Module 2

Neutron Reactions

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Notes & References

2.1 MODULE OVERVIEW

The purpose of this module is to describe some of the more important reactions that neutrons can induce when they come into contact with nuclei of the fuel, moderator or reactor structure. The reactions to be considered are:

Elastic scattering

a) *Elastic scattering:* The main process by which neutrons are slowed to energies at which they will have a high probability of inducing fission in the fuel. The purpose of a moderator is to slow the neutrons down as rapidly as possible. A knowledge of the mechanism of elastic scattering is necessary if you are to understand the design features of the CANDU (or any other) reactor.

Inelastic scattering

b) *Inelastic scattering:* While this is less important than elastic scattering, it does contribute to neutron slowing down in the fuel, and you should be aware of how it operates.

Nuclear transmutation

c) *Nuclear transmutation:* This is important in the operation of reactor equipment such as ion chambers and start-up instrumentation, and in the production of certain radiation hazards such as nitrogen-16.

Radiative capture

d) *Radiative capture:* The principal mechanism by which neutrons are lost before they can cause fission. The design of a successful reactor is crucially dependent on minimizing parasitic absorption due to radiative capture. From the operational point of view, this prolific source of activation products can present a serious radiation hazard.

Fission

e) *Fission:* The importance of understanding the fission process is obvious, since it is this reaction that leads both to a release of useful energy and the possibility of the chain reaction itself. Among the aspects of direct operational significance are: fission in U-235 and Pu-239, and the role of Pu-239 in extending the life of the fuel; the production of fission products and their importance as neutron absorbers; prompt and delayed fission neutrons, which determine how reactor power responds to changes imposed by the control system.

2.2 MODULE OBJECTIVES

After studying this module, you should be able to:

i) Explain what is meant by elastic scattering, and how this process depends on the angle of scattering and the mass of the scattering nucleus.

ii) Describe what happens in inelastic scattering, giving an example of the process.

iii) Describe what happens in nuclear transmutation, giving an example of the process.

iv) Describe what happens in radiative capture, giving an example of the process.

v) Differentiate between spontaneous and induced fission.

vi) Describe what happens in induced fission.

vii) Differentiate between thermal and fast fission, and list the nuclides for which each type is important in CANDU fuel.

viii) Describe the process by which Pu-239 is produced in a CANDU reactor.

ix) State why fission products frequently decay by beta emission.

x) State five important consequences of fission product production in the fuel.

xi) Differentiate between prompt and delayed fission neutrons, and explain how the latter originate.

xii) Define the quantities β and ν.

xiii) Calculate the energy released in a specified mode of fission, given the masses of the nuclei involved.

xiv) List the various forms in which the energy released in fission may appear, and the parts of the reactor in which each is deposited.

xv) Estimate the rate at which uranium is being burnt up for a given reactor power level.

xvi) Explain the process of photoneutron production and state why it is important in reactor operation.

2.3 NUCLEAR REACTIONS

Nuclear reactions can occur when a particle, such as a neutron, proton, or a gamma photon, strikes a nucleus. Charged nuclear particles, such as protons, deuterons (deuterium nuclei), and alpha particles need to have a large amount of energy (several MeV) before they are able to overcome the Coulomb repulsive forces and enter a nucleus.

Neutrons, however, are not charged and are therefore able to interact with nuclei very effectively, even when they have very little energy. In fact, generally speaking, there is a greater chance that a reaction will occur with low rather than high energy neutrons, because the former are in contact with the nucleus for a greater length of time.

The operation of a reactor depends on how neutrons react with nuclei in the reactor. It is therefore necessary to analyze these reactions, called neutron reactions, in some detail. Although there are well over a dozen known neutron reactions, we need only consider the five covered in the following sections.

2.4 ELASTIC SCATTERING

For reasons that will be discussed in Module 4, it is necessary to incorporate in the reactor a substance called a ***moderator***. By a process known as ***elastic scattering,*** the moderator slows high-speed neutrons generated in the fission process to much lower energies.

Elastic collision

In an elastic collision of a fast neutron with a nucleus, the neutron strikes the nucleus and rebounds with reduced kinetic energy. The distinguishing feature of an elastic collision, however, is that the ***total kinetic energy before the collision is equal to the total kinetic energy after the collision.***

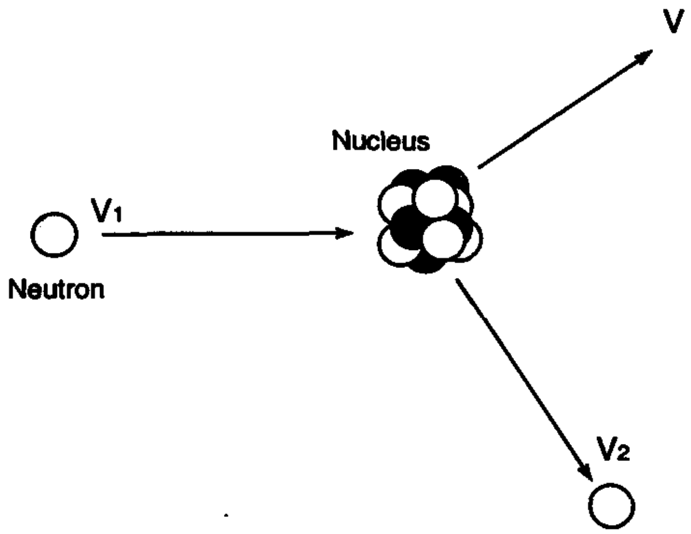

Figure 2.1: Elastic Collision

In Figure 2.1, for example, a neutron travelling at a speed v_1 strikes a stationary nucleus of mass A and bounces off with a speed v_2. Since some of the kinetic energy of the neutron is transferred to the nucleus, which recoils at speed v, v_2 must be less than v_1. Since the collision is elastic, however, the kinetic energy gained by the nucleus must be equal to the kinetic energy lost by the neutron.

The fraction of the initial energy that a neutron transfers in such a collision depends on two factors:

i) the angle through which the neutron is deflected;

ii) the mass, A, of the target nucleus.

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Elastic scattering

Maximum energy transfer occurs when the neutron hits the nucleus head-on, and the minimum amount of energy is transferred when the collision is a glancing one. The pool sharks among you will be well aware of how one can select the amount of energy transfer by choosing the angle of collision appropriately. When neutrons bounce around in a reactor, of course, the angles at which they hit the nuclei are quite random. That is why we say that the neutron is *scattered* in the process. Collisions in which ***kinetic energy is conserved*** are therefore instances of ***elastic scattering***.

The lighter the target nucleus, the greater (on average) the fraction of energy that a neutron will lose in these collisions. This is the reaction by which fast neutrons are slowed down in the moderator. Thus, the moderator nuclei should be light (that is an atomic mass number less than 16 or so) in order to slow the neutrons in as few collisions as possible. Otherwise the neutrons must travel large distances before they are slowed down and the reactor must therefore be larger. To emphasize this point, Table 2.1 shows the average number of elastic collisions neutrons must make in various materials to slow down from 2 MeV (the average energy with which they are produced at fission) to thermal energy, that is, the energy at which the neutrons have the same average kinetic energy as the atoms or molecules with which they are colliding. At room temperature, thermal energy is about 0.025 eV. Neutrons at this energy are known as ***thermal neutrons***, in contrast to those produced in the fission process, which are called ***fast neutrons***.

Thermal neutrons

Note that for a heavy nucleus, such as U-238, a very large number of elastic collisions would have to occur before the neutron would be slowed down to thermal energy. For heavy water on the other hand, the neutron loses on average about 40% of its kinetic energy in each collision, and so an average of only 36 collisions is required to thermalize neutrons (that is, to slow them down to 0.025 eV).

Table 2.1

Average neutron energy loss per collision during thermalization.

Material	Average energy loss per collision(%)	Average number of collisions to thermalize
^{1}H	65	18
^{2}H (deuterium)	52	25
H_2O (light water)	60	20
D_2O (heavy water)	40	36
^{12}C (graphite)	15	115
^{238}U	0.8	2172

2.5 INELASTIC SCATTERING

This is a more complicated process than elastic scattering because, instead of simply bouncing off the nucleus, the neutron actually enters the nucleus to form, for a very brief period (about 10^{-14} seconds), what is called ***a compound nucleus***. For a collision of a neutron with U-238, as shown in Figure 2.2, the short-lived compound nucleus is U-239. This immediately emits a neutron (any one) and ***a gamma photon***, reverting to U-238. The process still results in a slowing down of the neutron, since the energy associated with the gamma ray must be obtained at the expense of the kinetic energy of the neutron.

Compound nucleus

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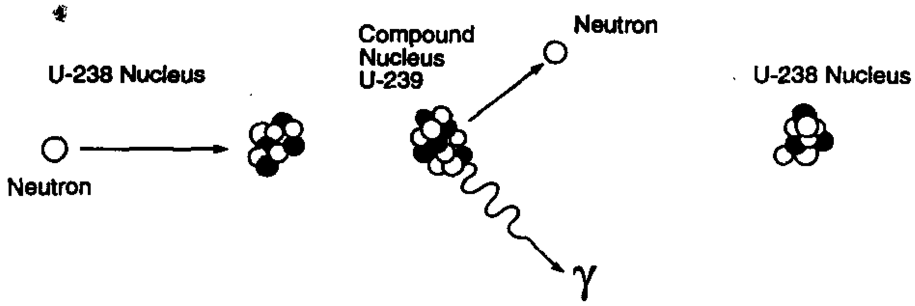

Figure 2.2: Inelastic scattering

The word "scattering" is again appropriate because the direction of the emitted neutron is quite arbitrary. An important distinction between elastic and inelastic scattering is that the former can occur at any neutron energy, whereas inelastic scattering can only take place if the initial kinetic energy of the neutron exceeds some **threshold energy** (44 keV for U-238 and 14 keV for U-235). For heavy nuclei, the threshold energy tends to be around 0.1 MeV or less, but for light nuclei it can be in the order of several MeV. The reason for the existence of a threshold level is that the neutron must contribute at least enough energy to raise the nucleus to its first excited state for a gamma ray to be emitted. The excited levels in a heavy nucleus are relatively closely spaced and consequently it has a lower threshold energy than light nuclei.

We can ignore inelastic scattering everywhere except in the fuel itself, since only there will the neutron energies be large enough for it to occur.

2.6 NUCLEAR TRANSMUTATION

Instead of re-emitting a neutron as in inelastic scattering, the compound nucleus formed by neutron capture may emit a proton or an alpha particle. Since the nucleus loses one proton in the first case and two in the second, the residual nucleus is that of a different element. The process is therefore known as ***nuclear transmutation.*** As an example, the transmutation of oxygen-16 to nitrogen-16 by high-energy neutrons is shown in Figure 2.3.

$^{1}_{0}$ neutron + $^{16}_{8}O$ $\qquad$ $^{16}_{7}N$ + $^{1}_{1}$ proton

Figure 2.3: Transmutation (n,p)

This reaction may be written as

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

or, more briefly, by the notation

$$^{16}O\ (n,p)\ ^{16}N$$

This reaction is of some importance in the operation of the reactor since the nitrogen-16 emits beta particles and associated high-energy gamma radiation (up to 7 MeV). The resulting radiation hazard is present in any region containing oxygen-16 that has recently been exposed to high-energy neutrons. For example, oxygen-16 is present in water (either H_2O or D_2O), and if this water has recently flowed through the reactor, some of the oxygen-16 will have been changed to nitrogen-16. Fortunately, the half-life of nitrogen-16 is only about 7 seconds, so that its activity decays rather rapidly.

Although transmutation reactions—(n,p) or (n,α)—are relatively rare, two other examples are of particular interest to us:

B-10 (n, α) Li-7: Reactor instrumentation (ion chambers) which monitors the neutron population in a reactor utilizes this reaction. The reaction releases 2.5 MeV of energy, which appears as kinetic energy of the lithium nucleus and the alpha particle. The nuclear and alpha particles lose this energy by producing a large amount of ionization in the counter—ionization that can easily be detected, even in the high gamma radiation background of a reactor environment. Boron is also used for reactivity control, and allowance has to be made for the gradual change in the proportions of the boron isotopes since boron-10 is burnt up by this reaction.

He-3 (n,p) H-3: This is an alternative to the boron reaction and is used as the basis for high-sensitivity instrumentation sometimes employed for the initial start-up of a reactor.

2.7 RADIATIVE CAPTURE

The most common neutron reaction of all is ***radiative capture,*** abbreviated as (n,γ). Here, the compound nucleus (formed when the neutron is captured) gets rid of all its excitation energy by emitting a gamma ray. Since, in contrast to the inelastic scattering case, the neutron is not re-emitted, the nucleus ***is converted into a higher isotope of the same element.*** Radiative capture can occur for practically all types of nuclei and at all neutron energies. Generally speaking, this is more probable for slow neutrons than for fast neutrons.

An example of this type of reaction is shown in Figure 2.4, which explains how tritium (hydrogen-3) is produced in heavy water reactors.

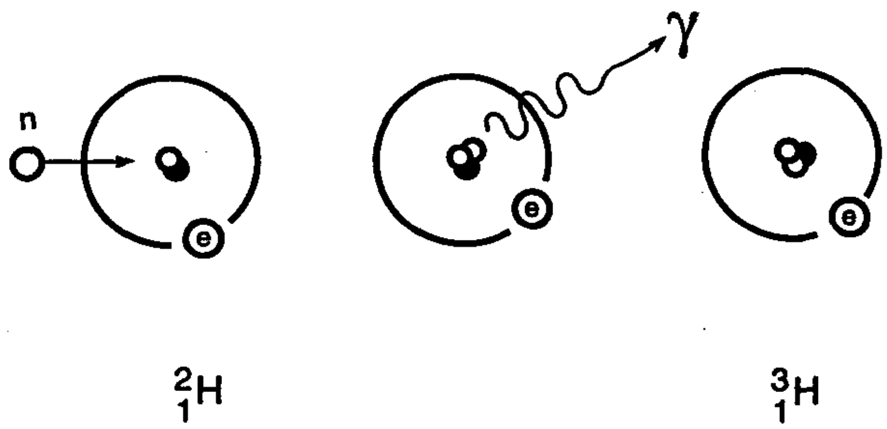

Figure 2.4: Radiative Capture (n,γ)

Radiative capture is important for three reasons:

Parasitic absorption

a) **Parasitic absorption**

Many substances absorb neutrons fairly readily by this process. Non-fuel materials used in a reactor must be restricted to those which have very low probabilities of radiative capture, otherwise too many neutrons would be lost and the chain reaction could not be maintained.

b) **Conversion of U-238 to Pu-239**

Pu-239 production

Generally, non-fission neutron capture in core materials is undesirable. However, if the non-fission capture is by U-238 (producing U-239), there is a bonus in the subsequent transformation of the U-239 to Pu-239 by two successive beta decays, as shown in Figure 2.5. Pu-239 is a fissile nuclide and its creation extends the life of the fuel.

$$^{238}_{92}U + n \longrightarrow {}^{239}_{92}U + \gamma$$

$$^{239}_{92}U \xrightarrow{\text{Half life = 24m}} {}^{239}_{93}Np + \beta$$

$$^{239}_{93}Np \xrightarrow{\text{Half life = 2.4d}} {}^{239}_{94}Pu + \beta$$

Figure 2.5: Production of Pu-239

Activation

c) **Activation of materials in core**

The nucleus formed by radiative capture is usually radioactive and may present a radiation hazard. For example, corrosion products circulated by the heat transport system will be activated as they pass through the reactor core. When they later plate out, the entire heat transport system becomes a radioactive hazard and will remain so even if the reactor is shut down (that is, if the neutron source is removed). In CANDU reactors, cobalt-60, manganese-56, and copper-64 are the three most troublesome of these plated-out activation products.

2.8 FISSION

The final reaction we will consider is called ***fission***. The word is borrowed from the biologists, who use it to describe the breaking up of a cell into two new ones. Production of nuclear power relies on the fact that some nuclei will undergo fission, and that energy is released during this fission process because some mass is converted to energy and then released (see Section 1.7). There are two types of fission: ***spontaneous*** and ***induced***.

2.8.1 Spontaneous Fission

In this reaction, a nucleus breaks up entirely spontaneously, without any external cause. It is quite a rare reaction, generally only possible for nuclei with atomic masses of around 232 u or more. As the atomic mass number, and therefore the charge on the nucleus increases, spontaneous fission becomes more and more probable. (One could argue that there is an infinite number of heavy elements which do not exist, because they are not stable against spontaneous fission decay). Table 2.2 shows the spontaneous fission and, for comparison, the alpha decay rates of the U-235 and U-238 isotopes.

TABLE 2.2

	Half-life (α) (years)	Half-life (s.f.) (years)	α decay rate (atoms/s/kg)	s.f. decay rate (atoms/s/kg)
U-235	7.1×10^8	1.9×10^{17}	79×10^6	0.3
U-238	4.5×10^9	8.0×10^{15}	12×10^6	6.9

The rate of spontaneous fission is so low that it plays an insignificant part in the production of power (it contributes only about 10^{-12}% of the full power output of a typical reactor). Nevertheless, it is of some importance because the spontaneous fission of a uranium nucleus is usually accompanied by the emission of a few neutrons, which constitute a small neutron source which is always present, even when the reactor is shut down. This has implications for processes such as the first fuel loading of a reactor core. If the system were inadvertently to become supercritical (see Section 4.3), the spontaneous fission neutrons would provide a source which could be multiplied to give a large and rapid power excursion. A similar situation could occur if supercriticality were attained in a fuel storage facility for enriched fuel.

On the other hand, the presence of a neutron source from spontaneous fission can be useful in maintaining a sufficient fission level to preserve a reading on the power-monitoring instrumentation in the shut-down condition, an important safety feature.

2.8.2 Neutron-Induced Fission

Certain heavy nuclei can be induced to undergo fission as a result of capturing a neutron. When a neutron is captured, the compound nucleus formed acquires an internal excitation energy (see Section 1.7) equal to the sum of the binding energy of the neutron to that nucleus and any kinetic energy the neutron possessed before its capture. If this internal excitation energy exceeds a threshold value known as the ***critical energy,*** the nucleus will undergo fission.

Critical energy

For most heavy nuclei, the kinetic energy of the incoming neutron must be very high before fission can occur. We will therefore restrict our discussion to those nuclei which can undergo fission with neutrons having the sort of kinetic energy typical of a reactor neutron. In practice, we are then dealing with neutrons ranging from 10 MeV down to thermal energies.

2.8.3 Practical Fission Fuels

Thermal fission

The only nuclei of practical importance to us are the U-235 and U-238 isotopes of uranium and the Pu-239 and Pu-241 isotopes of plutonium. U-235 will undergo fission with neutrons of ***any*** energy; in fact, fission of U-235 with thermal neutrons (***thermal fission***) is much more probable than fission with fast neutrons (fast fission). Pu-239 and Pu-241 also undergo fission with neutrons of any energy. These three nuclides are said to be ***fissile***. U-238, on the other hand, will only undergo fission with neutrons whose kinetic energy is greater than about 1.2 MeV; U-238 and other nuclides with a similar threshold energy are said to be ***fissionable***. U-238 makes only a small direct contribution (about 3%) to the power produced in a CANDU reactor.

Fissile nucleus

> **Note: Fissile describes a nucleus that can be fissioned by thermal neutrons. A fissile nucleus can also be fissioned by neutrons of any energy.**

Natural uranium contains only U-235 (0.72%) and U-238 (99.28%). Over a period of reactor operation, Pu-239 as well as some Pu-241 will be built up in the fuel as a result of neutron capture.

Pu-239 is fissile like U-235. If it does not undergo fission, it may capture a neutron to form Pu-240. Although this is fissionable, it is much more likely to capture another neutron to form fissile Pu-241. A significant fraction of the total power produced by fuel during its life in our reactors is due to fission of the fissile plutonium isotopes. We will deal with this in more detail later in the course.

2.8.4 Fission Products

A typical neutron-induced fission is shown in Figure 2.6. A neutron is captured by a nucleus of U-235, forming the compound nucleus U-236 which, in this particular case, splits up immediately to yield a nucleus of Xe-140 and a nucleus of Sr-96. The particular mode of fission illustrated is only one of many possible ways in which the nucleus may split up. The ***fission fragments*** Xe-140 and Sr-96 are two of about 300 nuclides that are known to be formed in fission. Figure 2.7 shows the relative frequency of occurrence for nuclides of specific mass numbers produced as fission fragments. Such a curve is known as a ***fission yield curve*** (as two fragments are produced per fission, the area under the curve adds up to 200%). You can see that both fission fragments are likely to consist of a substantial piece of the original nucleus. They are likely to have mass numbers between 70 and 160, with those around 95 and 140 the most probable. Note that symmetrical fission (equal fragments) is quite rare.

Fission fragments

Fission yield curve

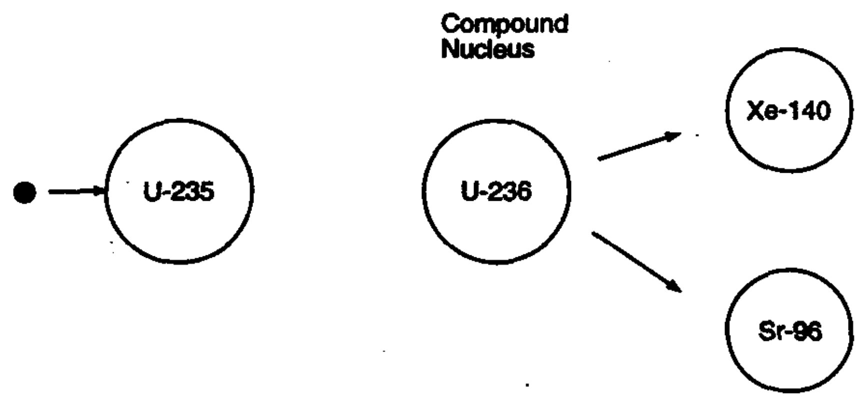

Figure 2.6: One of many possible ways in which a U-235 nucleus can split as a result of fission

Notes & References

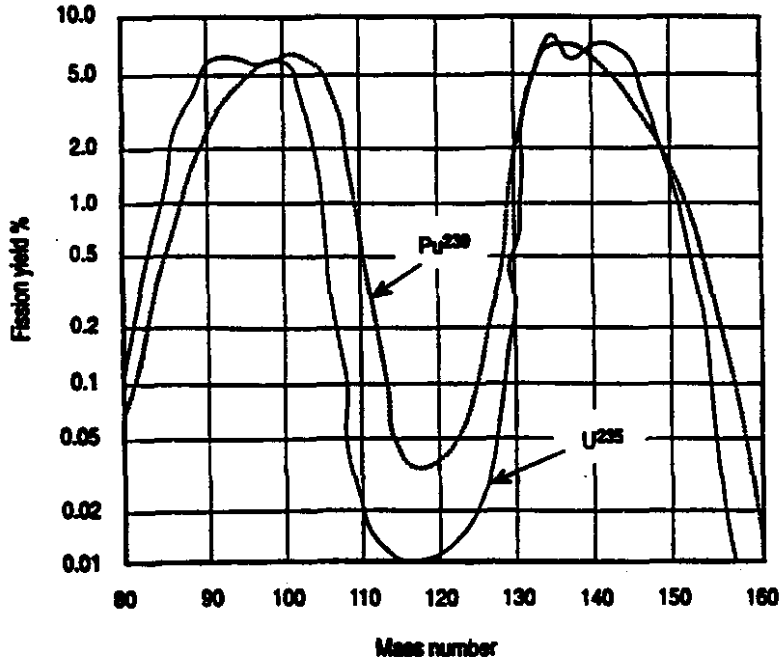

Figure 2.7 Fission Yield of U-235 and Pu-239

Decay chain

The fission fragments are almost invariably radioactive. The reason for this is that the neutron/proton ratio of the fragments is about the same as that of the fissioned nucleus, and this is too high for stability at medium mass numbers (see Figure 1.5). The fragments will therefore try to reduce their N/Z ratio by successive beta decays (accompanied by gamma emission) until stability is reached. A typical decay chain is shown in Figure 2.8. (Ignore the two neutrons resulting from the fission for the moment – we will deal with these in the next section).

$$n + {}^{235}_{92}U \rightarrow {}^{139}_{54}Xe \xrightarrow[\gamma]{\beta-} {}^{139}_{55}Cs \xrightarrow[\gamma]{\beta-} {}^{139}_{56}Ba \xrightarrow[\gamma]{\beta-} {}^{139}_{57}La$$

$$n + {}^{235}_{92}U \longrightarrow n + n + \gamma \text{ (from precursors of Xe-139 and/or Sr-95)}$$

$$n + {}^{235}_{92}U \rightarrow {}^{95}_{38}Sr \xrightarrow[\gamma]{\beta-} {}^{95}_{39}Y \xrightarrow[\gamma]{\beta-} {}^{95}_{40}Zr \xrightarrow[\gamma]{\beta-} {}^{95}_{41}Nb \xrightarrow[\gamma]{\beta-} {}^{95}_{42}Mo$$

Figure 2.8: Fission Product Decay Chains

Practical importance of fission products

All members of the decay chains (including the original fission fragments) are known as ***fission products***. The great majority of fission products have half-lives in the range from fractions of a second up to about 30 years. (It is this activity that causes so much concern in atomic bomb fallout). There are five important consequences of the production of fission products in the fuel:

a) Fission products must be held in the fuel by an encasing sheath, so that they do not enter the heat transport system and hence leave the reactor core. As long as the fission products remain in the fuel and the fuel remains adequately shielded, there is no biological risk. Since many fission products have long half-lives, their presence in the heat transport system would be a radiation hazard which would prevent access to equipment even when the reactor is shut down;

b) Heavy shielding is required around the reactor to avoid exposure to gamma radiation emitted by the fission products;

c) Fuel must be changed remotely and special precautions must be taken in handling and storing spent fuel;

d) Some of the fission products have a high affinity for neutrons and thereby ***poison*** the reactor. The two most important poisons are Xe-135 and Sm-149. They are produced in a relatively high percentage of fissions, and capture a significant number of neutrons;

e) The creation of ***delayed neutron precursors*** (see next section).

2.8.5 Prompt and Delayed Neutron Emission

Prompt neutrons and gammas

Chain reaction

The fission fragments are produced in an excited state and will tend to get rid of their excitation energy by emitting gamma rays and neutrons (known as ***fission neutrons***). When fission takes place in a reactor, most of the fission neutrons will be emitted almost immediately ($\sim 10^{-17}$ sec) after the fission takes place; these are known as ***prompt neutrons***, while the gamma rays are called ***prompt gammas***. The number of neutrons emitted following fission is variable, but the most probable yields are 2 or 3 neutrons per fission. It is the existence of these fission neutrons that makes it possible to maintain a self-sustaining ***chain reaction***, in which the fission neutrons from one generation of fissions are used to ***cause the next generation of fissions***.

The energy distribution of prompt neutrons is shown in Figure 2.9. The average kinetic energy with which these are emitted is about 2 MeV, although the most probable energy is only 0.72 MeV.

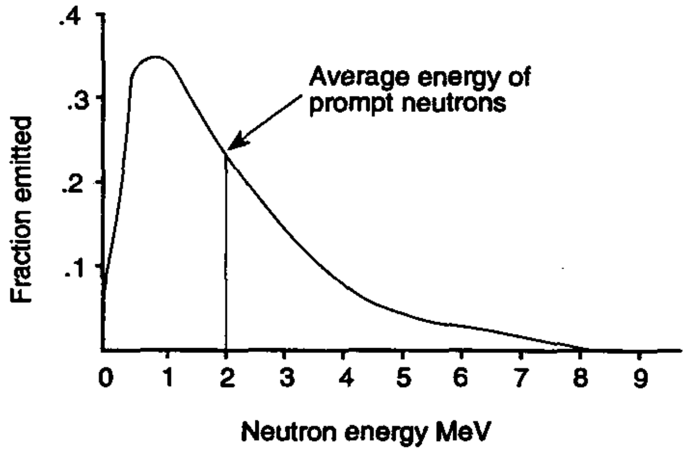

Figure 2.9: Energy distribution of prompt fission neutrons

In most cases, a nucleus formed by fission does not have sufficient excitation energy to throw off a neutron, but instead decays by beta emission (as mentioned earlier, this method of changing one neutron into a proton is an alternative way for the nucleus to reduce its complement of neutrons). In a few cases, a neutron may be emitted ***by the nucleus created by the beta decay***. One sequence which can lead to this process starts with Br-87, which decays with a half-life of 55 s to form Kr-87 (Figure 2.10). The excitation energy of the Kr-87 nucleus resulting from the beta decay may be high enough to allow it to decay by emitting a neutron. Although this will happen virtually instantaneously with the formation of the Kr-87, there will obviously be a delay between the occurrence of the ***fission*** and the emission of the neutron. Neutrons produced by this process, in fact, will appear at times governed by the 55-second half-life of the Br-87. Nuclides such as Br-87 are therefore known as ***delayed neutron precursors***. The neutrons which are produced by nuclides such as Kr-87 are called ***delayed neutrons***.

Delayed neutron-precursors

Delayed neutrons

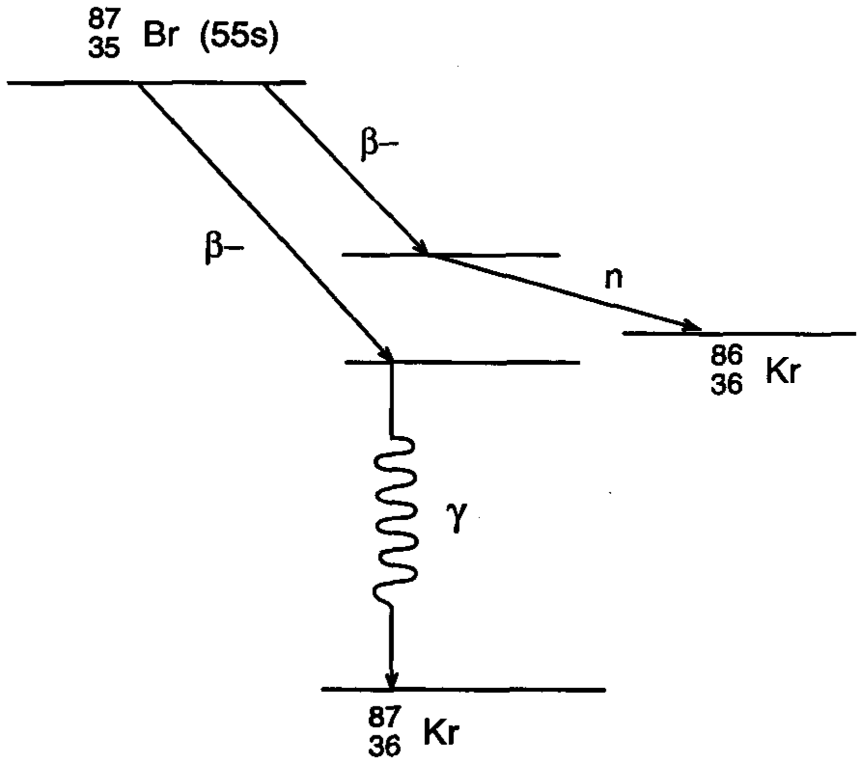

Figure 2.10: Production of delayed neutron from Br-87 (normal beta-gamma decay mode also shown)

Thus far, only about a dozen of the delayed neutron precursors have been identified. Among those most well known, in addition to Br-87, are Br-88 (half-life 16 s) and I-137 (half-life 24 s). As we will see in Module 8, the kinetic response of the reactor depends strongly on the proportions and half-lives of the various delayed neutron precursors. Fortunately, although it has not been possible to identify all of these, it turns out that their overall behavior can be approximated well enough by an empirical division into six groups, each with a single half-life. The half-lives and percentage yields of the six groups assumed for the fission of U-235 are given in Table 2.3.

Table 2.3

Delayed neutron precursor groups for U-235 fission

Precursor group	Half-life (seconds)	Relative yield (%)
1	55	4
2	22	21
3	6	19
4	2.2	40
5	0.5	13
6	0.18	3

Note: Average half-life is approximately 8.9 s, corresponding to an average decay constant (λ) of approximately 0.08 s^{-1}.

Delayed neutron fraction

For thermal fission of U-235, the total contribution of all the delayed neutrons (called the ***delayed neutron fraction, β***) is only 0.70% of the total neutrons produced in fission. For Pu-239, the delayed neutron fraction is even smaller, at 0.23%. Despite the fact that these fractions are so small, the very existence of delayed neutrons is the factor that makes it possible to control a nuclear reactor at all, as will be seen in Module 8.

As mentioned earlier, the number of neutrons emitted following a fission is a variable quantity. Table 2.4 provides the probability of a particular number of neutrons being emitted in the thermal fission of a nucleus of U-235. These numbers include both prompt and delayed neutrons.

Notes & References

Table 2.4

Neutron emission in thermal fission of U-235

Number of neutrons emitted	Number of cases per 1000 fissions (approximate)
0	32
1	154
2	341
3	321
4	128
5	22
6	2

Neutrons per fission, ν

The average number of neutrons emitted per fission is a very important quantity in reactor physics (Table 2.5). It is usually denoted by the Greek letter ν (pronounced "new"). For thermal fissions of U-235, ν=2.43. (Fast fissions, i.e., fissions caused by fast neutrons, usually produce marginally more fission neutrons.) It is also interesting to compare the number of neutrons released per thermal fission of Pu-239 and Pu-241 since both of these plutonium isotopes build up in our fuel after a while.

Table 2.5

Value of ν for thermal fissions

Fissile Nucleus	ν
U-235	2.43
Pu-239	2.89
Pu-241	2.93

Notes & References

Neutrons per fission, ν

Table 2.4

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Number of neutrons emitted	Number of cases per 1000 fissions (approximate)
0	32
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Pu-241	2.93

2.8.6 Energy Release in Fission

A little over 200 MeV of energy is liberated, on average, when a U-235 nucleus undergoes fission. We can illustrate this by considering the typical fission described in Figure 2.6:

$$^{235}_{92}U + ^{1}_{0}n \rightarrow ^{95}_{38}Sr + ^{139}_{54}Xe + 2^{1}_{0}n$$

We can find the energy release in the fission by calculating the total mass on each side, subtracting the final mass from the initial one, and converting to MeV. The masses of the ***nuclei*** involved (obtained from standard tables of ***atomic*** masses by subtracting the total mass of the electrons in each case) are:

$^{235}_{92}\text{U}$	234.9934 u
$^{95}_{38}\text{Sr}$	94.8984 u
$^{139}_{54}\text{Xe}$	138.8890 u

Taking the mass of a neutron to be 1.0087 u, the masses involved are:

Energy release calculation

Total mass before fission	= 234.9934	+ 1.0087
	= 236.0021	u
Total mass after fission	= 94.8984	+138.8890 + 2.0174
	= 235.8048	u
Hence, mass loss	= 0.1973	u
Energy released when fission occurs		= 0.1973 × 931.5
		= 184 MeV

What we have calculated is the energy released ***immediately after the fission occurs.*** This energy appears in the form of recoil kinetic energy of the fission fragments, kinetic energy of the prompt neutrons, and energy of the prompt gammas. In order to find the ***total*** energy generated as a result of the fission having occurred, we have to add in the energy that is gradually released in subsequent beta decays of the fission products. This energy appears in the form of the kinetic energy of the beta particles and neutrinos produced in the decays, together with any gamma rays emitted in the process. In the example considered here, it can be shown that the amount of energy generated by the subsequent radioactivity of the fission products amounts to about 24 MeV. The total energy released as a result of the fission is therefore 184 + 24 = 208 MeV.

When averaged over all the various possible modes of disintegration, the ***mean*** energy release from U-235 fission is approximately 205 MeV.

The way this energy is partitioned among the various components of the fission process is shown in Table 2.6.

Distribution of fission energy release

Table 2.6

Approximate distribution of fission energy release in U-235

Kinetic energy of lighter fission fragment	100 MeV
Kinetic energy of heavier fission fragment	69 MeV
Kinetic energy of fission neutrons	5 MeV
Energy of prompt gamma rays	6 MeV
Beta particle energy gradually released from fission products	8 MeV
Gamma ray energy gradually released from fission products	6 MeV
Neutrinos (energy escapes from reactor)	11 MeV
Total	205 MeV

Although the neutrinos escape without depositing any energy in the reactor, this is partly compensated by the fact that some of the fission neutrons, even after losing all their kinetic energy, produce (n, γ) reactions with materials in the reactor. An average energy per fission of about 8 MeV is released in such reactions. Consequently, the total amount of energy produced in a reactor per fission is always within a few MeV of 200 MeV.

Not all of the energy generated by fissions in the reactor is available as ***useful*** heat, since some of it is not deposited in the fuel or coolant. The way in which the energy from the various sources is distributed in the reactor is summarized below.

Location of heating

Source	**Location of heating**
Fission fragments	Fuel pellets (fragments slow down in a distance of a few μm).
Kinetic energy of fission neutrons	Mostly transferred to moderator by collisions with moderator atoms.
Beta particles from fission product decay	Fuel pellets and cladding.
Prompt gammas, and gammas from fission product decay	Throughout reactor and its shielding. (About one third deposited in moderator).

Decay heating

The energy transferred to the moderator is essentially waste heat which has to be rejected. Once the reactor has been running steadily for some time, the fission products will have built up to a level such that approximately 7% of the heat in the reactor is generated by fission product decay. This has a major effect on reactor design since, even when the fission reaction is shut down, there is a substantial heat source in the core which decays relatively slowly over a long period of time. The cooling system must be adequate to cope with this decay heat under normal shutdown conditions. In addition, it is essential that in the event of a loss of the normal cooling, a reliable emergency core cooling system is available to prevent damage to the fuel from this decay heating. It was a failure to maintain this emergency core cooling that led to extensive damage to the core of the Three Mile Island PWR in 1979.

2.8.7 Reactor Power and Fuel Consumption

Taking the useful energy release per fission of a U-235 nucleus to be 200 MeV, it can be deduced that the rate of fissioning required to maintain each watt of power is

$$1 \text{ watt} \rightarrow \quad 3.1 \times 10^{10} \text{ fission/s}$$

A convenient practical unit for specifying heat *energy* production is the megawatt-day (MWd). Using the above conversion, it can be shown that the production of 1 MWd of heat energy requires the fissioning of approximately 1g of U-235.

Assume that we have a CANDU reactor which, at full power, generates 1,744 MW from fission (540 MW gross electrical power). The quantity of U-235 that would have to undergo fission to maintain the reactor at full power for a day would therefore be 1,744 g or 1.74 kg.

Uranium consumption

Since the proportion of U-235 in natural uranium is only 0.72%, it might, at first sight, seem that the rate of consumption of natural uranium for the reactor would be 1.74/0.0072 or 242 kg per day (about 13 bundles per day). Two factors modify this estimate, however:

i) As we shall see later, only about 86% of the neutron absorptions by U-235 lead to fission; the remainder lead to the production of U-236 by radiative capture. The rate of destruction of U-235 is therefore greater than the fission rate by a factor of $1/0.86 = 1.16$. The total rate at which natural uranium is being burnt up if it produces power at the given rate is therefore $242 \times 1.16 = 280$ kg per day (about 15 bundles per day).

ii) The second factor affecting the rate at which new fuel must be supplied to the reactor is the build-up of Pu-239 in the fuel as a result of neutron capture in U-238. As time goes on, an increasing proportion of the power is generated from fission of Pu-239 rather than U-235. This effect reduces the rate at which uranium must be supplied to the reactor. Typically, as about 40% of the energy output is obtained from Pu-239 fission, the natural uranium requirement would drop to about $0.60 \times 280 = 168$ kg per day (9 bundles per day).

2.9 PRODUCTION OF PHOTONEUTRONS

Another source of neutrons exists in an operating reactor in addition to those produced by fission. This source is particularly important because it persists for quite a long time after the reactor is shut down. (Production of prompt fission neutrons ceases immediately upon shutdown and the delayed neutron production dies out after a few minutes as their precursors decay.) The neutrons which comprise the long-lived source are known as ***photoneutrons***.

Photodisintegration

Photoneutrons are unique to reactors which use heavy water as a moderator or heat transport fluid. They are produced when photons with energies greater than 2.2 MeV are captured by deuterium nuclei causing the break-up of the deuterium nucleus (***photodisintegration***):

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

After the reactor has been operating for a while, it will have built up an inventory of fission products in the fuel, some of which emit gamma rays with an energy greater than 2.2 MeV. When the reactor is shut down, this photoneutron source will persist because gamma rays from decaying fission products can still produce photoneutrons in heavy water present in the core. Even if the moderator has been dumped, heavy water will always be found in the core as heat transport fluid. Therefore, in our heavy water cooled reactors, we always have a relatively large neutron source (compared to the spontaneous fission source) with which to start the reactor up again after a shutdown.

ASSIGNMENT

1. Explain what is meant by an ***elastic collision*** of a fast neutron with a nucleus. In any individual collision, which two factors determine the fractional loss of kinetic energy of the neutron?

2. Explain what happens in an ***inelastic collision*** of a fast neutron with a nucleus, and give an example of the process. Why is it not possible for inelastic scattering to take place unless the neutron has a certain threshold energy? In general, will the threshold energy be higher for low-mass or high-mass nuclei? Why?

3. Explain what is meant by (a) ***nuclear transmutation*** and (b) ***radiative capture*** reactions by neutrons, and give an example of each. Give three reasons why the radiative capture reaction is important in the operation of nuclear reactors.

4. a) Why are fission products nearly always radioactive?

   b) Explain the process leading to the production of ***delayed neutrons*** from fission.

5. For each of the following reactions, give the nuclide X and briefly explain the significance of each reaction in the operation of a CANDU plant.

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

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

$$ {}_{8}^{16}O + {}_{0}^{1}n \rightarrow X + {}_{1}^{1}p $$

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

6. Taking the mass of one atom of U-235 as 235 u, calculate the total energy (in MWd) released by the fission of all the U-235 atoms in 1 kg of natural uranium. (Take 1 atomic mass unit = 1.66×10^{-27} kg and energy/fission = 200 MeV).