Module 4

Thermal Reactors (Basic Design)

July, 1997 (R-1)
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4.1 MODULE OVERVIEW

This module covers basic design requirements for a ***thermal neutron reactor***. If the fuel is to be natural uranium, the two most important features are the inclusion of a ***moderator*** and ***"lumping"*** of the fuel. In order to understand the function of the moderator, we will identify characteristics that make a material a good moderator, and based on these, consider how the choice of a particular moderator influences the design of the reactor using it. Finally, we will look at how the ***multiplication factor*** of a CANDU reactor varies as a function of the ***lattice pitch***.

4.2 MODULE OBJECTIVES

After studying this module, you should be able to:

i) Define the terms: critical, subcritical, supercritical and neutron multiplication factor (k).

ii) List the desirable properties of a moderator with particular reference to the role of the scattering and absorption cross-sections, and the number of collisions required to thermalize a neutron.

iii) Explain why "lumping" the fuel is necessary in a reactor fuelled with natural uranium.

iv) Define the roles of elastic and inelastic scattering in the reactor.

v) State why heavy water is used as the moderator in CANDU reactors.

vi) Describe the effect of downgrading the moderator or heat transfer fluid.

vii) State how and why the multiplication factor varies with lattice pitch, and why CANDU reactors are overmoderated.

4.3 MAINTAINING A CHAIN REACTION

The principle of a nuclear power reactor is that the fission rate must be maintained at a steady level which is high enough to maintain the fuel at a temperature enabling the required power to be supplied to the heat transport system and hence to the turbine. The ***thermal*** power for a CANDU 600 at full power is about 2,160 MW. The average energy release for a single fission is approximately 200 MeV, or 3.2×10^{-11} J. The rate of fissioning required at full power is therefore:

$$\mathit{Rate\ of\ fissioning} = \frac{2160 \times 10^{6}}{3.2 \times 10^{-11}} \frac{J/s}{J/\mathit{fission}} = 7 \times 10^{19}\ \mathit{fissions/second}$$

Each fission which occurs releases approximately 2.5 fission neutrons (Section 2.8.5). In order to maintain the fission rate at a constant value, exactly one of these must survive to cause a further fission. The remainder are lost either by ***capture*** (radiative capture in fuel or other reactor materials) or ***leakage*** (escape from the reactor). Under these circumstances, a self-sustaining chain reaction is set up where the number of fissions from one generation to another remains constant. A reactor operating in this condition is said to be ***critical***. Please note that a reactor can be critical ***at any power level.***

Critical chain reaction

If the neutron losses by non-fission capture or leakage are reduced (for example, by moving a control absorber out of the reactor), more neutrons will be available to cause fissions. The number of fissions occurring in any one generation will be greater than the number occurring in the previous generation. The chain reaction is then said to ***diverge,*** the reactor power continues to increase, and the reactor is said to be ***supercritical.*** Conversely, if neutron losses are increased until the number in each generation is less than in the one before, the reactor is said to be ***subcritical.***

Neutron multiplication factor (k)

To describe the condition of the reactor in a quantitative way, we introduce a factor known as the ***neutron multiplication factor (k)***, which is defined as

$$k = \frac{\text{number of neutrons in one generation}}{\text{number of neutrons in the previous generation}} \tag{4.1}$$

For the critical reactor, of course, k will be equal to 1. We should note at this point that the definition of k is only valid if the number of ***source neutrons***, that is, photoneutrons and neutrons from spontaneous fission, is negligible compared to those produced by the fission process. For the moment, we are ignoring the fact that some fission neutrons are delayed, and so are treating all the neutrons as if they were prompt.

4.4 THE MODERATOR

In order to reduce the loss of neutrons from one generation to another sufficiently to achieve a critical reactor, we must minimize leakage and radiative capture. As we will see in a later chapter, the first objective is accomplished simply by making the reactor large enough and of the right shape. Reducing radiative capture, especially in the U-238 which constitutes the overwhelming proportion of the fuel, is a more complicated problem. Neutrons produced by fission have average energies of around 2 MeV. They bounce around inside the reactor and lose energy by a combination of inelastic and elastic collisions with whatever nuclei are present.

Since the fission cross-section of U-235 is so much larger for slow neutrons than for neutrons of 2 MeV energy (580 barns at 0.025 eV, compared with 1 barn at 2 MeV), it seems reasonable to try to slow the fission neutrons down as rapidly as possible to increase their chance of causing fission rather than being lost in other processes. As we saw in Module 2, this can be done most effectively by letting the neutrons undergo collisions with light nuclei such as those of hydrogen, deuterium or carbon (graphite). A material added to the reactor for the purpose of slowing the fast neutrons down is known as a ***moderator.***

Moderator

If we use natural uranium (99.3% U-238, 0.7% U-235) as fuel, however, it is not enough simply to mix the moderator and fuel in a homogeneous mixture. This is apparent when you follow the history of a fast fission neutron as it slows down in the reactor. While its energy is still high the chance of it undergoing radiative capture is fairly low, since the cross-sections of the reactor materials for this reaction are small at high neutron energies. As it slows down to the energy region below about 1 keV, however, it begins to run into the very large U-238 resonances mentioned in the previous module. If the natural uranium is mixed uniformly with the moderator, the probability of neutrons being captured in the U-238 resonances is so high that the number surviving is insufficient to give criticality with any moderator other than heavy water; even with D_2O, the margin is too small to permit the design of a ***practical*** reactor.

Fuel lumping

The solution is to "lump" the fuel, that is, assemble it in bundles which are separated by regions of moderator. The main reason why this reduces the resonance absorption is because the absorption cross-section at the resonance energies is so large that any neutrons that come out of the moderator and enter the fuel bundle with an energy in the resonance range will almost certainly be absorbed before penetrating very far into the bundle. All the fuel in the inner region of the bundle is thereby effectively screened from resonance neutrons. The effect of lumping is to reduce the amount of fuel exposed to neutrons in the resonance range. An additional but minor advantage is that some of the fast neutrons will slow down through the whole range of the resonance energies while still in the moderator and only come into contact with fuel when their energy has fallen below the undesirable value.

Lumping reduces the overall resonance absorption and allows more neutrons to survive to low energies and cause sufficient fissions to maintain the multiplication factor at or above one. The combination of a moderator and fuel lumping permits criticality to be achieved even when natural uranium is used as fuel. Great care must still to be taken, however, to select a moderator with the right combination of properties, and we will now look at the criteria that a moderating material must satisfy to fulfill its function.

4.5 MODERATOR PROPERTIES

4.5.1 The Mean Log Energy Decrement

We might first recall that there are two types of scattering that can lead to a loss of energy on the part of the neutrons involved. These are:

1. **Inelastic scattering**

2. **Elastic scattering**

As mentioned in Module 2, ***inelastic*** scattering is relatively unimportant in slowing down neutrons. ***Elastic*** scattering in ***fuel*** is unimportant because the neutron's energy loss in a collision with a heavy nucleus is so small. Consequently, the only important mechanism for neutron energy loss is ***elastic scattering from the moderator nuclei.***

A good moderator must have two basic properties: it must slow fast neutrons down as rapidly as possible through the resonance energy range, and it must have a low absorption cross-section so that it does not itself soak up neutrons required to maintain the chain reaction. Let's look at the first of these properties to begin with.

Mean logarithmic energy decrement per collision

In an elastic collision, the energy lost by the neutron depends on the mass of the target nucleus and the angle of collision. Since the angle of collision is totally random, we have to treat the problem statistically. The effectiveness of a moderator in slowing down neutrons can most conveniently be expressed as a quantity called the ***mean logarithmic energy decrement per collision***. This quantity is given the symbol ξ (Greek xi). The formal definition of the log decrement is given by the equation

$$N_C \xi = ln \frac{E_i}{E_f} \tag{4.2}$$

where

ξ = mean log energy decrement

E_i = initial energy of fission neutrons (2 MeV average)

E_f = final energy of thermal neutrons (0.025 eV at room temperature)

N_c = average number of collisions required to reduce the energy of the neutron from its initial to its final energy.

Using the given values of E_i and E_f in equation (4.2) gives

Collisions to thermalize

$$N_c = \frac{18.2}{\xi} \tag{4.3}$$

It can be shown fairly easily that the closer the mass of the scattering nucleus to the mass of the neutron itself, the greater the average energy loss per scattering event, and the smaller the average number of collisions required to take the fission neutron down to thermal energy. This is illustrated in Table 4.1, which shows the values of the log energy decrement for several materials, including the commonly-used moderators.

Table 4.1

Mean logarithmic energy decrements

Material	ξ	Collisions to thermalize
^{1}H *	1.000	18
^{2}H *	0.725	25
^{4}He *	0.425	43
^{9}Be	0.206	83
^{12}C	0.158	115
H_2O	0.927	20
D_2O	0.510	36
BeO	0.174	105

*Gases at standard temperature and pressure

Although it is not obvious without further analysis, we can relate these values of ξ to the average percentage loss of energy that will occur in a collision. For heavy water, for example, the average ***percentage*** loss of kinetic energy of a neutron is 40% per collision; stated differently, the average ratio of the energy after the collision to the energy before the collision is equal to 0.6. Since $(0.6)^{36}$ is equal to 10^{-8}, the 36 collisions quoted in the table is indeed the average number required to reduce the neutron energy from 2 MeV to about 0.025 eV (a factor of 10^{-8}).

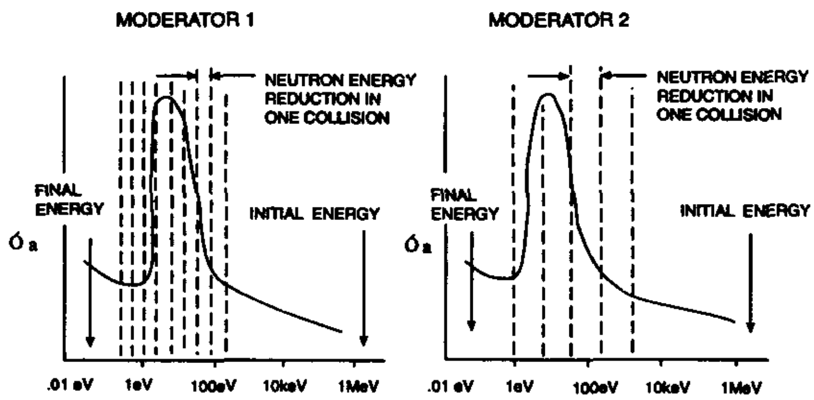

Figure 4.1: The fraction of the neutron's energy that is lost per collision is small on the left and large on the right

The importance of having as large an average energy loss per collision as possible is illustrated in Figure 4.1. Here, we are comparing the situations in two moderators, one of which has a small value of ξ, and the other a large one. (For simplicity, the U-238 resonance region has been smoothed out into a single resonance.) In the first case, it is clear that the neutrons will spend a fair amount of time going through the resonance region, while in the second, the time spent will be reduced since only a few collisions are required to take them through. The chance of a neutron encountering U-238 while its energy is in the resonance band will therefore be lower for the second moderator.

Notes & References

4.5.2 Slowing Down Powers And Moderating Ratios

A small number of collisions to thermalize the neutrons is obviously a desirable property for a moderator, but this in itself is of little use unless the collisions actually ***do*** occur. This implies that the macroscopic scattering cross-section of the moderator should be high, to increase the probability of collisions. We recall that

$$\Sigma_s = N\sigma_s \qquad (4.4)$$

This immediately rules out gases as moderators, since the atom number density (N) would be too small to allow the neutrons to be slowed down in a reasonable distance.

Since both ξ and Σ_s should be large for a good moderator, the overall effectiveness of a material in slowing down neutrons may be specified by quoting the ***product*** of these two quantities, known as the ***slowing down power***,

$$Slowing\ down\ power = \xi\Sigma_s \qquad (4.5)$$

Slowing down power

Table 4.2 shows the slowing down powers of the solid and liquid moderators. The value of the slowing down power of helium is also shown to demonstrate the unsuitability of a gas.

Not only must the moderator be effective in slowing down neutrons, but it must also have a ***small capture cross-section***. Neutrons are slowed down to decrease U-238 resonance captures compared to fission captures, and obviously the whole purpose of moderation would be defeated if the moderator nuclei themselves captured too many neutrons.

A reasonable indiction of the overall quality of a moderator is therefore the ***moderating ratio***, which combines the slowing down power and the macroscopic capture cross-section

Moderating ratio slowing down

$$\textit{Moderating ratio} = \frac{\xi \Sigma_s}{\Sigma_a} \qquad (4.6)$$

Table 4.2

Slowing down powers and moderating ratios

	ξ	$\Sigma_s (cm^{-1})$[a]	$\xi\Sigma_s$	Σ_a (cm^{-1})	$\xi\Sigma_s/\Sigma_a$
He[b]	0.425	21×10^{-6}	9×10^{-6}	? very small	? large
Be	0.206	0.74	0.15	1.17×10^{-3}	130
C[c]	0.158	0.38	0.06	0.38×10^{-3}	160
BeO	0.174	0.69	0.12	0.68×10^{-3}	180
H_2O	0.927	1.47	1.36	22×10^{-3}	60
D_2O	0.510	0.35	0.18	0.33×10^{-4}[d]	5500[d]
D_2O	0.510	0.35	0.18	0.88×10^{-4}[e]	2047[e]
D_2O	0.510	0.35	0.18	2.53×10^{-4}[f]	712[f]

a) average value for neutrons between 1 and 1000 eV

b) at standard temperature and pressure

c) reactor-grade graphite

d) 100% pure D_2O

e) 99.75% D_2O by weight

f) 99.0% D_2O by weight

The higher the moderating ratio, the better the material as a moderator. As can be seen by the moderating ratios presented in Table 4.2, heavy water is by far the best moderator and so it was adopted for the CANDU line of reactors. The value of Σ_a for heavy water is so low that natural uranium can even be used in compound form as UO_2, whereas reactors moderated by light water require the use of uranium enriched in the U-235 isotope.

The substance used as a moderator must be very pure, since it is usually present in larger amounts than any other material in the reactor. The concern in using D_2O as a moderator is to limit the amount of H_2O impurity to the lowest practical level. The quality of the D_2O moderator is specified by a parameter called the ***isotopic***, which is defined as the ***weight of D_2O divided by the total weight of D_2O plus H_2O*** in a given sample. For instance, if in a 20 g sample we have 19.6 g of D_2O and 0.4 g of H_2O, the isotopic (expressed as a percentage) will be

Moderator isotopic

$$\frac{19.60}{19.60+0.40}\times 100 = 98\%$$

Reactor grade D_2O

The acceptable value of isotopic in CANDU plants is 99.50% and up. While the reference value is set at 99.75%, some stations typically operate at about 99.9%. Heavy water with an isotopic of 99.75% or more is known as ***reactor grade*** D_2O. Because the absorption cross-section of hydrogen is so much larger than that of deuterium, any significant downgrading of the moderator purity produces a marked effect on the neutron multiplication factor (defined in Section 4.3). At 99.8% isotopic, for example, half of the absorption in the moderator will be due to the 0.2% light water impurity. Figure 4.2 shows the change in multiplication factor with moderator isotopic for a typical CANDU. Downgrading the moderator by only 0.1% reduces the multiplication factor from 1.000 to 0.997. By contrast, downgrading of the heavy water in the heat transport system produces a much smaller effect, simply because there is so much less of it in the core. The isotopic of the coolant would have to be downgraded to about 97% in order to produce the same change in multiplication factor.

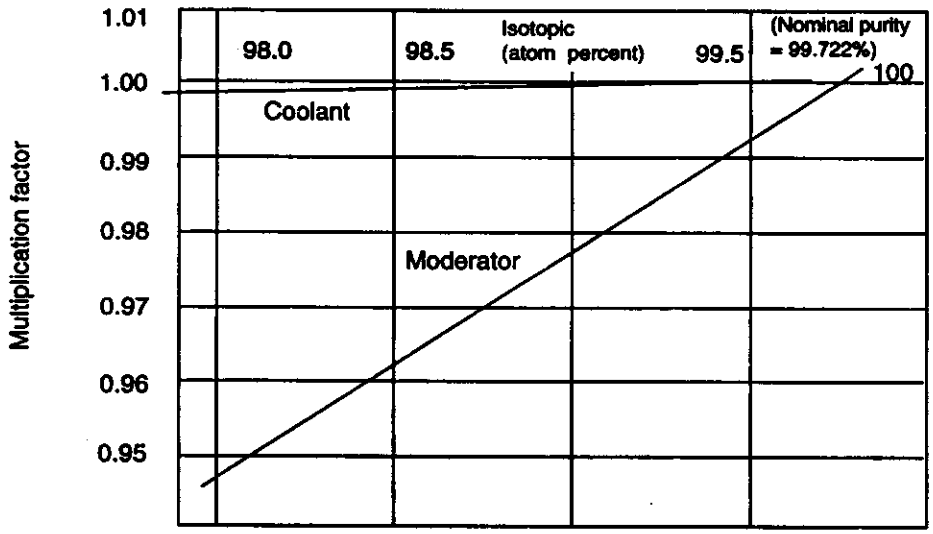

Figure 4.2: Change of multiplication factor with moderator isotopic

4.6 VARIATION OF K WITH LATTICE PITCH

Neutron diffusion

The distance between adjacent fuel channels in a CANDU reactor is called the ***lattice pitch***. To understand how the lattice pitch is chosen, we should first review how neutrons move through the moderator as they go from birth as fission neutrons to absorption by fuel or other materials. This process is known as ***neutron diffusion***. As shown in Figure 4.3, a neutron will follow a zigzag path, since its direction of motion is changed in a random manner at each scattering collision. The straight portions of path between two scattering collisions are about 1 to 3 cm. in length in most media. (Neutrons virtually never collide with each other because the neutron density is so very much smaller than the atomic density of the medium).

Notes & References

In Figure 4.3, a fission neutron born at point A is slowed down as a result of several collisions and reaches thermal energy at point B. In a CANDU reactor, the average distance between A and B is about 25 cm. The neutron then diffuses, as a thermal neutron, to point C where it is absorbed. The average distance between B and C is about 40 cm. The values quoted are "crow-flight" distances; the total path lengths actually followed by the neutron are considerably larger.

Overmoderation and undermoderation

The lattice pitch must be large enough to allow for sufficient moderator to thermalize the neutrons thoroughly and so minimize the U-238 resonance absorption. The variation of the neutron multiplication factor, k, with lattice pitch is shown in Figure 4.4. Note that k has a maximum value at a particular lattice pitch, and that it decreases steadily if we move to either smaller or larger lattice pitches. If we increase the lattice pitch above the optimum value, k decreases because the extra moderator absorbs too many neutrons before they have a chance of being absorbed in the fuel. In this case, the reactor is said to be ***overmoderated.*** If we decrease the pitch below the optimum, there is too little moderator to thermalize the neutrons properly and more of them are absorbed in the U-238 resonances before they have a chance to become thermal. In this case, the reactor is ***undermoderated.***

All large CANDUs are slightly overmoderated to meet physical rather than nuclear limitations. The pressure tubes must be sufficiently separated to allow fuelling machine access to the fittings on either end of the fuel channel, and the calandria tubes must have sufficient separation to allow horizontal and vertical space for the control mechanism guide tubes. As you can see in Figure 4.4, overmoderating the reactor by a small amount has little effect on the neutron multiplication factor, since the curve of k plotted against lattice pitch is fairly flat near the optimum value.

Notes & References

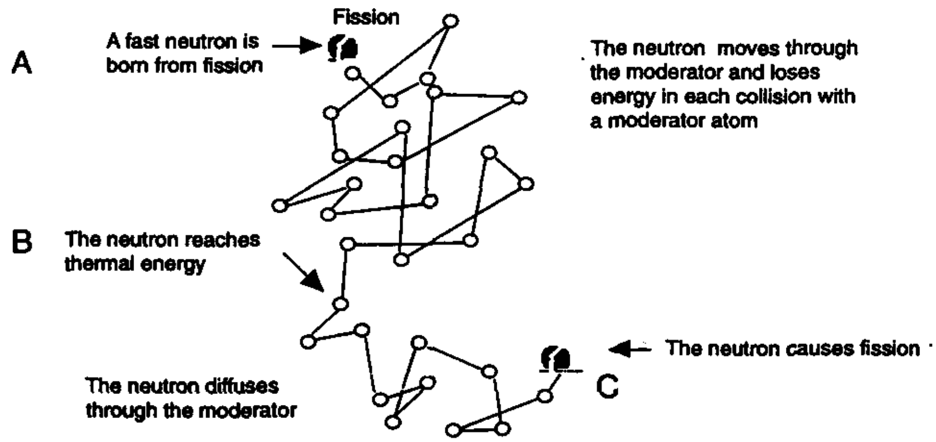

Figure 4.3: Path of a neutron from birth to absorption

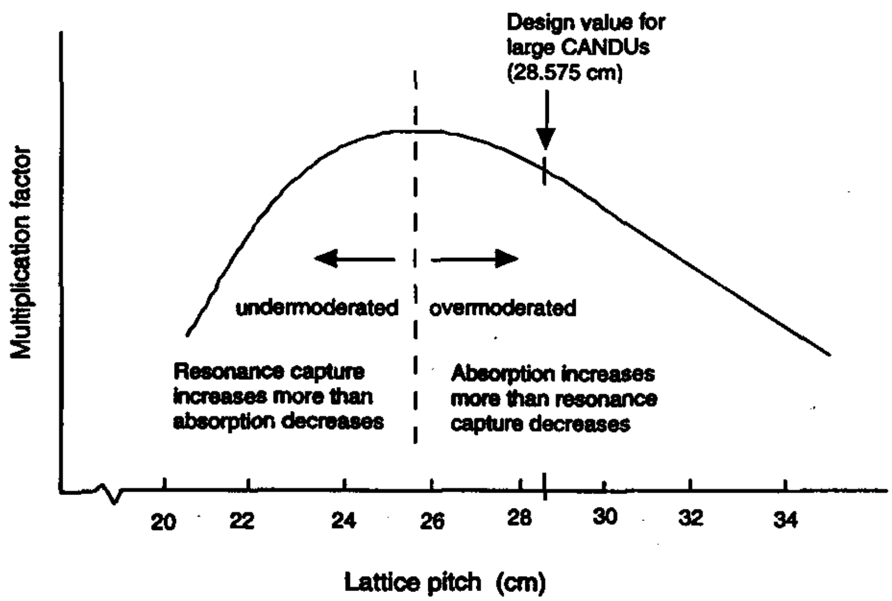

Figure 4.4: Variation of k with pitch of lattice

Notes & References

ASSIGNMENT

1. Explain why "lumping" the fuel is necessary to make a natural uranium reactor critical.

2. Discuss the relative importance of elastic and inelastic scattering in the fuel and moderator in a CANDU reactor.

3. The value of ξ for the nuclide boron-10 is 0.187. On average, how many collisions are required to thermalize fission neutrons in boron-10? How useful would it be as a moderator?

4. Define the following terms:

   i) slowing down power;

   ii) moderating ratio.

5. What are the relative advantages and disadvantages of H_2O and D_2O as moderators?

6. Explain: (a) why there is an optimum lattice pitch that gives a reactivity maximum for the core; and (b) why the CANDU pitch is greater than this value.