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

$$\text{from, } E = Mc^2$$

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

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

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.

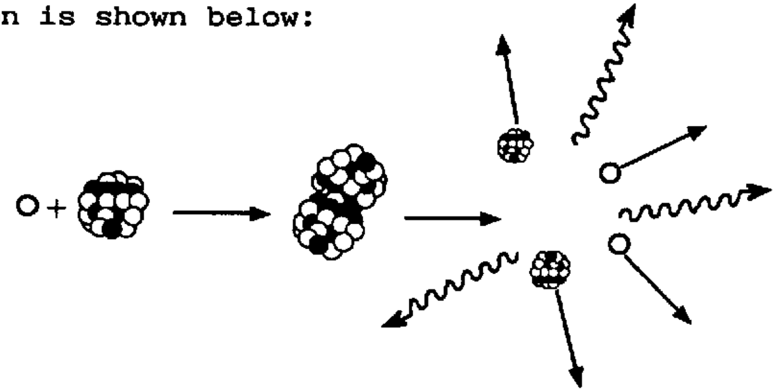
Energy from complete fissioning of 1 kg of U-235 = 8.2×10^{13} joules.

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

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

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:



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

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 x 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.

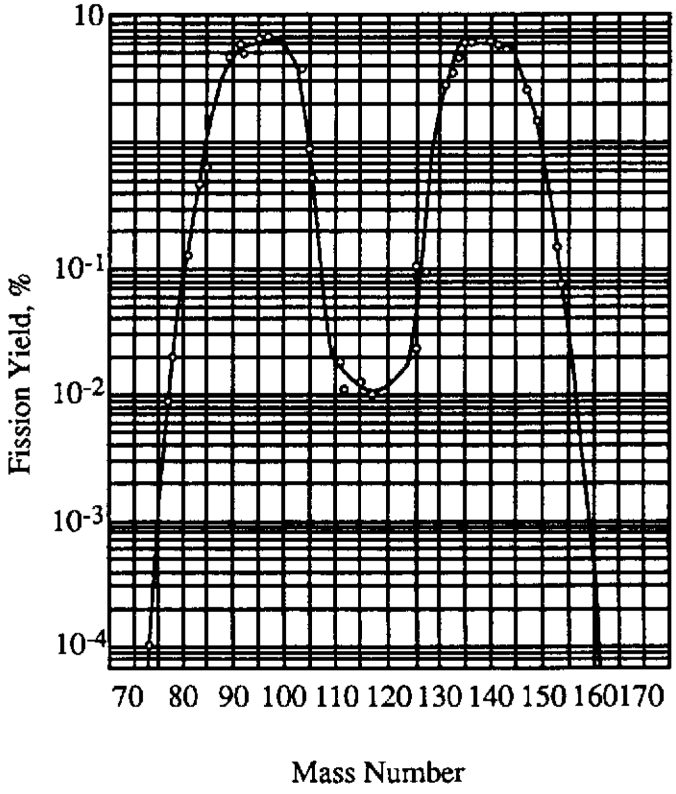


Figure 6.1: Yield of Fission Fragments

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.

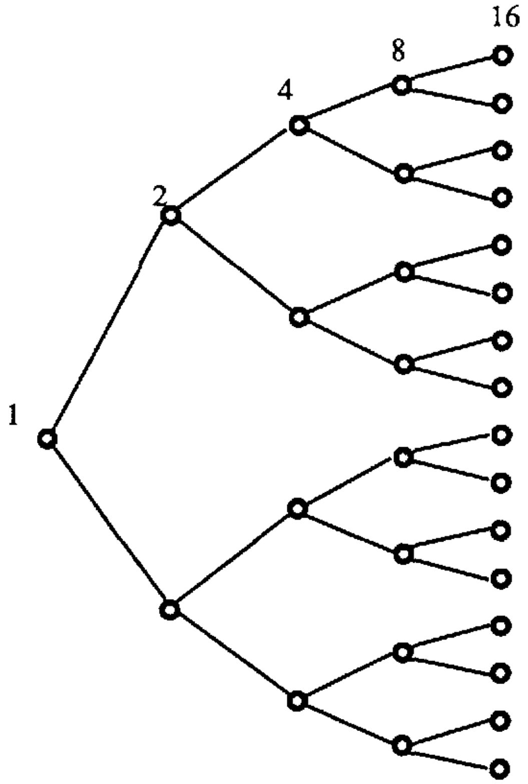


Figure 6.2: The Chain Reaction

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.

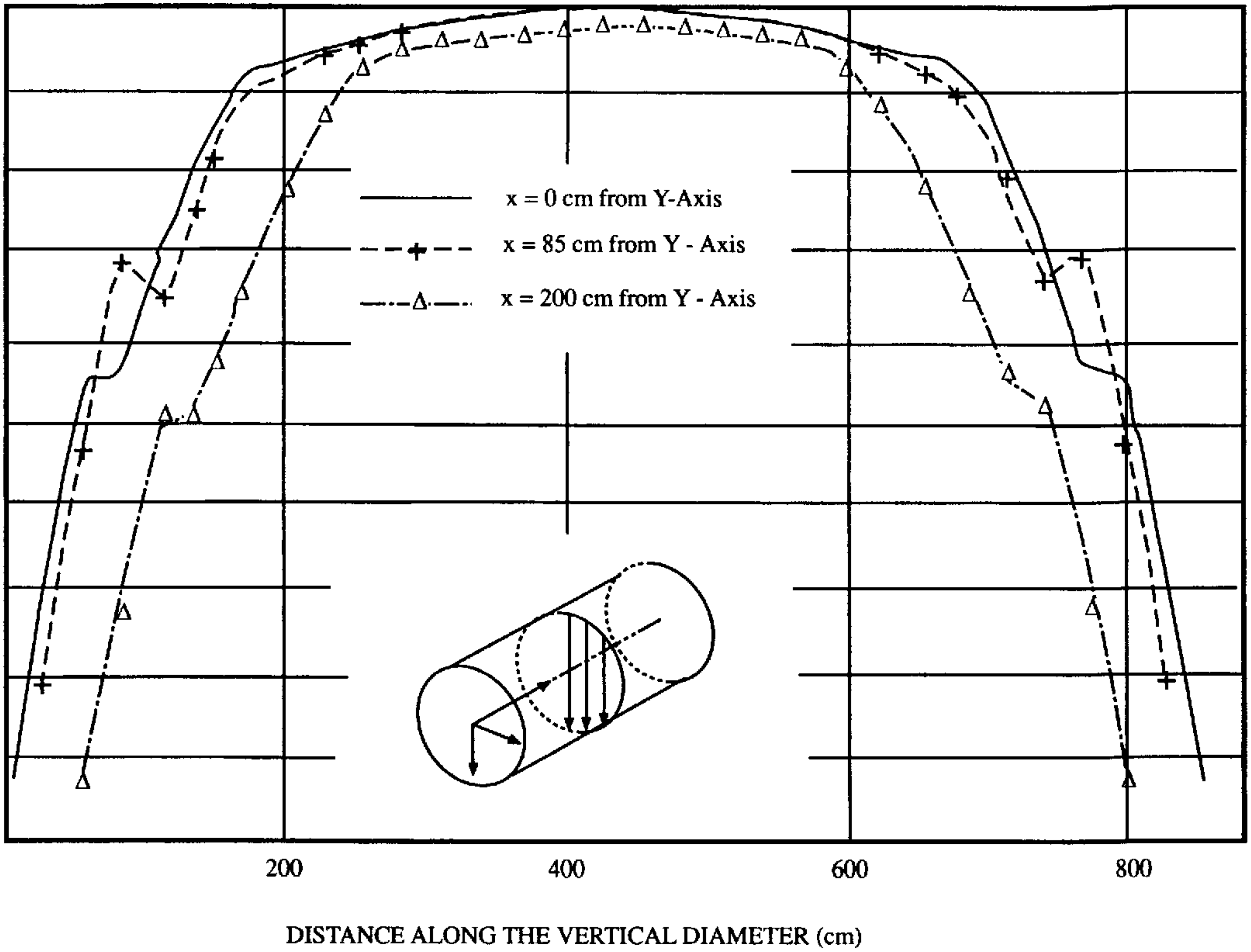


Figure 6.3: Neutron Flux

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.

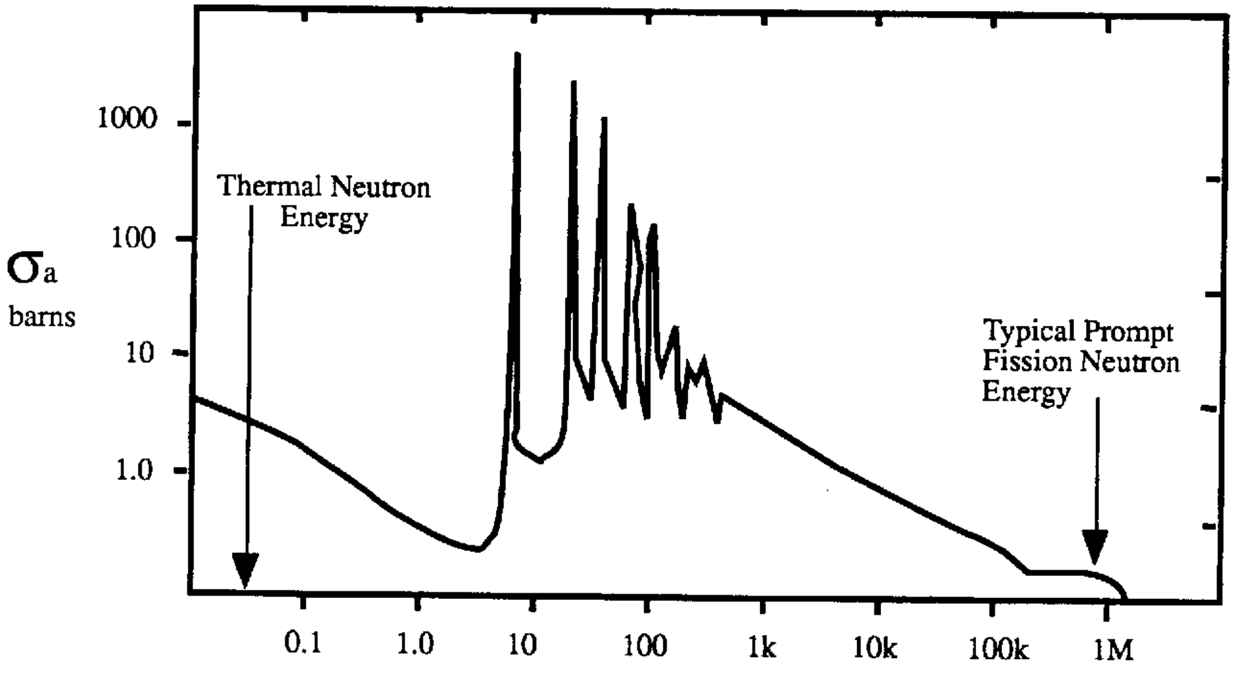
Neutron Cross-Section denotes the probability of a reaction occurring when a target nucleus is bombarded with neutrons. It has the dimensions of area and can be thought of as the effective target area of the nucleus for an incoming neutron, although the cross-section has no simple relationship with the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



Neutron Energy (eV)
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?

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