ATOMIC ENERGY OF CANADA LIMITED
Power Projects, Sheridan Park, Ontario

Lecture 4

Nuclear power symposium

REACTOR PHYSICS

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ATOMIC ENERGY OF CANADA LIMITED
Power Projects

NUCLEAR POWER SYMPOSIUM

LECTURE NO. 4: REACTOR PHYSICS

by

E. Critoph

1. "SPLITTING THE ATOM": A SOURCE OF ENERGY

In 1937-8 when I was a young boy, 8 or 9 years old, living in B. C., I had a friend who was considerably older attending high school. He was a constant source of new information, and I can still remember him telling me several times that if a way to "split the atom" were discovered it would make an enormous store of energy available. I, of course, found this very interesting but at the same time extremely baffling. I couldn't understand (a) how "splitting the atom" could produce energy, (b) how anyone knew that "splitting the atom" would produce energy if they didn't know how to do it and hence presumably had never done it, and (c) why someone didn't just go ahead and "split the atom" in a brute force way (maybe with a small sharp axe).

In case any of you have had similar questions and have not been fortunate enough to have them entirely resolved, I would like to take the first few minutes of my talk to recall the situation in 1937-8, and try to show you the answers to this type of question.

1.1 Equivalence of Mass and Energy

Well before 1937-8 Einstein's famous formula for the equivalence of mass and energy, which was first proposed in 1903 as a consequence of his Special Theory of Relativity, was generally accepted.

In this formula, $E = mC^2$, if the mass, m, is in grams, and the velocity of light, C, is in cm/sec, then the energy, E, will be in ergs.

Therefore, since the velocity of light is 3×10^{10} cm/sec,

1 lb (454 g) of mass is equivalent to 4.1×10^{23} ergs
or 4.1×10^{16} watt-sec
or 1.14×10^{10} kWh
or 4.75×10^{5} MWd

To put this in perspective: to get 1.14 x 10^{10} kWh of heat from bituminous coal requires the burning of 1,500,000 tons.

1.2 Mass Defect

To see how this fits in with the possibility of acquiring energy by "splitting the atom" we must now consider a phenomenon called the mass defect.

One might think offhand that, since any atom is composed entirely of neutrons, protons and electrons, the mass of the atom would be the sum of the masses of the neutrons, protons and electron present in the atom. However, careful measurements of atomic masses show that they are always less than this and the difference is called the mass defect for that atom.

The mass defect arises from the fact that there are strong nuclear forces holding the protons and neutrons in a nucleus of an atom together. The mass defect represents the energy which would be released in the process of assembling the particular atom from the requisite number of protons, neutrons and electrons. The same amount of energy would have to be supplied to break up an atom into its constituent parts (protons, neutrons and electrons). The energy equivalent of the mass defect is appropriately called the binding energy of the particular nuclide.

1.3 Binding Energy Per Nucleon

If the binding energy of a nuclide is divided by the number of nucleons (i.e., protons plus neutrons), the result is the binding energy per nucleon. Figure 1 is a plot of the binding energy per nucleon as a function of mass number (i.e., number of protons plus neutrons in the nucleus).

1.4 Energy From "Splitting The Atom"

Let us examine this curve of binding energy per nucleon vs mass number a little more closely and see what it tells us. Notice that the curve starts at quite low values for light elements, climbs to a maximum for mass numbers in the range 50-100 and then decreases slowly for further increases in mass number. It is this last slow decrease which explains why energy can be obtained by "splitting the atom".

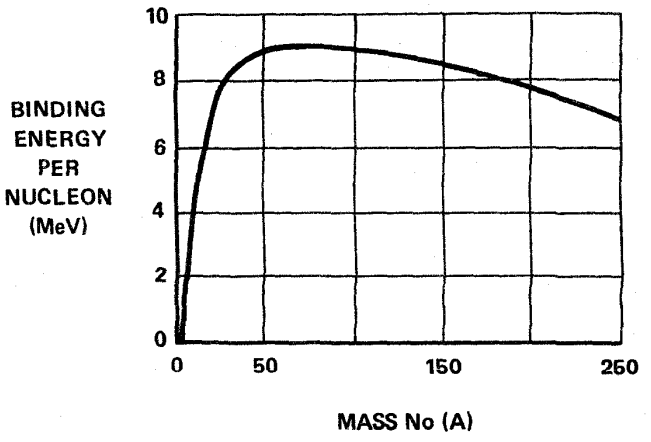

Figure 1 Binding Energy Per Nucleon Vs. A

As an example let us calculate the energy released if we take a pound of material with mass number 250 and split each nuclide into two nuclides of mass number 125.

Total number of nucleons (protons + neutrons):

$\approx 454 \times 6 \times 10^{23}$

$\simeq 2.72 \times 10^{26}$ nucleons

From Figure 1 the binding energy per nucleon goes from about 7.3 MeV for mass number 250 to 8.4 MeV for mass number 125. Therefore, the total increase in binding energy is:

$(8.4 - 7.3)(2.72 \times 10^{26})$ MeV

$= 3 \times 10^{26}$ MeV.

(The MeV is a unit of energy:

1 MeV = 1.6×10^{-6} ergs

$= 4.5 \times 10^{-20}$ kWh)

Therefore, the total energy released in "splitting the atoms" in 1 pound of mass number 250 material is:

$(3 \times 10^{26})(4.5 \times 10^{-20}) = \underline{1.35 \times 10^{7} \text{ kWh}}$

(or the equivalent of burning 1750 tons of coal).

Note that in doing this we have only converted a little over 0.1% of the original mass into energy.

1.5 Fusion

Note also in Figure 1 that if we were to combine the nuclides of a very light element together to form nuclides with twice the number of nucleons, much more energy would be released per nucleon than is obtained by splitting heavy atoms. This process is referred to as fusion and has, of course, great potential for power production. There are still very difficult problems to surmount in using fusion on a large scale, and since this is not the subject of this talk, I will say no more about it.

We have now answered the questions of how "splitting the atom" can lead to energy release, and how people realized this was so before knowing how to do it. We will now go on to a discussion of a discovery which made it feasible to use this process as a basis for commercial power production.

2. FISSION

In 1939 a way to split the nucleus of certain heavy isotopes was discovered by Hahn and Strassmann. They found that when these isotopes were bombarded by neutrons (which were discovered in 1932 by Chadwick and about which we will say more later), some of the nuclei divided into two more or less equal fragments with, of course, an associated release of energy. This process of division was called fission.

Fission can also occur spontaneously in a number of materials, particularly in uranium and plutonium isotopes.

However, this is a rare occurrence as can be seen from the fact that for ^{238}U (which is the uranium isotope which undergoes spontaneous fission most easily) it takes 8×10^{15} years for half the nuclides in a large sample to fission spontaneously. The spontaneous fission half-lives of a number of isotopes are given in Figure 2.

There are three very important characteristics of fission, namely:

(i) Two nuclides, of approximately equal mass, are produced and these are usually radioactive. These are called fission products or fission fragments. Figure 3 shows the distribution of fission products obtained when ^{235}U is fissioned by neutrons.

ISOTOPE	S.F. 1/2 LIFE
^{233}U	$\geq 3 \times 10^{17}$ YRS.
^{234}U	1.6×10^{16} YRS.
^{235}U	2×10^{17} YRS.
^{236}U	2×10^{16} YRS.
^{238}U	8×10^{15} YRS.
^{239}Pu	5.5×10^{15} YRS.
^{240}Pu	1.2×10^{11} YRS.
^{242}Pu	6.6×10^{10} YRS.

Figure 2 Spontaneous Fission Half-Lives

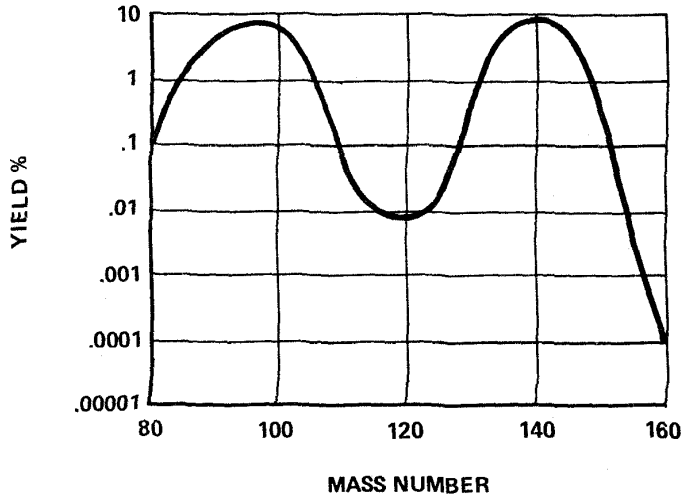

Figure 3 Fission Product Yields

(ii) Energy is released for the reasons we have discussed. Most of this energy is in the form of kinetic energy of fission fragments but some is also in the form of β -and γ -ray energy, neutrinos, and neutron kinetic energy. Figure 4 is a tabulation of the distribution of energy release among these forms, for fission of ^{235}U caused by neutrons.

FORM	ENERGY PER FISSION (MeV)
1. K.E. FISSION FRAG.	167.0
2. K.E. NEUTRONS	4.8
3. PROMPT γ'S	7.2
4. F.P. γ'S	6.1
5. F.P. β'S	7.4
6. NEUTRINOS	10.4
	202.9

Figure 4 Distribution of Energy Release from ^{235}U Fission

(iii) Typically 2 or 3 neutrons are produced by the fission process. Figure 5 lists the average number of neutrons produced per fission in several fissile isotopes when fission is caused by "thermal" neutrons (the term "thermal" neutron will be explained later).

ISOTOPE	ν
^{233}U	2.49
^{235}U	2.43
^{239}Pu	2.87
^{241}Pu	2.97

Figure 5 Average Number of Neutrons per Fission (ν)

The bulk (>99%) of the neutrons produced as a result of fission are "prompt"; that is, they are produced during the actual fission process. A small fraction of the neutrons are produced later by the decay of fission products. These are referred to as "delayed" neutrons and we shall see later that these are very important when we come to thinking of devices for commercial power production. Figure 6 shows the percentage of neutrons released as delayed neutrons for neutron-induced fission of several isotopes. Six groups of delayed neutrons have been identified, each having a characteristic half-life associated with their production. Figure 7 shows the distribution among these groups of the delayed neutrons from ^{235}U fission as well as the half-lives associated with the groups.

The prompt neutrons produced in fission are quite energetic. Figure 8 shows the distribution of neutron energies from thermal neutron induced fission of ^{235}U. The average kinetic energy is about 2 MeV and the most probable kinetic energy around 1 MeV. (An energy of 1 MeV corresponds to a velocity of about 10^7 m/s). Delayed neutrons have definite energies associated with them depending on which group they belong to. These energies are lower than the average energy for prompt neutrons, being in the neighbourhood of a few tenths of an MeV.

ISOTOPE	% NEUTRONS DELAYED
^{233}U	.265
^{235}U	.650
^{239}Pu	.212

Figure 6 % Delayed Neutrons from Neutron-Induced Fission

DELAYED NEUTRON GROUP	HALF LIFE (SEC)	%OF ALL NEUTRONS IN GROUP
1	55.7	.021
2	22.7	.142
3	6.2	.128
4	2.3	.257
5	.61	.075
6	.23	.027

Figure 7 Distribution of Delayed Neutrons from Neutron-Induced Fission of ^{235}U

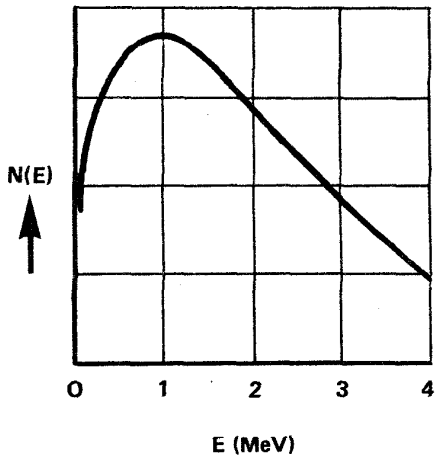

Figure 8 Fission Neutron Energy Spectrum

3. THE CHAIN REACTION

In providing a method of "splitting the atom", nature has been kinder to us than was strictly necessary. The fission process not only solves the problem of how to "split the atom" but does so in a way which makes it easy to achieve large scale power production by means of a chain reaction.

The fission process, which is initiated by a neutron, itself produces more than 2 neutrons. Therefore, as long as we have a neutron to trigger the first fission we are automatically supplied with more neutrons to produce further fissions which, in turn, produce further neutrons and so on. Provided we are not too wasteful of the neutrons (i.e., by losing them in other ways than by causing fission) the process can continue until most of the fissile material has been consumed. Hence we can easily derive most of the stored energy from the fissile material without supplying any external energy (since enough neutrons to start the process are available naturally).

4. NEUTRONS

By now I am sure most of you are convinced that it would be a relatively simple task to design a reactor (i.e., a device for using fission to produce neutrons and energy by means of a chain reaction) and we have not yet started to consider the main body of reactor physics. I do not want to disillusion you, since reactor design is in principle simple, but there are a few details which I have rather glossed over which must be understood before one could successfully design a good reactor. Since neutrons are the key agent in making a reactor work, let us start by considering the properties of neutrons and the ways in which neutron interact with matter.

The neutron has no electrical charge and therefore a neutron travelling through a material does not interact with the electrons and nuclei through their electric fields, but only by "direct collision" with the nuclei. Since nuclei are very small compared to atoms, neutrons generally travel relatively long distances (usually measured in cms) between such interactions.

4.1 Neutron Reactions With Materials

When a neutron does collide with a nucleus there are three main reactions which can take place: scattering, capture or fission.

In a scattering reaction the neutron essentially bounces off the nucleus, changing its direction of motion. There are two kinds of scattering, elastic and inelastic. In elastic scattering the neutron and nucleus obey the classical laws of collisions between rigid spheres whereas in inelastic scattering there is an additional energy loss. Capture means just what it says - the neutron collides with the nucleus and combines with it to form a new nucleus. Usually β and γ -rays are emitted from the nucleus after a neutron capture. We have already discussed the fission reaction fairly thoroughly.

4.2 Neutron Cross Sections

The mean distance a neutron travels through a material before undergoing a given type of reaction, or the mean free path for the reaction, is related to the neutron cross section of the material for that reaction. Neutron cross sections have the dimensions of area and are usually quoted in barns (1 barn = 10^{-24} cm^2). The relationship is as follows:

$$\ell = \frac{1}{N\sigma}$$

where

ℓ = the mean free path,
N = the number of nuclei per unit volume,
σ = neutron cross section.

($N\sigma$) is often denoted by Σ and is called the macroscopic cross section for the reaction. It has the dimensions of inverse length and is the probability per unit length of neutron travel of the reaction occurring (in that material). Neutron cross sections have been compiled in many publications, the best known of which is probably BNL-325.*

For example, iron (Fe) has the following neutron cross sections for neutrons with an energy of .025 eV:

Capture:	σ_c	= 2.62 barns
Scattering:	σ_s	= 11 barns
Fission:	σ_f	= 0 barns
Total	σ_t	= 13.62 barns

Since iron has a density of 7.86 g/cm^3 and a gram atomic weight of 55.85 g, the number of nuclei/cm^3 is:

$$\frac{6.02 \times 10^{23}}{55.85} (7.86) = 8.48 \times 10^{22}/\text{cm}^3$$

* BNL = Brookhaven National Laboratory

(6.02×10^{23} is Avagadro's number or the number of atoms in a gram atomic weight.)

Therefore, the macroscopic cross sections are:

$$\Sigma_c = 0.222 \text{ cm}^{-1} \left[\text{i.e. } (8.48 \times 10^{22})(2.62 \times 10^{-24})\right]$$
$$\Sigma_s = 0.933 \text{ cm}^{-1}$$
$$\Sigma_f = 0$$
$$\Sigma_t = 1.15 \text{ cm}^{-1}$$

and the mean free paths are:

$$\ell_c = 4.5 \text{ cm}$$
$$\ell_s = 1.07 \text{ cm}$$
$$\ell_f = \infty$$
$$\ell_t = 0.87 \text{ cm}$$

(The total mean free path, ℓ_t, is the mean distance between collisions of any type.)

Neutron cross sections are functions of neutron energy. Capture and fission cross sections usually have the general behaviour shown in Figure 9. At low neutron energies the cross section is inversely proportional to the neutron velocity. Going up in energy we enter the "resonance region" in which the cross section fluctuates widely. At higher energies it becomes quite small and decreasing.

Scattering cross sections also vary with energy but usually far less than fission and capture cross sections.

4.3 Neutron Flux and Reaction Rates

In a nuclear reactor producing significant power, as you can well imagine, there are neutrons flying about in all directions. If we could take an instantaneous picture we could count the number of neutrons per unit volume. This is called the neutron density denoted by ρ (see Figure 10). If now we were able to take a time exposure we could measure the total neutron track length in a unit volume in a unit time. This is called "neutron flux", denoted by ϕ (see Figure 11).

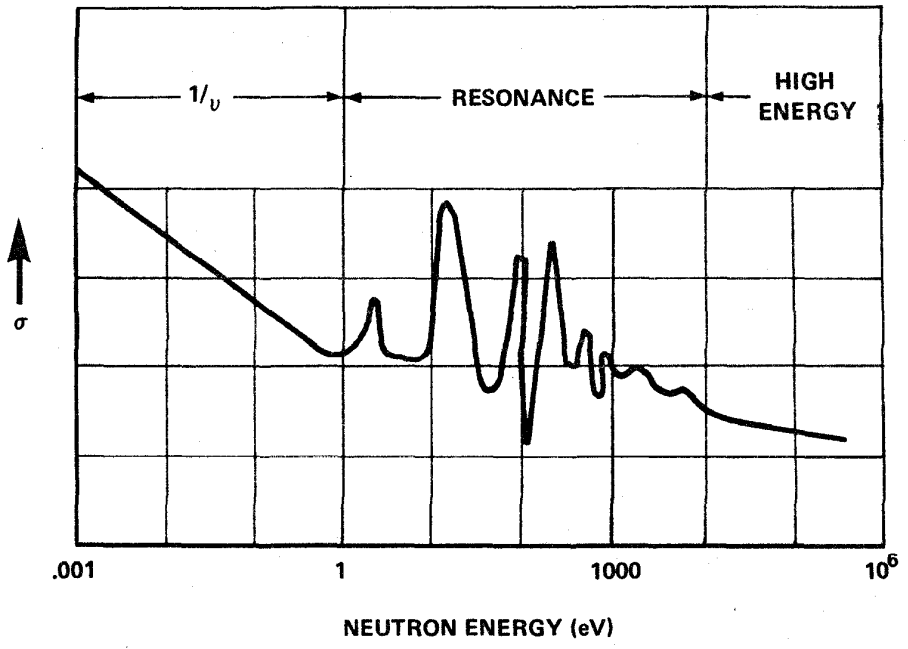

Figure 9 Typical Behaviour of Neutron Capture or Fission Cross Section With Energy

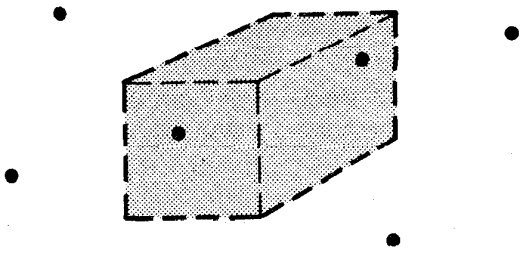

Figure 10 Neutron Density, ρ

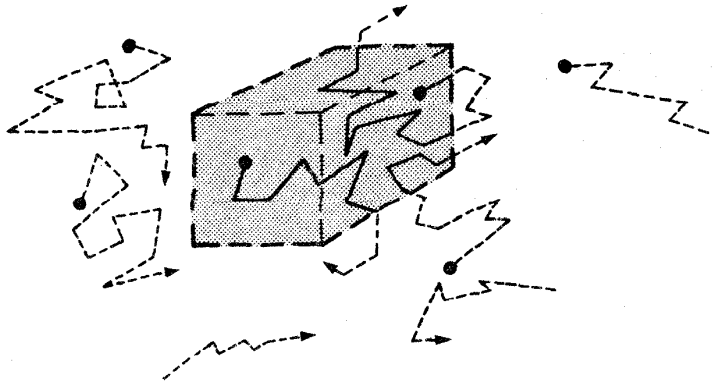

Figure 11 Neutron Flux, ϕ

$$\phi = \rho \bar{v}$$

where $\bar{v}$ is the average neutron velocity.

The reaction rate per *unit volume (i.e., the number of reactions* taking place per second per unit volume) for any particular neutron reaction is given by:

$$\text{Reaction rate} = N\sigma\phi$$

where N is the number of nuclei per unit volume,
σ is the neutron cross section of the nuclide for that reaction (suitably averaged over the range of neutron energies present),
and ϕ is the neutron flux.

4.4 Neutron Slowing Down

We have seen that the neutrons emitted in fission are energetic, with high velocities. For such energetic neutrons, in most materials the scattering cross section is much higher than the fission or capture cross sections. In these materials the neutron, on the average, undergoes *many scattering collisions* before it is captured or causes fission. The question is "What effect do these scattering collisions have on the neutron?" The answer is that the neutron slows down.

Since in most cases elastic scattering is the dominant process we can easily apply the conservation laws for *collisions between rigid spheres* to calculate the extent of this slowing down.

The results are fairly simple. If the neutron is scattered by a nucleus with mass number A, and if E_i is the energy before the collision and E_f the energy after the collision:

$$\left(\frac{A-1}{A+1}\right)^2 E_i \leq E_f \leq E_i$$

For example:

(i) After a collision with a deuterium nucleus (A = 2) the neutron energy lies between the initial energy and one-ninth of the initial energy.

(ii) After a collision with a mercury nucleus (A = 200) the neutron energy lies between the initial energy and 0.98 times the initial energy.

The probability of any final energy between these limits is almost constant.

There is a lower limit to the energy to which the neutron will be slowed down, which depends on the temperature of the medium. The atoms of the medium are in motion and so the neutron eventually reaches a range of energy where it is just as likely to gain energy in a collision with a nucleus as lose it. When this happens the neutron is in thermal equilibrium with the medium and such neutrons are referred to as "thermal neutrons". Such neutrons have a distribution in energy called a "Maxwellian" distribution (see Figure 12). At room temperature the most probable neutron velocity in such a distribution is 2200 m/s corresponding to an energy of .025 eV.

The number of elastic collisions required to reduce the neutron energy from fission energies ($\approx 2 \times 10^6$ eV) to thermal energies ($\approx .025$ eV) is given by:

$$N_{COLL} = \frac{18.2}{\xi}$$

where ξ is the average logarithmic energy decrement per collision and is given by:

$$\xi = 1 + \frac{(A-1)^2}{2A} \ln \frac{A-1}{A+1}$$

$$\approx \frac{2}{A + 2/3} \quad (\text{for } A > 10)$$

Figure 13 shows ξ vs A.

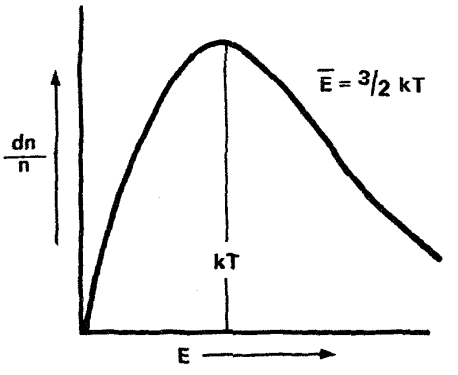

Figure 12 Maxwellian Distribution

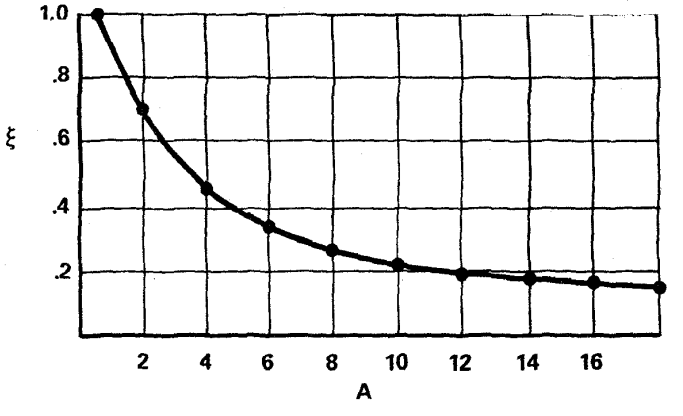

Figure 13 ξ vs A

Examples:

(i) For fission neutrons slowing down in deuterium (A = 2):

$$\xi = 0.725 \text{ and } N_{COLL} = 25$$

(ii) For fission neutrons slowing down in carbon (A = 12):

$$\xi = 0.158 \text{ and } N_{COLL} = 112$$

4.5 Neutron Migration

The other important effect of scattering collisions is that they change the direction of motion of the neutron. On each scattering the direction is changed by any amount up to 180^{o} and as a result the path traced out by the neutron is very erratic.

We already know how to calculate the mean path length of neutrons between different types of reactions. However, it is obvious that in order to understand reactors in detail we must have a statistical knowledge of the spatial distribution of the reactions arising from neutrons born at a certain point. Therefore, we must relate our knowledge of mean free paths to spatial distances, taking into account the erratic paths caused by scattering.

There are several ways of doing this which differ in accuracy and difficulty of application. Unfortunately we do not have time to go into these in detail, and the best I can do is to give you a vague feeling for three of them: diffusion theory, transport theory and Monte Carlo methods.

Diffusion Theory makes two very sweeping assumptions:

(i) The probability of scattering is very much larger than that for capture or fission.

(ii) Scattering isotropic (i.e., the probability of the scattered neutron going off in any direction) is constant.

With these assumptions a differential equation can be derived which relates the value of neutron flux at any position in a region to the material properties of the region (macroscopic cross sections), neutron sources in the region, and neutron fluxes and currents at the boundaries of the region. This is called the diffusion equation.

$$\nabla^2 \phi - \frac{3}{\ell_s \ell_a} \phi + \frac{3}{\ell_s} S = \frac{3}{\ell_s} \frac{\partial \rho}{\partial t}$$

where S is the source per unit volume per second.

Diffusion theory also relates the net neutron current at a boundary to the shape of the flux distribution. In diffusion theory the relationship between path lengths and spatial distances turns out to be:

$$\text{Diffusion length} = \sqrt{\frac{\overline{r^2}}{6}} = \sqrt{\frac{\ell_s \ell_a}{3}}$$

where $\overline{r^2}$ is the root mean square distance travelled.

Transport theory does not make the two simplifying assumptions of diffusion theory and therefore gives more accurate results. On the other hand it is more difficult and time consuming to apply. Fortunately modifications to diffusion theory based on transport theory can be made which increase the accuracy without making application more difficult. Diffusion theory with transport corrections is probably the most widely used method.

I thought Monte Carlo methods worth mentioning since they are conceptually so simple, although in practice often too time consuming to be practical. As suggested by the name, the method embodies the use of chance. In essence the histories of many neutrons are followed. All decisions as to distance between reactions, type of reactions, and change in neutron direction are made by selecting random numbers. Information on spatial distribution of events is then based on these histories.

So far we have treated neutrons slowing down and migration separately, whereas they occur at the same time.

Multi-group theory is usually used to combine them. The neutrons are divided into energy groups. Within any group the neutrons are assumed to obey diffusion theory with transport corrections, or transport theory. Neutrons slowing down below the lower energy limit of any group are treated as being captured with respect to that group and as a source in one of the lower energy groups.

5. A SIMPLE REACTOR

Let us now consider the simplest reactor imaginable - basically just a sphere of ^{235}U.

5.1 Critical Size

The first question we must answer is how big to make the sphere in order to just sustain a chain reaction.

Suppose a fission takes place in the sphere producing 2.6 neutrons. These neutrons will eventually either escape from the sphere or be absorbed (i.e., a capture or fission reaction) by the ^{235}U, although they may first undergo several scatterings. Let the probability that a neutron will not escape from the sphere before being absorbed be P. Since for fission energy neutrons the ratio of fission to absorption cross section in ^{235}U is 0.85, the neutrons from the original fission will cause (2.6 x 0.85 x P) fissions.

If the neutrons from one fission cause on the average one more fission a chain reaction is just sustained and the reactor is said to be "critical". In our simple reactor the "criticality condition" is:

$$2.6 \times 0.85 \times P = 1$$

or $$P = 0.453$$

The neutrons from the first fission are said to belong to one generation and the neutrons from the fissions which they cause belong to the next generation and so on.

The "infinite multiplication factor" (k_∞) is the ratio of the number of neutrons in one generation to the number in the previous generation, neglecting leakage from the system (or assuming an infinite system). The "effective multiplication factor" (k_{eff}) is the ratio of the number of neutrons in one generation to the number in the previous generation taking leakage into account. The criticality condition in general is given by $k_{eff} = 1$.

In our simple case then:

$$k_\infty = 2.6 \times .85 = 2.21,$$

$$k_{eff} = k_\infty P = 2.21\ P$$

and the criticality condition is:

$$k_{eff} = 2.21\ P = 1,$$

or $$P = 0.453.$$

Obviously P is a function of the size of the sphere and provided we can calculate this function we can find the size for which P = .453. This is the "critical size".

The accurate calculation of "non-leakage probability" (P) requires the use of one of the methods mentioned under neutron migration and we do not have time to discuss this in detail.

However, let us make a very crude estimate of P for the purpose of illustration using only the knowledge we have acquired.

To do this assume:

(i) that there is no scattering, and

(ii) that the leakage probability is the same for all neutrons in the sphere independent of position.

For a neutron at the centre of the sphere the leakage probability, 1 - P, is the probability that the neutron will traverse a thickness R of ^{235}U (where R is the radius of the sphere) without being absorbed.

To calculate this probability we observe that if n neutrons are perpendicularly incident on a thin slice of material dx the number of neutrons absorbed in passing through the slice, -dn, is given by:

$$-dn = n\Sigma_a dx$$

Integrating $$\int_{n_o}^{n} \frac{dn}{n} = -\int_{o}^{R} \Sigma_a dx$$

Therefore $$\frac{n}{n_o} = e^{-\Sigma_a R} = 1 - P$$

For a fission spectrum the absorption cross section of ^{235}U is 1.65 barns.

Assuming a density of 18.7 g/cm^3 for ^{235}U, the number of atoms of ^{235}U per cm^3 is given by:

$$\frac{6.02 \times 10^{23}}{235} (18.7) = 4.8 \times 10^{22}/cm^3,$$

Therefore $$\Sigma_a = (4.8 \times 10^{22})(1.65 \times 10^{-24}) = .079 \text{ cm}^{-1}$$

Therefore $$P = 1 - e^{-.079R}$$

The value of R for which P is 0.453 is 7.7 cm

Therefore, this crude estimate gives a sphere with a critical radius of 7.7 cm.

The critical mass is:

$$4/3\ \pi(7.7)^3\ (18.7 \times 10^{-3}) = \underline{35.8\ kg}$$

A reactor, GODIVA, which was very similar to our simple reactor was built and operated in the U. S. The main difference was that it was made of 93.9% ^{235}U (plus 6.1% ^{238}U) whereas we have assumed 100% ^{235}U. Experimentally it was found that 48.8 kg of ^{235}U was needed for criticality.

The difference in U isotopic composition would lead to a small change compared to the errors in our crude estimate due to the assumptions we made. In fact, the reason we came as close as we did is that the two assumptions tend to lead to compensating errors - assuming no scattering tends to overestimate leakage whereas assuming all neutrons have the same leakage probability as those at the centre tends to underestimate it.

5.2 Neutron Flux vs Power

As a matter of interest let us calculate the average neutron flux in the sphere if it is operating at a power of 100 W.

$$P = E_f\,(NV)\,\sigma_f\,\phi$$

where P is the power,
E_f is the energy produced/fission
(NV) is the total number of ^{235}U atoms in the sphere,
σ_f is the fission cross section,
ϕ is the neutron flux.

Substituting:

$$E_f = 3.2 \times 10^{-11}\ \text{watt - s per fission},$$

$$(NV) = \frac{48,800}{235}\,(6.02 \times 10^{23}) = 1.25 \times 10^{26}$$

$$\sigma_f = 1.4 \times 10^{-24}\ cm^2$$

gives $$\phi = 1.8 \times 10^{10}\ n/cm^{-2}s^{-1}$$

If we solve the transport equation to obtain the flux distribution in this reactor we find that the flux peaks in the centre and is approaching zero at the surface.

5.3 Control

In our simple reactor we have discussed the condition for criticality ($k_{eff} = 1$).

If $k_{eff} > 1$ the reactor is said to be supercritical and the neutron flux, fission rate, and power will increase with time. If $k_{eff} < 1$ the reactor is sub-critical and the neutron flux, fission rate, and power will decrease with time.

Obviously in order to use a reactor we must be able to control it; making it supercritical when we want to start up or raise power, and sub-critical to shut down or lower power. We must do this by changing the reactivity, i.e., the amount by which k_{eff} differs from 1. In Canada the most common unit of reactivity is the milli-k which is 10^{-3} in k_{eff}.

There are several ways in which we could provide a variation in k_{eff} in our simple reactor in order to permit us to go supercritical, critical, or sub-critical at will.

(i) We could make the sphere a little bigger than required for criticality but split it in half and arrange to make the distance between the two hemispheres variable. Then with the hemispheres together the reactor would be supercritical. As the hemispheres were separated a position would be reached where the reactor was just critical and further separation would make it sub-critical. This, we control the reactor by varying the leakage. This was actually the method used in GODIVA.

(ii) A second method of control would be to again make the sphere a bit larger than required for criticality and then to drill a hole through it. The sphere radius and hole size would be chosen so that the reactor was a bit supercritical with the hole empty but sub-critical when a rod of purely absorbing material was fully inserted in the hole. The rod would change the ratio of fission to absorption cross section in the reactor and hence k_{eff}. By adjusting the position of the rod in the hole we could again control the power of the reactor.

Similarly we could make the rod of fissile material and attain control but the sign of the reactivity change with direction of motion of the rod would be reversed.

5.4 Kinetics and Safety

We have seen how to control our simple reactor but we have no idea yet as to the sensitivity of the reactor to changes in reactivity.

Suppose we are using leakage control, with the reactor running at steady power, and suddenly we move the hemispheres closer together by enough to increase the reactivity by 1 mk (i.e., increase k_{eff} to 1.001).

Let us first assume that all the neutrons are prompt and see what would happen. The absorption mean free path of fission neutrons in pure ^{235}U is about 12.5 cm, and their average velocity is about 10^7 m/s. Therefore the average lifetime of a neutron between birth and absorption is about 10^{-8} seconds. Thus, in 1 millisecond (10^{-3} s) there are about 10^5 neutron generations and the power would increase by a factor of $(1.001)^{105} \approx 2.7 \times 10^{43}$.

(If $k_{eff} - 1 \equiv \rho_{ex}$, and the neutron lifetime is denoted by ℓ, the power as a function of time would be given by:

$$P(t) = P_o(1 + \rho_{ex})^{\frac{t}{\ell}} \approx P_{oe}^{\frac{\rho_{ex} t}{\ell}}$$

(since $\ell n\,(1 + x) \approx x$ for x small)

$$P(t) = P_{oe}^{\frac{t}{T}}$$

where $T \equiv$ the reactor period $= \ell / \rho_{ex}$

Such extreme sensitivity to changes of k_{eff} would make such a reactor dangerous and very difficult to control. Fortunately it is not this bad. In the above analysis we neglected the delayed neutrons and these, although only making up about 0.7% of the neutrons from ^{235}U fission, have a tremendous effect on the kinetics. How does this come about?

Since the delayed neutrons are emitted with half-lives of the order of seconds, for times short compared to this, they do not reflect the sudden change in reactivity. Therefore, in the above example the multiplication factor between successive generations is initially not 1.001 but

$$\left(\frac{.007}{P/P_o} + .993\right)(1.001),$$

where P/P_o is the ratio of power to initial power.

Thus, we can see that for $P/P_0 = 1.17$ the multiplication factor is only 1. This means that in the first second or so the power cannot rise above 1.17 times its initial value (compared with the previously calculated factor of 2.7×10^{43} in 1 ms when we neglected delayed neutrons). In the general case, if β is the delayed neutron fraction the initial multiplication factor between generations will be:

$$\left[\beta \frac{P_0}{P} + (1 - \beta)\right](1 + \rho_{ex})$$

Note that if $\rho_{ex} > \beta$ the multiplication factor > 1 for any value of P/P_0, so again we would get into very fast power rises (i.e., we would be critical on prompt neutrons alone). The fast initial rise when $\rho_{ex} < \beta$ can be seen to be limited to a power ratio of

$$\frac{\beta}{\beta - \rho_{ex}} .$$

After the initial fast rise the power increase settled out into an exponential rise. The period of this rise is greatly affected by the delayed neutrons. Figure 14 shows reactor period vs excess reactivity taking delayed neutrons into account.

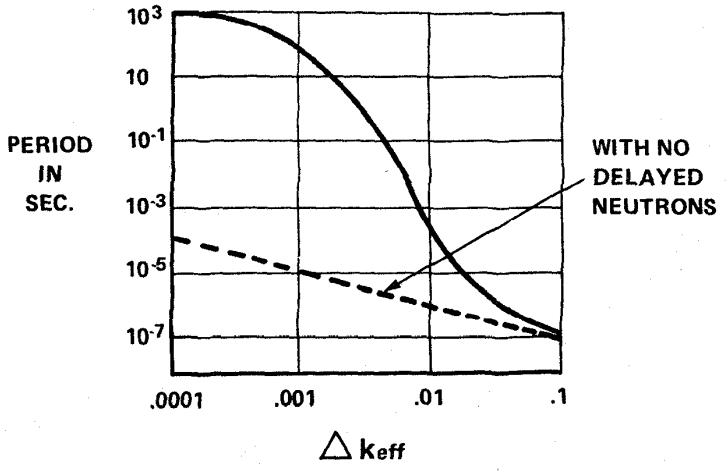

Figure 14 Reactor Period Vs Excess Reactivity
(For Simple Reactor)

We have only discussed the case of increased reactivity so far. The same type of conclusions hold for decreases in reactivity. As long as the reactivity decrease is less than β the rates of change of power are greatly affected by the delayed neutrons and are much slower than they would be if there were no delayed neutrons.

In summary, as long as sudden reactivity changes are kept small compared to the delayed neutron fraction, the kinetics of the reactor are quite reasonable. Safety considerations usually require that any sudden reactivity changes larger than this be of extremely low probability.

6. PHYSICAL PARAMETERS AND COMPARATIVE PHYSICS OF DIFFERENT REACTOR TYPES

6.1 Changes to Simple Reactor Required for Power Production

The simple reactor which we have been talking about is not suitable for use as a power reactor for the reasons discussed below.

(i) The power which can be extracted is too low. We would be lucky if we were able to extract a few kilowatts of heat from it and yet the capital investment, with separated ^{235}U costing about \$12/g, would be over \$0.5 million. Obviously the direction to go is towards that of heat exchanger design. We must increase the surface area of the heat producing elements and force coolant past these surfaces. This means the addition of coolant to the reactor core.
(NOTE: It should be mentioned here that some materials, notably ^{238}U and ^{232}Th, are fertile, i.e., when they capture a neutron they become fissile - ^{238}U goes to ^{239}Pu and ^{232}Th goes to ^{233}U. The abundance of the naturally occurring fertile material is several hundred times that of fissile material. For example, natural uranium contains roughly 137 parts of ^{238}U to one part of ^{235}U. Most practical power reactors use fuel containing some fertile material since this reduces separation costs and also makes efficient use of neutrons.)

(ii) It turns out the uranium metal fuel does not give very good performance in a reactor if long burnup is desired. This difficulty can be alleviated by adding alloying materials but the better solution is to use ceramic fuel. Currently oxide fuel is favoured although carbide has advantages in some cases and its use is being developed. Ceramic uranium fuels have a lower uranium density than metallic uranium.

(iii) The heat producing fuel elements must be clad to prevent fission products from entering the coolant and contaminating the coolant circuits. Cladding also prevents corrosion of the fuel by the coolant. Therefore, cladding material must be added to the core.

(iv) (Another point which will be mentioned here although it does not affect our further discussion is that the whole reactor must be shielded with a sufficient thickness of heavy material to reduce radiation fields in the vicinity to a tolerable level.)

6.2 Effects of Adding Materials to the Core

The major changes required to make our simple reactor suitable for power production involve the addition of materials to the core. What effects will they have?

The first effect is that addition of non-fissile material (and the change to ceramic fuel) will decrease the overall density of fissile material in the core. For this reason alone the core will have to be made larger.

All materials capture neutrons to a certain extent and so the addition of non-fissile material will decrease the fission to absorption ratio and hence reduce the infinite multiplication factor.

Finally all materials slow down neutrons to a certain extent and so the addition of materials will tend to degrade the neutron spectrum (i.e., lower the average energy of the neutrons in the reactor).

6.3 Types of Reactors

The above situation brings us to a cross-road, as I will now attempt to show. We have already seen the general behaviour of capture and fission cross sections with neutron energy - they both tend to increase as the neutron energy decreases. In fact, the fission cross section for ^{235}U goes from about 1.4 barns for a fission neutron spectrum to about 580 barns for a Maxwellian spectrum at room temperature. Obviously there are advantages to having higher fission cross sections.

Now observe Figure 15, which shows the average number of neutrons emitted per absorption, η, in the fissile isotopes ^{235}U, ^{239}Pu, and ^{233}U. η is the average number of neutrons per fission times the ratio of fission to absorption cross section, and is a key parameter in the

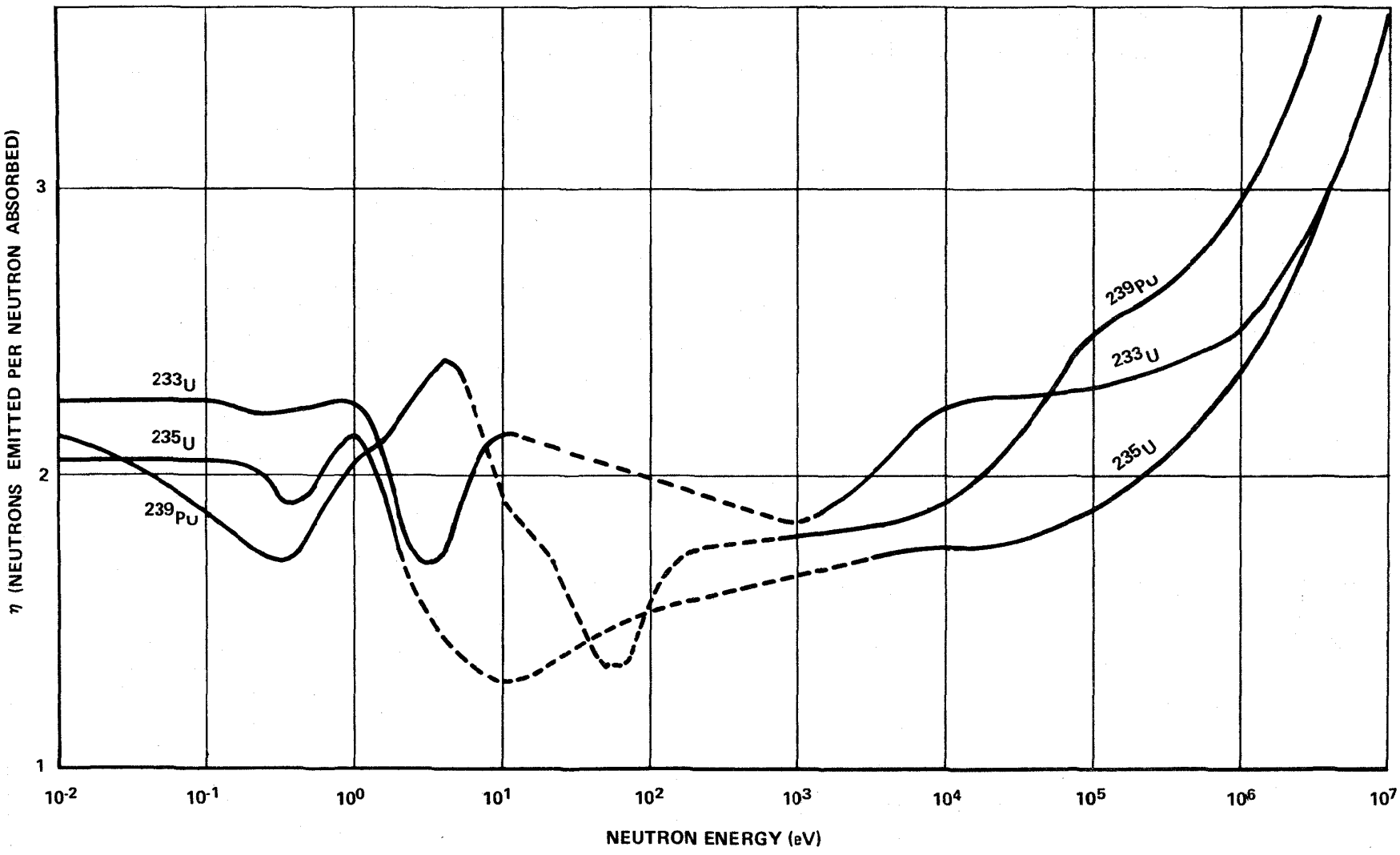

Figure 15 η Vs Neutron Energy

efficient use of fissile material. Figure 15 shows that η is high for a fission spectrum, decreases to a minimum for intermediate energies, and increases again at low energies (i.e., at thermal energies).

We can divide reactors into three general types characterized by the neutron spectrum; fast, intermediate, and thermal. In a fast reactor the neutron energies are kept as high as feasible to take advantage of the high η value. In a thermal reactor the neutron energies are reduced to the thermal region as completely and quickly as feasible to take advantage of the relatively good η value and the high fission cross section. In between, with the neutrons at intermediate energies, lies the intermediate reactor, and this design is to be avoided for power production because of its inefficient use of neutrons.

It turns out that, to design a fast reactor, great care must be taken in the choice of coolants. The currently favoured choice is liquid metal (Na or NaK) although gas is a possibility. Any significant amount of water added to the core moves the reactor into the unfavourable intermediate range (unless enough is used to make a thermal reactor). This restriction in the choice of coolant, and the fact that the technology of liquid metal coolants is difficult, explains partially why development of commercial fast power reactors has been so slow.

On the other hand if one opts for a thermal reactor it is advantageous to move the neutrons through intermediate energies as quickly as possible and one purposely adds efficient slowing down materials as moderators. It is important that the moderator have a low neutron absorption cross section. Favoured moderators are heavy water, graphite, ordinary water, and beryllium.

Thus we see that there is no practical continuous gradation from fast reactors to thermal reactors. One must decide which design to go for and one does exactly the opposite for each with respect to moderating materials.

6.4 General Features of Fast Reactors

There are several test and experimental fast reactors in operation, and a few prototype power reactors being designed and built. Commercial fast power reactors are probably at least a decade away. Present fast power reactor designs are all very similar and use PuO_2-UO_2 ceramic fuel and liquid sodium cooling. The fuel cladding is usually a steel alloy.

In the core of these reactors the fuel is in the form of elements or pencils with an outer diameter of about 1/4". The ratio of coolant to fuel plus cladding is in the neighbourhood of 1:1 and average power densities in the range 1/3 to 1/2 MWt per litre or about 1 MWt per kg of fissile material are anticipated. The ratio of fissile to fertile material is usually about 15% and the fuel must be taken to burnups in the range 50,000 to 100,000 MWd per tonne of fissile plus fertile material for the concept to be economically competitive.

The core of the reactor is surrounded by a blanket containing mostly fertile material so that the neutrons escaping from the core can be used efficiently in producing fissile material. Powers of 1000 MWe are thought to be required for competitive units.

The characteristic most often associated with fast reactors is their ability to breed, i.e., to produce more fissile material (from fertile material) than they consume. Thus the world's fertile resources (which are several hundred times the fissile resources) can be used to produce power.

Some of the physics problems associated with fast reactor design have to do with safety and fuel management. The sodium void coefficient (i.e., the effect of losing sodium coolant) can be positive, leading to safety problems. Fuel management is very complex. Until recently the calculations of critical size and breeding ratio (extent of breeding) were inaccurate, but experiments on full-scale core mock-ups have improved the situation. The ability to accurately estimate the neutron spectrum is important.

Fast reactors are characterized by very high fuel inventories. A 1000 MWe fast power reactor will contain about 3 tonnes of fissile Pu, worth at least $10/g. This high inventory contributes to the power cost and also limits the rate at which fast reactors can be introduced, even though they breed Pu.

6.5 General Features of Thermal Reactors

In sharp contrast to fast reactors, where there is close to unanimity in choice of type, there are many types of thermal reactors already developed or being developed. These differ mainly in choice of moderator - heavy water, ordinary water and graphite. However, even within the class of heavy water moderation AECL is developing the use of three coolants - pressurized D_2O, boiling H_2O, and organic.

Rather than treat all these types I will give only a brief comparison between light water and heavy water thermal reactors and then concentrate on the heavy water type.

Light water reactor cores resemble in some ways those of fast reactors, in that the UO_2 fuel is in the form of a more or less uniform array of elements or pencils. These have a diameter of 0.4" to 0.6" and the ratio of coolant to fuel is about 1.7:1 for the pressurized water version (PWR) and about 2.4:1 for the boiling light water version (BWR). The average core power densities are about 90 kW per litre for the PWR and 50 kW per litre for the BWR. Light water reactors cannot operate with natural uranium fuel but use fuel enriched to about 2.3% in ^{235}U and achieve burnups of about 20,000 MWd/te fissile plus fertile. The power obtained from the fissile material is about 1.2 MWt per kg fissile material for the PWR and about 0.8 MWt per kg fissile for the BWR.

Since the water is heated to 300°C or so the whole core is contained in a pressure vessel. Mainly as a result of this the reactor must be shut down for refuelling.

The combination of enrichment and relatively high neutron absorption in ordinary water leads to relatively high fuel costs. From an engineering point of view, the core is relatively compact with a high power density. The coolant is one for which the technology is well developed.

Heavy water does not slow neutrons down as quickly as light water. On the other hand it has an extremely low neutron absorption cross section and on balance, from the physics point of view, is a much better moderator than light water. Much more moderator volume is provided in a heavy water reactor than in a light water one; the ratio of moderator to fuel volumes being typically 20:1.

In a typical Canadian design the fuel is in the form of Zircaloy clad, UO_2 elements or pencils with an overall diameter of 0.6". Nineteen to thirty-seven of these elements are made up into a bundle about 50 cm long. Nine to twelve bundles are then put end-to-end in a pressure tube ($3\frac{1}{4}$" or 4" inner diameter). Each pressure tube has a calandria tube around it to insulate the moderator from the hot coolant. The channels are arranged in a square lattice at a pitch of $9\frac{1}{2}$" to $11\frac{1}{4}$" and the interstitial space is filled with heavy water. The coolant, of course, flows inside the pressure tubes. A feature made possible by this type of design is that refuelling may be done at power.

Heavy water reactors can use natural uranium as fuel and conversion ratios (i.e., ratio of fissile atoms produced per fissile atom destroyed) of 0.83 are currently achieved. This permits burnups of about 10,000 MWd/te fissile plus fertile to be achieved, about half of the energy coming from Pu.

Although the overall power density in a heavy water core is relatively low - typically 9 kWt/litre - the power per unit of fissile material is higher than in either fast or light water reactors - typically approaching 3 MWt/kg fissile material.

The result is a very low fuel cost, and although heavy water is expensive, the contribution to energy costs is low as long as losses due to leakage are kept small. The reactor core is considerably larger than a light water reactor core due to the higher moderator volume.

In the following sections some of the physics aspects of heavy water reactor design will be discussed. While the discussion will be restricted to heavy water reactors, most of the points have their counterpart in other reactor designs.

7. PHYSICS ESTIMATES FOR PRACTICAL POWER REACTORS

Obviously many more difficulties are encountered in estimating the characteristics of practical power reactors than those discussed for our simple reactor. Some of these are summarized below.

(i) Variety of Materials Present:

There are several materials present in a practical power reactor rather than the single isotope in the example we treated. This, in itself, would not be too serious since we can easily take suitable averages of properties such as cross sections. However, it does create difficulties in that properties of these materials are often significantly different than those for ^{235}U and introduce effects which could be neglected in that particular case (e.g. slowing down, resonance capture). Some of these will be discussed below.

(ii) Fast Fission:

In the simple reactor which we considered, fast fission in ^{235}U was the main process of interest. In a natural uranium, heavy water moderated reactor this effect can be almost neglected.

However, fast fission in ^{238}U becomes significant. Although ^{238}U cannot be fissioned by thermal neutrons, above a threshold of about 1.4 MeV, fast neutrons can cause it to fission. This effect must be taken into account.

(iii) Slowing Down:

Obviously the slowing down process must be taken specifically into account in calculations for any thermal reactor.

(iv) Resonance Absorption:

^{238}U has very substantial capture resonances in the intermediate energy region. Even though there is a lot of moderator, which moves the neutrons quickly through this region, a substantial number are captured. This complex effect must be treated quite accurately.

(v) Heterogeneity:

The heterogeneous nature of the core presents calculational difficulties. Since there is a fine structure in neutron flux from material to material in the same neighbourhood, straight homogenization cannot be used. Instead a detailed flux calculation for a small region is usually done first and the results used to suitably weigh properties for use in a homogeneous reactor model.

(vi) Reactivity Coefficients:

In a power reactor there is feedback between power and reactivity. Macroscopic cross sections depend on material densities which in turn depend on temperatures and hence powers. Resonance absorption can depend on material temperatures (Doppler effect). The thermal neutron spectrum depends on the material temperatures. The effects of power, temperature and density on reactivity are important from the point of view of control and safety.

(vii) Reactivity Worths:

In a reactor which uses on-power fuelling it is important to be able to estimate the effect of removing fuel from a channel. Practical reactors usually have some structural materials in the core (such as spray nozzles, control rod guides, flux

detector strings, etc.) which must be accounted for. The effects of control and shut-off devices must be estimated.

I will mention one useful result which is derived from perturbation theory. This is that the reactivity effects of a given absorber placed at different positions in a homogeneous reactor are proportional to the flux squared at the different positions. This means that an absorber placed at the centre of a typical reactor decreases the reactivity by far more than the same absorber at the outside. One of the flux factors accounts for the different absorption rates at the two positions and the other for the importance of the neutrons in sustaining the chain reaction.

(viii) Non-uniformities:

In addition to being heterogeneous, most power reactors have large regions which differ physically. For example most reactors have a reflector, i.e., a region around the outside containing no fuel whose purpose is to prevent as many neutrons as possible from escaping by reflecting them back into the core.

(ix) Reactor Kinetics:

Even in practical power reactors a point kinetic model is very useful, i.e., a model which treats changes in power, etc., as an overall effect and neglects time changes in the distribution of power. As reactors get larger it becomes more important to have a proper spatial dynamics treatment.

8. SOME PHYSICS CHARACTERISTICS OF HEAVY WATER REACTORS

8.1 Critical Sizes and Reactivities

Using the type of lattice which has been described for a heavy water reactor with fresh fuel, room temperature conditions, and D_2O coolant, a cylinder 3 m in diameter and 3 m high could be made critical. The core in a Bruce station reactor unit is cylindrical with a diameter of about 8.5 m and a length of 5.95 m. The excess reactivity associated with this size is about 110 mk. Of this 110 mk, about 35 mk is used in overcoming short term fission product poisons such as xenon and samarium (see below). Another 15 mk is lost in going to operating temperature conditions and a few more in structural features and control margins. This leaves some 50 - 60 mk in hand to look after reactivity changes due to burnup (see below). This is enough to achieve a burnup of close to 10,000 MWd/TeU from the fuel.

For the initial charge this excess reactivity presents some problem. If it were compensated by operating at reduced moderator level the reactor power output would have to be reduced. The methods used in Canadian reactors have included depleted uranium (i.e., uranium with less fissile content than natural uranium) in some of the bundles of the initial charge, additions of boron to the moderator (which can be removed as required), and addition of removable cobalt absorbers.

8.2 Burnup

One of the most important features of the CANDU reactor is the burn-up attainable from natural uranium fuel. Reactivity change as a function of burnup is a very difficult effect to estimate for natural fuel since it is a very fine balance between the effects of burning ^{235}U, producing ^{239}Pu, and absorption in fission products. The success of the Canadian system has hinged on very careful attention to neutron economy. Before the successful operation of Canadian demonstration and prototype power reactors there was widespread doubt that the predicted burnups could be attained. Since the fuelling costs are roughly inversely proportional to the burnup, the whole concept depended on reaching close to these burnup levels. Many experiments were done on which to base the estimates until a sufficient level of confidence was reached.

To put the burnup of 10,000 MWd/TeU in perspective, the amount of energy which can be derived from burning all the ^{235}U in a tonne of natural U is only 6000 MWd. In fact, more than half the energy comes from ^{238}U fast fissions and ^{239}Pu fission.

One of the secrets of obtaining such a high burnup from natural uranium is the use of short fuel bundles which can be shifted in the axial direction. If full length fuel were used, the fuel at the ends would receive a lower than average burnup whereas that at the centre would be higher than average. Since the centre portion is by far the most important in determining the overall reactivity effect this would lead to a lower average burnup than in the bundle fuelling case where it can be arranged to obtain the same average burnup from all bundles by appropriate shifting.

The fuel bundles removed from the reactor after producing 10,000 MWd/TeU in energy, contain about 3 g of Pu per kg of U. This Pu is valuable either for recycling in a thermal reactor or to provide Pu inventory for a fast reactor. While at present no credit is taken for the value of this Pu, it is estimated that in the future there will be a market for it and that this will further reduce fuelling costs for heavy water reactors.

8.3 Power Shape and Flux Flattening

In thermal reactors, such as existing CANDU reactors, the reaction rates are determined predominantly by the thermal neutron flux. Therefore, in what follows, flux should be taken to mean thermal neutron flux unless otherwise specified.

The spatial flux distribution in a CANDU reactor has a good deal of structure. It is convenient and useful to divide this structure into three levels - macroscopic distribution, fine structure, and hyperfine structure.

The macroscopic flux distribution is the overall flux distribution which would be obtained in a homogeneous reactor. Or in a heterogeneous reactor with a uniform lattice the distribution obtained from equivalent points (e.g. centre of fuel bundles) in each lattice cell. The macroscopic flux distribution in a cylindrical, uniform, unreflected reactor has the shape of a cosine in the axial direction (with the zeroes of the cosine lying just beyond the ends of the reactor) and a Bessel function (J_0) in the radial direction (with the zero lying just beyond the edge). This radial distribution is shown in Figure 16. For this macroscopic distribution the ratio of average flux in the reactor to flux at the centre is about 0.275; the axial factor being 0.637 and the radial 0.432. Since the power of the reactor is basically limited by the central bundle rating, there is considerable incentive to increase these factors in order to increase the power density per unit of fuel.

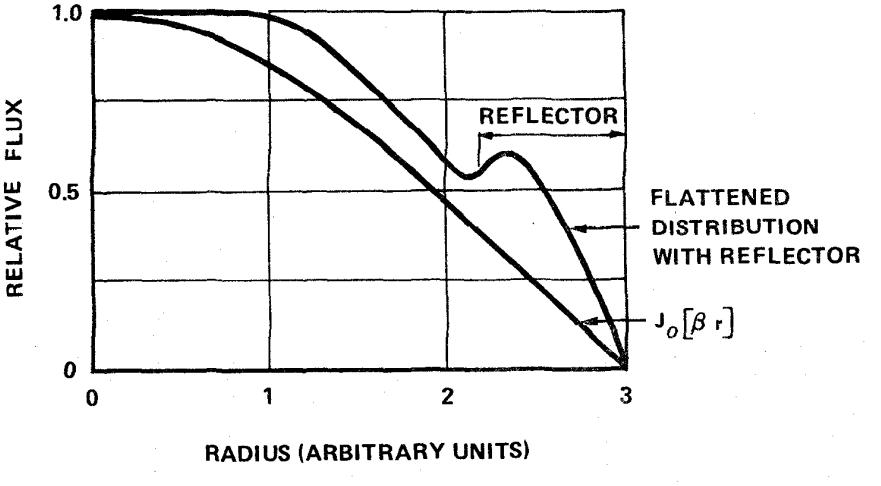

Figure 16 Typical Radial Macroscopic Flux Distributions

In CANDU two methods are used to increase the radial factor. The first is by providing a reflector, which is done essentially by not putting in some of the outer rods, and the second is by flattening the radial flux distribution in a central region. This flattening is achieved by having a lower reactivity in the central core region than the outer region either by adjusting the burnups so that the average burnup is higher in the central region than the outer region, or by introducing absorbers in the central region. Figure 16 shows schematically a typical radial flux distribution in a CANDU reactor. By these means the radial factor is increased to 0.8 or higher.

The fine structure of the flux distribution refers to the distribution within a lattice cell (assuming a homogeneous rod or taking equivalent points within the fuel bundle such as average flux in an element). The hyperfine structure refers to the detailed distribution within the fuel channel itself. Figure 17 shows the hyperfine structure in more detail. Note that the flux in the moderator is a factor of about 2 higher than that in the fuel. There is also fine structure in the axial direction, caused by the fact that short regions at the bundle ends do not contain fuel. This type of fine structure is shown schematically in Figure 19.

Obviously this flux distribution information is very important in fuel design since limits are often related to the power production at some specific position in a single element. This must be related to bundle power through a knowledge of the detailed flux distributions.

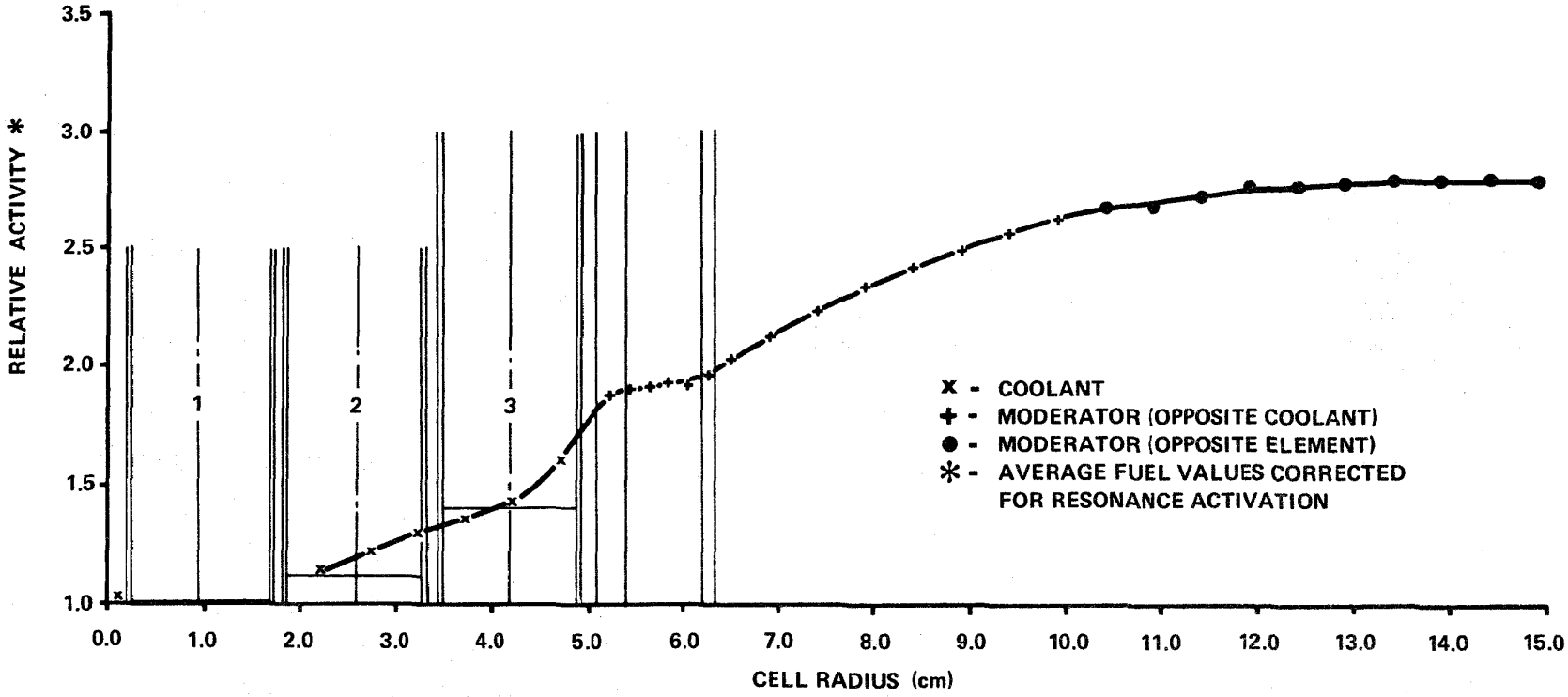

Figure 17 Fine Structure in Lattice Cell
(28 cm Pitch, D_2O Cooled)

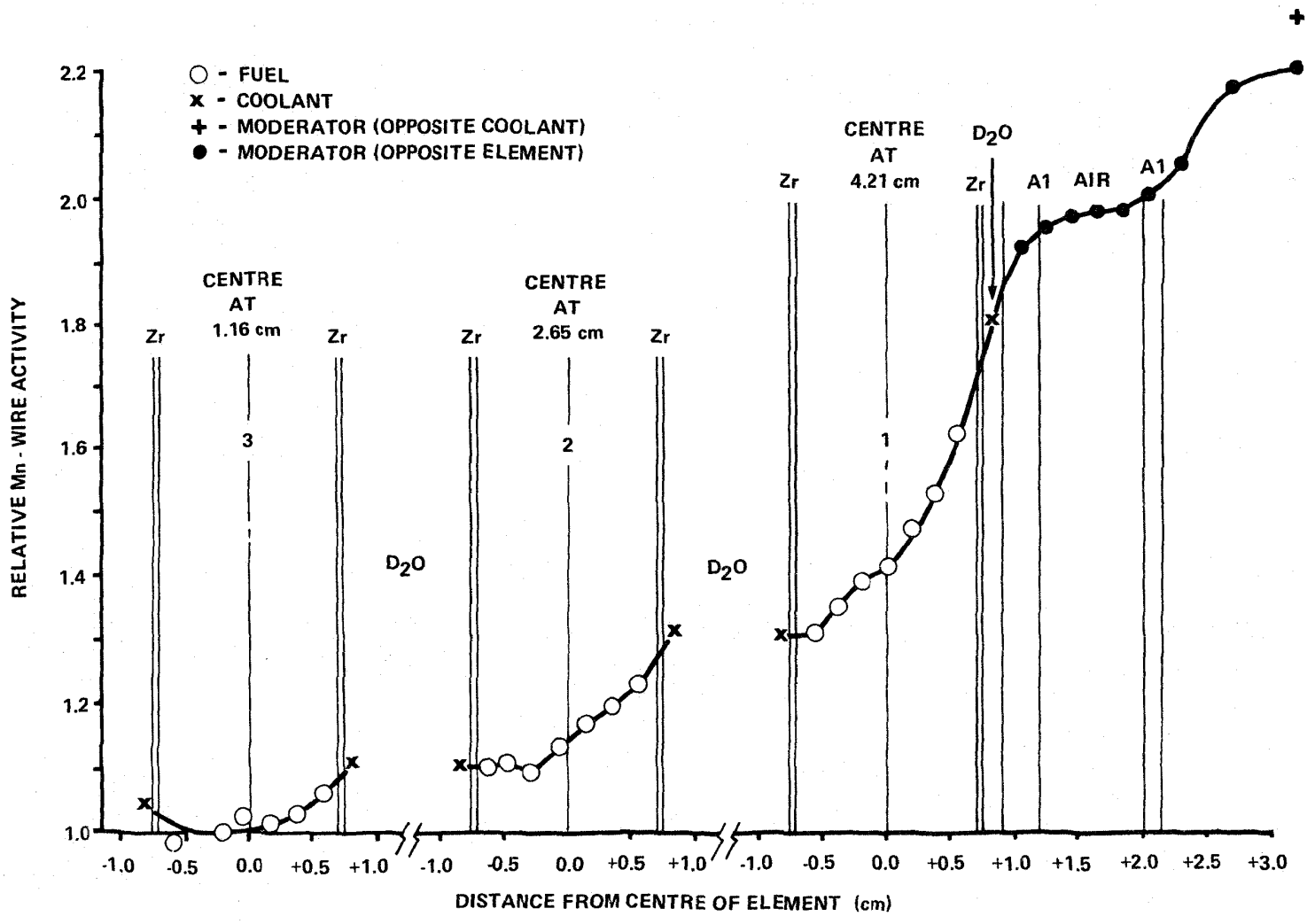

Figure 18 Hyperfine Structure in Fuel Assembly
(28 cm Pitch, D_2O Cooled)

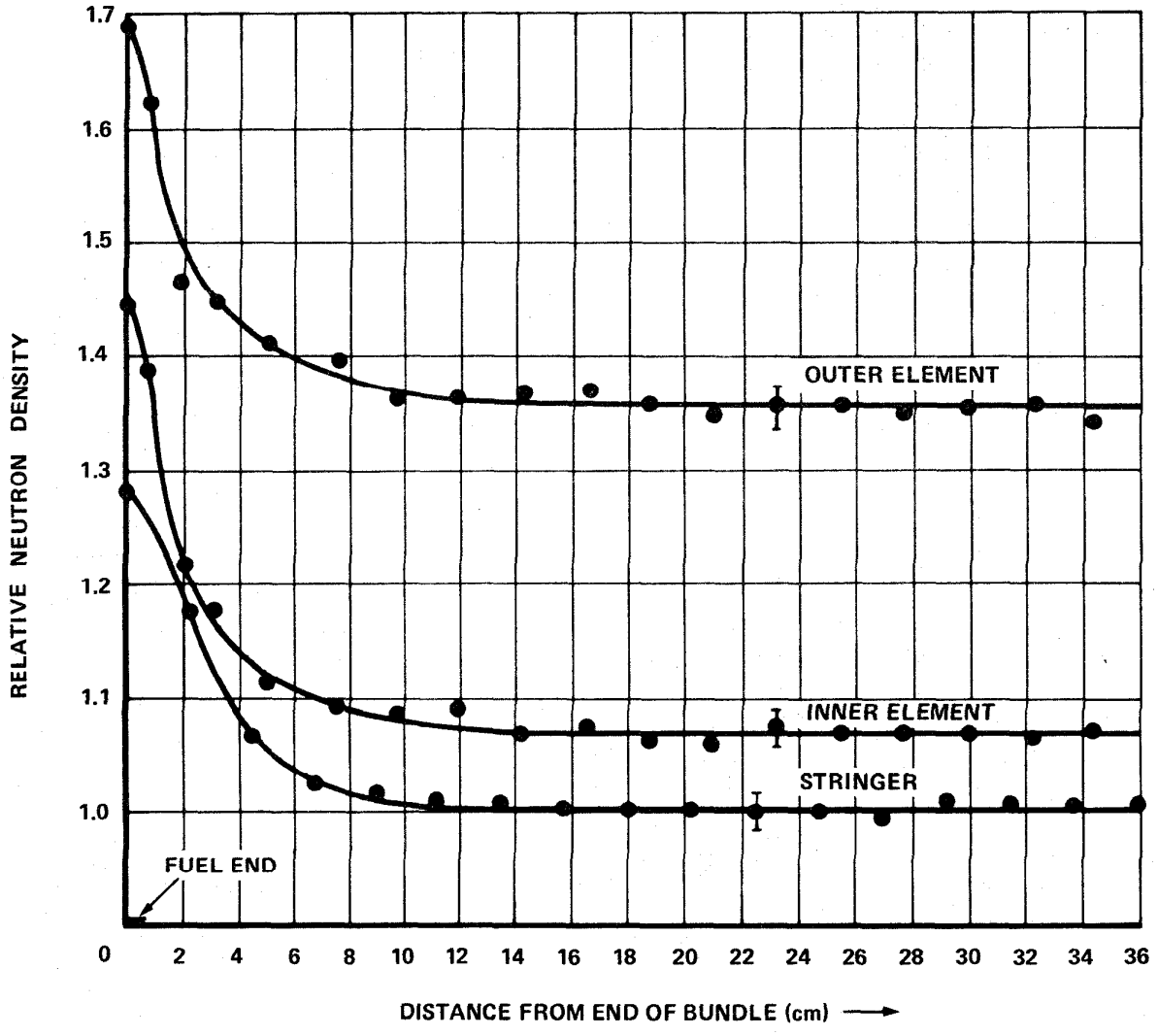

Figure 19 Flux Peaking When Two BLW Bundles Are In Contact - Air Coolant

8.4 Control and Safety

There are many reasons for requiring a fairly wide range of control of reactivity in a CANDU reactor:

(i) Power Regulation:

It must be possible to manoeuvre the power, up or down, at any time as well as to start up and shut down.

(ii) Power and Temperature Effects:

It must be possible to compensate for changes in reactivity due to changes in operating conditions.

(iii) Fuel Management:

Reactivity changes due to fuel burnup must be compensated as well as changes occurring during fuelling operations.

(iv) Xenon Transients:

These are discussed below and put additional requirements on the control system.

(v) Spatial Flux Distribution:

In large reactors there must be some scope provided for changing the spatial flux distribution to combat xenon oscillations (see below) or other potential spatial instabilities.

To provide the necessary control many different methods are used in CANDU reactors; often several in a single reactor. Some of these are briefly discussed below.

(i) On-Power Refuelling:

Fuel management provides a very slow control on reactivity. It also can be used to provide slow adjustments on the spatial power distribution.

(ii) Boron in Moderator:

As has already been mentioned, this means is used for compensating the large reactivity excess associated with the initial charge of fresh fuel.

(iii) Absorbers:

Solid or liquid movable absorbers are used for fast reactivity control and spatial flux control.

(iv) Booster Rods:

This is the name given to rods containing fissile material which can be inserted or removed from the core. They are usually used for overcoming xenon transients (see below).

(v) Miscellaneous:

Other methods of control are also used occasionally. For example, in Gentilly changes in coolant flow cause changes in reactivity and flow control is used to a certain extent.

(vi) Moderator Level:

Adjustment of moderator level has been used in some of the CANDU reactors as a fast reactivity control.

In addition to reactivity control means must be provided to insert negative reactivity quickly to shut down the reactor safely in the event of some abnormal occurrence (e.g., loss of coolant). The methods most commonly used are fast moderator dump, shut-off (absorber) rod insertion, and fast poison injection into the moderator.

8.5 Xenon Effects

^{135}Xe is a fission product with a relatively high yield ($\approx 6\%$ from ^{235}U fission) and a very high neutron absorption cross section ($\approx 3 \times 10^6$ barns). ^{135}Xe decays with a half life of 9.2 hours into a low cross section. ^{135}Xe is formed (mainly) indirectly from the decay of ^{135}I (≈ 6.7 hour half life) which has a low neutron absorption cross section.

These properties of ^{135}Xe lead to some rather strange effects which will be summarized below.

Equilibrium Xenon:

The cross section of ^{135}Xe is so high that in a high flux power reactor such as CANDU most of the ^{135}Xe is destroyed by neutron

capture rather than by decay. When equilibrium has been established the reactivity effect of this xenon is a loss of about 25-30 mk.

Xenon Transients:

When a reactor with fresh fuel is first started up to full power there is no ^{135}Xe present. ^{135}I builds up first as operation continues and then ^{135}Xe as the ^{135}I decays. Thus the reactivity loss starts very slowly, builds up more quickly, and then levels off at its equilibrium value. This takes somewhat over a day in a high flux reactor. At equilibrium there is much more ^{135}I present than ^{135}Xe.

Now consider what happens if the flux is suddenly lowered. The ^{135}Xe present increases at first since less is being destroyed by capture and the production from ^{135}I initially is constant. Later as the amount of ^{135}I decreases (since the formation rate of ^{135}I is lower) the ^{135}Xe decreases. Therefore, on a shutdown after long operation, the reactivity decreases to a minimum value (after about 10 hours) and then slowly increases again back to the equilibrium value (after some 30 hours), and finally after considerably longer back to the ^{135}Xe-free value. This is called a xenon transient.

If enough reactivity is available to start up at the peak of the xenon transient, once full power has been reached, the reactivity will increase very rapidly, since the Xe is burning out and there is very little ^{135}I to supply more, level out, and then decrease to the value set by equilibrium xenon.

In any slow power manoeuvering ^{135}Xe plays a strong role in determining reactivity requirements.

Xenon Oscillations:

One can see how these properties can lead to spatial flux instability by the following argument. Suppose we start with a fixed flux distribution and equilibrium distributions of ^{135}I and ^{135}Xe. Now assume a flux perturbation whereby the flux at one side of the reactor increases a bit and at the other side decreases a bit. On the side with the slight increase in flux the ^{135}I production rate will go up, but this will not be immediately reflected in the Xe production rate due to the delay in the decay of the I. The destruction rate of Xe by neutron capture will go up due to the flux increase. Hence initially the amount of Xe present will drop

and since the Xe is a heavy parasitic absorber this will cause the flux to rise further. On the other side of the reactor by similar arguments the amount of xenon will be going up thereby decreasing the flux still further. Therefore the flux perturbation will initially be amplified. Eventually the Xe production rate is affected by the I production rate and this starts to turn the process around and drive the fluxes in the other direction. If the effects of changes in Xe on changes in flux are large enough, the result is an oscillation in flux of growing magnitude, with cycle lengths of the order of 24 hours. All types of thermal reactors are susceptible to this form of instability if they are made large enough.