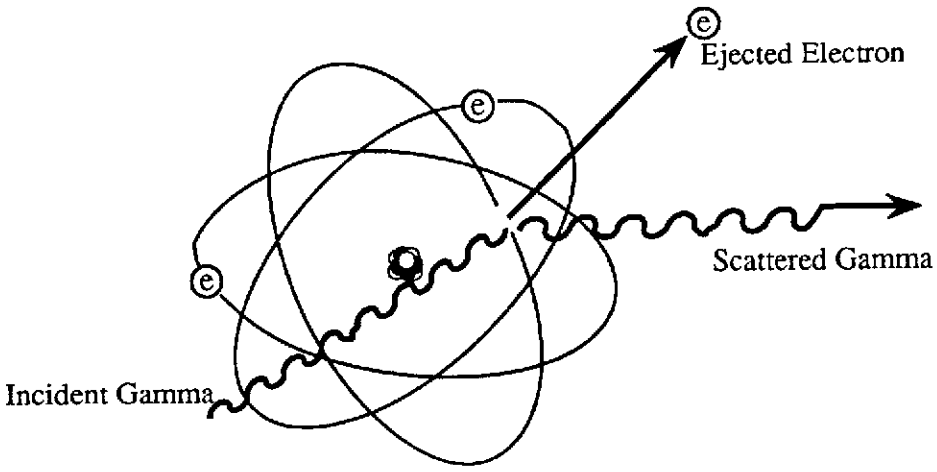


Figure 2.3: Compton Effect

3. Pair Production

This gamma ray interaction always occurs near an atomic nucleus which recoils. The gamma ray gives its energy to the creation of an electron-positron pair. (A positron is a positively charged electron!) The minimum gamma photon energy that can do this is 1.02 MeV (the energy equivalent of 2 electron masses). The process most often happens at higher energies.

The positive and negative electrons created both cause ionization but their fates differ. The positron will meet with another atomic electron and they will "mutually annihilate". Both cease to exist but 2 gamma rays of 0.511 MeV each are created. These gammas will go on and cause one of the other possible gamma ray interactions. The electron will eventually settle down with some accommodating atom and become a normal atomic electron.

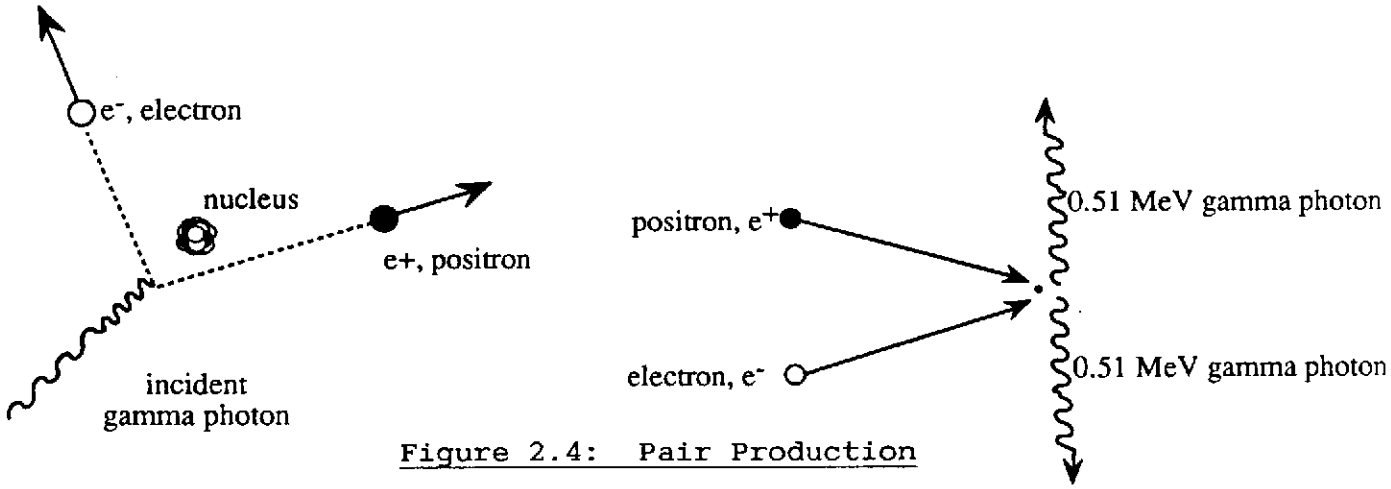


Figure 2.4: Pair Production

Direct and Indirect Ionization

Alphas and betas cause direct ionization. Each ion-pair created takes a small amount of energy and therefore slows the alpha or beta a little bit. Eventually the particle will be stopped. Alphas of a given energy would all travel the same straight line distance (range) in a given material. Similarly betas of a given energy would all have about the same range in a given material. By contrast the gamma rays cannot be assigned a range as they may interact immediately or travel a very long distance between interactions. The gamma ray energy is transferred in large chunks and is deposited in the material by indirect (i.e. secondary) ionizations near each of the interactions.

SHIELDING

It is easy to shield against alphas or betas, we simply need material of thickness equal to or greater than their range.

Shielding against gamma rays is not so easy. No matter how thick the shielding some of the gamma rays can still penetrate. For any particular energy of gammas we can always find the amount of material that will reduce the intensity to half. We call this the half value layer (HVL). Two half value layers would reduce the intensity to ¼ of the original.

As an example, for gamma rays from fission products about 15 cm of water is a half value layer. In the irradiated fuel bays, water is maintained at least 4.5 m (30 HVL's) depth above the fuel. That means that the γ ray intensity reaching the surface of the bay will be reduced by a factor of 2^{30}, that is, it has been halved 30 times. In round numbers that is a reduction of 10^{-9} or one billionth of the original intensity. (You should check these numbers on your calculator.)

Gamma rays are shielded most effectively using materials made from heavy nuclei. Lead is often used where there is very little room for shielding. Where lighter materials (e.g. concrete or water) are used, greater thicknesses are needed.

Type of Radiation	Approximate Mass (AMU)	Charge	Energy Range (MeV)	Remarks
α	4	+2	4 to 8	Very short range, highly ionizing
β	0.0005	-1	0.5 to 3.5	Short Range
γ	0	0	Up to 10 (most below 3)	Long Range

Table 2.1

ASSIGNMENT

1. Using ${}^{A}_{Z}X$ notation, write equations for Alpha, Beta and Gamma decay.

2. Briefly explain how Alpha, Beta, and Gamma deposit their energy in matter.

3. List the masses and charges for α and β particles.

4. What is ionization?

5. Why is it said that γ rays do not cause direct ionization?

6. Describe methods used to shield against α or β particles.

7. What type of material makes good gamma ray shielding?

8. For a material of half value thickness of 6 cm, shielding 1 MeV gamma rays, calculate the thickness needed to reduce the intensity by 1 000.

A. Broughton

NUCLEAR STABILITY AND INSTABILITY

OBJECTIVES

At the conclusion of this course the trainee will be able to:

1. Discuss the stability of nuclides in terms of neutron proton ratios and nuclear forces.

2. From a plot of n against p state the emission a given nuclide is likely to undergo.

3. Given a chart or table of nuclides, list all members of the decay chain of a given radioactive nuclide.

NUCLEAR STABILITY AND INSTABILITY

A plot of all the stable nuclides is shown in Figure 3.1. Each black dot represents a stable nuclide. Where more than one dot appears for a particular atomic number those dots represent stable isotopes. For example if we look at atomic number one, we will see two dots, these represent hydrogen one (H-1) and hydrogen two (H-2). If we look at atomic number 8, we will see 3 dots representing O-16, O-17 and O-18.

Looking at the general shape of the dot distribution we can draw the simple conclusion that for light nuclides the number of neutrons is approximately equal to the number of protons. There are of course some exceptions, such as hydrogen-1 which has only 1 proton. For intermediate mass nuclides the neutron to proton ratio tends to be higher say about 1.3, e.g. for Rh-103, n:p = 1.29. For the heavy nuclides the ratio goes up to about 1.5, e.g. for gold, Au-197, n:p = $\frac{118}{79}$ = 1.5.

In general we can say that if the neutron-proton ratio is outside of this range the nuclide is unstable. We find for instance that two protons cannot combine to form a nucleus without the aid of neutrons. If we look at the nucleus of He-3 we see that it contains two protons and one neutron. The neutron helps "dilute" the electric force which tends to push the protons apart. In a sense the neutron "glues" the protons together. Excess neutrons permit the short range attractive force between nucleons to overcome the long range repulsive electric force between the protons. In He-4 the 2 neutrons and 2 protons give a nucleus which is particularly stable in terms of nuclear behavior.

Adding more neutrons won't always increase the stability. There is no evidence that $^{5}_{2}He$ can exist and $^{6}_{2}He$ (which has been observed) has less than a 1 s half-life. In general, extra neutrons aid stability but too few or too many neutrons will cause instability. Unfortunately we can't specify it much closer than that.

Suppose we look at Cu-64, (Z = 29). You will not find a dot for it on the graph because it is unstable, yet if we do plot it we find it sits right between Cu-63 and Cu-65, both of which are stable. Having a n:p ratio in the right range is important for stability but some unstable nuclides also have n:p ratios in the right range. We are safe however in saying that the neutron proton ratio must be "right" for a stable nuclide. If it is "wrong" the nuclide is certainly unstable.

We've already seen that the very heavy nuclides (Z > 83), are all unstable. If they have a suitable n:p ratio (and thus do not beta decay) they will α decay because of the large repulsive electric force. If extra neutrons are present to dilute this electric force, α decay may be prevented but beta decay occurs because of the high n:p ratio.

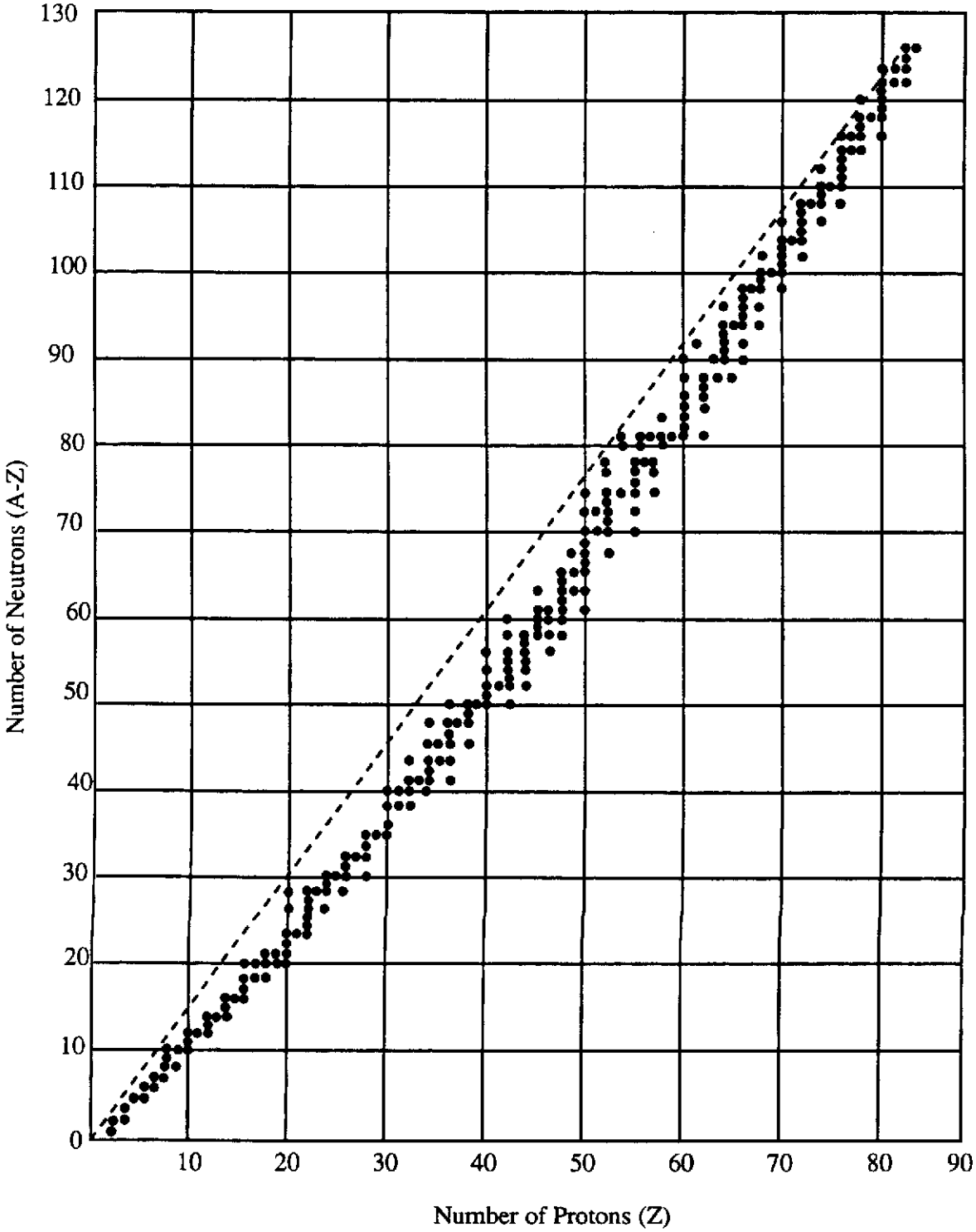

130
120
110
100
90
80
70
60
50
40
30
20
10
0
Number of Neutrons (A-Z)
10
20
30
40
50
60
70
80
90
Number of Protons (Z)

Figure 3.1

Neutron Rich Nuclides

For lighter nuclides a relatively easy way to get a "wrong" n:p ratio is to add a neutron to a stable nuclide (by neutron absorption). For example adding a neutron to the nucleus of O-18 would give O-19 which is unstable. This process is called activation. Neutron absorption does not always cause activation e.g. add a neutron to H-1 and get the stable H-2 nuclide.

A second way to get nuclides which are neutron rich is to split (fission) a heavy nuclide into two intermediate mass nuclides. These fission products are almost certain to be unstable. Consider splitting a nucleus of U-236. It has an n:p ratio of 1.56 and the two fragments will likely also have n:p ratios around 1.56. This is too high for nuclides around the middle of the mass range. The dashed line in Figure 3.1 shows the position of all nuclides with a neutron to proton ratio of about 1.5. Only the heaviest stable nuclides lie squarely on this line.

Suppose for example that two fission products are $^{95}_{36}Kr$ and $^{139}_{56}Ba$. If we plot these on the graph we find that they are quite a way off the stability zone. They are unstable. They will decay or disintegrate by emitting a particle. Will the particles be alphas or betas? Well we can make a guess. Perhaps $^{95}_{36}Kr$ is an alpha emitter.

$$^{95}_{36}Kr \longrightarrow {}^{4}_{2}\alpha + {}^{91}_{34}X \quad ?$$

Where would this new nuclide plot? Is it more stable, less stable or about the same?

You should suspect that its stability is about the same and alpha decay is unlikely. So let's second guess (it's usually more reliable) and try a beta decay.

$$^{95}_{36}Kr \longrightarrow {}^{0}_{-1}\beta + {}^{95}_{37}X \quad ?$$

How does this one look on the graph? Yes, it looks better. We can assume therefore that beta emissions are likely. This is confirmed by experiment. The fission products are nearly all neutron rich beta emitters. A few turn out to be stable. A very small number follow a beta decay with the immediate release of a neutron. (These are called delayed neutrons.) There are no fission products at all that emit alpha particles.

Interchangeable Nucleons

In the case of beta emissions you will notice that the number of nucleons remains constant even though the nuclides have changed. What has happened is that a neutron has changed into a proton.

$$^{95}_{36}Kr \longrightarrow \beta + {}^{95}_{37}Rb$$

$$\left.\begin{matrix}36 \text{ protons} \\ +59 \text{ neutrons}\end{matrix}\right\} \longrightarrow \beta + \left\{\begin{matrix}37 \text{ protons} \\ +58 \text{ neutrons}\end{matrix}\right.$$

As you will see in the next section the reverse process can also happen in some nuclides. A proton can turn into a neutron either by emitting a positive electron or by capturing an orbiting electron.

Neutron Deficient Nuclides

Many neutron deficient nuclides are known, but we are not likely to meet with any in our reactor technology. These nuclides will plot beneath the curve in Figure 3.1. What type of emissions should we expect from them? Try some guesses as we did in the last section. Are they alpha emitters? Are they beta emitters? You should be able to satisfy yourself that neither of these seems to be reasonable.

These neutron deficient nuclides decay by either positron emission or electron capture or both. Positron? What is that? The positron is a positively charged electron. It is exactly like an electron in every way, except for its positive charge. Putting this into the usual symbolic form:

$$^{A}_{Z}X \longrightarrow {}^{0}_{1}\beta + {}^{A}_{Z-1}X$$

If you plot ${}^{A}_{Z-1}X$ on the graph, Figure 3.1, you will see that it is closer to the stability region then ${}^{A}_{Z}X$.

For electron capture (also called K-capture because the electron comes from the K-shell orbit) we write:

$$^{A}_{Z}X + {}^{0}_{-1}e = {}^{A}_{Z-1}X$$

The nuclide produced is the same one produced by positron emission.

Heavy Nuclides

For nuclides above atomic number 83 we find there are several possible decays. Most of the naturally occurring heavy nuclides are alpha emitters, some are beta emitters and a few may also undergo spontaneous fission. Some of the manmade heavy nuclides are also positron emitters (rare) or undergo electron capture (more likely).

ASSIGNMENT

1. Why do all nuclides except H-1 have neutrons in their structure?

2. For these nuclides guess which type of particle they will emit. Use the graph of Figure 3.1 and then check your answers from a chart of the nuclides or from the Table of Isotopes in the Science Data Book.

   Sr-90, Br-87, Xe-135, I-135, I-131, Sm-149, Co-60, B-10, N-16, U-238, Pu-239, Cu-64, Mn-56, H-3, Cs-137.

3. What happens if the neutron to proton ratio of a nuclide is too high or too low?

4. For the natural heavy nuclide U-238 write down the stages in its decay to become a stable nuclide. Use a chart of the nuclides or the Science Data book for this. The exercise can be repeated for U-235 as the starting nuclide.

A. Broughton

ACTIVITY

OBJECTIVES

At the conclusion of this course the trainee will be able to:

1. Define the units of activity, the Becquerel, and the Curie.

2. Define the half-life and discuss activity in terms of half lives.

3. State the activity relationship $A_t = A_o(\frac{1}{2})^n$ and solve simple activity and half-life calculations.

ACTIVITY

The activity of a radioactive material is measured in terms of the number of nuclei which decay per unit of time. This can be expressed in a number of ways. The simplest is perhaps to express it as "disintegrations per second" or dps. In the SI system this unit is called the becquerel (Bq). The becquerel is defined as being one radioactive disintegration (decay) per second. Another unit, widely used, is the curie. One curie is equal to 3.7×10^{10} Bq, which is the activity of 1 g of radium-226.

The Law of Radioactive Decay

A pure radioactive substance decays at a fixed fractional rate, i.e., in each second a constant fraction of the total amount present decays. Thus we can see that the activity, the actual number of atoms decaying per unit time, is proportional to the amount of the substance.

If we consider a particular sample of a radio-nuclide, the continual decay will diminish the quantity of the sample and so the activity will also diminish. This process will continue until the radiactive material is gone. Figure 4.1 is a plot of the quantity of the sample Q against time.

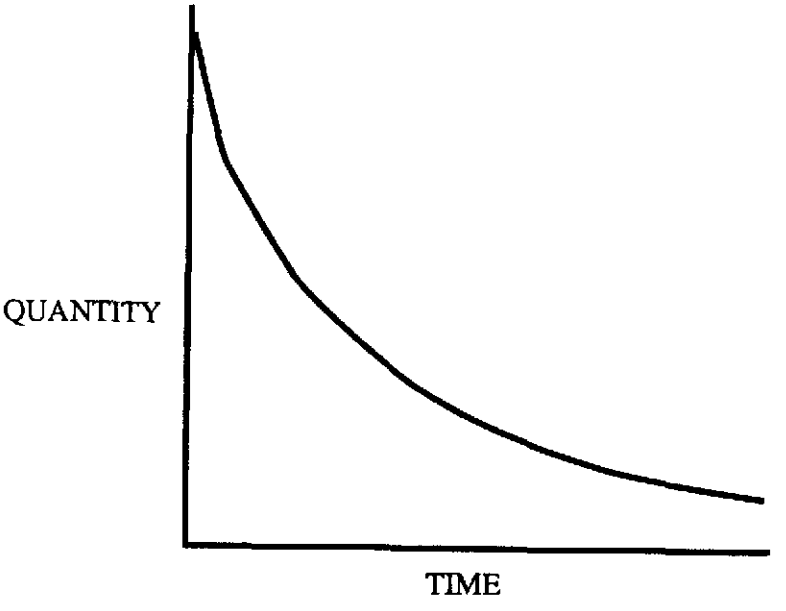


Figure 4.1: Quantity v Time for a Radionuclide

We have already seen that the activity is proportional to the quantity of the radioactive substance so we can also plot the activity A against time (Figure 4.2).

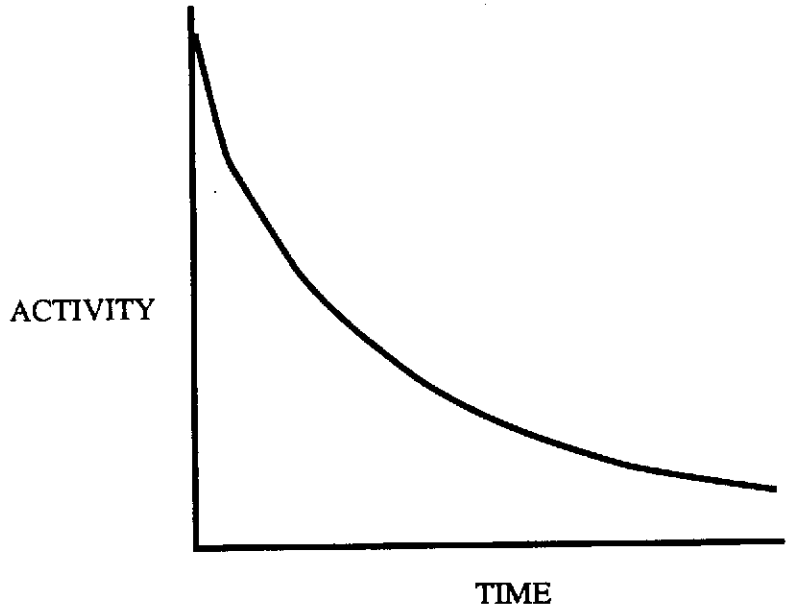


Figure 4.2: Activity v Time for a Radionuclide

These two graphs (Figures 4.1 and 4.2) are mathematically identical and only differ in that they have different vertical scales. In practice we prefer to use the second form because activity is the quantity usually measured and indeed activity is what we are most often interested in. By contrast we seldom can measure, and often don't care about the actual quantity of the radioactive substance. For example the activity of the moderator, quoted in curies per kilogram, is very important information but tells us nothing directly about the number of tritium atoms in the moderator.

Half-life

If we plot graphs of activity vs time (Figure 4.3) for different radioactive materials we find that they have different rates of change. To distinguish between the different rates we use the concept of the half-life (Figure 4.4). The half-life ($T_{\frac{1}{2}}$) is the time interval that it takes for the activity of a specimen to fall to half of its original value, i.e., the time interval between activity Ao and activity $\frac{A_o}{2}$. For an exponential decay curve (which these are) it does not matter where we start with Ao. For any starting point on the curve the time to $\frac{A_o}{2}$ will be the same (Figure 4.4).

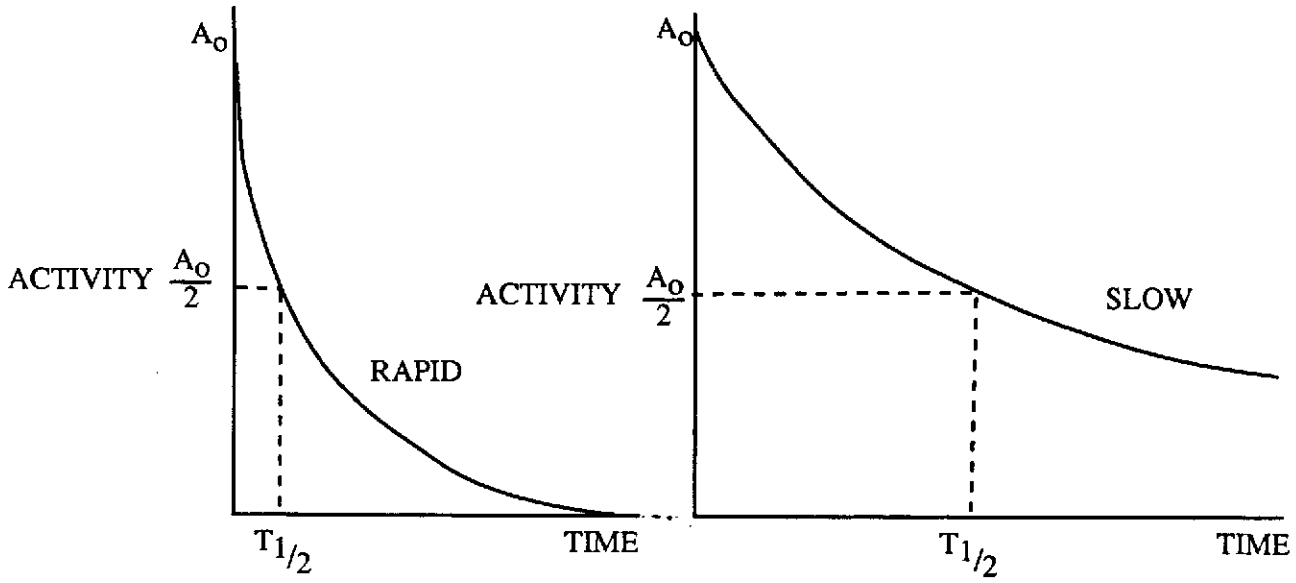


Figure 4.3 a&b:
Activity Plots for Two Different Radioactive Materials

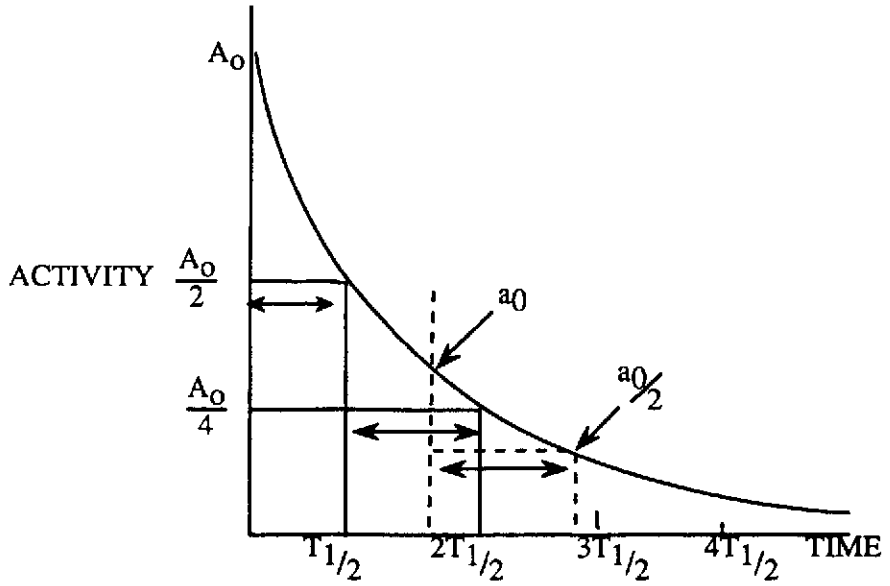


Figure 4.4: Activity and Half-life

The time from Ao to $\frac{Ao}{2}$ is the same as from $\frac{Ao}{2}$ to $\frac{Ao}{4}$. It also takes the same time from ao to $\frac{Ao}{2}$. This leads to the formula

$A_t = Ao\ (\frac{1}{2})^n$ where n is the number of half lives,

i.e., $n = t/T_{\frac{1}{2}}$

In this relationship we usually take n to be an integer but it need not be.

Another form of the equation is

$$\frac{Ao}{A_t} = 2^n$$

Before reviewing the following examples try the end-of-chapter exercises. Many people find these calculations easier to do then to read about.

Examples

1. Suppose a radioactive substance has an activity of 6144 Bq (this is a high figure). How many half-lives will it take for the activity to fall to 6 Bq?

$\frac{Ao}{At} = 2^n$

$\frac{6144}{6} = 2^n$

$1024 = 2^n$

$\therefore \quad n = 10$

Answer

It will take 10 half-lives for the activity to fall from 6144 to 6 Bq.

2. What will the activity be 6 half-lives later for a radioactive substance which has an activity now of 192 Bq?

$A_t = Ao\ (\frac{1}{2})^n$

$= 192\ (\frac{1}{2})^6$ i.e. $(192 \times \frac{1}{2} \times \frac{1}{2} \times \frac{1}{2} \times \frac{1}{2} \times \frac{1}{2} \times \frac{1}{2})$

or $= \frac{192}{2^6} = \frac{192}{64}$

$= 3$ Bq

Answer

After 6 half-lives an original activity of 192 Bq falls to 3 Bq.

3. If the half-life in example 1. is 25 minutes what would be the time t?

$t = nT_{\frac{1}{2}}$

= 10 half-lives x 25 minutes per half-life

= 250 m

= 4h 10 m

Range of Half-lives

Half-lives vary from very short (fractions of a second) to very long (billions of years). From an operational point of view these values are important in comparison to reactor life, operational times, outage times, fuel life, etc.

For example, fresh CANDU fuel is made with natural uranium. The half-life of U-238 is 4.5 billion years and for U-235 is 700 million years. Although both of these are decaying by α emission we never notice any change in their activity over the lifetime of the reactor. Thus we can say that fresh fuel will be the same no matter how long we keep it. By contrast, the half-life of N-16 (produced by activation in the reactor core) in only 7s and the activity changes faster than most of us can calculate.

In fuel that has been irradiated there are isotopes of uranium and neptunium that decay to make fissile plutonium. The beta decays of U-239 and Np-239 have half-lives of 23m and 2.3d respectively. Thus they convert into the fissile Pu-239 on a quite short time scale. The Pu-239 decays by α decay with a half-life of 25 thousand years so its quantity does not change due to α decay in the 1 to 2 years the fuel is in the reactor.

ASSIGNMENT

1. What is the relationship between disintegrations per second and the becquerel?

2. What is the usual way we express the characteristic rate of decay of a radionuclide?

3. Fe59 has a half-life of 45 days. If a sample has an activity of 1000 dps what will its activity be after one year.

4. The activity of a radioactive specimen is 2×10^7 DPS. After 20 days the activity is 2×10^4 dps. What is the half-life of this specimen? (Calculate to the nearest whole number of half-lives.)

NEUTRONS AND NEUTRON INTERACTIONS

OBJECTIVES

At the conclusion of this lesson the trainee will be able to:

1. Write equations for:
   a) Transmutation
   b) Radiative capture
   c) Photoneutron reaction with deuterium.
2. Describe elastic and inelastic scattering of neutrons.

NEUTRONS AND NEUTRON INTERACTIONS

A nuclear reactor will not operate without neutrons. It is important to have a thorough knowledge of those reactions which produce neutrons and those interactions which neutrons undergo.

NEUTRON PRODUCTION

Most of the neutrons in a CANDU reactor come directly from fission. In addition, about ½% of the neutrons in a reactor at power are emitted by the fission products. These two important types of neutrons (known as prompt and delayed neutrons respectively) are discussed in Module 6. The only other important neutron source in an operating CANDU is the photoneutron reaction.

THE PHOTONEUTRON REACTION

A gamma ray with an energy of 2.2 MeV or greater can interact with a deuterium nucleus removing a neutron. The deuterium nucleus becomes a normal hydrogen nucleus and the neutron is free to move around.

$$ {}^{2}_{1}H + \gamma \longrightarrow {}^{1}_{1}H + {}^{1}_{0}n $$

NEUTRON INTERACTIONS

1. Elastic Scattering

   This resembles a billiard ball collision. A neutron collides with a nucleus, transfers some energy to it and bounces off in a different direction. The fraction of its initial energy lost depends on whether it hits the target nucleus dead-on or at an angle - exactly like the cue ball striking a ball on the billiard table. The energy lost by the neutron is gained by the target nucleus which then moves at an increased speed. Light nuclei are the most effective for slowing down neutrons.

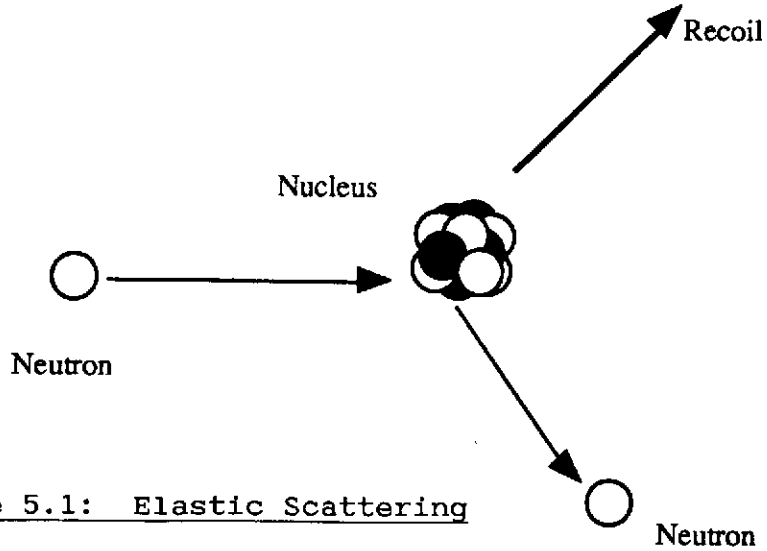


Figure 5.1: Elastic Scattering

A neutron colliding with a heavy nucleus rebounds with little loss of speed and transfers very little energy - rather like firing the cue ball at a cannon ball.

On the other hand neutrons will not be scattered by the light electron clouds surrounding the nucleus, but will travel straight on - much like baseballs through a fog.

2. Inelastic Scattering

A neutron may strike a nucleus and, instead of bouncing off, be temporarily absorbed, forming a "compound nucleus". This will be in an excited state. It de-excites by emitting another neutron of lower energy, together with a gamma photon which will take the remaining energy. This process is known as INELASTIC SCATTERING. It generally happens only when high energy neutrons interact with heavy nuclei and has little practical importance for reactor operation.

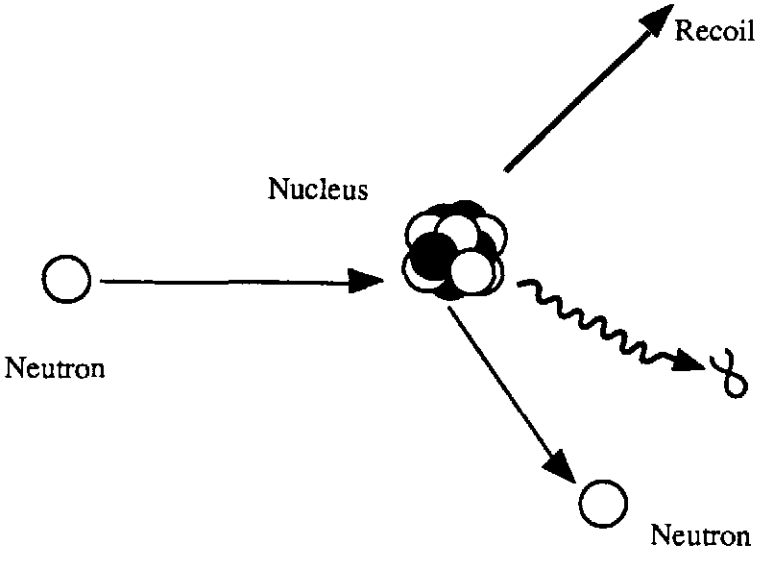


Figure 5.2: Inelastic Scattering

3. Transmutation

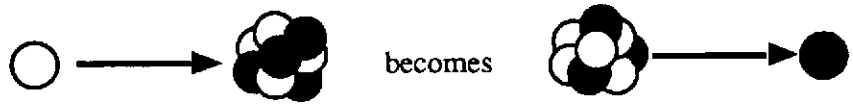
A neutron may be captured by a nucleus forming a compound nucleus which then de-energizes by emitting a charged particle, either a proton or an alpha particle. This produces a new nucleus of a different element. Such a nuclear reaction is called a TRANSMUTATION.

TRANSMUTATION is the transformation of one element into another by a nuclear reaction.

Examples:

a) Neutron-Proton Reaction (n,p)

Oxygen-16 captures a neutron and emits a proton to form nitrogen-16:

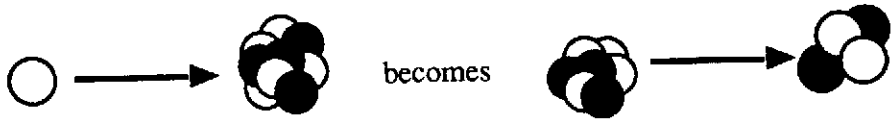


$^{1}_{0}$ Neutron + $^{16}_{8}$ 0 Nucleus $\rightarrow$ $^{16}_{7}$ Nucleus + $^{1}_{1}$ proton

The product, nitrogen-16, is radioactive with a half-life of 7.1 seconds; it is a beta emitter, but more important, it also emits very high energy gammas.

b) Neutron-Alpha Reaction (n,α)

Neutrons captured by boron-10 cause the following reaction:



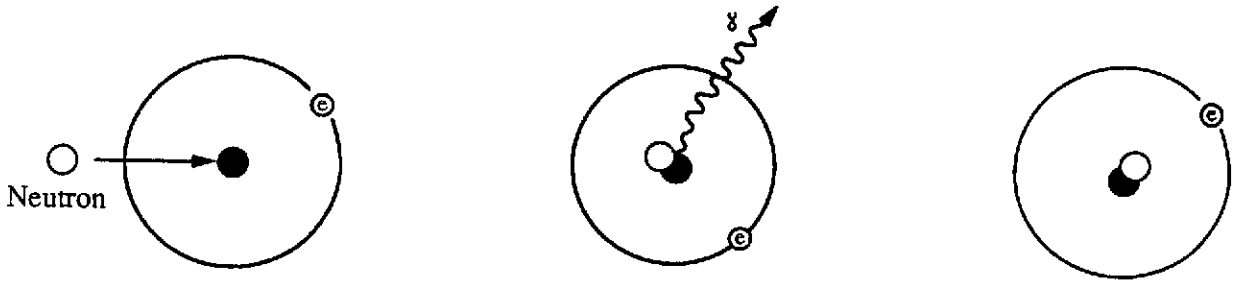
$^{1}_{0}$ Neutron + $^{10}_{5}$ B Nucleus $\rightarrow$ $^{7}_{3}$ Li nucleus + $^{4}_{2}$ He

4. Radiative Capture (n,γ)

This is the most common nuclear reaction. The compound nucleus formed emits only a gamma photon; in other words the product nucleus is an isotope of the same element as the original nucleus (its mass number will have increased by one).

Examples

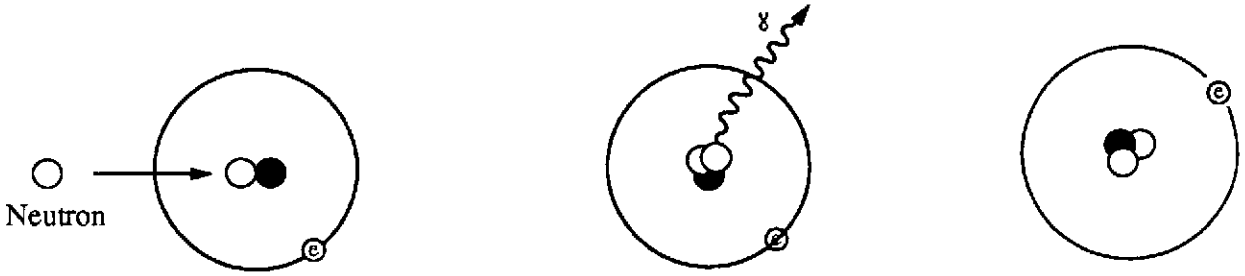
a) The simplest neutron-gamma reaction occurs with hydrogen to produce deuterium (heavy Hydrogen);



$^{1}_{1}H$ $^{2}_{1}H$

The deuterium formed is a stable nuclide. However, many radiative capture products are radioactive and are beta-gamma emitters.

b) Deuterium itself undergoes a radiative capture reaction to form tritium;



$^{2}_{1}H$ $^{3}_{1}H$

The tritium isotope is unstable and is one of our major radiation hazards.

c) Cobalt-59 undergoes radiative capture to form highly radioactive Co-60:

$$^{59}_{27}Co + ^{1}_{0}n \longrightarrow {}^{60}_{27}Co + \gamma$$

Cobalt-60 has a long half-life (5¼ years) and very penetrating gamma radiation, making it a serious hazard among activated corrosion products. The concentration of cobalt in reactor grade materials is limited to trace amounts.

Cobalt-60 is also the isotope most commonly used in radiation treatment of cancer.

5. Fission

This most important reaction is the subject of the next lesson.

ASSIGNMENT

1. Write the equation for the photo-neutron reaction with H-2.

2. Describe elastic and inelastic scattering of neutrons.

3. Identify the following reactions:

   a) $^{40}_{18}Ar + ^{1}_{0}n \longrightarrow {}^{41}_{18}Ar + \gamma$

   b) $^{16}_{8}O + ^{1}_{0}n \longrightarrow {}^{16}_{7}N + ^{1}_{1}p$

4. List the examples of neutron reactions is this chapter which are also activations.

FISSION

OBJECTIVES

At the conclusion of this lesson the trainee will be able to:

1. Explain where the energy released by fission comes from (mass to energy conversion).

2. Write a typical fission reaction.

3. State how much energy is released per fission and how the major portion of that energy is carried away.

4. Define:

   a) thermal neutrons and fast neutrons

   b) prompt neutrons

   c) delayed neutrons

5. Discuss neutron cross-sections and neutron flux.

6. Explain a self-sustaining chain reaction.

FISSION

In 1939 Hahn and Strassman discovered that when U-235 nuclei were bombarded with neutrons some would split into two nuclei of medium mass with two important results:

1) Energy was produced

2) More neutrons were released.

The process was called fission which can be defined as, "The splitting of a heavy nucleus into two lighter nuclei".

Energy Released by Fission

Prior to the twentieth century physicists believed that mass and energy were separate non-related quantities, each of which was governed by a fundamental law:

a) The Law of Conservation of Mass which states that mass cannot be created or destroyed.

As an example consider the burning of carbon. A carbon atom reacts with a molecule of oxygen according to the equation:

$$C + O_2 \longrightarrow CO_2$$

Each reaction releases approximately 5 eV of energy. It was believed that if all the carbon dioxide (CO_2) gas could be collected and weighed its weight would be equal to the combined weight of the carbon (C) and the oxygen (O_2).

b) The Law of Conservation of Energy which states that energy cannot be created or destroyed.

One form of energy can be changed to another but no energy disappears in the transfer. Chemical potential energy is converted to heat in burning. Heat energy can be used to change water to steam (i.e. to provide increased molecular energy). The steam can, in turn, be used to produce mechanical energy in a turbine. The turbine can drive a generator to produce electrical energy which can then be used to produce light energy. Any loss of energy during a change from one form of energy to another can always be accounted for by heat losses, frictional losses, etc., which are all different forms of energy transfer. There is no net loss in energy.

Fission defies both of these laws, as energy is created and mass is lost. Conversion of mass to energy had been predicted by Einstein. He had observed contradictions in the laws of classical physics and had postulated that mass and energy were related by the formula:

$$E = Mc^2$$

where: E = energy (joules)
M = mass (kilograms)
c = speed of light (3×10^8 m/s)

To demonstrate why this relationship has not been observed in chemical reactions look at the complete combustion of one kilogram of coal:

Energy from complete combustion of 1 kg of coal = 3.36×10^7 joules

from, $E = Mc^2$

Therefore; $M \text{ (converted)} = \frac{E}{c^2} = \frac{3.36 \times 10^7}{(3 \times 10^8 \text{ m/s})^2}$

$= 3.7 \times 10^{-10}$ kg

This is a very small fraction of the initial 1 kg and impossibly small to measure. A similar calculations shows that a few billionths of a mass unit is converted to energy in burning one atom of carbon. Present technology is not quite able (yet) to measure such a small mass loss.

Now consider the complete fissioning of 1 kilogram of uranium-235.

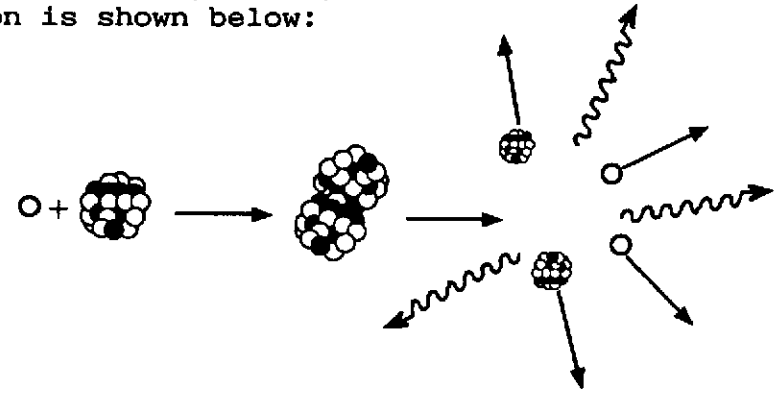
Energy from complete fissioning of 1 kg of U-235 = 8.2×10^{13} joules.

$$M \text{ (Converted)} = \frac{E}{c^2} = \frac{8.2 \times 10^{13} \text{ J}}{(3 \times 10^8 \text{ m/s})^2}$$

$$= 9 \times 10^{-4} \text{ kg}$$

This is nearly 0.1% of the original mass which has been converted to energy and is more easily measured.

Now we will look at the fission of a single U-235 atom in more detail. A neutron enters the U-235 nucleus to form a highly excited compound nucleus, U-236, which in turn fissions. A typical fission is shown below:



$$^{1}_{0}n + {}^{235}_{92}U \rightarrow ({}^{236}_{92}U)^{*} \rightarrow {}^{95}_{37}Rb + {}^{139}_{55}Cs + 2^{1}_{0}n + \gamma$$

The mass of the reactants	236.05u
The mass of the products	235.86u
Mass converted	0.19u

This yields approximately 200 MeV if energy from the subsequent radioactive decay is included. Thus, each fission produces about 200 MeV of energy as a result of the conversion of some of the original mass to energy. While one fission does not create a significant amount of energy, each kilogram of natural uranium contains 1.8×10^{22} U-235 atoms, most of which can be fissioned.

Fission Fragments

The general equation for the fissioning of uranium is:

$$^{235}_{92}U + ^{1}_{0}n \rightarrow (^{236}_{92}U)^* \rightarrow 2\ F.F. + 2.5\ ^{1}_{0}n + \gamma$$

The two fission fragments (F.F.) leave the fission site with velocities around 9×10^6 m/s (that is 32 million kilometers per hour). They are highly positively charged and deposit their energy by ionization in a very short distance (5×10^{-4} cm).

Most of the energy from fission (≈84%) is in the form of kinetic energy of the fission fragments. The mass of the fission fragments falls within the narrow range shown in figure 6.1.

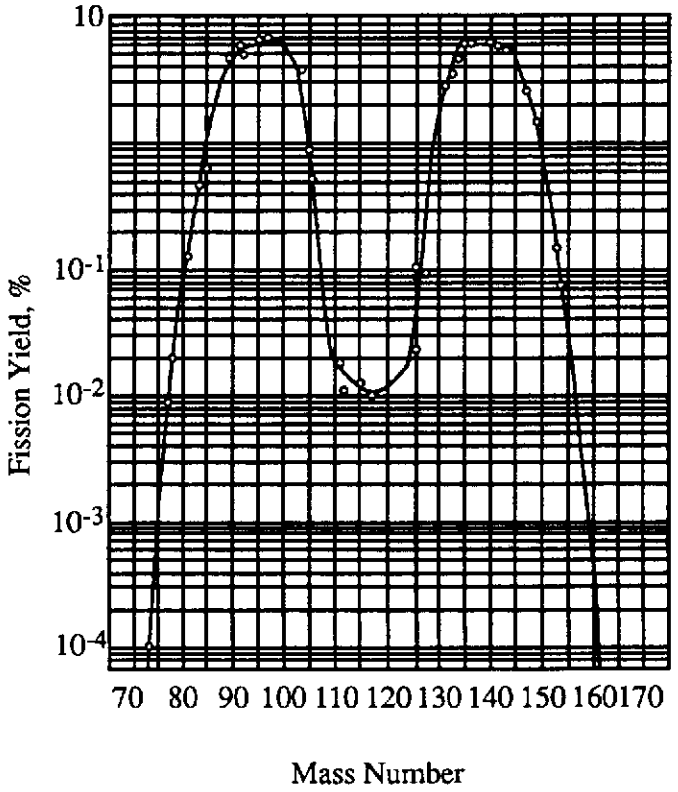


Figure 6.1: Yield of Fission Fragments

The fission fragments have about the same neutron/proton ratio as the parent U-236 excited nucleus, yet are much lighter nuclei, (n:p = 144/92 = 1.57). Their stable neutron/proton ratio is smaller, (about 1.3 for the light fragment and 1.4 for the heavier one). They are "neutron rich". As a result they normally undergo beta-gamma decay.

Chain Reaction

Each individual fission will produce between 0 and 5 neutrons; however, the average is approximately 2.5. It is these neutrons which can, under the right circumstances, go on to produce further fissions. As Figure 6.2 shows, one fission will give two, two gives four, four gives eight and so on. This would give over one thousand fissions in just ten generations.

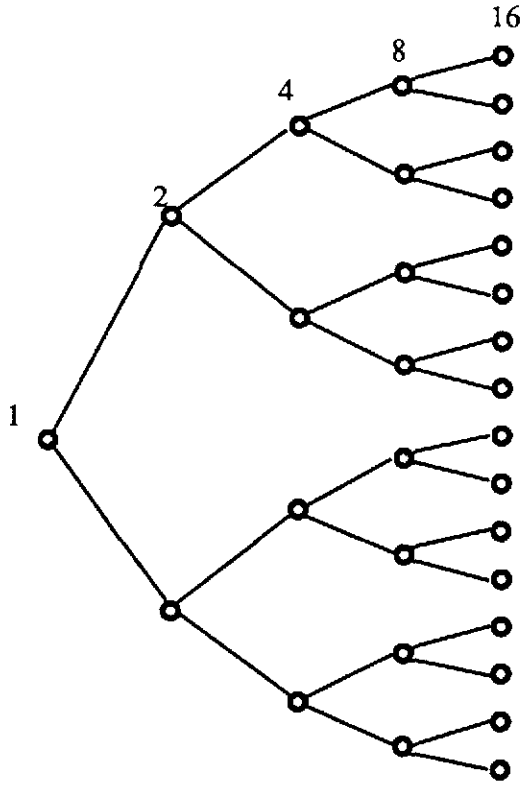


Figure 6.2: The Chain Reaction

This type of multiplication is unsuitable for a power reactor where steady power production is required. For a power reactor we want each fission to cause just one other fission; thus 1.5 neutrons must meet some fate other than causing fission. This special condition, where each fission causes one more, is called a "self-sustaining chain reaction" and will be discussed in detail in a later module.

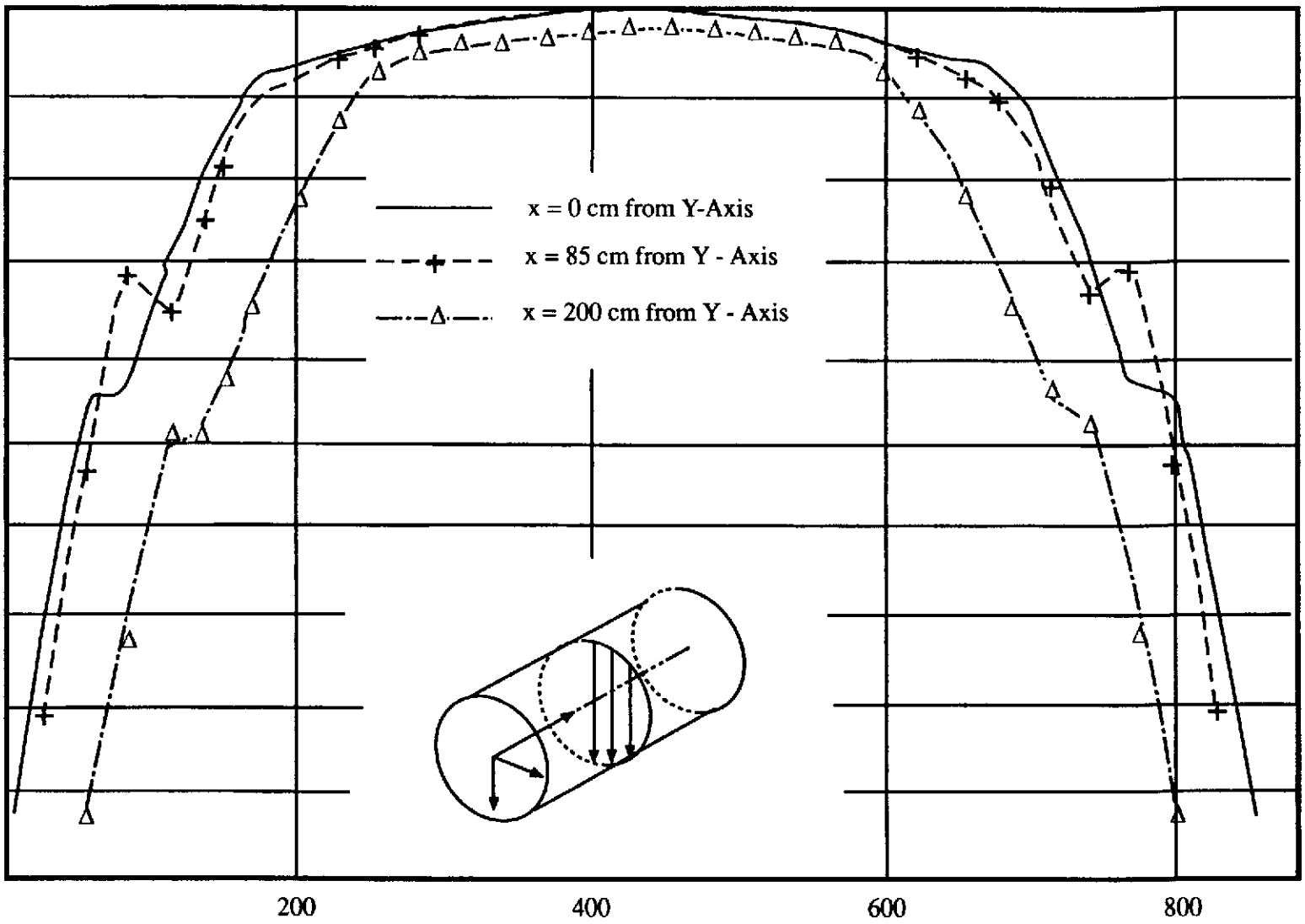


Figure 6.3: Neutron Flux

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Prompt and Delayed Neutrons

Most (99.35%) of the neutrons from fission are born at the time of the fission (10^{-14} seconds after neutron absorption by U-235). A very small number of the fission fragments emit a neutron while decaying. These decaying fission fragments yield 0.65% of the neutrons from the fission of U-235.

The neutrons born "instantly" at the time of fission are called Prompt Neutrons. The average lifetime before neutron emission from the fragments is 13 seconds. These neutrons are called Delayed Neutrons and will be seen to be indispensable to reactor control.

Neutron Energy

Neutrons from fission have relatively high energies in the vicinity of 2 MeV. High energy neutrons travel at speeds a few percent of the speed of light and are called fast neutrons. They slow down by undergoing elastic and inelastic collisions with surrounding nuclei until they reach an energy equilibrium with their surroundings.

Once slowed down the neutrons diffuse through the core, jostled by surrounding molecules. (In subsequent collisions with neighbouring molecules the neutron is just as likely to pick up a bit of energy as to lose some). Such neutrons are called thermal neutrons. A thermal neutron has an energy of 0.025 eV at 20°C. Thermal neutrons are also called slow neutrons.

Neutron Flux

Thermal neutrons are much more likely to interact with nuclei than are fast neutrons. The effect of the thermal neutrons at any point in the reactor depends on both the number of neutrons and their speeds. The quantity that relates these properties is the Thermal Neutron Flux, represented by the Greek letter ϕ (phi). In this course neutron flux can be thought of as a neutron population function. (i.e. higher flux means more neutrons). Figure 6.3 shows the thermal neutron flux in a Bruce reactor. Flux distributions will be discussed later in more detail.

Neutron Cross-sections

In lesson five we examined the two basic reactions which neutrons can undergo, scattering and absorption. (Fission is a special case of absorption). However, a given target nucleus does not have an equal likelihood of undergoing either of these two reactions nor do different nuclei have the same probability of reacting with a neutron.

Neutron Cross-Section denotes the probability of a reaction occurring when a target nucleus is bombarded with neutrons. It has the dimensions of area and can be thought of as the effective target area of the nucleus for an incoming neutron, although the cross-section has no simple relationship with the actual geometric area of the nucleus.

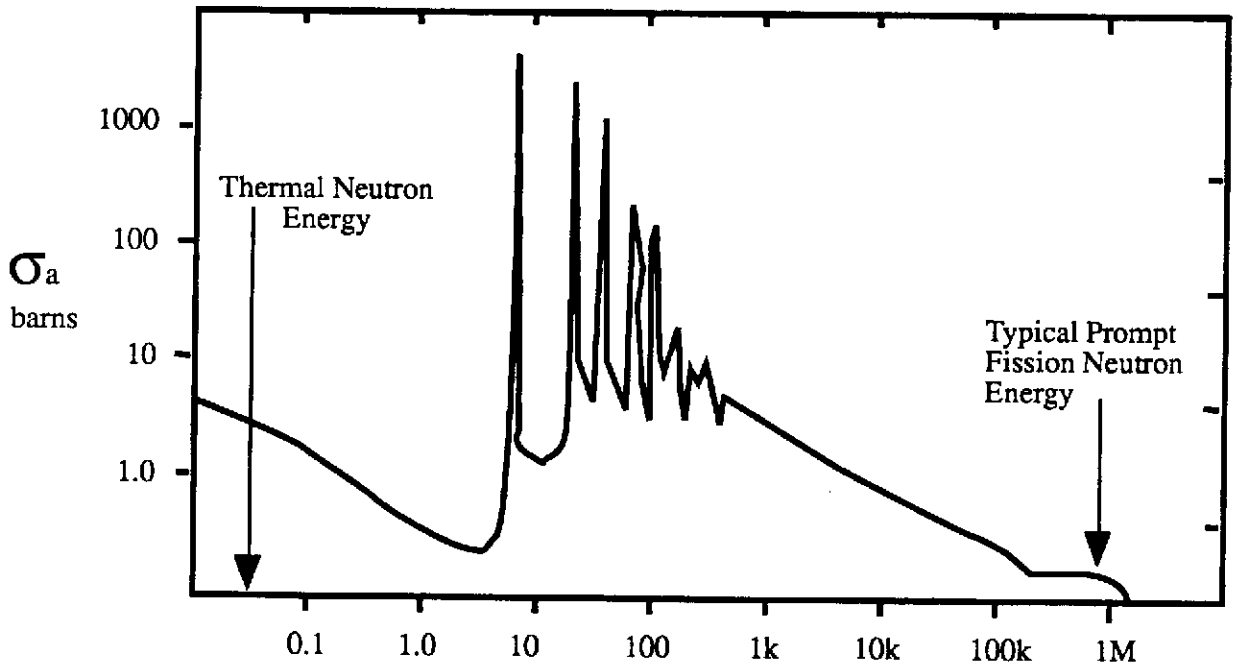
The microscopic neutron cross-section is denoted by the Greek letter sigma (σ). Subscripts denote the type of cross-section e.g. σ_a is the absorption cross-section while σ_f is the fission cross-section. The unit for cross-section is a barn (1 barn = 10^{-24} cm^2). To a neutron an area of 10^{-24} cm^2 appears "as easy to hit as the broad side of a barn".

The actual value of the cross-section is dependent on many factors including:

1. The composition of isotopes in the target.

2. The energy of the incoming neutron.

These two effects are covered in the next two sections.



Variation of the absorption cross section of U238 with neutron energy

Figure 6.4:

1. Effect of Composition

Uranium-235 has a fission cross-section of 580 barns for thermal neutrons, but it makes up only 0.7% of natural uranium. The other 99.3% is U-238, which has a zero fission cross-section for thermal neutrons. Thus, the fission cross-section of natural uranium (used in CANDU fuel) can be written:

$$\sigma_f^{\text{Nat. U}} = .993\ (0) + .007\ (580)$$

$$\approx 4 \text{ barns}$$

If we enrich the fuel to 2% U-235 (typical for U.S. light water reactors) the fission cross-section would be:

$$\sigma_f^{\text{2\% Enriched}} = .98\ (0) + .02\ (580)$$

$$\approx 11.6 \text{ barns}$$

As you can see, enrichment increases the fission cross-section of the fuel. Fission will be a more probable fate for a neutron entering enriched fuel, (almost 3x as likely as for CANDU fuel).

2. Effect of Neutron Energy

For most reactions the cross-section decreases with increasing neutron energy; thus the fission cross-section for U-235 for thermal neutrons is 580 barns while the fission cross-section for fast (2 MeV) neutron is only 2 barns.

The absorption cross-section for U-238 is shown in Figure 6.4. Note that the only time absorption in U-238 is significant is in the energy range of ≈10 eV to ≈1 keV. The peaks shown are called "Resonance Absorption Peaks".

ASSIGNMENT

1. Explain where the energy released by fission comes from.

2. Write the general fission reaction for $^{235}_{92}U$.

3. State how much energy is released per fission and how the majority of this energy shows up.

4. Explain a self-sustaining chain reaction.

5. Define:

   a) thermal neutron

   b) prompt neutron

   c) delayed neutron

6. Define neutron cross-section and state its units.

7. How does the probability of fission in U-235 vary with neutron energy?

J.E. Crist

FUEL, MODERATOR, AND REACTOR ARRANGEMENT

OBJECTIVES

At the conclusion of this lesson the trainee will be able to:

1. Explain the purpose of a moderator.
2. Sketch the basic arrangement of fuel and moderator in a CANDU reactor and explain why this arrangement is desirable.
3. State the basic differences between fresh and equilibrium fuel.
4. Compare the moderating properties of heavy water, light water and graphite.

FUEL, MODERATOR, AND REACTOR MANAGEMENT

An assemblage of material which will just give a self-sustaining chain reaction is called a critical mass. If we had a small pile of pure U-235 and initiated fission, many of the neutrons would escape before they could cause further fissions. Thus the chain reaction would die away. As more U-235 is added to the pile, fewer neutrons would escape before causing fission and at some point the pile would support a self-sustaining chain reaction. A pile of that size is the critical mass of U-235.

It is important to note that natural uranium (0.7% U-235) cannot be made critical without the help of a moderator. Too many of the neutrons are absorbed by the resonance peaks in U-238 and are thus unavailable to cause fission.

In order to obtain a self-sustaining chain reaction with natural uranium it is necessary to take the fission neutrons away from the fuel, slow them down to thermal energy then return them to the fuel. By using this process the neutrons are:

1. Away from the U-238 when they have slowed to the resonance absorption energies, and

2. They return to the fuel at thermal energy where they are far more likely to be absorbed by U-235 and cause fission.

Moderator

The function of the moderator is to slow down the fission neutrons without absorbing them. In order to perform this function adequately a moderator must:

1. Thermalize the neutrons in as few collisions as possible over a short distance,

2. Not absorb too many of the neutrons.

Neutrons lose most of their energy by elastic collisions with other nuclei. Elastic scattering with light nuclei is a more efficient method of moderation than elastic scattering with heavy nuclei. It takes an average of only 18 collisions to thermalize a neutron in pure hydrogen but it takes 2172 collisions to thermalize the same neutron in U-238. Thus only light nuclei are considered suitable as moderators.

The second point is low absorption. Boron ($^{10}_{5}B$) could thermalize a neutron in ≈90 collisions but it has an absorption cross-section of 3840 barns and would therefore absorb most of the neutrons it thermalized.

As a result of these nuclear considerations and other economic and engineering considerations, only three moderators are considered suitable for thermal reactors; light water (H_2O), heavy water (D_2O) and graphite (C). Table 5.1 summarizes the properties of each.

Moderator	Average Number of Collisions to Thermalize	σ_s (barns)	σ_a (barns)
H_2O	20	103	0.664
D_2O	36	13.6	0.001
C	115	4.8	0.0034

Table 5.1

Clearly light water will thermalize a neutron faster than either heavy water or graphite (higher scattering cross-section coupled with fewer collisions to thermalization). However, light water's absorption cross-section is 664 times that of heavy water and 195 times that of graphite. Due to light water's neutron absorption, it is impossible to obtain a self-sustaining chain reaction with natural uranium fuel and a light water moderator. Light water moderated reactors must use 2 to 3% enriched fuel. (Uranium in which the percentage of U-235 has been increased from 0.7% to 2 or 3%.)

Most reactor designs, including the CANDU, use UO_2 rather than uranium metal for fuel. Ceramic fuel (UO_2) has excellent corrosion resistance and is very stable in a radiation environment, making it a good choice for reactor fuel. However, it is impossible to obtain a critical mass of unenriched UO_2 with a graphite moderator. Only heavy water is suitable as a moderator for a reactor using unenriched UO_2.

Fresh Fuel and Equilibrium Fuelling

When a reactor is first fuelled the fuel is called fresh fuel. This initial fuel load will operate the reactor for about 6 months after which we remove and replace a few fuel bundles each day. This state is called equilibrium fuelling.

Radical changes occur in the composition of the fuel between the fresh and equilibrium conditions. The most significant are the depletion of the U-235, mostly by fission, the buildup of fission products and the buildup of Pu-239 (a fissile fuel) by the following scheme:

$$^{238}_{92}U + ^{1}_{0}n \rightarrow ^{239}_{92}U + \gamma$$

$$^{239}_{92}U \rightarrow ^{239}_{93}Np + \beta + \gamma \qquad (T_{\frac{1}{2}} = 23.5 \text{ m})$$

$$^{239}_{93}Np \rightarrow ^{239}_{94}Pu + \beta + \gamma \qquad (T_{\frac{1}{2}} = 2.35 \text{ d})$$

Fresh fuel contains 0.7% U-235 and no Pu-239. By the time fuel is removed from the reactor the U-235 is depleted to near 0.2% and there is an approximately equal amount of Pu-239. The fissioning of Plutonium provides a significant portion of the power produced by a CANDU reactor.

Each atom that fissions produces two new atoms so the fission products build up to a concentration just over 1%. The content of U-238 in the fuel changes very little.

Reactor Arrangement

Figure 7.1 and 7.2 show the axial and radial arrangement of the fuel in the moderator. This arrangement permits the fast neutrons from fission to leave the fuel and enter the moderator before significant resonance absorption occurs. The neutrons are then thermalized in the moderator before re-entering the fuel. This arrangement accomplishes two goals:

1. slowing down neutrons to thermal energy where the fission cross-section is significantly higher, and

2. minimizing neutron captures by:

    a) keeping the neutrons away from the U-238 while they are passing through resonance energy.

    b) returning the neutrons quickly to the fuel to reduce absorption in the moderator.

The channel spacing shown in Figure 7.2 is very important and is carefully chosen for CANDU reactors such that any significant increase or decrease in this spacing will decrease the probability of sustaining a chain reaction.

An important safety feature is that our fuel can only be made critical in heavy water in an arrangement similar to the one used. Thus there is no chance of a criticality accident in the handling or storage of CANDU fuel.

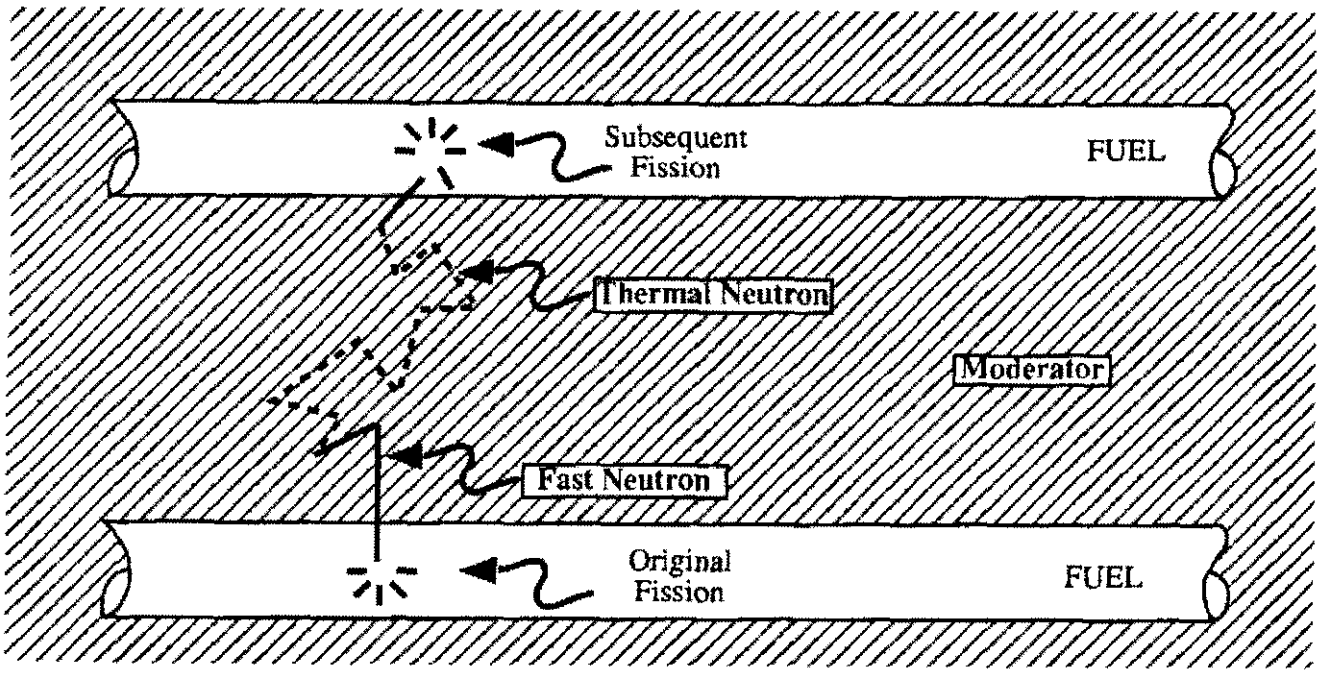


Figure 7.1: Axial Reactor Arrangement

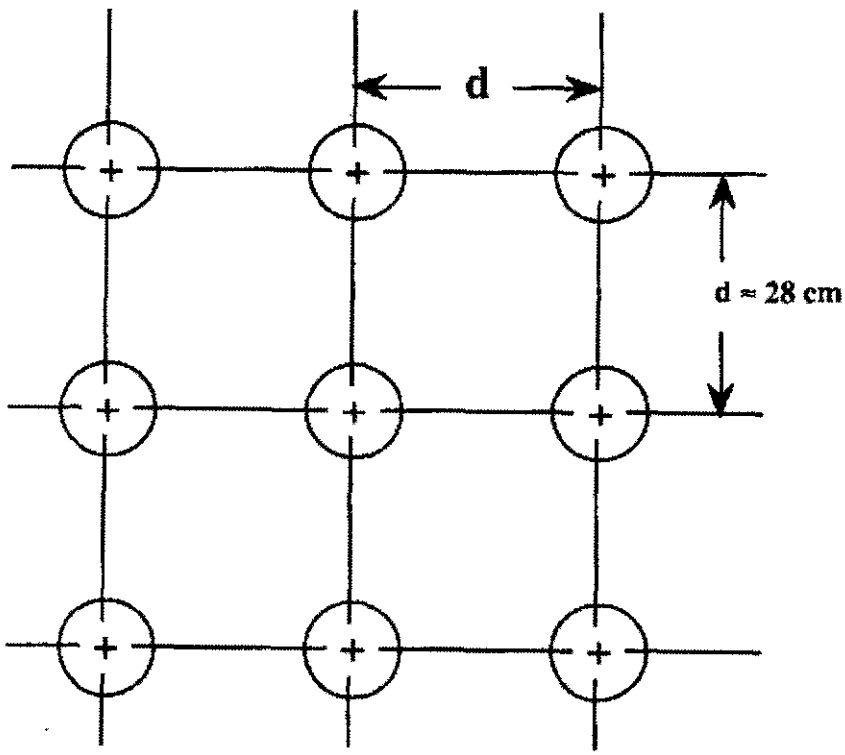


Figure 7.2: Radial Fuel Arrangement

ASSIGNMENT

1. Describe the arrangement of the fuel and moderator in a CANDU reactor.

2. Explain why heavy water (D_2O) is a better choice as a moderator than light water (H_2O).

3. Explain the differences between fresh and equilibrium fuel.

J. Crist
A. Broughton

NEUTRON LIFE CYCLE

THIS SECTION IS NOT REQUIRED FOR MECHANICAL MAINTAINERS

OBJECTIVES

At the conclusion of this lesson the trainee will be able to:

1. Sketch the life cycle of a neutron including all possible fates of the neutron.

2. Discuss why our reactors use reflectors.

NEUTRON LIFE CYCLE

In a CANDU reactor a prompt neutron has a lifetime of approximately 0.001 seconds. During this time it will travel about 25 cm while slowing to thermal energy. Once thermalized it diffuses about 30 cm before it is absorbed in the fuel. The time from birth as a fission neutron to absorption by the fuel is called the neutron lifetime. If a neutron is not absorbed in the fuel then it must be absorbed elsewhere, or escape into the shielding. Figure 8.1 give a pictorial view of the possible fates of a neutron.

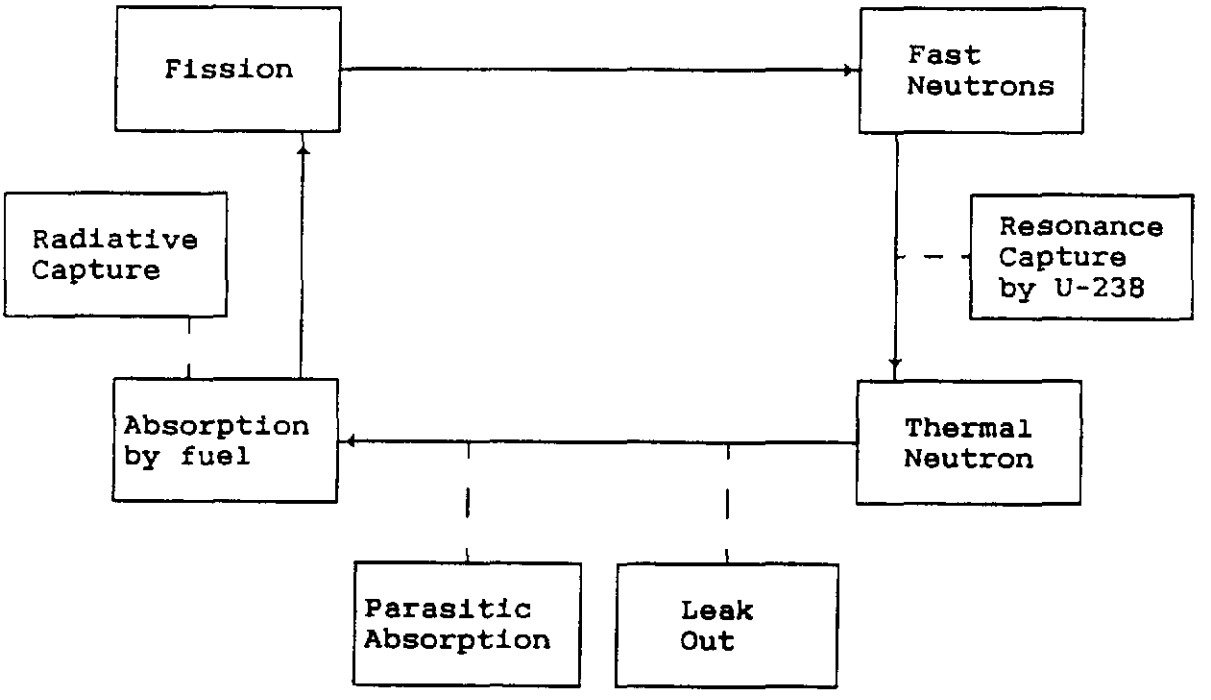


Figure 8.1: Neutron Life Cycle

We shall now examine each of the possible fates a neutron faces.

Resonance Capture

Recall from module 6 that in the energy range of about 10 eV to 1 keV U-238 has several extremely high absorption peaks with cross-sections as high as 6,000 barns. Virtually any neutron which returns to the fuel while it is slowed through these energy peaks will be absorbed.

In our reactor about 10% of the neutrons undergo resonance capture.

Leakage

If while travelling approximately 40 cm from birth to death a neutron reaches the boundary of the reactor it may "leak" out, never to return. Essentially three things influence how many neutrons actually leak out: size of the reactor, shape of the reactor, and what happens at the boundary. The designer can adjust these effects to minimize leakage into the shielding.

In our reactors leakage accounts for the loss of about 2.5% of the neutrons.

Size and Shape

Figure 8.2 shows three spherical reactors. Assuming that some neutrons travel 50 cm, a neutron born at any location in reactor 'A' has a possibility of escaping. As we increase the size of the reactor to 'B', the neutrons born inside the dotted circle normally won't leak before they are captured. By increasing the size again to 'C' a still smaller percentage of the neutrons can leak out.

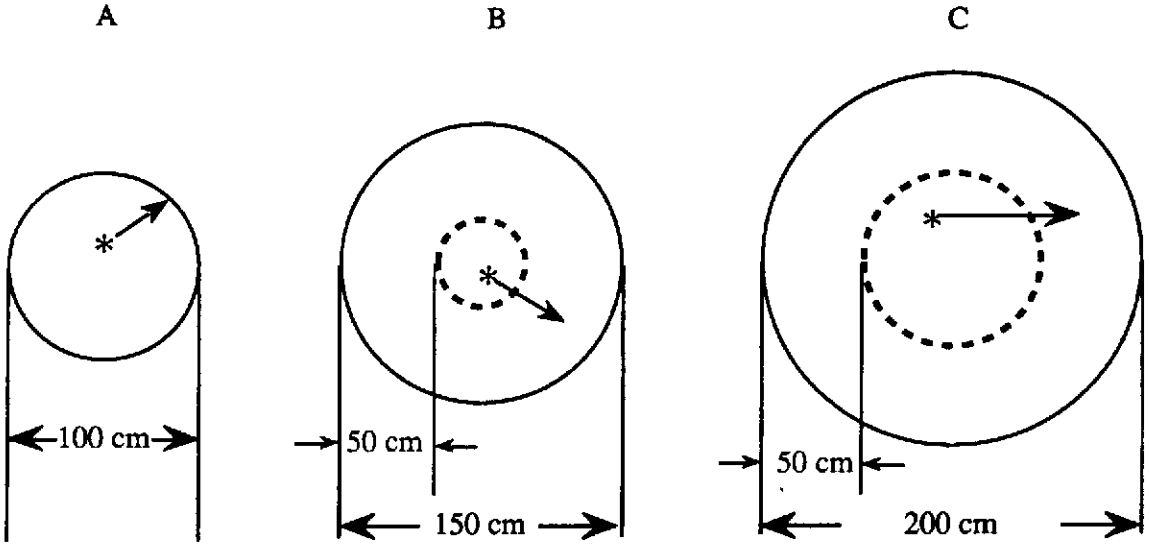


Figure 8.2: Effect of Size

Similar arguments can be made concerning the shape of a reactor. It can be shown that for a given volume of fuel and moderator a sphere will always have the smallest leakage. A sphere is not a practical shape from an engineering point of view. Instead we use approximately cylindrical reactors which have the diameter slightly bigger than the length. The actual shape is a compromise between engineering and nuclear considerations.

Reflectors

The final thing that affects leakage is what happens to a neutron when it reaches the boundary. If we surround the reactor with a material which will "bounce" some of the leaking neutrons back into the reactor, the loss due to leakage is reduced. We call this surrounding material a reflector.

An ideal reflector has a high probability of scattering neutrons and a low probability of absorbing them. Heavy water is an excellent reflector and is used in all our rectors. The reflector is merely an extension of the moderator as shown in Figure 8.3.

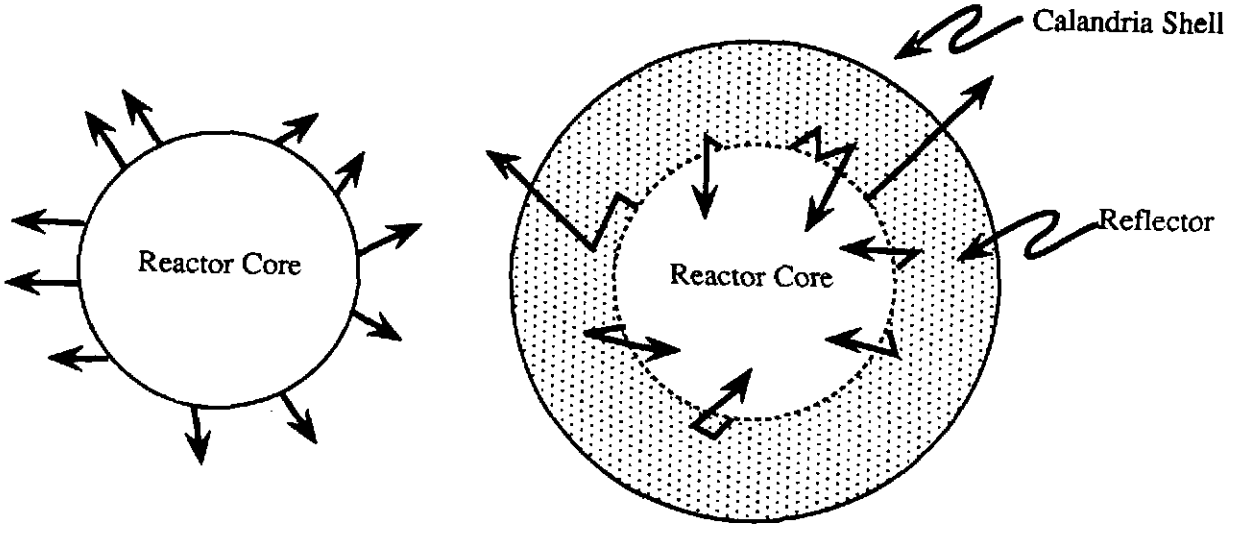


Figure 8.3: Reflector Location

The zone between the dotted line and the calandria shell serves as the reflector.

Parasitic Absorption

A neutron absorbed by something other than U-235 is unavailable to cause a fission. Neutrons can be absorbed by any of the following:

a) Fuel Sheath

b) Coolant, Moderator and Reflector

c) Pressure tubes and calandria tubes

d) Incore guide tubes

e) Various rods and control zone compartments

In total the materials on this list absorb about 5% of the neutrons, most of them in items (b) and (c).

Absorption by Fuel

The remaining neutrons are absorbed by the fuel which contains U-235, U-238, various isotopes of plutonium and a variety of fission products. Approximately 50% of the neutrons absorbed by the fuel simply undergo radiative capture. The remaining 50% cause fissioning of the U-235 and Pu-239.

The net result is that we get about 1.2 fast neutrons per thermal neutron absorbed by the fuel.

Overall Cycle

Roughly 20% of the neutrons are lost and do not return to the fuel. If about half of those remaining cause fissions (i.e. 40%) and each fission produces an average of 2.5 neutrons, this will restore the total to 100%. The cycle keeps going and the neutron number can be increased or decreased by adjusting one or more of the loss mechanisms.

ASSIGNMENT

1. Sketch the neutron life cycle.
2. Discuss each of the possible fates of a neutron.
3. Why do our reactors have reflectors?

J.E. Crist

CRITICALITY AND NEUTRON MULTIPLICATION

THIS SECTION IS NOT REQUIRED FOR MECHANICAL MAINTAINERS

OBJECTIVES

At the conclusion of this lesson the trainee will be able to:

1. Define the neutron multiplication constant (k).
2. Define reactivity (Δk) and state its common units.
3. Discuss what is meant by sub-critical, critical and super-critical in terms of the values of k and Δk and state whether power is increasing, decreasing or remaining constant.
4. State and understand that the reactor can be critical at any power level.
5. Given a method of criticality control, discuss how it affects the neutron cycle.

CRITICALITY AND NEUTRON MULTIPLICATION

In the chain reaction illustrated in Figure 9.1, only one neutron is available each time to cause fission. Therefore, the number of fissions occurring per second remains constant.

The power produced depends on the number of fissions per second (each watt requires 3.1×10^{10} fissions per second). If a reactor is producing one watt of power steadily, then 3.1×10^{10} fissions will occur each second. 3.1×10^{10} neutrons are available from these fissions to produce 3.1×10^{10} fissions during the next second, and so on. There is no multiplication of neutrons.

When the chain reaction is being maintained steady like this, the power level is steady and the reactor is said to be CRITICAL.

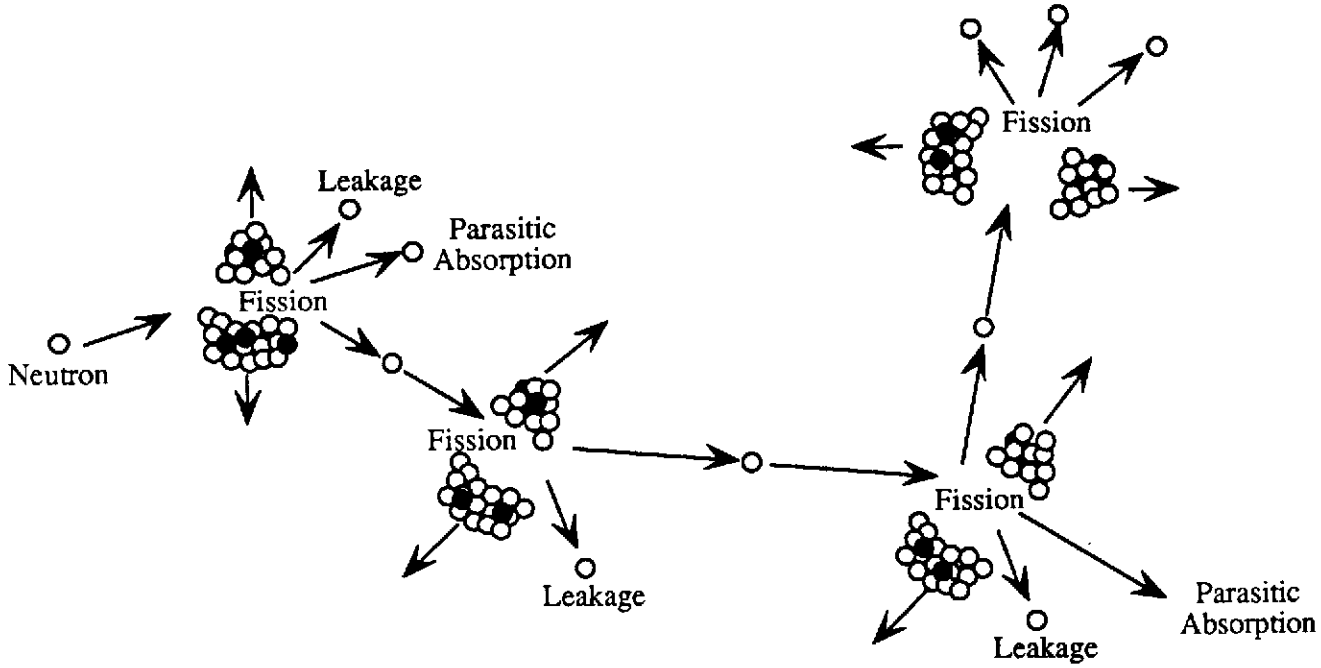


Figure 9.1: A Chain Reaction

The rate of neutron multiplication will not be constant if the power is being increased or decreased.

Neutron multiplication is conveniently expressed in terms of the neutron multiplication factor k based on the neutron cycle introduced in the preceding module.

$$k = \frac{\text{Number of neutrons in a generation}}{\text{Number of neutrons in the preceding generation}}$$

A nuclear reactor can operate with its power steady, increasing or decreasing. In order to show how these three different conditions can be described by the multiplication factor, let us suppose that to start with we have 100 neutrons, which is our first generation. Some of these 100 neutrons will be lost by absorption or leakage and the remaining ones will be available for fission. In a certain time (the generation time) these neutrons will cause fissions and neutrons of the second generation will be produced.

If $k = 1$, there will be 100 neutrons at the beginning of the second generation, 100 at the third, and so on, and fissions will continue at the same rate as at the beginning. The power will be steady and the reactor is said to be in the critical condition. Notice from this definition that the reactor may be critical at any power level.

If $k > 1$ (greater than one), say 1.05, the 100 neutrons of the first generation would produce 100 x 1.05, i.e, 105 neutrons at the beginning of the second generation. Again this would lead to a greater number of induced fissions and consequently to a larger neutron population. The number of neutrons would thus increase from one generation to the next. After 100 generations, for example, the number of neutrons present would be 13150 (100×1.05^{100}). The arithmetic is just like compound interest buildup in a daily interest bank account. A few neutrons could thus initiate a growing chain of fissions. The power would be increasing and the reactor is said to be super-critical.

In the above example, with $k = 1.05$, the power increased 131 times in about one tenth of a second. This is too fast a rate to control and in practice the multiplication factor is never allowed to become so large.

If $k < 1$ (less than one), 0.95 for instance, the number of neutrons would be reduced from 100 at the beginning to 95 in the second generation. In this situation, the original 100 neutrons would be reduced to one in about 90 generations (100×0.95^{90}). It is obvious that the chain reaction cannot be maintained under this condition. As the neutron population decreases, so will the number of fissions and the power decreases. The reactor is then said to be sub-critical.

Often a term called reactivity Δk is used in place of the neutron multiplication factor k. It is defined by the following equation:

$$k = 1 + \Delta k$$

k is always very near to 1 so Δk takes on small positive or negative values. We can say that the reactor is:

critical if $\Delta k = 0$,

super-critical if $\Delta k > 0$ (positive reactivity)

sub-critical if $\Delta k < 0$ (negative reactivity).

Reactivity is normally given in units of milli-k, where
$1 \text{ mk} = 10^{-3}$ k.

Example

given $k = 1.004$

$\Delta k = 1.004 - 1$

$= 0.004$ or 4 mk

It is important to stress that neither k nor Δk give any information concerning the power level in the reactor. They simply tell you whether the current power level is constant, increasing or decreasing.

Reactivity Control

Reactivity must be controlled for three basic reasons:

1. Maintain the reactor critical and the power level steady,

2. Increase or decrease power to match the demand,

3. Reduce power quickly in response to an upset.

There must always be excess positive reactivity available in case we need to raise power. Several things influence the excess reactivity such as the burnup of U-235, the production of Pu-239, the production of neutron absorbing fission products, and changes in the temperature of the fuel, coolant, and moderator. Before we look at how Δk can be adjusted we will discuss fuel burnup effects which cause slow long term reactivity changes. The fission product and temperature effects are discussed in separate modules later.

The effect of the burnup of U-235 and the buildup of Pu-239 is illustrated by Figure 9.2. The graph assumes a freshly fuelled (new) reactor at day zero.

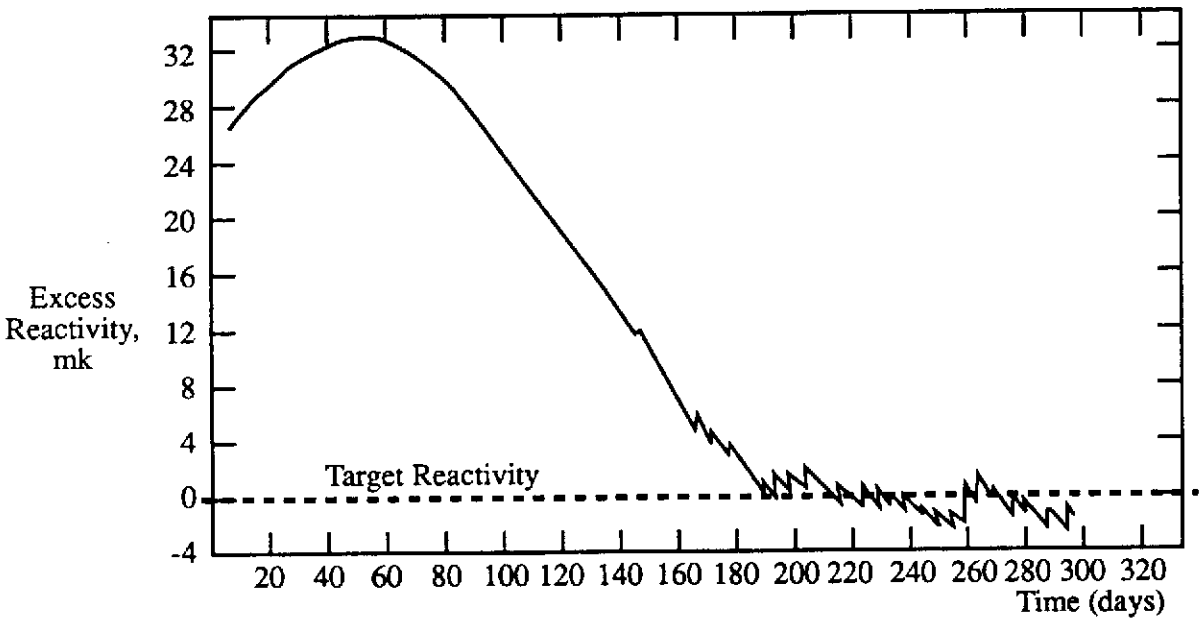


Figure 9.2: Excess Reactivity

As the reactor is operated at power, fissile atoms will be consumed causing reactivity to decrease. When the overall reactivity gets close to zero, fissile atoms must be replaced at the rate at which they are consumed (on-power refuelling).

The initial increase in reactivity worth occurs because Pu-239 is initially produced more rapidly than U-235 and Pu-239 are being consumed. The production of Pu-239 levels off after a while. The burnup of U-235 and Pu-239 is then higher than Pu-239 production and reactivity decreases. In operating the reactor we must adjust the reactivity to compensate for these reactivity changes.

There are three basic methods available to control reactivity:

1. Adjusting the amount of fissile material in the reactor.

2. Adjusting the amount of parasitic absorber in the reactor.

3. Adjusting the neutron leakage from the reactor.

Adjusting Amount of Fissile Material

If more U-235 is inserted into the reactor, more neutrons will be absorbed by U-235 compared to those absorbed in other materials. Thus inserting fissile material is an addition of positive reactivity (+ Δk). We do this in two ways:

1. On-power refuelling. (Used in all CANDU's.)

2. Booster rods (≈95% U-235, available only at Bruce A).

Adjusting the Amount of Parasitic Absorber

If a neutron absorbing material is introduced into the reactor, it will absorb neutrons which otherwise could have been absorbed by U-235. Thus insertion of absorbers adds negative reactivity (- Δk). One liquid absorber and three types of solid absorbers are used:

1. Liquid Zone Compartments (used in all CANDU's).

2. Adjuster Rods, made of cobalt or stainless steel, (used in all CANDU's except Bruce A).

3. Absorber Rods, made of cadmium or stainless steel, (used in all CANDU's except Pickering "A").

4. Shutoff Rods, made of cadmium encased in stainless steel, (used in all CANDU's).

Light water is used in the liquid zones. A tube is partially filled with light water. Increasing the water level causes more neutrons to be absorbed (- Δk). Decreasing the level causes fewer neutrons to be absorbed (+ Δk). Figure 9.3 shows a simplified sketch of a liquid zone control compartment.

The solid rods are all physically similar. Their names come from the specific purposes for which they are used.

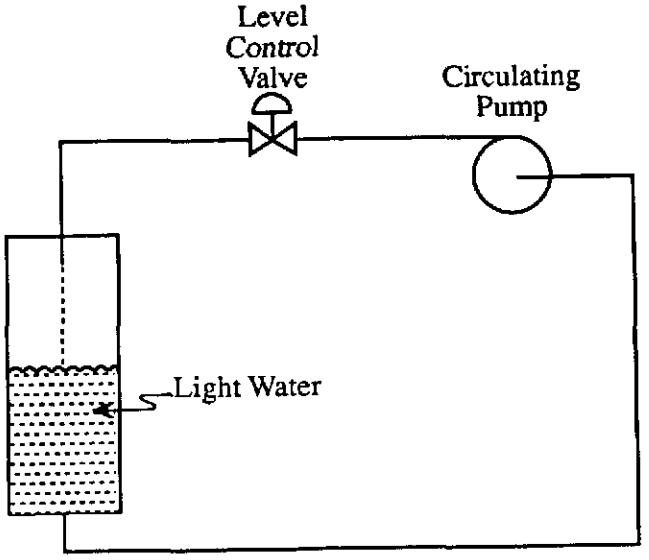


Figure 9.3: Liquid Zone Control

In addition to these absorption devices, parasitic absorption by dissolved neutron absorbers is used in two ways.

5. Neutron absorbers are dissolved in the moderator. The absorbers used, called poisons, are boron and gadolinium. These can be added gradually by the poison addition system or removed by the purification system to adjust Δk. All CANDU's use dissolved poisons.

6. All CANDU's (except Pickering "A") are able to inject a gadolinium solution rapidly into the core for a fast shutdown.

Adjusting Neutron Leakage (Not used except at Pickering "A")

If we can cause a larger fraction of the neutrons to leak out of the reactor, negative reactivity is inserted ($- \Delta k$). Leakage can be increased by lowering the level of the moderator in the calandria. This reduces the effectiveness of the reflector. Level control works by varying leakage: increasing level inserts positive reactivity, decreasing level inserts negative reactivity.

In addition the moderator can be dumped rapidly out of the core, stopping the fission process. As level drops leakage increases and unthermalized neutrons are less likely to be absorbed. They just leak away.

ASSIGNMENT

1. Define the neutron multiplication constant.

2. Complete the chart below.

	k $\begin{pmatrix} > 1 \\ < 1 \\ = 1 \end{pmatrix}$	Δk $\begin{pmatrix} + \\ - \\ 0 \end{pmatrix}$	Power $\begin{pmatrix} \text{Increasing} \\ \text{Decreasing} \\ \text{Constant} \end{pmatrix}$
Super-critical			
Critical			
Sub-critical			

3. If k = 0.997, find Δk in units of milli-k.

4. List the three basic methods of reactivity control and explain how each works.

J. Pachner
J.E. Crist

CHANGES IN REACTOR POWER WITH TIME

THIS SECTION IS NOT REQUIRED FOR MECHANICAL MAINTAINERS

OBJECTIVES

At the conclusion of this lesson the trainee will be able to:

1. Define Reactor Period.
2. Explain why and how delayed neutrons affect changes in reactor power.

CHANGES IN REACTOR POWER WITH TIME

The two preceding modules discussed how reactivity changes can be used to increase or decrease neutron flux and hence change the thermal power output from the fuel. We saw how the neutron population can change from one generation to the next.

The rate of change of power is the factor that determines how difficult a reactor may be to regulate, or whether it can in fact be regulated at all. This lesson will consider the rate of change of reactor power.

Effect of Neutron Lifetime on Changes in Reactor Power

We have seen how neutron density, neutron flux and reactor power increase or decrease over a number of generations. If $k > 1$ an initial power level of P_0 will increase to $P_0 k$ in one generation, to $(P_0 k) \times k$ in two generations, to $P_0 k^3$ in three and after N generations to $P_0 k^N$. This can be written:

$$P = P_0 (k)^N = P_0 (1 + \Delta k)^N \qquad (1)$$

This tells us that if we started with power P_0 and reactivity Δk, then N generations later the power would have changed to P, as given above. This gives the power change in terms of the number of generations that have elapsed, but not in terms of time.

The time t required for N generations to elapse is merely:

$$t = \ell . N \text{ or } N = \frac{t}{\ell} \qquad (2)$$

In this equation ℓ is the average time for one neutron generation. Under normal operating conditions $\ell = 0.1$ s for a CANDU reactor, as we shall show later. Equations (1) and (2) can be used to calculate the power increase in time t.

Example

Suppose reactor power is steady at 60% F.P. when $\Delta k = +0.5$ mk is inserted (i.e. $k = 1.0005$). How high will the power go in 100 seconds?

Solution

From (2)

$$N = 100 \text{ s}/0.1 \text{ s} = 1000 \text{ generations}$$

From (1)

$$P = 60\% \times 1.0005^{1000} = 60\% \times 1.65 = 99\%$$

Reactor Period

For mathematical convenience (especially in the days before calculators) equation (1) is usually written in a different way:

$$P = P_0 e^{t/T} \qquad (3)^1$$

where the constant T is called the reactor period.

In practical terms, to get an idea of how fast power is changing, we could talk of the length of time it takes for the power to double, or increase ten-fold or whatever. Due to the nature of equation (3), it is simplest to think in terms of the length of time it takes for the power to change by a factor of e. This is our definition of reactor period. In equation (3), for the power to increase by a factor of e, i.e., $P = eP_0$, the time t must equal the reactor period T.

For small values of reactivity (Δk) encountered in normal operation, equation (1) and (3) will give identical results provided:

$$T = \ell/\Delta k \qquad (4)$$

Using values from our earlier example T = 0.1/0.0005 = 200 s

$$P = 60\% \times e^{100/200} = 60\% \times e^{0.5} = 99\%$$

Note that the larger Δk is the shorter the reactor period becomes and the faster the power changes will be.

The Effect of Delayed Neutrons on Power Changes

For fission of U-235, 99.35% of the neutrons produced are prompt neutrons, and 0.65% are delayed neutrons emitted by fission products. The time for one generation of prompt neutrons is 0.001 s. The average lifetime of the delayed neutrons is a little over 13 seconds. The average lifetime, ℓ, for all the neutrons, prompt and delayed is then:

$$\ell = 0.9935 \times 0.001 \text{ s} + 0.0065 \times 13 \text{ s} = 0.085 \text{ seconds}$$

For the sake of simplicity we usually round off the value of ℓ to 0.1 s, as was done in the earlier example.

---

[1] For mathematicians in the group, to show (3) and (1) are the same you need to know that $\ell n(1 + \Delta k) = \Delta k$ for small values of Δk.

Although the delayed neutrons represent only a small fraction (0.65%) of the neutrons generated by fission, they increase the average lifetime of all neutrons, ℓ, from 0.001 s to 0.085 s, i.e., by a factor of 85. From formula (4) for reactor period we see this makes the period 85 times longer than it would be for ℓ = 0.001 s. The initial rate of power rise will also be reduced by a factor of 85.

In summary, the effect of the delayed neutrons is to make the rate of power changes reasonably slow for small additions of positive reactivity. It is the delayed neutrons which make regulation and protection practical.

The Effect of Prompt Neutrons Alone

The formulas for reactor period (4) and for power change (1 and 3) accurately predict power changes provided Δk is a small value, typical of reactivity additions used in normal reactor regulation. These formulas do not work at all for large $+\Delta k$ insertions such as would be used to calculate possible upset or accident conditions.

The behavior of a reactor when large amounts of positive reactivity are suddenly inserted was tragically demonstrated by the Chernobyl reactor. In that accident (discussed in the Reactor Safety course) power increased from a low level to an estimated 10 000% full power in less than 2 seconds. Why didn't the delayed neutrons limit the rate of power increase? In the remainder of this chapter we will describe the effect (or non-effect) of delayed neutrons in more detail, to be able to answer this question.

Consider first the role of the delayed neutrons in a constant power reactor ($k = 1$). In the core 99.35% of the neutrons are prompt and 0.65% are delayed. Suppose that somehow we could "shut off" the delayed neutrons. Starting with 100 neutrons, after one generation this would drop to 99.35 (since we are assuming the delayed neutrons are not showing up). In the second generation this drops to 98.7 and by the third generation it has dropped to 98. The power is decreasing as if the reactor is sub-critical.

In fact, the reactor depends on the arrival of the delayed neutrons to "top up" the neutron population and stay critical. When $+\Delta k$ is added, as long as Δk is not too big, the power cannot rise very quickly until the extra delayed neutrons from the extra fission products at the higher power level begin to show up, and this takes several seconds. The slow arrival of the delayed neutrons controls the rate of power increase.

Figure 10.1 illustrates the power increase for a reactivity of + 0.5 mk considering only promt neutrons and considering delayed neutrons.

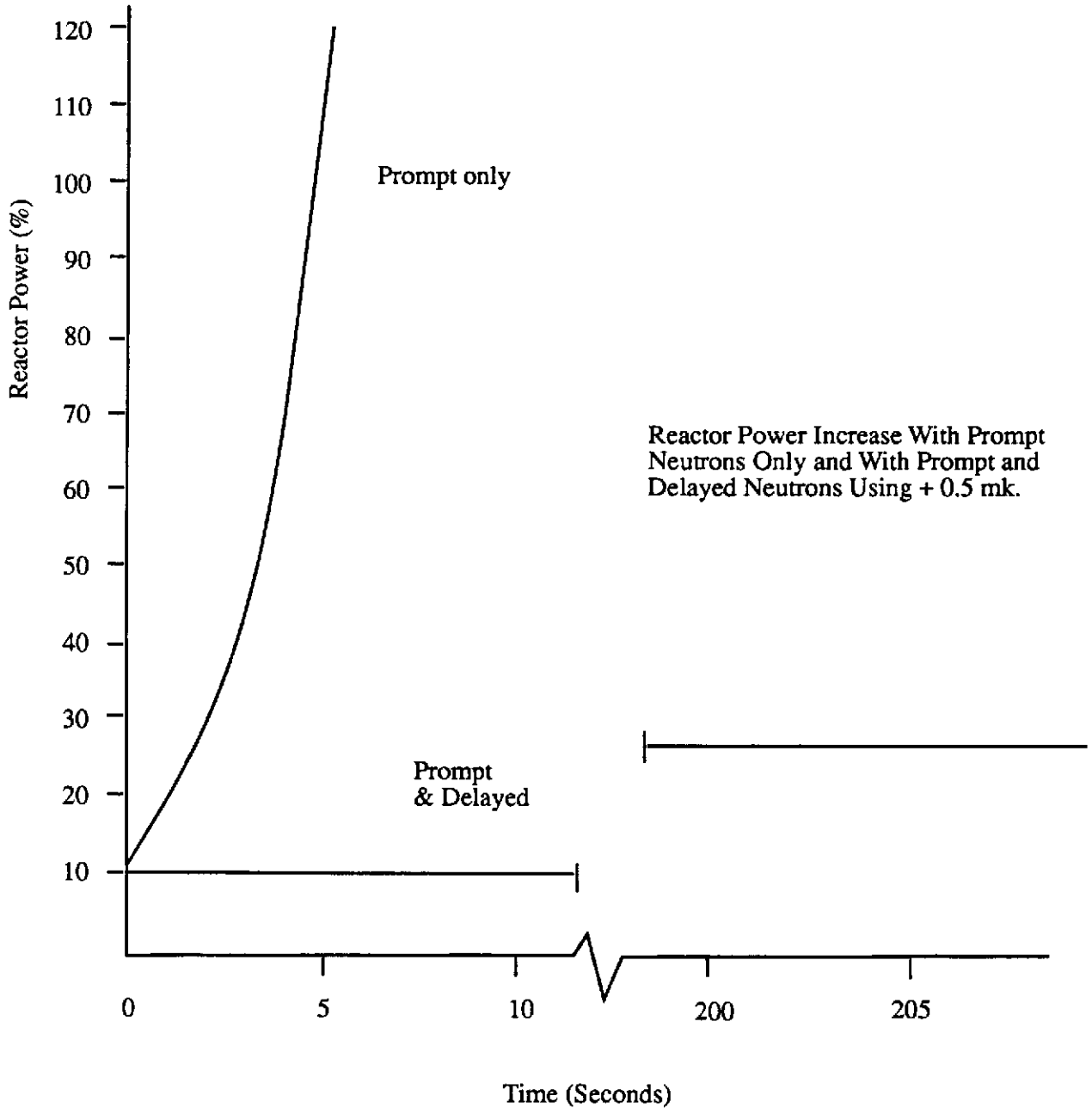


Figure 10.1

Now suppose a large $+\Delta k$ is inserted in the reactor core. The prompt neutrons (multiplied by k) will increase enough from generation to generation that the power increases even without the delayed neutrons. The prompt neutron population "takes over" and power rises as though the neutron generation time is $\ell = 0.001$ s, the lifetime of the prompt neutrons, and not $\ell = 0.085$ s, the average lifetime we used before.

The behavior of such a reactor can be illustrated using the earlier example with $\ell = 0.001$ s. Also we will consider the power increase in one second instead of one hundred seconds.

With a positive reactivity of 0.5 mk, the reactor period would be given by:

$$T = \frac{\ell}{\Delta k} = \frac{0.001}{0.0005} = 2 \text{ seconds.}$$

In one second, the power would increase as given by equation (3), i.e.,

$$P = P_0 e^{t/T} = P_0 e^{\frac{1}{2}} = P_0 \times 1.65$$

For $P_0 = 60\%$ this gives a power rise to almost 100% in 1 s instead of 100 s.

This example shows how rapid the power increases would be, even for small reactivity changes of the order of a mk, if all the neutrons were prompt neutrons.

Effective reactor regulation would not be possible under these circumstances, because the changes to be regulated would be far too fast. Emergency shut-down of the reactor would be an even greater problem, since even very fast protective systems need of the order of a second or two to become effective. In this relatively long period of time, severe damage would result from the excessive power levels reached.

Figure 10.1 illustrates the power increase for a reactivity of + 0.5 mk considering only prompt neutrons and considering delayed neutrons.

ASSIGNMENT

1. Define reactor period.

2. Explain why delayed neutrons are important for reactor control.

J.U. Burnham
J.E. Crist
J.L. Groh

XENON, A FISSION PRODUCT POISON

THIS SECTION IS NOT REQUIRED FOR MECHANICAL MAINTAINERS

OBJECTIVES

At the conclusion of this lesson the trainee will be able to:

1. Explain why xenon is the most important fission product poison.
2. Explain how xenon is produced in, and how it is removed from the reactor.
3. Sketch xenon concentration as a function of time for a shutdown/trip from full power.
4. Discuss "xenon poison out".

XENON, A FISSION PRODUCT POISON

Many fission products will absorb neutrons. Most have small absorption cross sections and can be regarded as insignificant in short term operation. However, Xenon-135 has a cross section of approximately 3,000,000 barns, over 4000 times that of U-235. About 6.6% of all fissions produce a nuclide of Xenon-135, either directly as a fission fragment or indirectly as a fission product daughter. Because of its large yield and its ability to absorb neutrons, xenon is a major problem in our reactors.

Xenon Production

Xenon is produced directly from fission. 0.3% of the fission fragments are Xe-135. It is also produced by the decay of iodine via the following decay scheme:

$$^{135}_{53}\mathrm{I} \xrightarrow[t_{1/2} = 6.6\ \mathrm{hr}]{} \beta + {}^{135}_{54}\mathrm{Xe}$$

Iodine-135 constitutes 6.3% of all fission fragments. Thus, approximately 95% of the total production of xenon is due to the decay of iodine. (6.3/6.6 = 0.95). Iodine-135 does not absorb neutrons.

Xenon Loss

Xenon is "removed" from the reactor by decay:

$$^{135}_{54}\mathrm{Xe} \xrightarrow[t_{1/2} = 9.1\ \mathrm{hr}]{} {}^{135}_{55}\mathrm{Cs} + \beta$$

or by neutron absorption:

$$^{135}_{54}\mathrm{Xe} + {}^{1}_{0}\mathrm{n} \rightarrow {}^{136}_{54}\mathrm{Xe} + \gamma$$

For CANDU reactors at full power neutron absorption constitutes about 90% of the loss of Xe-135. Neither Xenon-136 or Cesium-135 is a neutron absorber.

Equilibrium Xenon Load

When a reactor has been shutdown for a long time (or has never been operated) there is no xenon present. After the reactor is started up, the xenon will slowly build up to an equilibrium level which will depend on the steady state power of the reactor. Figure 11.1 is a graph of xenon load in milli-k versus time for various power levels. For CANDU reactors the full power xenon load will build up to approximately -28 mk in about 50 hours.

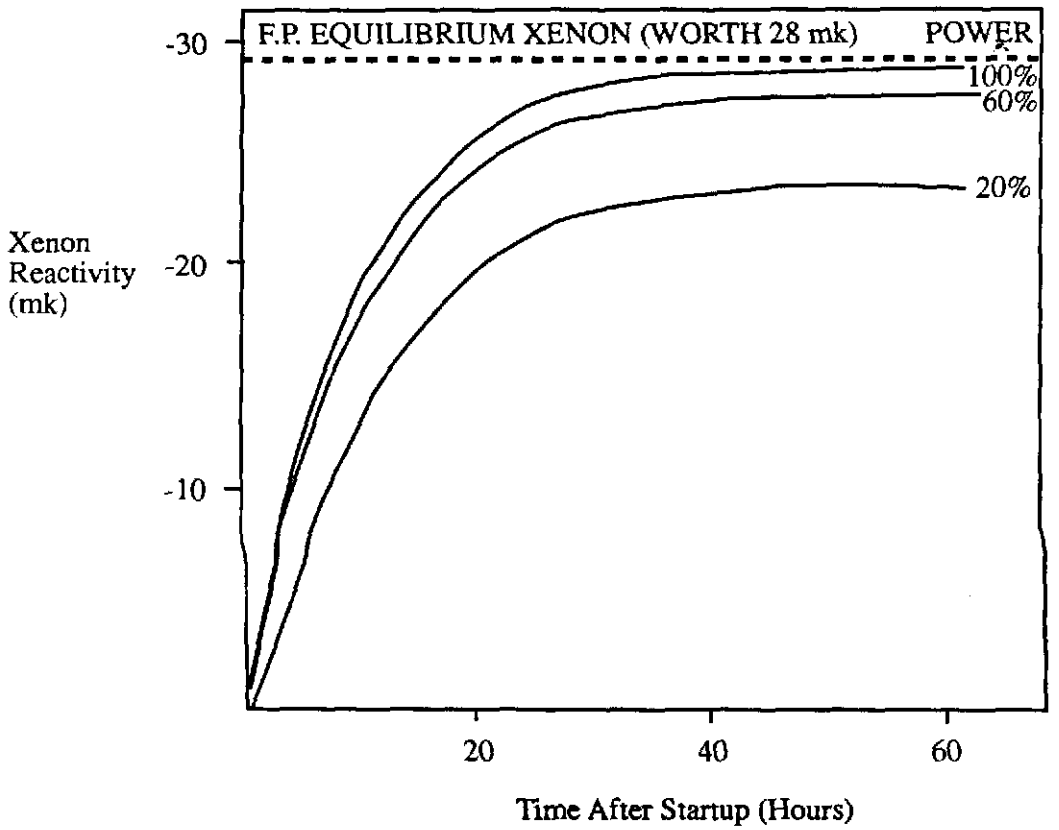


Figure 11.1: Xenon Buildup to Equilibrium

This negative reactivity (-28 mk) will always be present in normal steady operation, except during the first several hours after startup. There is enough excess positive reactivity designed into our reactors to compensate for the -28 mk load. This excess reactivity creates an operational problem in the first hours of operation when the xenon is not present.

Negative reactivity worth must be added to the reactor on startup to compensate for the lack of xenon. To accomplish this a soluble poison (Boron or Gadolinium) is added to the moderator. As xenon levels increase the poison is removed by burnout or ion exchange purification.

Xenon Transients

After operating for around 50 hours, xenon has built up to it's equilibrium level. It causes little problem unless the reactor power level is changed. Consider what happens to the production and loss of xenon following a reactor trip (a fast power reduction to 0%).

a) Production

- from fission (5%) - stops immediately
- from decay of iodine (95%) - continues

b) Loss

- by decay (10%) - continues
- by neutron absorption (90%) - stops immediately

Thus, immediately following a trip, xenon production continues virtually undiminished while xenon loss is drastically reduced. Figure 11.2 is a graph of xenon load versus time after a trip from full power.

Note the negative reactivity from xenon peaks approximately 10 hours after shutdown, at a level considerably higher than the equilibrium level.

Each reactor has a maximum excess positive reactivity (15 to 20 mk) which can be achieved by withdrawing adjuster rods (inserting booster rods at Bruce "A"). If the negative reactivity due to xenon exceeds this, the reactor will be sub-critical with no way to restart it. It is said to be "poisoned out". Figure 11.3 shows the xenon transient reactivity and the maximum reactivity available.

The reactor will be poisoned out 30 to 40 minutes after a trip and remain shutdown for up to 40 hours. If the reactor is started up during the 30 to 40 minute poison override time and brought up to power before poisoning out, the xenon will be burned up rapidly and a poison out may be prevented.

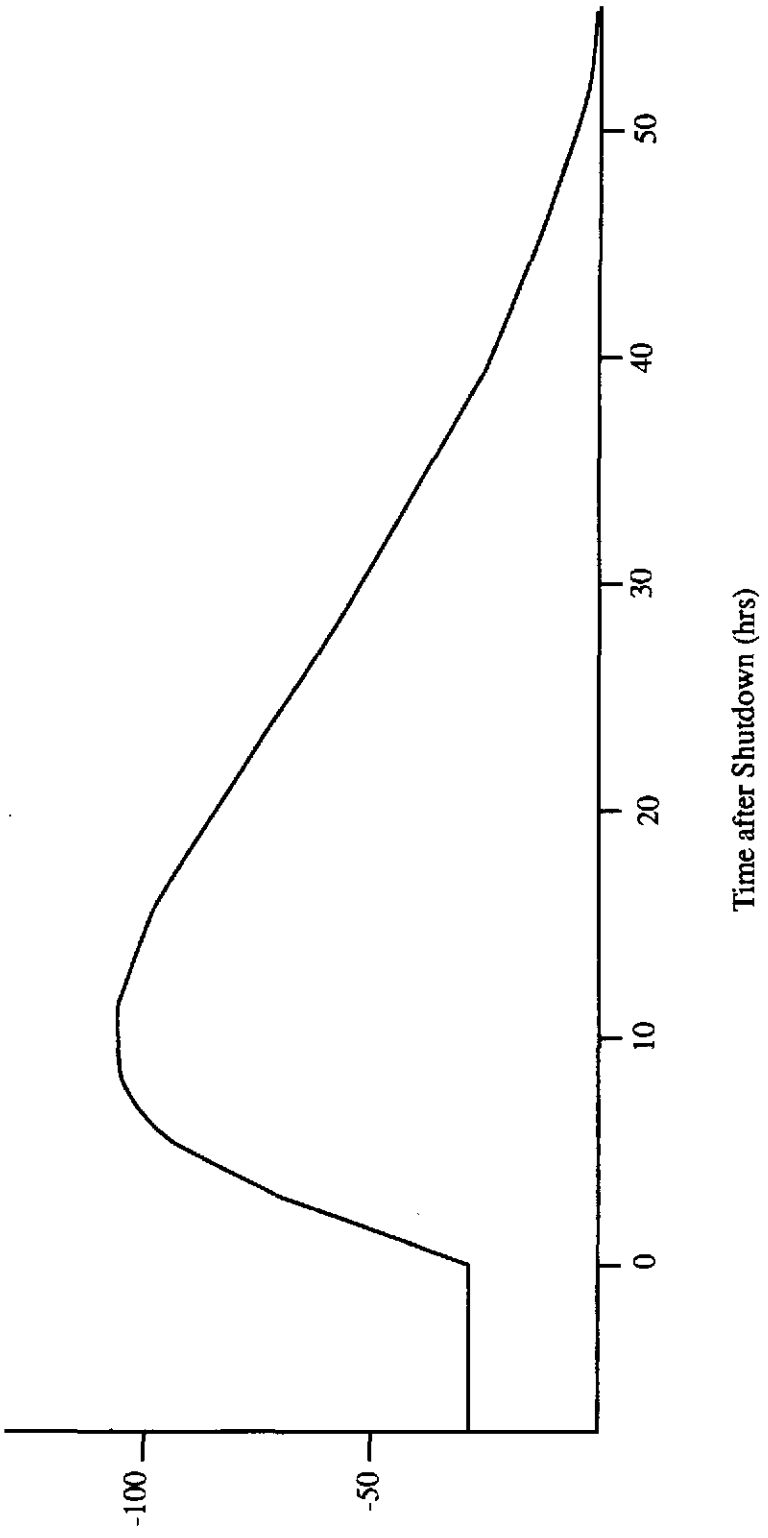


Figure 11.2: Behaviour of Xenon

Any reduction in power level will cause the xenon to peak; however, the smaller the power reduction the smaller the xenon peak. For example a power reduction of 40% from 100% to 60% still leaves a significant neutron flux available to burn out the xenon thereby reducing the magnitude of the peak. Figure 11.3 shows a typical reactivity variation for a power reduction to 60%.

It is important to realize that on a turbine trip it is economically sound to exhaust steam to atmosphere or a condenser in order to maintain reactor power greater than 60% thereby preventing a poison out. This mode of operation is called 'poison prevent'.

On a power increase after steady low power operation (say from 60% to 100%) the reverse effect occurs. Xenon burns out rapidly while production from iodine decay continues low. Reactivity increases and the control system must add negative reactivity to compensate.

In a large high flux reactor it is possible to have flux increasing in one part of the reactor while it is decreasing elsewhere. This operational problem caused by xenon is discussed in module 13.

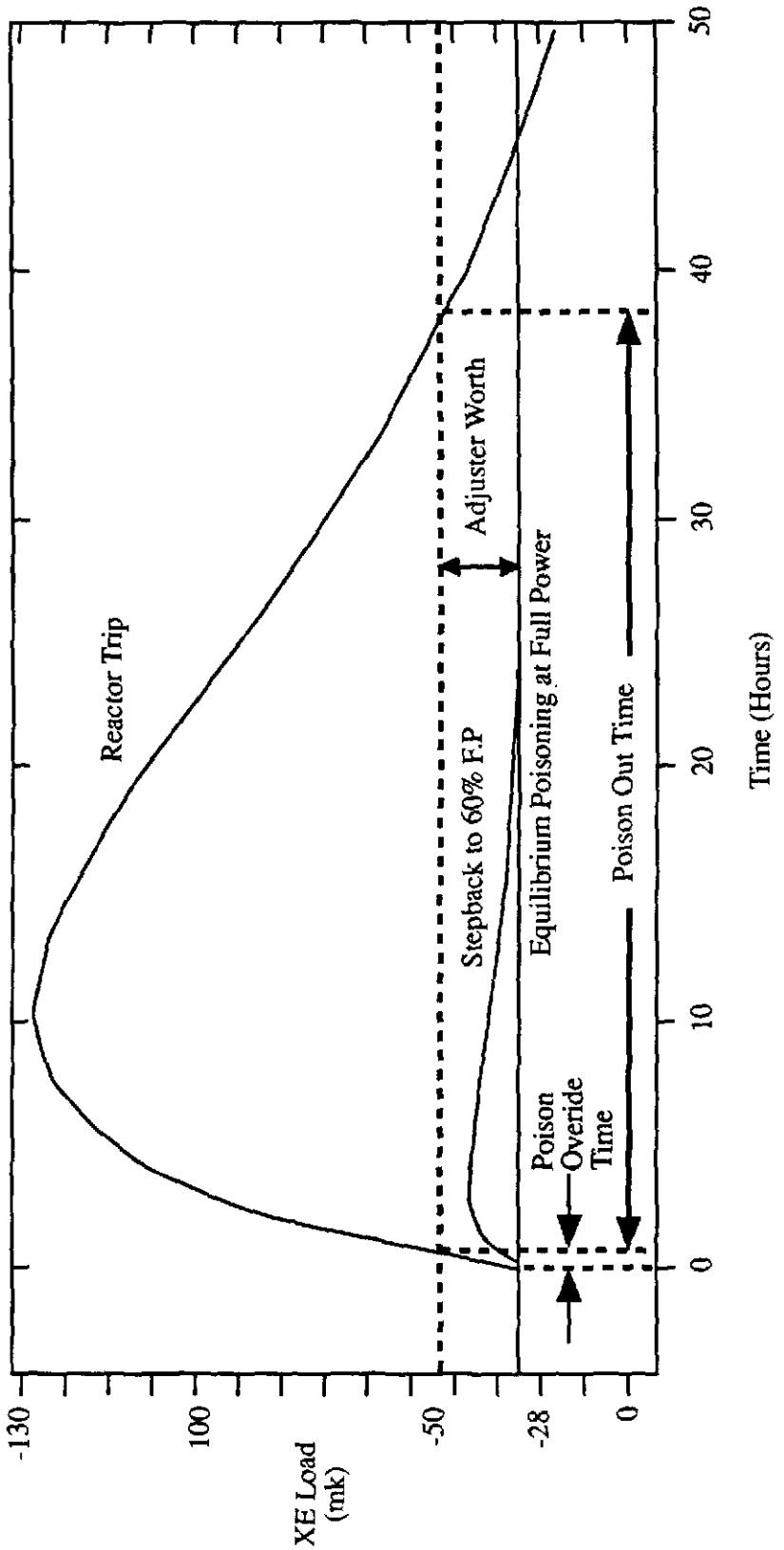


Figure 11.3: Transient Xenon Reactivity

ASSIGNMENT

1. Sketch the behavior of xenon on a reactor trip from full power.

2. Explain why a xenon poison out occurs.

3. Discuss the production and loss of xenon including the relative magnitude of each term.

J.E. Crist

REACTIVITY EFFECTS OF TEMPERATURE CHANGES

THIS SECTION IS NOT REQUIRED FOR MECHANICAL MAINTAINERS

OBJECTIVES

At the conclusion of this lesson the trainee will be able to:

1. Define:
   a) temperature coefficient of reactivity
   b) void coefficient
   c) power coefficient
2. Explain why and how reactivity changes when the temperature of the fuel changes.
3. Explain why a negative temperature coefficient is desirable.

REACTIVITY EFFECTS OF TEMPERATURE CHANGES

The NRX Experiment

In 1949, the NRX reactor at the Chalk River Nuclear Laboratory was allowed to "run away". NRX is a heavy water moderated experimental reactor which uses control rods for reactor regulation. The heavy water level was set 3 cm above the height at which the reactor would be critical at low power with the rods withdrawn. The reactor power was allowed to increase unchecked, and the manner in which it increased is rather unexpected (see Figure 12.1).

The power initially increased. However, it did not increase indefinitely as you might have expected. As the temperature of the fuel rods increased, the reactivity decreased and this caused the rate of power increase to slow down. Later the reactivity decreased at a faster rate as the heavy water got warmer. The total decrease in reactivity was enough to make the reactor sub-critical, and the end result was that the power reached a maximum value and then started to decrease.

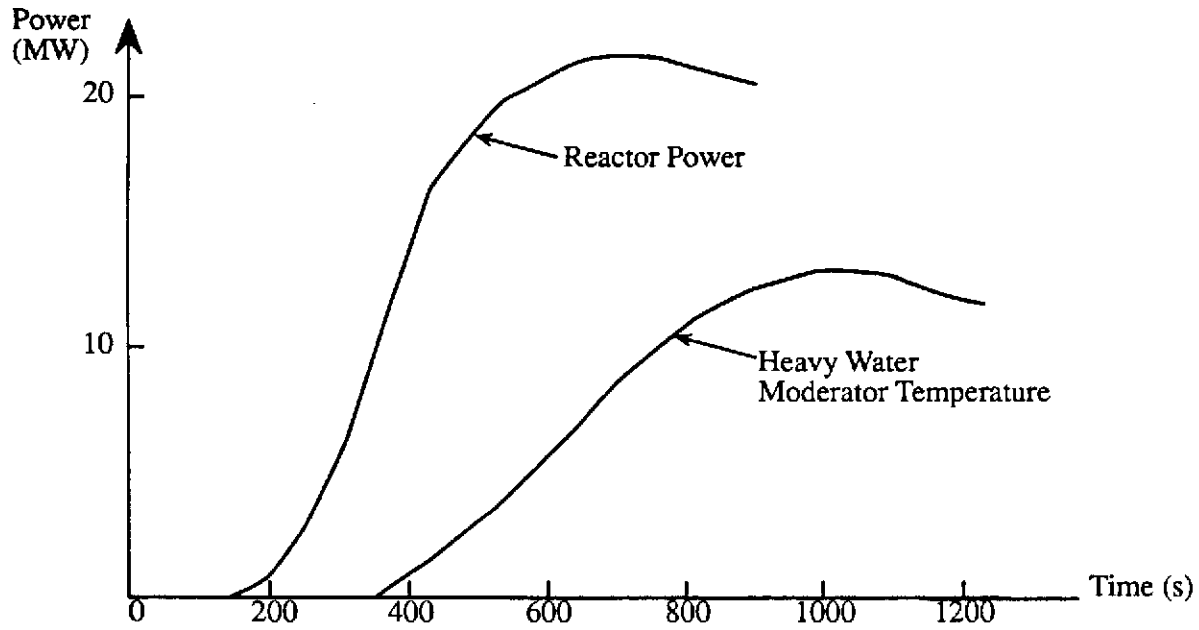


Figure 12.1: The NRX Experiment

Thus for small positive reactivity insertions the reactor is self-regulating because the temperature increases reduce reactivity, preventing the power from continuing to increase. In this experiment the initial excess reactivity was quite small. If more reactivity had been inserted initially it is quite possible that the power would have continued to rise. The point of this example is not to demonstrate that reactor power would never increase continuously (it well might), but to show that there is a loss in reactivity due to the increase in the temperatures of fuel and heavy water.

The temperature coefficient of reactivity is defined as the change in reactivity per unit increase in temperature. Its units are mk/°C. This coefficient may be positive or negative. In the example just described, it was negative because an increase in temperature led to a loss of reactivity.

Temperature changes occur, more or less independently, in the fuel, the heat transport system and the moderator, and there will therefore be a temperature coefficient of reactivity associated with each of these. It is very desirable for the overall temperature coefficient of a reactor to be negative to provide the self-regulating feature illustrated by NRX.

In this course only the effects caused by the temperature changes in the fuel will be examined in detail.

The total change in reactivity due to normal coolant temperature changes is small and an explanation of the effects causing it is beyond the scope of this course. Moderator temperature changes can cause reactivity changes but we generally control the moderator temperature constant. Furthermore, changes in moderator temperature occur at a relatively slow rate (because of the large amount of D_2O) and therefore the effects are not nearly as immediate as those in the fuel.

Fuel Temperature Coefficient of Reactivity

There are two primary causes of the fuel temperature coefficient:

1. Increasing fuel temperature causes increased resonance capture in U-238.

2. The ratio of fissions to absorptions in the fissile material changes with fuel temperature. (The direction and magnitude of change depend on whether the reactor is at fresh or equilibrium fuel).

Both of these effects will be examined.

Increased Resonance Absorption

In lesson 6 the presence of certain resonance absorption peaks in U-238 was discussed. The width and height of these peaks depend on the temperature of the U-238. Figure 12.2 shows one particular resonance peak at 20°C and 800°C. At the higher temperature, the peak is lower but covers a wider range of energies. Even the low peak is high enough that virtually any neutron within this energy range will be captured. Therefore, increasing the fuel temperature increases resonance absorption by increasing the range of energies which can be captured. Resonance absorption is always the most important fuel temperature effect.

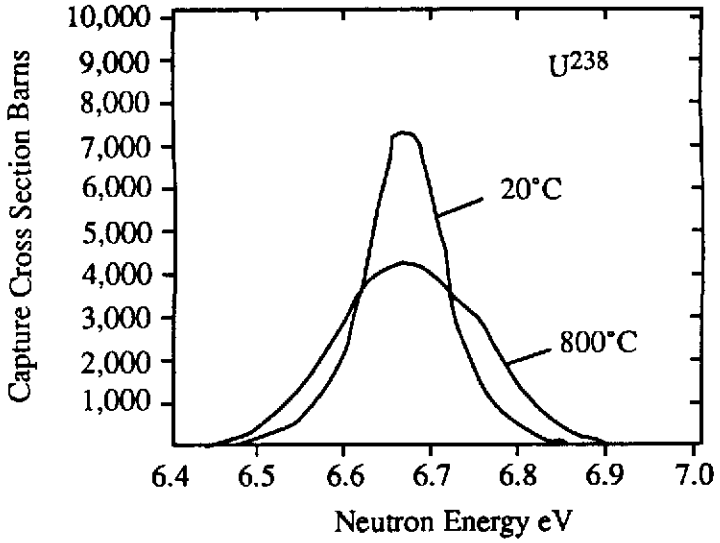


Figure 12.2: Resonance Broadening

Ratio of Fissions to Absorptions

In an earlier lesson the variation of neutron cross-section with temperature was discussed. Consequently the ratio of fissions to absorption $\frac{\sigma f}{\sigma a}$ may be expected to change if the speeds of the thermalized neutrons change. For U-235 this ratio goes down with increasing temperature. For Pu-239 the ratio increases. Thus, for fresh fuel where U-235 is the only fissile fuel, increasing the temperature tends to decrease reactivity. In equilibrium fuel where Pu-239 is a significant portion of the fissile material, increasing the fuel temperature tends to increase reactivity.

The overall change in reactivity is a combination of both these effects. The resonance absorption effect is always larger than the ratio effect. Clearly for fresh fuel, the fuel temperature coefficient of reactivity is negative, since both effects are negative. A typical value is -0.013 mk/°C. For equilibrium fuel, the effect of the plutonium partly cancels the resonance effect and reduces the magnitude of the fuel temperature coefficient to about -0.004 mk/°C.

Power Coefficient

Operationally the reactor power is measured and fuel temperature is not. It is convenient to define the "power coefficient", the overall effect of temperature increases. The power coefficient is defined as "the change in reactivity due to temperature effects when power is increased from 0% to 100%". A typical value for CANDU reactors is ≈-5 mk, a decrease in reactivity. The exact value depends on the reactor design and fuel condition. For Bruce NGS the values are about -9 mk for fresh fuel and -3.5 mk for equilibrium fuel.

Void Coefficient

If boiling occurs in a coolant channel, steam will gradually displace the coolant. This effect is called "voiding". If a channel is partially or totally voided there will be a number of reactivity effects. An exact discussion of most of the effects is beyond the level of this course. The void coefficient is defined as "the change in reactivity for 100% voiding of all coolant channels". It is always positive for our reactors. The actual value varies from reactor to reactor but is about +10 mk for the Bruce reactors.

10 mk is a very large positive reactivity. It should be noted that this definition is "theoretical" in that it is not ordinarily possible for all the coolant to flash rapidly to steam. Even on a large pipe break the rate of voiding is limited. The safety systems are designed to detect and stop the power rise before the +10 mk of reactivity is inserted.

ASSIGNMENT

1. Define:

   a) temperature coefficient,

   b) power coefficient,

   c) void coefficient.

2. Discuss the reasons why the fuel temperature coefficient is negative and why the magnitude is lower for equilibrium fuel than it is for fresh fuel.

3. Why is a negative temperature coefficient desirable?

J.E. Crist

NEUTRON FLUX CONTROL

THIS SECTION IS NOT REQUIRED FOR MECHANICAL MAINTAINERS

OBJECTIVES

At the conclusion of this lesson the trainee will be able to:

1. Explain why a flat flux distribution is desirable.
2. Explain how each of the methods used in CANDU reactors flattens the flux.
3. Explain what flux oscillations are and how liquid control zones are used to prevent them.

NEUTRON FLUX CONTROL

If nothing is done to flatten the flux in our reactors it would look something like the distribution shown in Figure 13.1. The flux would be a maximum in the center of the reactor (where neutrons are moving in from all directions) and decrease toward the boundaries (where neutrons are escaping into the shielding).

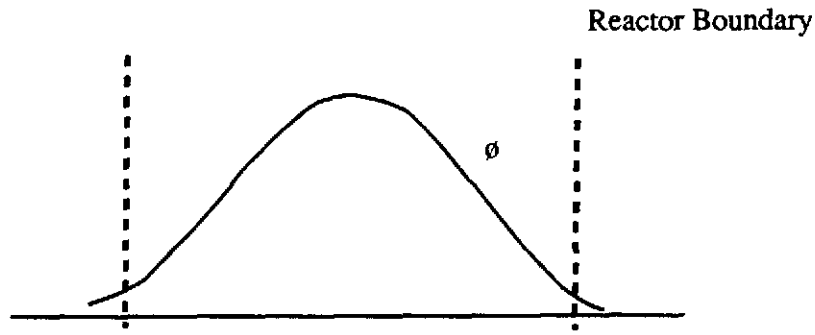


Figure 13.1: Unflattened Flux Distribution

Unfortunately with a distribution like this, the average flux is only about 30% of the maximum flux. This means that when the fuel bundle in the center of the reactor is producing the maximum power it can safely produce, the reactor is producing only 30% of the power it could safely produce if the flux were evenly distributed ($\emptyset_{avg} = \emptyset_{max}$).

While it is not possible to achieve a completely flat flux distribution, CANDU reactors achieve an average flux which is about 60% of the maximum flux, using the methods we are about to discuss.

You should appreciate that increasing the average flux without increasing the maximum flux has enormous economic benefits. For example, without flattened flux, Pickering NGS would be producing only half the power if now produces for roughly the same capital investment.

Reflectors

In module 8 we noted that adding a reflector to our reactors reduced leakage. That is only part of the advantage of reflectors. They also help flatten the flux distribution in the radial direction. Figure 13.2 shows the flux distribution in a reactor without a reflector and with a reflector added. With the same maximum flux, (limited by the maximum allowed power level for the fuel) the average flux has been increased due to the neutrons reflected back into the reactor.

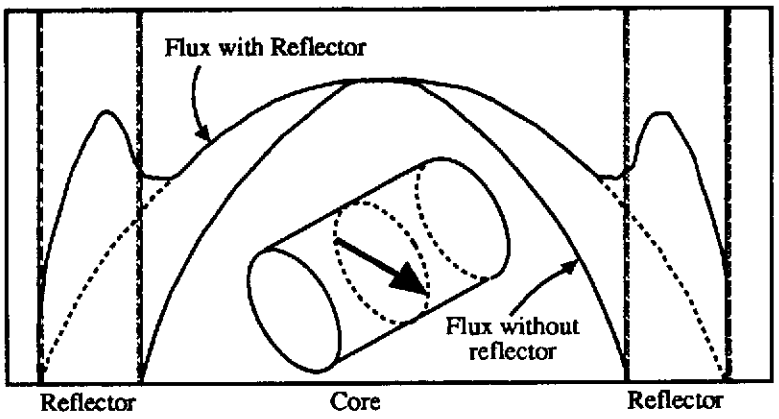


Figure 13.2: Effect of Adding a Reflector

Bi-Directional Fuelling

If adjacent fuel channels are fuelled in opposite directions as they are in all of our reactors an automatic flux flattening effect arises in the axial direction. This is shown in Figure 13.3.

We do not change all the fuel bundles when a channel is refuelled so the newer fuel (at the input end of the channel) generates more neutrons than the highly burned up fuel at the exit end. How much flattening is obtained in this way actually depends on how many bundles are fuelled in each visit to the channels. The less the better, from this point of view.

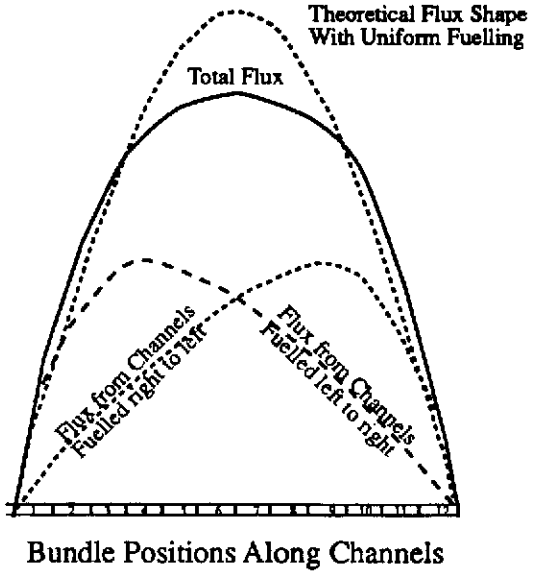


Figure 13.3: Effect of Bi-Directional Fuelling

Adjuster Rods

18 Adjuster rods are normally fully inserted in the central regions of the core to absorb thermal neutrons and in this way depress or 'adjust' the flux both radially and axially. Figure 13.4 shows the basic effect adjuster rods have on flux distribution. (Note: Bruce A reactors do not have adjuster rods).

Flux flattening with the use of these rods is quite effective but it does represent a loss in fuel burnup. We accept this because the benefits of increased power production greatly outweigh the higher fuel cost.

Several of the Ontario Hydro reactors use cobalt as the neutron absorbing material in the adjuster rods. The adjuster rods are replaced periodically, and the cobalt-60 is processed and marketed by AECL.

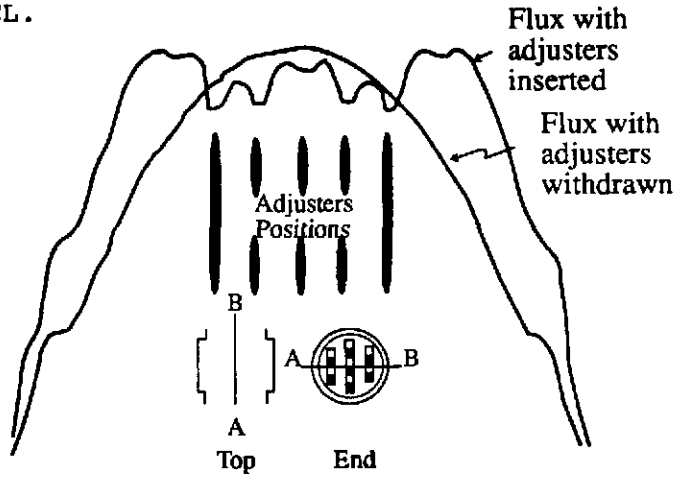


Figure 13.4: Effect of Adjuster Rods

Differential Fuelling

A different method of flux flattening was chosen for the Bruce "A" reactors, which use boosters rather than adjusters for xenon override. Differential fuelling means that the bundles in the central channels are left to reach higher than average burnups while bundles in the outside channels are removed at lower burnup. The central bundles therefore generate relatively fewer neutrons from fission, because they contain fewer fissile nuclei than the outer bundles.

Figure 13.5 illustrates this method of flux flattening.

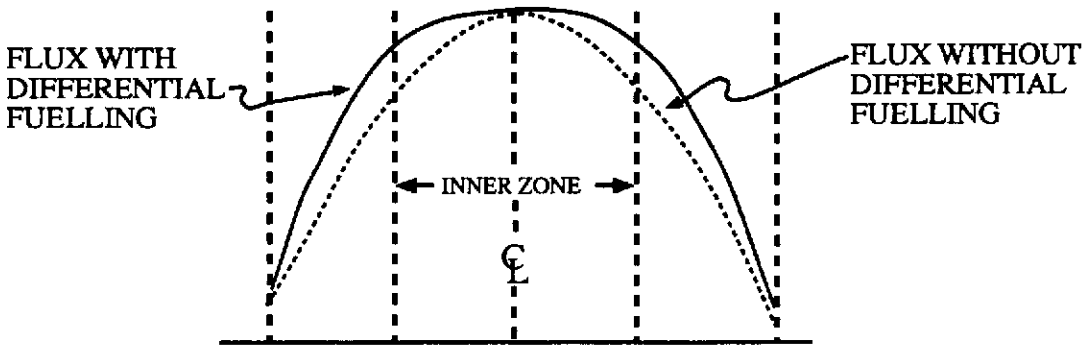


Figure 13.5: Effect of Differential Fuelling

The fuelling engineers at all our reactors plan the fuelling to maintain an optimum flat flux shape. For reactors with adjusters the amount of differential fuelling required to achieve this is relatively small.

Flux Oscillations

So far we have assumed the flux distribution is static. Suppose now that without changing the total power of the reactor, the flux is increased in one region of the reactor. This typically happens when a channel is refuelled. In the region of increased flux, the xenon now burns out more rapidly than it did prior to the change, and its concentration decreases. This decrease in xenon concentration leads to a higher reactivity in this region, which, in turn, leads to another increase in flux. This again leads to increased local xenon burnup, increased local reactivity, increased flux, and so on.

Meanwhile the control system is keeping bulk power constant so the flux away from the "hot spot" is lower than before. In the region of decreased flux, the xenon concentration increases due to reduced burnup while iodine continues to decay. This increased xenon concentration decreases the reactivity in this region, which reduces the flux, in turn increasing the xenon concentration, and so on. The thermal flux, and hence the power density decreases in this region while it increases in the other, the total power of the reactor remaining constant.

These local power excursions do not continue without limit. In the region of increased flux, the production of xenon from iodine, which is now being formed more rapidly, ultimately reduces the reactivity there. The flux and power eventually decrease. Similarly, in the region of reduced flux, the accumulated xenon eventually decays, increasing the local reactivity and reversing the flux and power transient in that region.

In this way, the flux and power of a reactor may oscillate between different regions (end to end, side to side, top to bottom) unless action is taken to control them. Calculations show that these xenon oscillations (also called flux tilts) have a period of 15 to 30 hours.

Since xenon oscillations can occur at constant overall power they may go unnoticed unless the flux distribution is monitored at several points in the reactor. This must be done in order to prevent such oscillations, since they represent something of a hazard to the safe operation of a reactor. Conceivably, they may lead to dangerously high local fuel temperatures.

One of the purposes of the liquid zone control system is to limit such oscillations. Each reactor is controlled using 14 zones. Each zone has a flux detector which via the Digital Control Computer, controls the light water level in the zone control compartment.

As an example of how light water zones may be used, look at Figure 13.6. Assume there are only two zones and a flux tilt is developing such that the flux in Zone I is increasing and the flux in Zone II is decreasing. By raising the water level in Zone I control compartment more neutrons are absorbed. Conversely, lowering the level in the Zone II compartment reduces the neutron absorption in that zone. Thus, the action of the two zone control compartments returns the flux to a normal flat distribution.

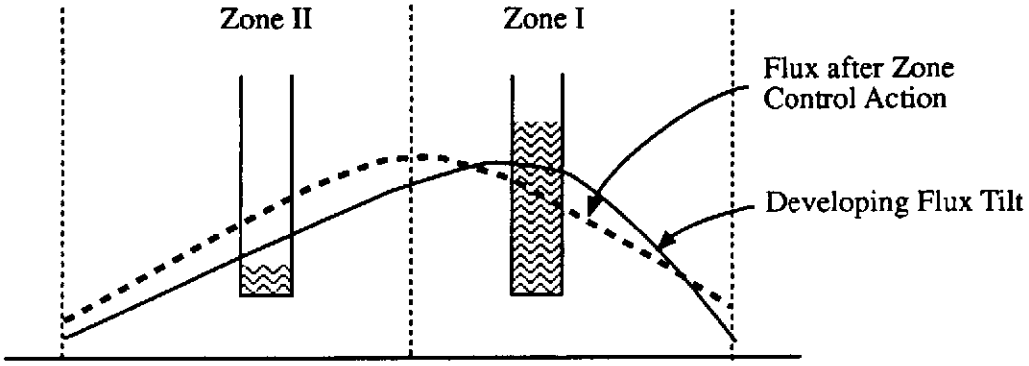


Figure 13.6: Zone Control System

ASSIGNMENT

1. List and briefly describe the four methods of flux flattening used in CANDU reactors.

2. Why is flux flattening desirable?

3. Explain how light water control zones are used to prevent flux oscillations.

J.E. Crist