Nuclear Theory - Course 227

# PROMPT, DELAYED & PHOTO-NEUTRON PRODUCTION

The most important aspect of the design of a nuclear reactor is CONSERVATION OF NEUTRONS. Energy is released during fission and neutrons are used to cause Uranium-235 to fission. As many neutrons as possible must be conserved to cause fission, and the choice of moderator, reactor materials and the arrangement of fuel in a reactor are decided primarily with neutron conservation in mind.

As will be seen later, these neutrons have an important bearing on reactor control. This lesson will discuss the sources and types of neutrons in a reactor.

## Production of Prompt Neutrons

In a nuclear reactor, energy is released during the fissioning of Uranium-235. The U-235 captures a neutron and becomes Uranium-236. The U-236 is so unstable that it splits or fissions into two parts known as fission products or fission fragments. At the instant the fission occurs, or very shortly thereafter, from one to three neutrons are released. On the average, two and one-half ($2\frac{1}{2}$) neutrons are released in this manner. These $2\frac{1}{2}$ neutrons are released within one ten-millionth of a second of the U-235 fissioning and, therefore, they can be regarded as being produced <u>during</u> the fission process. This is why they are called PROMPT neutrons. U-235 fission can, therefore, be represented diagramatically as follows: -

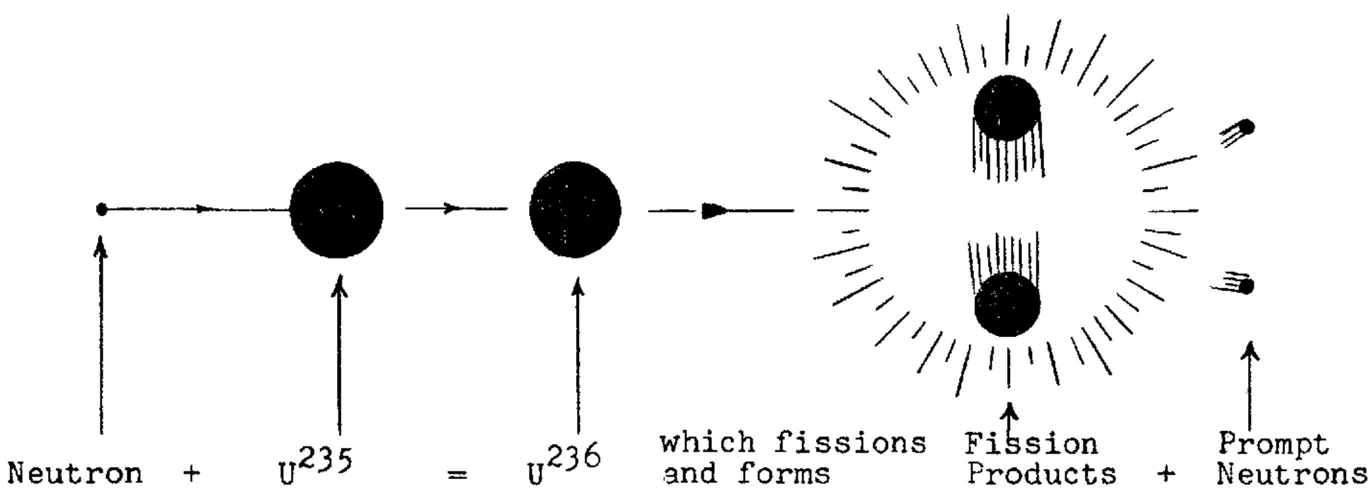


Fig. 1

The same process that is shown diagramatically in Figure 1 can also be represented by the following equations:

$$ {}_{92}U^{235} + {}_{o}n^{1} = {}_{92}U^{236} \longrightarrow {}_{38}Sr^{90} + {}_{54}Xe^{144} + 2\ {}_{o}n^{1} $$

$$ {}_{92}U^{235} + {}_{o}n^{1} = {}_{92}U^{236} \longrightarrow {}_{35}Br^{87} + {}_{57}La^{147} + 2\ {}_{o}n^{1} $$

The first equation shows Strontium and Xenon as the fission products, whereas the second equation shows Bromine and Lanthanum as the fission products. These are merely two of many such examples that can be given. In fact, the fission products can be any two of most of the elements as long as the equation balances.

## The Fission Products

During the fission or splitting of U-236, the fission products can be any two of most of the isotopes of the elements. Some isotopes are produced, as fission products, during only a small fraction of the fissions. Other isotopes occur, as fission products, during a much greater number of fissions, e.g. Xenon-140 and Strontium-94 are fission products during 6% of all fissions that occur.

Most of these fission products, whether they occur frequently or otherwise, are unstable isotopes and, therefore, radioactive. They usually decay by emitting beta particles which are often accompanied by gamma rays. The daughter of the radioactive fission product is frequently radioactive and a whole decay series results.

The following illustrates the Sr-94 decay chain:

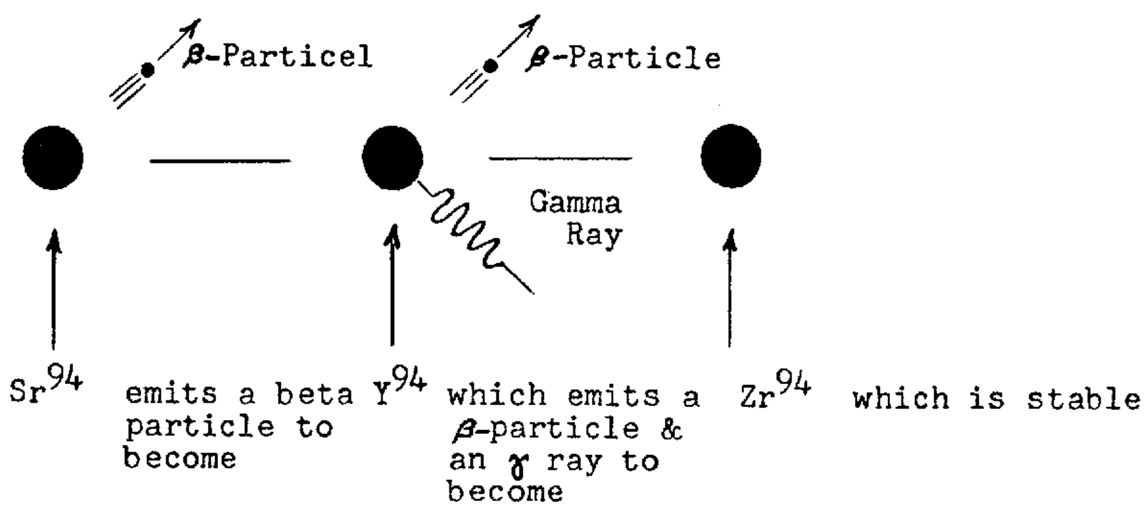


Fig. 2

or in equation form:

$$_{38}Sr^{94} \xrightarrow[\beta]{} {}_{39}Y^{94} \xrightarrow[\beta]{} {}_{40}Zr^{94}$$

Many such decay chains exist and many are larger than the one shown.

These fission products and their daughters are important in reactor operation. Their significance will be considered as various aspects of reactor operation are considered. One aspect will be considered in this lesson.

Production of Delayed Neutrons

Most of the fission products are radioactive. The example given above shows a fission product and its daughter decaying by beta particle and gamma ray emission; this is by far the most common method of decay.

However, there are some fission products or their daughters which are sufficiently excited or unstable to emit neutrons. For instance, Bromine-87, decays by beta emission to Krypton-87. The Kr-87, so formed, may be unstable enough to emit a neutron and become Kr-86. The following illustrates this decay chain diagramatically:

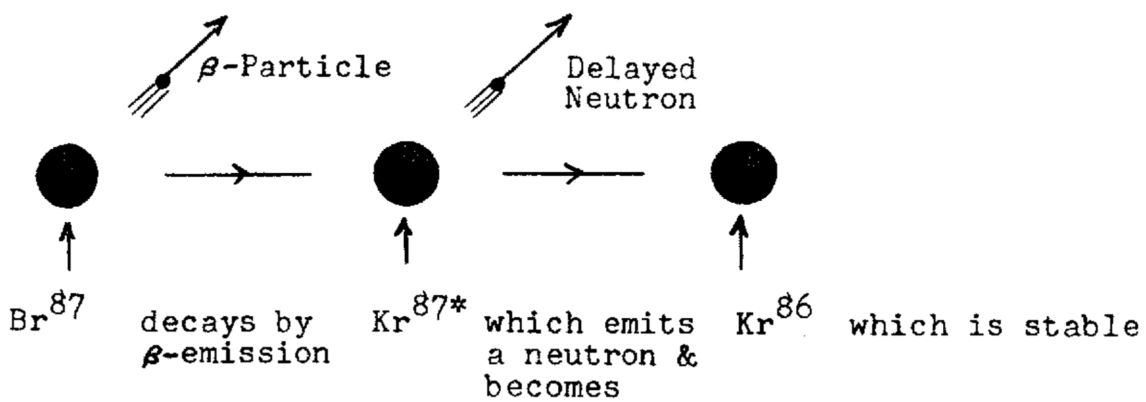


Fig. 3

or as an equation, it would be:

$$_{35}Br^{87} \xrightarrow[\beta]{} {}_{36}Kr^{87*} \xrightarrow[n]{} {}_{36}Kr^{86}$$

The asterisk, *, denotes that the Kr-87 is in a highly excited state.

The Kr-87 emits the neutron as soon as it has been formed but the rate at which the neutrons are emitted will depend on the rate at which the Kr-87 is formed. Now the Br-87 has a half-life of 55.6 secs and, therefore, these neutrons will appear to have an effective half-life of 55.6 secs as measured from the instant at which the fission occurs. In other words these neutrons are not emitted at the instant of fission, but their emission is delayed because of the delay in the decay of Br-87. They are therefore called DELAYED neutrons and the neutron is labelled as such in Fig. 3.

There are six groups of delayed neutrons with effective half-lives, varying from 0.05 secs to 55.6 sec. The delayed neutrons form 0.7% of the total neutrons produced as a result of fission, whereas 99.3% of these neutrons are prompt neutrons. Even though they form such a small percentage of the total, the delayed neutrons are very important in reactor regulation.

Production of Photo-neutrons

Prompt and delayed neutrons are produced as a result of fission. If no further fissions occur, no more prompt or delayed neutrons are produced. This is not the case with photo-neutrons.

Photo-neutrons are peculiar to reactors which have heavy water moderators or heat transport fluids. They are produced when gamma rays with energies greater than 2.2 Mev are captured by deuterium nuclei. Diagramatically the reaction would be represented as follows: -

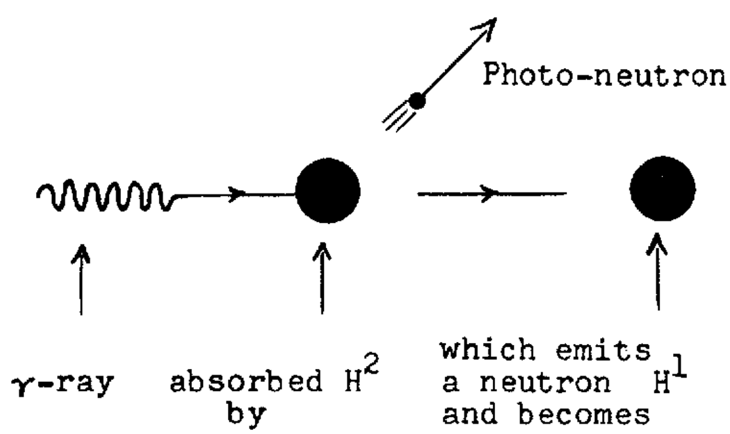


Fig. 4

The equation for this is: -

$$_1H^2 + \gamma = {}_1H^1 + {}_0n^1$$

This photo-neutron is not produced in a graphite moderated reactor because no deuterium is present. The reaction is insignificant in light water moderated reactors due to the extremely low deuterium content.

Since gamma rays are emitted in the core, the heat transport system and in the moderator system, external to the core, photo-neutrons will be produced in all these systems. The important ones, though, are those produced in any heavy water moderator or heat transport fluid in the core. Even when the reactor is not operating, and prompt and delayed neutrons are not being produced, gamma rays from fission product decay will still produce photo-neutrons in any $D_2O$ that might be present, providing a neutron source in the reactor to start the fission chain reaction.

## ASSIGNMENT

1. (a) Name the three types of neutrons produced in a heavy water moderated reactor.

   (b) Which type is <u>not</u> produced in a graphite moderated reactor?

2. (a) Which two of the three types are produced as a result of fission?

   (b) What fraction of the total does each type represent?

   (c) Of the types produced as a result of fission, when and how are the ones produced in greater number, produced?

   (d) Explain how the ones, produced in smaller number, are produced and why they have the particular name that they have.

3. (a) If the third type of neutrons are <u>not</u> produced as a result of fission, how are they produced?

   (b) Why are these neutrons produced in the reactor core even when the reactor is <u>not</u> operating and the other two types are <u>not</u> being produced?

A. Williams

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## NEUTRON BALANCE DURING STEADY REACTOR OPERATION

We have seen, in the previous lesson, what type of neutrons are produced and how they are produced in a reactor. We now take a look at this neutron population in a reactor, how this population varies, if at all, during steady power operation and the factors that can affect this neutron population. For the moment we will disregard the photo-neutrons since they form a very small fraction of the total. We will also group the prompt and delayed neutrons together and consider them both to be produced as a result of fission.

### Maintaining a Chain Reaction

We have seen that the energy released during one U-235 fission is not of any practical significance and that, to produce useful power, this one fission has to be duplicated millions of times over. Fig. 1, illustrates how each individual fission releases a minute quantity of energy locally, but when millions of these small amounts of energy are all released at the same time in a small volume of uranium the total amount of energy released becomes significant.

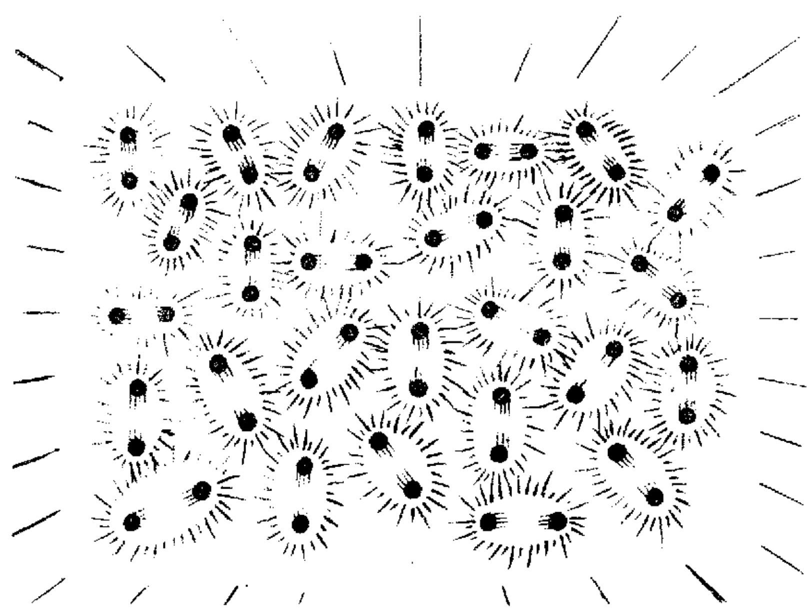

Fig. 1

The bulk of the fuel then heats up and the heat can then be used to produce useful power. It would require $3.1 \times 10^{10}$ (thirty one thousand million) of these fissions to occur every second to produce 1 watt of power.

We have also seen that, to produce steady power, the fission process must be continuous. It would be of no value to have thirty thousand million fissions occurring in one second and no fissions occurring after that. The number of fissions occurring per second must remain constant second after second, day after day while the reactor is producing steady power. Such a continuous repetition of the fission process is known as a chain reaction. Such a chain reaction can be maintained if one of the neutrons, produced at fission, is used to cause a further fission.

On the average, 2½ new neutrons are produced during each fission and one of these must be used to cause a further fission. A typical chain reaction is therefore a continuous series of fissions each one caused by a neutron from the previous one. Diagramatically a chain reaction would be as in Fig. 2. Here, one neutron from each fission is being used to cause a further fission. The other neutrons are lost by one of the following processes: -

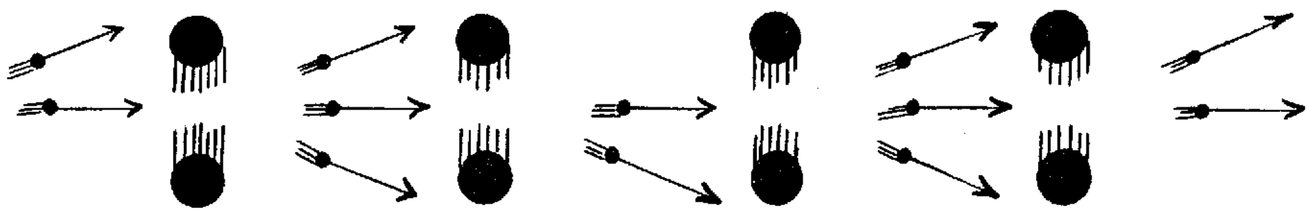

Fig. 2

(1) Capture in U-235, U-238, or the nuclei of other materials in the reactor without causing fission.

(2) Leakage or escape out of the reactor.

Non-fission captures can be reduced and fission made more likely by: -

(1) using a moderator to slow down the fission neutrons to thermal energies (0.025 ev)

(2) suitable arrangement of fuel in the reactor

(3) suitable choice of reactor material to minimize capture of neutrons.

Neutron leakage can be decreased by using a reflector around the reactor core. Let us now consider the neutron balance in more detail.

A neutron, shown in Fig. 2, may go through a typical cycle which can be described briefly as: -

## Neutron Balance and a Typical Neutron Cycle

A neutron, shown in Fig. 2, may go through a typical cycle which can be described briefly as: -

Neutron born or released, as a fast neutron, at fission → Neutron slowed down in the moderator until it becomes a thermal neutron → Neutron captured by U-235 to cause fission

From each fission the only neutron that goes through this cycle is the one that maintains the chain reaction. The others are lost by some means of other during the cycle. However, it is useful to know how they are lost since the knowledge will enable losses to be reduced and will also indicate how reactor power can be regulated. We therefore consider the histories of a number of neutrons. Fig. 3 shows a typical neutron balance cycle when losses are considered. The continuous line (——>——) show what happens to the neutrons that contribute to the chain reaction whereas the dotted line (---->) show what happens to the neutrons that are lost.

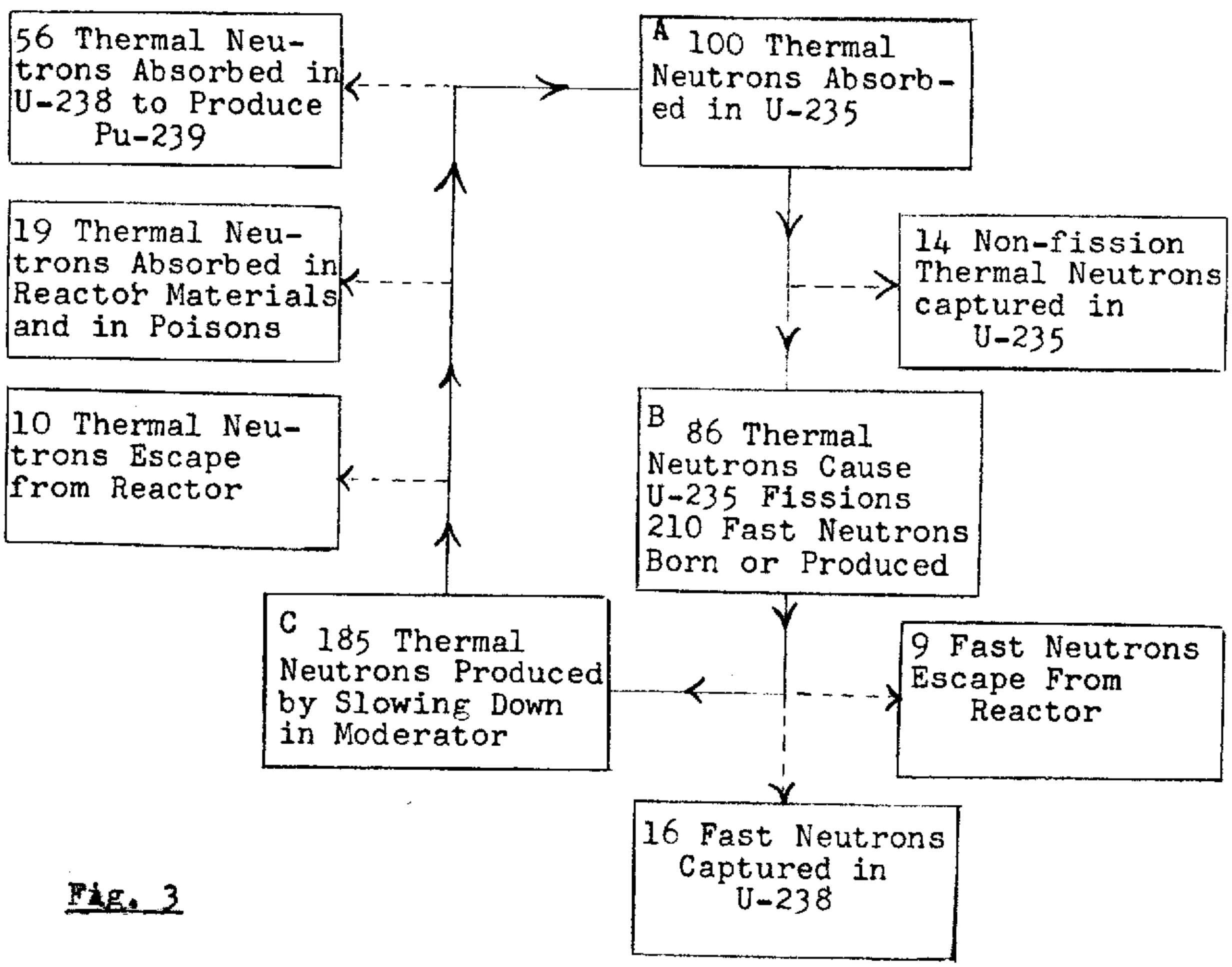


Fig. 3

The cycle can be started from any of the boxes marked A, B or C. If the cycle is to be continuous and the chain reaction just maintained (i.e. steady power operation), then, on completing the cycle back to the starting box, the same number of neutrons must appear in the box as were there at the beginning of the cycle.

For instance suppose we start at B and go around one cycle. Fig. 4 shows the same cycle in a different way and may help to follow the cycle.

86 Thermal Neutrons Cause U-235 Fissions ← First Generation

↓

210 Fast Neutrons Produced

9 Fast Neutrons Escape ← ↓ → 16 Fast Neutrons Absorbed in U-238

185 Thermal Neutrons Produced

10 Thermal Neutrons Escape ← ↓ → 16 Absorbed in Reactor Material
3 Absorbed in Poisons
56 Absorbed in U-238

100 Thermal Neutrons Absorbed in U-235

↓ → 14 Non-Fission Thermal Neutron Captures in U-235

86 Thermal Neutrons Cause U-235 Fissions | 2nd Generation

Fig. 4

At B, 86 thermal neutrons cause fission. These 86 fissions will be referred to as the FIRST GENERATION of neutrons. Since 2.5 fast neutrons are produced at each fission a total of 210 fast neutrons are produced. 9 of these are lost by leakage out of the reactor and 16 are lost by capture in U-238. Thus, only 185 become thermal neutrons. Before these 185 thermal neutrons can be used to cause fission, 10 of them escape and 75 are lost by non-fission captures. The remaining 100 are captured in U-235 nuclei but 14 of those captured do not cause fission. Therefore 86 fissions occur and these are referred to as the SECOND GENERATION of neutrons. At the end of a further cycle there would be a THIRD GENERATION of neutrons and so on.

If the number of fissions, in each succeeding generation remains the same, the chain reaction is just being maintained and the reactor is operating at steady power. If the number of fissions in succeeding generations decreases, because the neutron losses are greater, then the reactor power decreases and the chain reaction will eventually stop. If the number of fissions in succeeding generations increases, reactor power increases. This last condition is known as neutron MULTIPLICATION.

Factors Affecting Neutron Balance

Let us now examine how the number of neutrons, in the various boxes in the cycle, could be changed:

(1) Fast and thermal neutron leakage or escape. Neutron leakage can be decreased by increasing the size of the reactor core or by placing a reflector around the core to reflect the escaping neutrons back into the core. The reactor size is generally determined only once but a reflector is something that can be varied. Here, then, is a possible method of changing neutron leakage and therefore reactor power ie, a possible method of reactor regulation.

(2) Fast neutron capture in U-238. Since the U-238 is in the fuel, the only method of decreasing fast neutron capture in U-238 is to keep the neutrons away from the fuel while they are being slowed down. The fast neutrons, produced by fission, must be allowed to escape quickly from the fuel into the moderator and then slowed down to thermal energies before being allowed back into the fuel. This, then, is a neutron loss which can be reduced by suitable choice of moderator and fuel arrangement. The fuel arrangement is fixed in a reactor but the amount of moderator may be variable and is, again, a possible method of reactor regulation.

(3) Neutron absorption in reactor material is, of course, a loss that is kept to a minimum by suitable choice of materials. Substances with low neutron capture should be used for structural, moderator and heat transport materials, but these are basically design considerations.

(4) Neutron absorption in poisons. Poisons in a reactor may be substances which accumulate as a result of reactor operation, such as fission products, or substances deliberately introduced into the reactor to absorb neutrons. In either case the poisons are good absorbers of neutrons. Little can be done about the fission product poisons except to allow for them in the design of the reactor. However the introduction or removal of absorbers in the form of boron or cadmium rods provides another method of reactor regulation.

(5) Thermal neutron absorption in U-238. The U-238 forms 99.3% of the uranium in the fuel if natural uranium is used. Thus, if a system is committed to using natural uranium fuel, nothing can be done about thermal neutron absorption in U-238. However, absorption in U-238 can be decreased and absorption in U-235 increased by artificially increasing the U-235 content of the fuel. This is known as ENRICHMENT.

(6) Non-fission captures of thermal neutrons in U-235. The percentage of neutrons, captured by U-235, which do not cause fission does not vary except for a slight variation with temperature.

## ASSIGNMENT

1. What is a chain reaction and what is the minimum condition required to maintain it?

2. (a) Starting with fast neutrons produced at fission state how they can be lost while being thermalized?

   (b) Explain briefly, the factors that would decrease these losses.

3. (a) By what methods, other than leakage, can thermalized neutrons be lost?

   (b) Which of these form the basis for reactor regulation? Explain.

   (c) Which of them can be decreased by enrichment?

A. Williams

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## NEUTRON DENSITY, NEUTRON POWER AND NEUTRON FLUX

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In this lesson, we introduce a new quantity, called NEUTRON FLUX, which is closely connected with neutron production.

### Neutron Density

The amount of power produced in a reactor depends on the number of fissions that occur per second, ie, on the rate of fissioning of U-235. Let us be quite clear on the meaning of the terms we are using here By "power" we mean the rate at which heat is released, ie, we mean THERMAL power in watts or joules/sec or Btu/minute or any other appropriate units. This thermal power, then, is directly proportional to the number of fissions occurring per second.

eg: To produce 1 watt of power, we must have

$$3.1 \times 10^{10} \text{ fissions/sec}$$

To produce 100 Megawatts, we must have

$$3.1 \times 10^{10} \times 100 \times 10^{6},$$

or $$3.1 \times 10^{18} \text{ fissions/sec}$$

We are referring here to the <u>whole</u> reactor producing this power and the corresponding rate of fissioning being the <u>total</u> throughout the whole core.

If the power production is to remain constant or steady, the same rate of fissioning must be maintained continuously. Therefore, the number of thermal neutrons causing fission in the first generation must be the same as in the second generation and in all following generations. This means that the total number of neutrons in the reactor must remain constant, ie, the neutron POPULATION, as it is called, must remain constant.

Now the power produced and, therefore, the rate of fissioning varies from one point to another in a reactor. To allow for this variation, we talk in terms of the NEUTRON DENSITY in the reactor.

The neutron density is the number of neutrons per unit volume (eg, per cubic centimeter).

Therefore, although the total neutron population in the reactor remains constant, the neutron density differs from one point to another having a maximum value, usually, at the centre of the reactor core. However, the average neutron density throughout the core, is constant when the power is constant. If the power changes, the numbers of fissions per second changes and the average neutron density changes.

Neutron Power

The thermal power in a reactor could be measured from the temperatures and flow rates in the heat transport system, ie, from the thermal energy transported out of the reactor core. However, when the rate of fissioning and the neutron densities are changed, there is some delay before the temperatures settle down. It is, therefore, desirable that the neutron densities be measured also. This is done with electronic equipment which measured the neutron density at one point in the reactor. The equipment is usually calibrated, by comparison with the thermal power, in percent of full power. This measurement of neutron density is known as NEUTRON POWER. It has the advantage, over thermal power, of being an instantaneous measure of reactor power.

Neutron Flux

The quantity that determines how many fissions take place per second and what the neutron density is at any point, is known as the NEUTRON FLUX. It is the most difficult of these three quantities to understand. It may be sufficient to say that, the higher the neutron flux, at any point in the reactor, the higher the rate of fissioning at that point, the higher the neutron density at that point and the greater the power produced at that point. Like neutron density, the neutron flux is greatest at the centre of the reactor and, at constant reactor power, the average neutron flux, over the reactor core, is constant.

A physical meaning or definition of neutron flux can be arrived at as follows. We have said that, at constant power, the neutron density at any point in the reactor remains constant. However, these neutrons are not like peas packed in a tin, so that none of them leak out and no more peas can get into the tin. The neutrons are moving around all the time. Some neutrons move away from a particular location but an equal number move in to replace them, so that the total number in a particular volume remain constant.

Fig. 1 illustrates a 1 cm cube in the reactor with neutrons continually entering this volume from the rest of the core and neutrons escaping from this cubical volume to the rest of the core. The net number in this cube will remain constant.

If we take one face of the cube, neutrons are continually crossing this 1 sq cm of area in both directions and at all angles. The total number of neutrons crossing this unit area per second is the neutron flux at the centre of that unit area. This neutron flux can be defined as follows: -

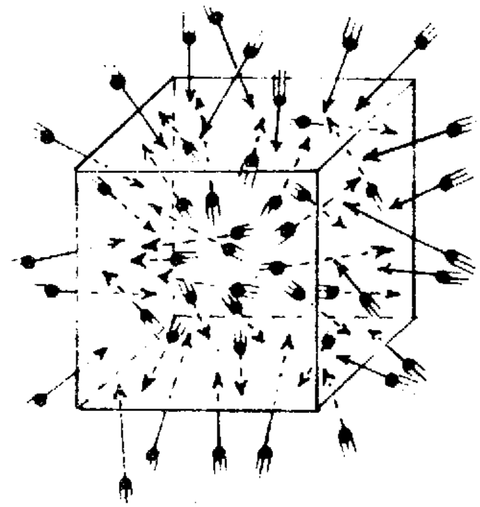

Fig. 1

> The neutron flux at a point in a reactor is defined as the number of neutrons per second, travelling in all directions, which cross a square centimeter area, placed at that point.

Therefore the flux will be measured in neutrons per square centimeter per second.

Neutron flux is usually denoted by the Greek letter $\phi$.

## ASSIGNMENT

1. What is meant by the term "Thermal Power" in a reactor?

2. (a) Define the term "Neutron Density" in a reactor.

   (b) If the thermal power in a reactor is constant, what can be said of (i) The average neutron density in the reactor?
   
   (ii) The neutron densities at different points in the reactor?

(iii) The neutron density at the centre of the reactor?

3. What is meant by "Neutron Power" in a reactor and what advantage does measurement of neutron power have over measurement of thermal power?

4. (a) Define "Neutron Flux".

   (b) How does the neutron flux affect the rate of fissioning, the neutron density and the power in a reactor?

Nuclear Theory - Course 227

# NEUTRON FLUX DISTRIBUTION AND ITS EFFECT ON POWER

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The meaning of neutron flux was introduced in the last lesson. It was also indicated that neutron flux was a maximum in the centre of a reactor and that the power produced in a reactor was proportional to the neutron flux.

We will now discuss, in greater detail, the manner in which neutron flux varies from one point to another and how this affects the power output of a reactor.

## Neutron Flux Distribution in a Reactor

We have seen that the flux is a maximum at the centre of the reactor but we do not know as yet how the flux varies across the reactor. In a reactor, we have fast neutrons produced during fission and the fast neutrons, that are not lost, are slowed down until they become thermal neutrons. Since it is the thermal neutrons that cause fission, it is the distribution of thermal neutron flux with which we are concerned.

The thermal neutron flux distribution (ie, the way the flux varies from one point to another in a reactor) will depend, very much on the shape of the reactor. Since the reactor is a solid, three-dimensional system, there could be different flux distributions in three perpendicular directions. Two reactor shapes only will be mentioned.

### (1) Cubical Reactor

Fig. 1 shows a cubical reactor, without a reflector, each side of which has length "a". The centre point of the reactor is at O. Taking O as the origin, three perpendicular axes Ox, Oy, and Oz have been drawn parallel to the sides of the cube.

In the case of a cube the flux distribution is the same in each of these three directions. Thus in the Ox direction it follows the relation:-

$$\phi = \phi_m \cos\left(\frac{\pi x}{a}\right) \quad . . . (1)$$

That is to say the flux $\phi$ at a distance x from O, along the Ox direction is given by equation (1). In this equation, $\phi_m$ is the maximum thermal neutron flux at O, the centre of the reactor.

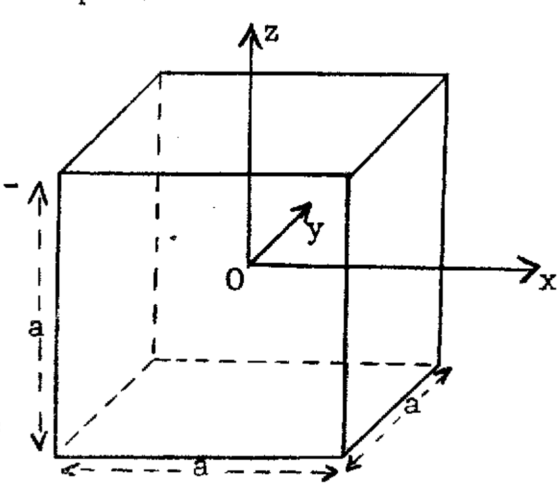


Fig. 1

If we substitute values of x in equation (1), and remember that $\pi = 180^o$ we can calculate the corresponding values of $\phi$. If $\phi$ is then plotted against x, the curve in Fig. 2 is obtained.

Fig. 2, then, shows the cosine distribution of the thermal neutron flux, not only along Ox, but also along Oy and Oz.

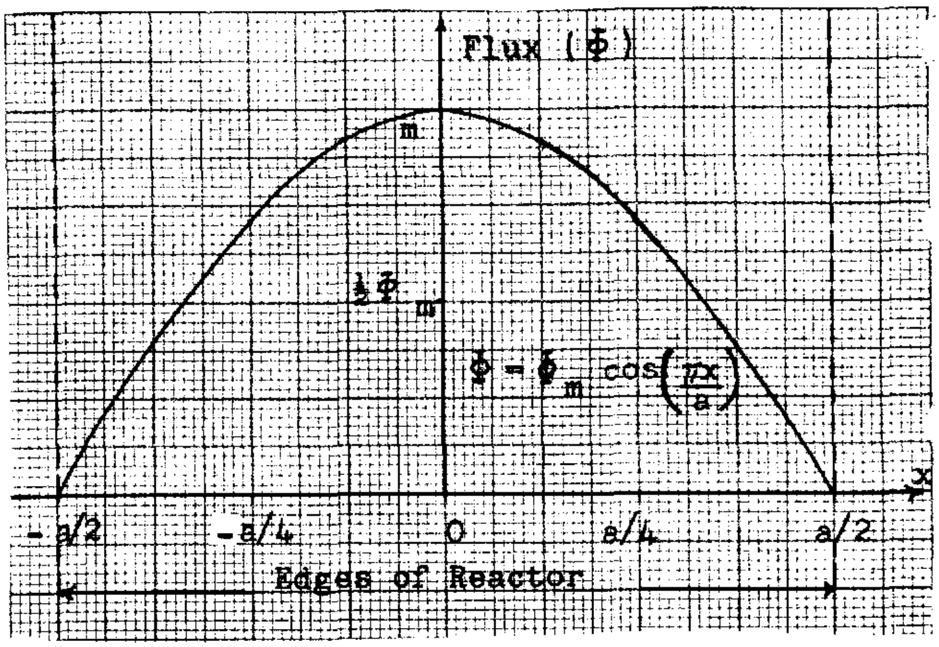


Fig. 2

(2) Cylindrical Reactor

Fig. 3 shows a cylindrical reactor of radius R and length L. Again O is the centre of the reactor. In this case there are only two directions along which the neutron flux distributions have to be considered. These directions are along any radius Or from O (known as the radial distribution) and along Oz, the axis of the cylinder (the axial distribution).

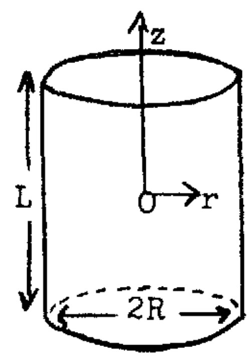


Fig. 3

The radial flux distribution is given approximately by

$$\phi = \phi_m \cos\left(\frac{\pi r}{2R}\right)$$

The flux distribution along the axis is given by $\phi = \phi_m \cos\left(\frac{\pi z}{L}\right)$ where $\phi_m$, in each case is the maximum flux at O.

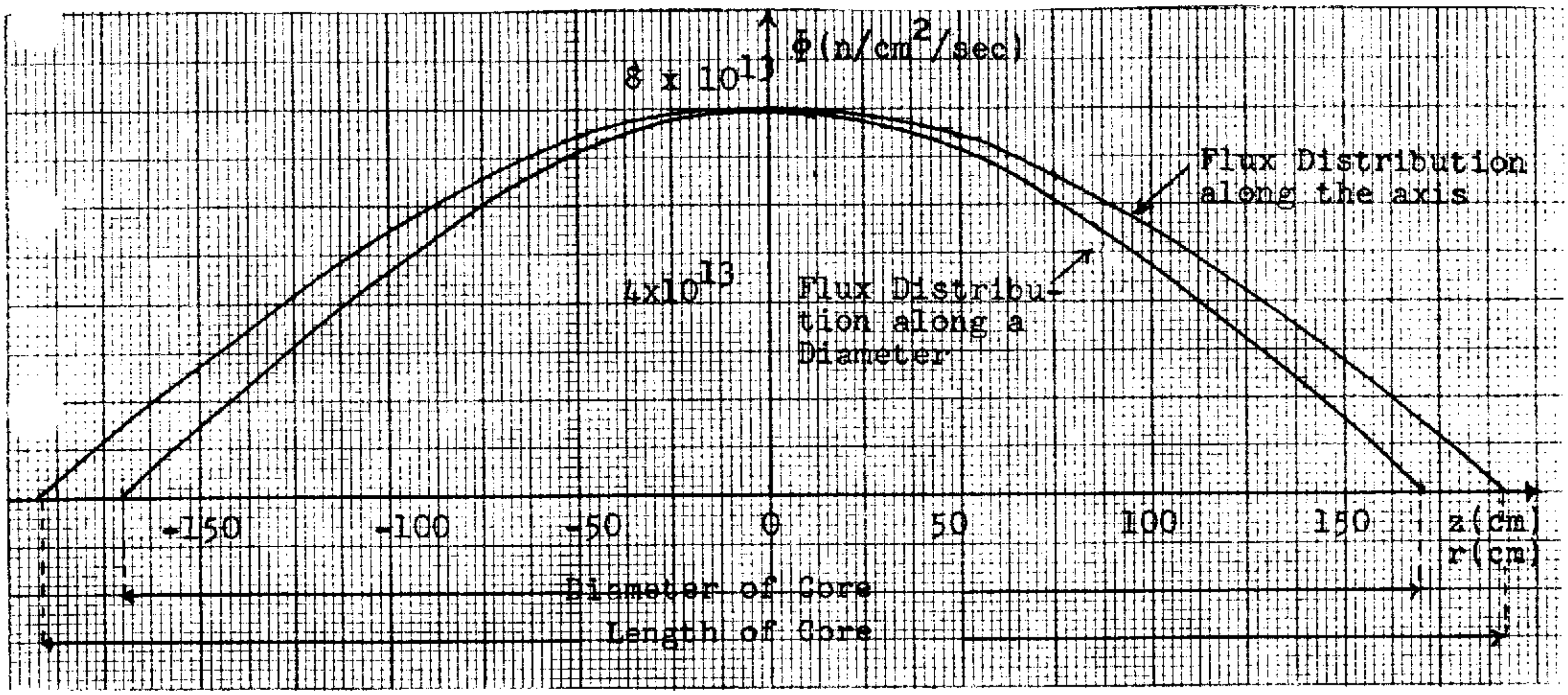


Fig. 4

Figure 4 shows the radial and axial flux distribution for the NPD reactor, without its reflector, for which R = 169 cm, L = 384 cm and $\emptyset_m$ is approximately $8 \times 10^{13}$ neutrons/cm$^2$/sec.

Again the thermal neutron flux at any point is given by a cosine formula.

From the curves in Figs. 2 and 4, we can list several interesting conclusions, as follows: -

(1) In all cases the neutron flux falls to zero at the edge of the reactor and therefore very little power is being produced by fuel in the outer regions of the reactor core.

(2) The flux has a very definite maximum value at the centre of the core and, therefore, the maximum power is being produced near the centre.

(3) The total power produced by the reactor depends on the average thermal neutron flux. Due to the type of flux distribution that we have in the cylindrical reactor, the average flux is only about 27.5% of the maximum flux or the maximum flux is 3.6 times greater than the average flux.

Thus the fuel in the centre is producing 3.6 times more power than the average fuel in the core.

(4) The only way to increase the total power produced is by increasing the average flux and this can only be done by increasing the maximum flux still further. However, this would increase the heat released in the fuel at the centre and the amount of heat that can be produced is usually limited by the fuel rating and the heat removal capacity of the heat transport system.

In summary, then, we can say that, in a reactor without a reflector, the thermal neutron flux distribution leads to the following disadvantages: -

(1) Poor use of fuel, in the outer regions of the core, to produce power.

(2) Limitation of the total power produced because of the high ratio of the maximum flux to the average flux, ie, the average flux and, therefore the total power, could be much higher but for the fact that the average flux cannot be increased without exceeding the limit on fuel rating for fuel in the centre of the reactor.

<u>Effect of a Reflector Around the Core</u>

A reflector is a substance, placed around the reactor core, to reflect neutrons back into the core. This makes more neutrons available for fission so that the critical size of the reactor is smaller or with the same size of reactor a higher fuel burnup can be obtained.

Let us now see how the thermal neutron flux distribution is affected by surrounding the core with a reflector. To illustrate the effect of the reflector we will again take, as a specific example, the radial flux distribution in NPD. Fig. 5 shows the same radial flux distribution, as in Fig. 4 without a reflector. It also shows, for comparison, the flux distribution when the core is surrounded by a reflector.

An examination of Fig. 5 will lead to the following conclusions: -

(1) The flux, at the edge of the core is no longer zero. It drops down to only 37% of the maximum flux. Therefore much more power is being produced by the fuel in the outer regions.

(2) The flux still has a maximum value at the centre but the average flux is now around 42% of the maximum when both radial and axial distributions are allowed for.

(3) Due to the higher average neutron flux, the power produced is much greater than without the reflector, even though we have the same size core and the same maximum flux.

(4) The graph representing the flux distribution is still a cosine curve but it now becomes zero outside the core ie, the reflector has increased the effective core diameter.

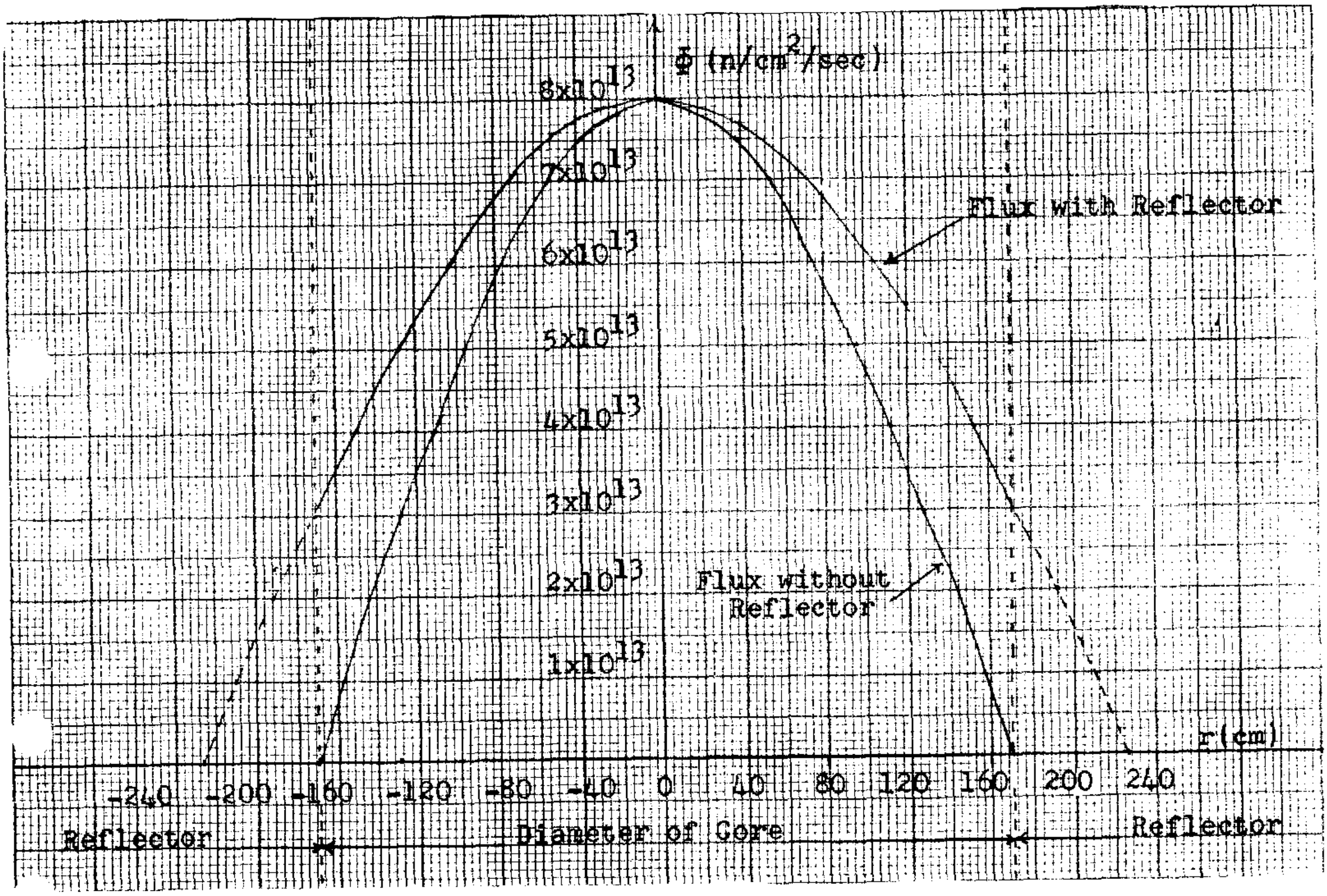


Fig. 5

## ASSIGNMENT

1. Write down the equations giving the radial and axial flux distributions in a cylindrical reactor, without a reflector, explaining the meaning of the terms used.

2. If graphs were drawn of the equations in question 1, what three facts about the flux values could be learned from the graphs?

3. What effects has this type of flux distribution on the effective use of fuel and on total power produced by the reactor?

4. If a reflector is placed around the reactor core, what effect will this have on:

   (a) Flux distribution or flux values?

   (b) Reactor power?

A. Williams

Nuclear Theory - Course 227

# MEANING OF CRITICALITY, MULTIPLICATION FACTOR, REACTIVITY AND NEUTRON LIFETIME

In the previous lessons, we discussed a reactor operating at steady power, in which there is an exact balance between the neutrons lost and neutrons produced through fission. The neutron population remained constant, the neutron flux remained constant and the chain reaction was just being maintained.

We will now turn our attention to changes in power and in neutron density, which may occur due to a variety of reasons.

## Criticality and Neutron Multiplication

In a critical chain reaction, the number of fissions in each succeeding generation of neutrons remains constant. There is no MULTIPLICATION of neutrons and the power remains constant.

Suppose that neutron losses, by non-fission capture or leakage, are reduced somehow or other. More neutrons are then available for fission and the number of fissions occuring in any one generation will be greater than in the previous generation. Thus more neutrons are produced, in any one generation, than in the previous generation. There is, therefore, a multiplication of neutrons.

Fig. 1 illustrates an extreme case of such neutron multiplication. The neutron losses have been reduced so much that two neutrons, from each fission, become available to cause further fissions. It shows the number of neutrons in succeeding generations, that would result from one original neutron. The just critical case is shown immediately underneath, for comparison. It is obvious from the figure that there is a doubling of the neutron population, and therefore a doubling of power, every generation. Since the time between one generation and the next is only about one-thousandth of a second, this means that the power would increase to 1000 times its value in one-hundredth of a second. This is obviously an extreme case, but it serves to illustrate the principle of neutron multiplication.

The ratio of the number of neutrons, available for fission, in one generation, to the number available in the previous generation is called the MULTIPLICATION FACTOR (k). Hence

$$k = \frac{\text{Number of neutrons causing fission in the 2nd generation}}{\text{Number of neutrons causing fission in the 1st generation}}$$

In the case illustrated, k = 2

The neutron multiplication factor is also the number of neutrons from each fission, which are available for further fission.

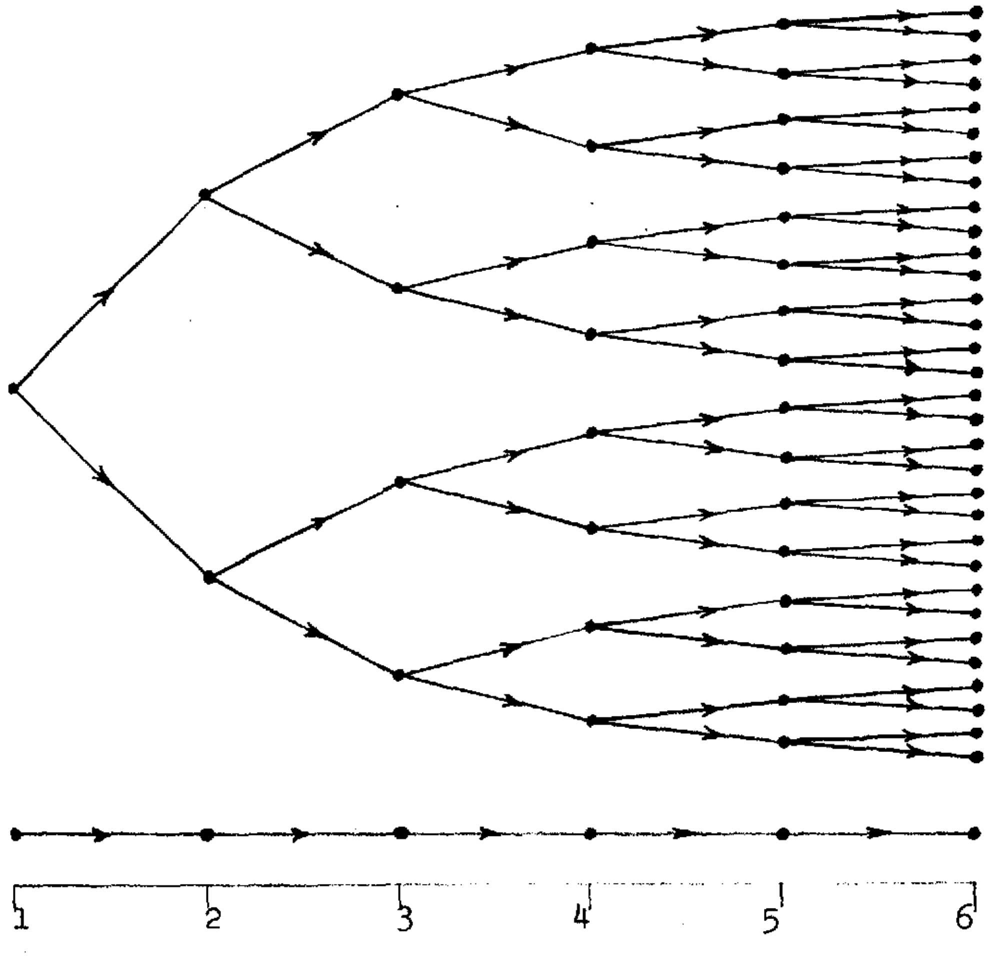


Fig. 1

Having defined the neutron multiplication factor, we can now add the following statements: -

(1) The chain reaction is just sustained, or the reactor is just critical or the reactor power remains unchanged if k = 1.

It should be noted that the reactor can be critical at any power, whether it be 1 watt or 500 Megawatts, as long as that power remains unchanged.

(2) The neutron population flux or neutron density increases and the power increases if k is greater than 1. The reactor is then said to be SUPERCRITICAL.

So, to increase reactor power, we have to reduce neutron losses or increase U-235 fissions to make k greater than unity, and, thus cause neutron multiplication.

(3) The neutron population, neutron flux, neutron density and the reactor power all decrease if k is less than 1. The reactor is then said to be SUBCRITICAL, ie, below critical.

Thus, to decrease reactor power, we have to increase neutron losses to make k less than unity.

The state of CRITICALITY in a reactor is the state achieved when k becomes equal to unity and the reactor becomes critical.

Reactivity

A reactor is critical when $k = 1$. The factor that would determine how subcritical or supercritical a reactor may be, is the amount by which k differs from 1. For this reason, the quantity $(k-1)$ may be more important than k itself, especially in reactor regulation considerations. We might say that changes in k are as important, if not more important, than k itself.

Now $k = 1$ while the reactor is just critical, but k, in a reactor, usually has a maximum value greater than unity, to permit reactor regulation and to allow for absorption of neutrons by poisons which accumulate during reactor operation. Thus k may not be 1 all the time. Changes in the value of k are referred to as REACTIVITY changes, and are normally measured in a unit called the milli-k or mk, which represents one thousandth of k.

The reactivity of a reactor can therefore be used to measure how far the reactor is from being just critical.

We can therefore say that: -

(1) the reactor is just critical when $k = 1$

(2) the reactor is supercritical when k is greater than one or when there is excess positive reactivity, ie, if $k = 1.003$, the reactor is 3 mk above critical.

(3) the reactor is subcritical when k is less than one or when the reactivity is negative, ie, if $k = .997$ the reactor is -3 mk or 3 mk below critical.

For some calculations it is convenient to use a value of reactivity which is expressed as follows:

$$\text{Reactivity} = \frac{k - 1}{k}$$

For most changes in a reactor where k is very close to 1, it can be seen that there is very little difference between this value and k - 1.

It should be understood, however, that within Ontario Hydro, when a reactivity value is given, it is expressed as a change in k ($\delta k$) or k-1 unless it is specifically identified otherwise.

## Neutron Lifetime

The average time between successive neutron generations will decide how fast multiplication of neutrons take place. For any particular value of k or reactivity, the smaller the time between one neutron generation and the next, the faster the neutrons multiply and the faster the power increases. This time between successive neutron generations is called the NEUTRON LIFETIME.

The neutron lifetime will include the time it takes for the nucleus to fission after it has captured a neutron, the time it takes for the fast neutron, which is produced during fission, to slow down and the time taken by the thermalized neutron to be captured.

For prompt neutrons, the neutron lifetime is one-thousandth of a second (0.001 sec). The lifetime of a delayed neutron will depend on the half-life of the nucleus producing it. The average lifetime of the delayed neutrons is about 0.09 sec and this is much higher than that of prompt neutrons.

## ASSIGNMENT

1. Define the neutron multiplication factor, k.

2. In terms of k, when is the reactor (a) just critical?
   (b) supercritical?
   (c) subcritical?

3. (a) Define or explain the term "Reactivity".

   (b) In what units will reactivity be measured?

   (c) When k = 1.005, calculate the reactivity.

4. In terms of reactivity, when is the reactor (a) just critical?
(b) supercritical?
(c) subcritical?

5. (a) What is meant by "Neutron Lifetime"?

(b) Why is there a difference between the lifetimes of the prompt and delayed neutrons and in round numbers, what are the average values of the two lifetimes?

A. Williams

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# CHANGE OF REACTOR POWER WITH REACTIVITY CHANGE

In this lesson, we will consider how neutron density, neutron flux and reactor power change when the multiplication factor, k, or the reactivity, $\delta k$, change. For the moment, we will ignore the effects of the accumulation of fission products in the fuel and the effects of using up the U-235 in the fuel. We are only concerned with how the reactor power changes immediately following a change in reactivity. This will enable us to decide, later, how changes in power can be achieved safely, ie, it will help us to determine how reactor power can be regulated.

## Effect of Reactivity on Neutron Multiplication

In the previous lesson, we saw how neutrons multiplied at a fantastic rate when the multiplication factor was equal to 2 or the reactivity was 1000 milli-k. This is, of course, an extreme case for two reasons: -

(1) A multiplication factor of 2 would be impossible to achieve in practice because neutron losses could not be cut down to this extent.

(2) Even if such a value of k was possible, the neutrons multiply so fast that the power would increase 1000 times in one-hundredth of a second. This type of power increase would be impossible to control and is, in fact, an explosive rate of increase.

Let us now look at more practical values of k and compare the neutron multiplications for various values of k.

Fig. 1 shows how the neutrons multiply for values of k from 1.0005 to 1.003 or for reactivity values from 0.5 milli-k to 3 milli-k.

When $\delta k$ = 0.5 mk, the neutron population or neutron density is doubled in 1400 neutron generations.

When $\delta k$ = 1 mk, the neutron population or density is doubled in 700 neutron generations, trebled in 1100 generations and is increased by a factor of 4 in 1400 generations.

When $\delta k$ = 2 mk, the neutron density is doubled in 350 generations and has to increase 15 times its original value in 1350 generations.

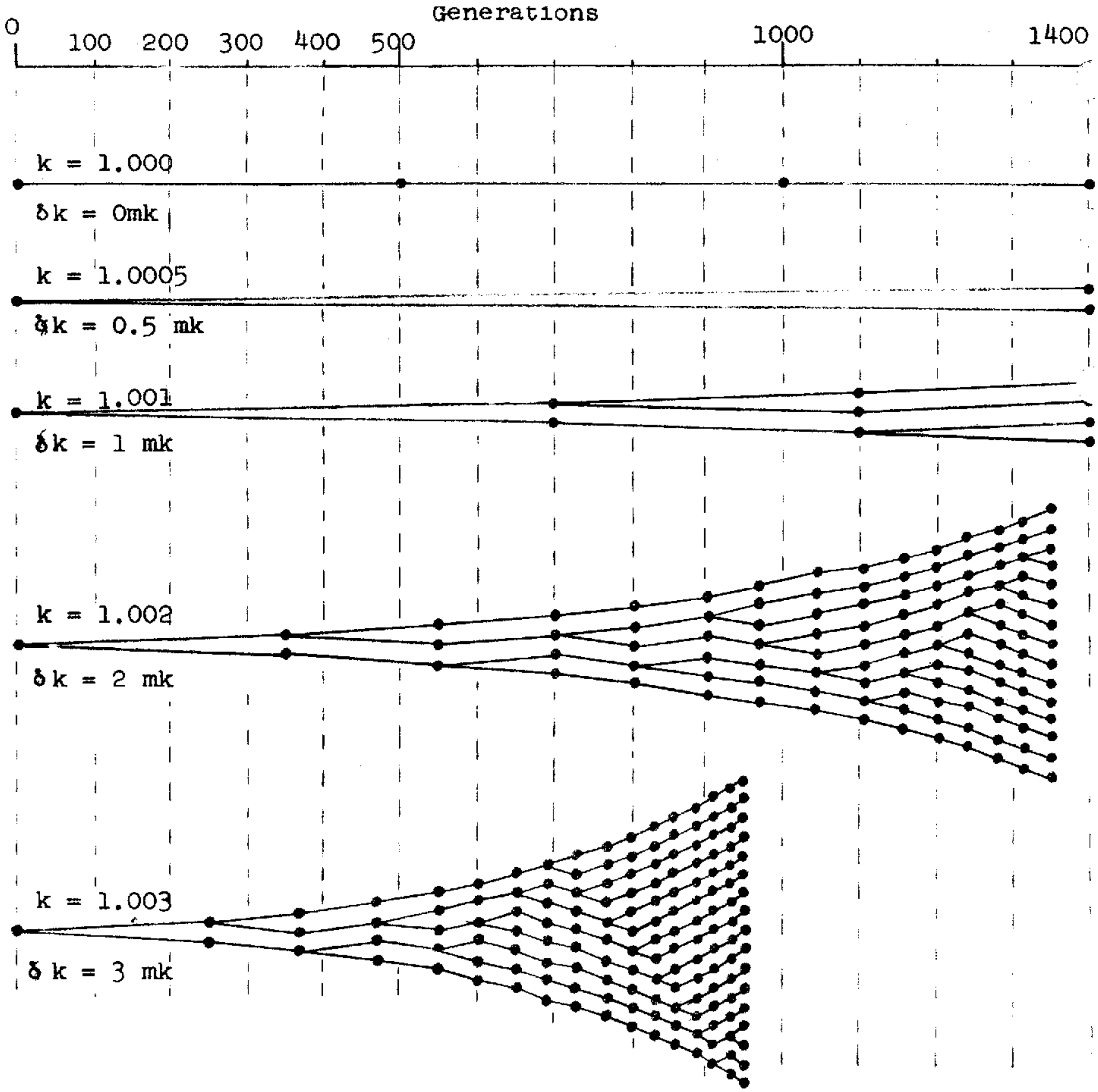


Fig. 1

Finally, when $\delta k = 3$ mk, the neutron density is doubled in only 250 generations and, in 950 generations, the neutron density would be 17 times its original value.

So, as k, and the reactivity increase in value, the number of neutrons grows progressively faster and faster. Also for any one value of the reactivity, the number of neutrons produced in successive equal numbers of generations, gradually increases.

eg, when $\delta k = 3$ mk.

New additional neutrons produced in first 250 generations is 1.12
New additional neutrons produced in second 250 generations is 2.36
New additional neutrons produced in third 250 generations is 5.02
New additional neutrons produced in fourth 250 generations is 10.5

In comparison, when $\delta k = 2$ mk,

New additional neutrons produced in first 250 generations is 0.65
New additional neutrons produced in second 250 generations is 1.07
New additional neutrons produced in third 250 generations is 1.76
New additional neutrons produced in fourth 250 generations is 2.9

Figure 2 shows, graphically, how the neutron density increases for various values of reactivities.

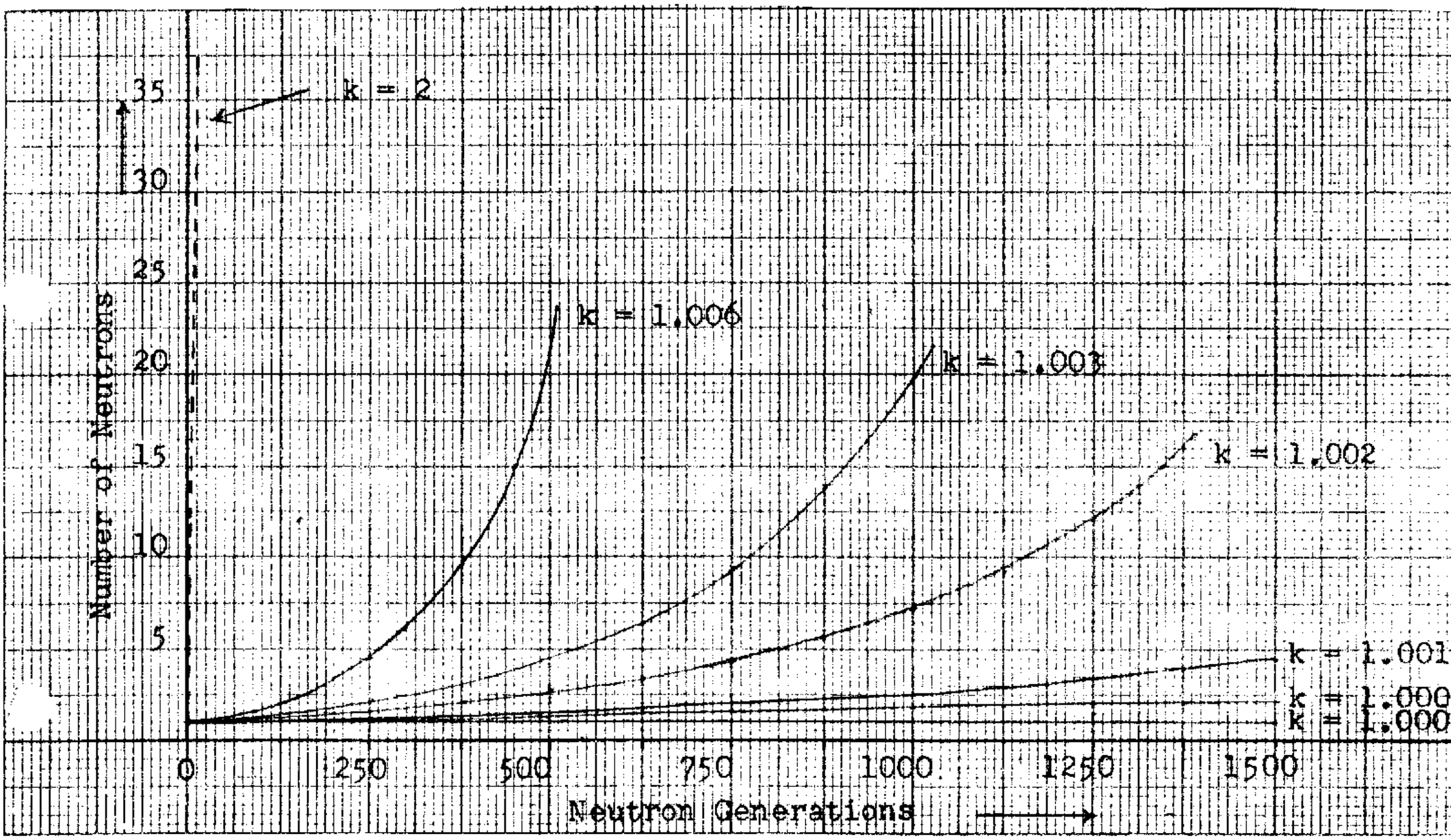


Fig. 2

The shape of each curve is the same as the shape of the envelope of the corresponding neutron diagram in Fig. 1 Such curves are known as EXPONENTIAL curves. Thus we can say that the neutrons population in a reactor increases exponentially.

If we had started with a neutron density $n_o$, in a reactor, instead of just one neutron, then the same would apply to every neutron we had initially. Therefore, the neutron density also increases exponentially.

Thus if the neutron density is n, then after N neutron generations: -

$$n = n_o e^{aN}$$

where "e" is the "exponential e" (e = 2.718) and a is some factor depending on the reactivity or the value of k. In fact, a = $\delta$k.

That is,

$$n = n_o e^{N.\delta k}$$

If $\delta k$ = 0.5 mk = 0.0005 and N = 1400 $\qquad n = n_o e^{0.7} = 2.01\ n_o$

If $\delta k$ = 3 mk = 0.003 and N = 240 $\qquad n = n_o e^{0.72} = 2.05\ n_o$

As can be seen, these are more accurate figures than those used in Fig. 1.

## Effect of Reactivity on Neutron Flux and Reactor Power

We have seen that both the neutron flux and the power level, in a reactor, are proportional to the neutron density. Therefore, reactor power and the neutron flux follow the same exponential law as neutron density. So we can write: -

$$P = P_o\ e^{N.\delta k} \quad \text{for the power}$$

and

$$\phi = \phi_o\ e^{N.\delta k} \quad \text{for the flux}$$

On plotting either power or flux against the number of neutron generations, N, we get the same shape curves as in Fig. 2.

In words, what happens is that, if $\delta$k is positive, more fissions take place than is just required to maintain the chain reaction. The neutron density, therefore, increases and still more neutrons become available to cause further fissions. Thus, the neutron density increases, the rate of fissioning therefore increases and consequently, the power increases. All these increases are exponential.

Effect of Negative Reactivity Changes

Suppose that a reactor was operating at steady power so that it was just critical (ie, $k = 1$, $\delta k = 0$). Now suppose that k was suddenly reduced below 1, say to $k = 0.999$. The reactivity, $\delta k$, is now -0.001, or -1 mk. What happens?

Obviously, the chain reaction can no longer be sustained so that the neutron density starts to decrease. We no longer have one neutron from each fission causing a further fission. So the reactor power decreases and the flux increases.

The neutron population decreases and we go from **right** left, in Fig. 1, instead of from left to right. The neutron density, neutron flux and reactor power again change exponentially except that $\delta k$ is now negative.

ie, $n = n_o e^{N.\delta k}$ $\phi = \phi_o e^{N.\delta k}$ and $P = P_o e^{N.\delta k}$

eg, if $\delta k = -1$ mk $= -0.001$ and $N = 1000$ $P = P_o e^{-1} = 0.037\ P_o$

So the power decreases to 37% of $P_o$ in 1000 neutron generations if $\delta k = -5$ mk $= 0.005$ and $N = 1000$ $P = P_o e^{-5} = 0.007\ P_o$ or 0.7% $P_o$

The more negative reactivity is introduced (ie, the lower **k** is made) the faster the power decreases.

ASSIGNMENT

1. How is reactor power affected by: -

   (a) increasing the positive reactivity in a reactor?

   (b) increasing the negative reactivity in a reactor?

2. (a) The reactivity is +3 mk. The number of neutrons produced in the first 250 neutron generations is 2.12, the number produced in the second 250 generations is 4.48, in the third 250 generations, 9.5 and in the fourth 250 generations, 20. What do these figures illustrate?

2\. (b) If the reactivity is reduced to zero at the end of these 1000 neutron generations, what would be the number of neutrons produced during the next 250 generations?

(c) If the multiplication factor, k, is made equal to 0.997, at the end of these 1250 generations, what would be the number of neutrons at the end of the next 500 generations?

3\. Write down the exponential equations connecting the neutron density, neutron flux and neutron power with reactivity.

A. Williams

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# CHANGE OF REACTOR POWER WITH TIME

We have seen, in the previous lesson, how reactivity affects neutron multiplication and reactor power. We saw how reactor power increases from one neutron generation to the next. However, it is not the actual change in reactor power that is always of importance, but the rate at which power increases or decreases.

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

## Effect of Neutron Lifetime on Changes in Reactor Power

We have seen how reactor power changes, for positive and negative reactivities, in terms of neutron generations. In other words, we have seen how the neutron population or neutron density changes from one generation to the next when the multiplication factor, k, is greater than one or less than one. In all cases an exponential law is followed which may be written: -

$$P = P_o e^{N.\delta k} \quad \text{or} \quad n = n_o e^{N.\delta k}$$

where $P_o$ and $n_o$ are the initial power and neutron density respectively and P and n are the power and neutron density values, N neutron generations later. $\delta k$ is the reactivity which is positive when k is greater than 1, zero when k = 1 and negative when k is less than one.

Thus we can find what the power would be after so many neutron generations. However, what we really want to know is what the power will be after a certain time. We want to know how fast or how slow the power changes for some definite value of $\delta k$. We want to know whether or not the increase in power is too fast to handle or whether or not decrease in power is rapid enough when the reactor is shut down. These are the facts that are of practical importance.

What other factor is required in order to determine how the power changes with time? We have to know the time between successive neutron generations. The average time between successive neutron generations determines how fast the neutrons multiply or how fast the power increases. It will also decide how fast the power decreases when $\delta k$ is negative. This average time between successive neutron generations has been defined as the NEUTRON LIFETIME.

If $\mathscr{L}$ is used to denote the neutron lifetime, ie, the time between successive neutron generations, the time, t, for N neutron generations is given by: -

$$t = N.\mathscr{L}$$

$$\text{or} \quad N = \frac{t}{\mathscr{L}}$$

Therefore $P = P_o e^{\frac{\delta k}{\mathscr{L}}.t}$ and $n = n_o e^{\frac{\delta k}{\mathscr{L}}.t}$

These are, then the equations that give the reactor power or the neutron density t secs after the reactivity is changed from zero to $\delta k$. From the equations we can see that: -

(1) the greater the positive value of the reactivity, $\delta k$, the faster the increase in neutron density and power.

(2) the greater the negative value of the reactivity, $\delta k$, the faster the decrease in neutron density and power.

(3) the greater the value of the neutron lifetime, $\mathscr{L}$, the slower the change in neutron density and power.

Since the reactivity can be controlled at any suitable value in a reactor, the value of $\mathscr{L}$ will eventually decide how fast the power changes; the value of $\mathscr{L}$ is very important.

Reactor Period

From Fig. 1 in the previous lesson or from the above equations, we can say that the reactor power doubles in so many generations or in such and such a time.

eg, when $\delta k$ = 0.5 mk, the power doubles in 1400 generations, or if one generation or neutron lifetime was 0.001 sec, then the power would double in 1.4 sec.

When $\delta k$ = 3 mk, the power doubles in 250 generations or 0.25 sec.

Alternatively we could specify the time that the power takes to increase tenfold ie, for P to become 10 $P_o$. These are both indications of how fast the power increases.

In practice, however, the rate of increase or decrease of power is always indicated by specifying the time taken for the power to change by a factor of e (the exponential e = 2.718), ie, for P to become $eP_o$ or $P_o/e$.

Thus when $\delta k$ = 1 mk the power changes by a factor of e in 1000 neutron generations or 1 second, if $\mathcal{L}$ = 0.001 sec.

When $\delta k$ = 3 mk the power changes by a factor of e in 333 generations or 0.33 sec., if $\mathcal{L}$ = 0.001 sec.

The <u>time</u> required for the power to change by a factor of e is called the REACTOR PERIOD and is denoted by the letter T sec.

Therefore, <u>if</u> the neutron lifetime is 0.001 sec and the reactivity is 1 mk, the reactor period is 1 second.

However, if the neutron lifetime is 0.1 sec. and $\delta k$ = 1 mk, then T = 100 sec.

It can be seen, from these examples, that the reactor period is very dependent on the neutron lifetime as well as on the reactivity. In fact, the period is connected with these two quantities by the formula: -

$$T = \frac{\mathcal{L}}{\delta k}$$

The equations connecting the neutron density and power with time can now be modified to: -

$$P = P_o e^{t/T} \qquad \text{and} \qquad n = n_o e^{t/T}$$

It is precisely because of the exponential form of these equations that the reactor period is defined as the time for the power to change by a factor e rather than 2 or 10 or 100 or some other nice round number.

From the equations, when t = T $P = eP_o$ and $n = en_o$

The shorter the reactor period, the faster the power changes will be.

## ASSIGNMENT

1. Give two reasons why it is desirable to know how reactor power changes with time.
2. How does neutron lifetime affect the way in which reactor power changes with time?

3. Define the term "Reactor Period"?

4. (a) On what two quantities does the reactor period depend?

   (b) Write down the equation connecting the period with these two quantities.

   (c) Write down the equation connecting the power with the reactor period.

A. Williams

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# EFFECTS OF PROMPT AND DELAYED NEUTRONS ON REACTOR POWER CHANGES

In the previous lesson, we saw that the reactor period and, consequently, the rate at which the power increases, depend on both the reactivity, $\delta k$, and on the neutron lifetime, $\mathcal{L}$.

The reactivity is a quantity which is, normally, controlled by the regulating system and which, therefore, can, normally, be given any value that is desireable and reasonable. However, the neutron lifetime, or the time between successive neutron generations, is a quantity which is characteristic of the reactor itself.

The neutron lifetime of prompt neutrons includes the time for fission to occur, the time required for the neutrons to become thermalized and the time taken for the thermalized neutron to be captured. The lifetime of the delayed neutron, on the other hand, depends on the half-life of the nucleus producing it. We shall now examine how the lifetimes of the prompt and delayed neutrons affect reactor power changes and the reactor period.

## Effect of Prompt Neutrons Only

Suppose that all the neutrons, in a reactor, were prompt neutrons. The neutron lifetime of prompt neutrons is around 0.001 sec (one-thousandth of a second) so that there would be 1000 successive neutron generations taking place every second.

Therefore, $\mathcal{L}$ = 0.001 seconds.

Thus, even if $\delta k$ was only 0.5 mk, the reactor period would be 2 secs, and the reactor power would double in 1.4 secs.

If $\delta k$ = 2 mk, the reactor period would be 0.5 sec (ie, the power would nearly triple in one-half a second). In 1 sec, the reactor power would increase by a factor of 7.4. This means that, if the reactor was operating at full power and the reactivity was suddenly increased by 2 mk, the power could be about 7 - 1/2 times full power in 1 second.

Fig. 1 shows how P/Po varies with time for various values of positive reactivity.

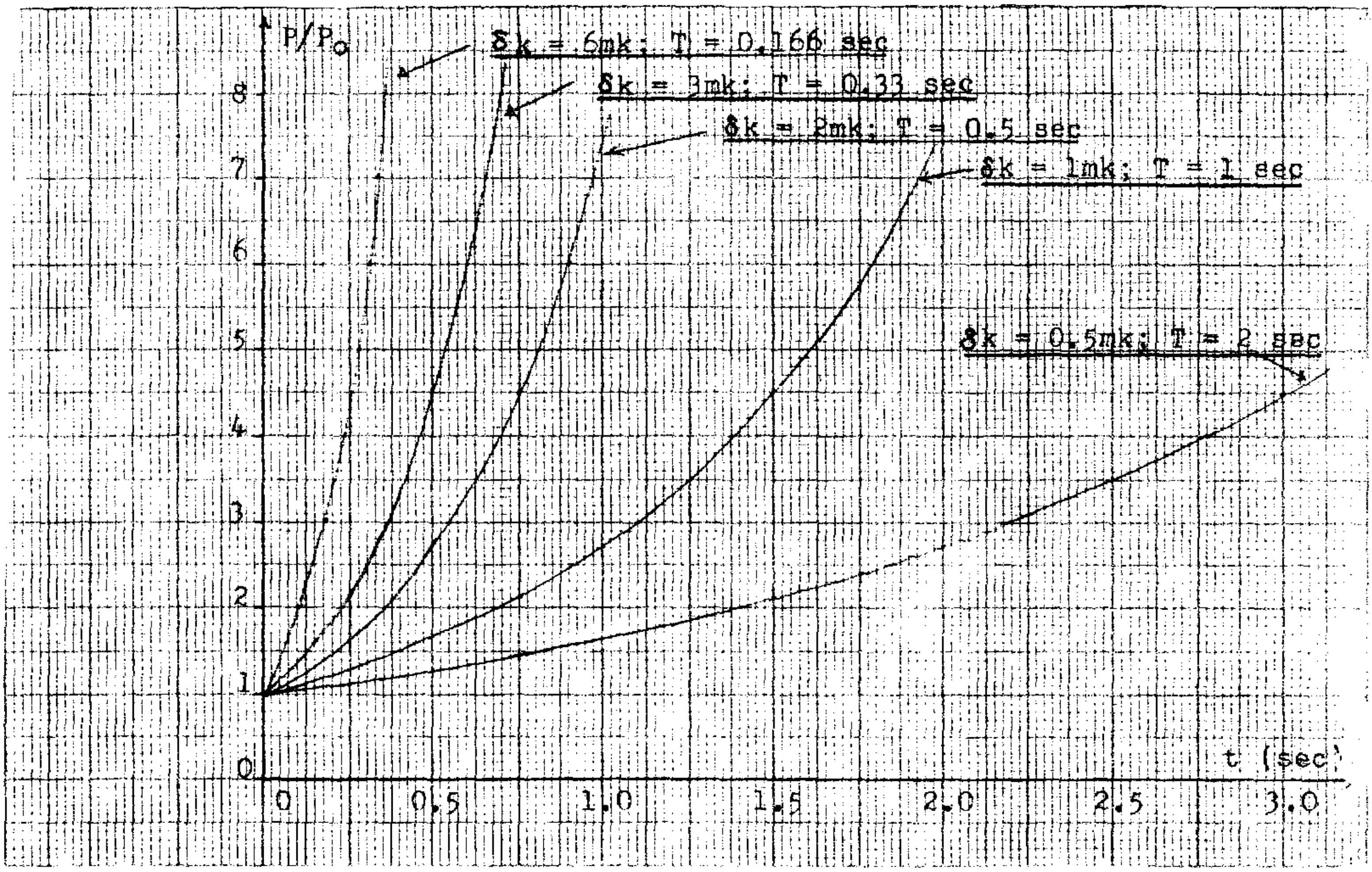


Fig. 1

A practical regulating system cannot cope with rapid increases in power of this kind. In fact, reactor regulation would not be possible if all the neutrons, in a reactor, were prompt neutrons. Even the fastest acting protective system will take at least 1 second to act. In this time, severe damage would have resulted from the high power level reached.

Now suppose that the reactor is operating at steady power and $\delta k$ is suddenly made negative, ie, the reactor is being shutdown. Let us see how the power decreases for various negative values of reactivity. Fig. 2 shows how the power would decrease if all the neutrons were prompt neutrons. Due to the wide range of power that has to be represented, the relative power, $P/P_o$, is plotted on a logarithmic scale. In this way, many decade of power (from one-tenth to one-millionth of full power or lower) can be covered on the same size sheet of graph paper that would only enable one decade (from full power to one-tenth of full power) to be covered, using a linear scale. It is interesting to note that an exponential graph, plotted on a logarithmic scale, becomes a straight line.

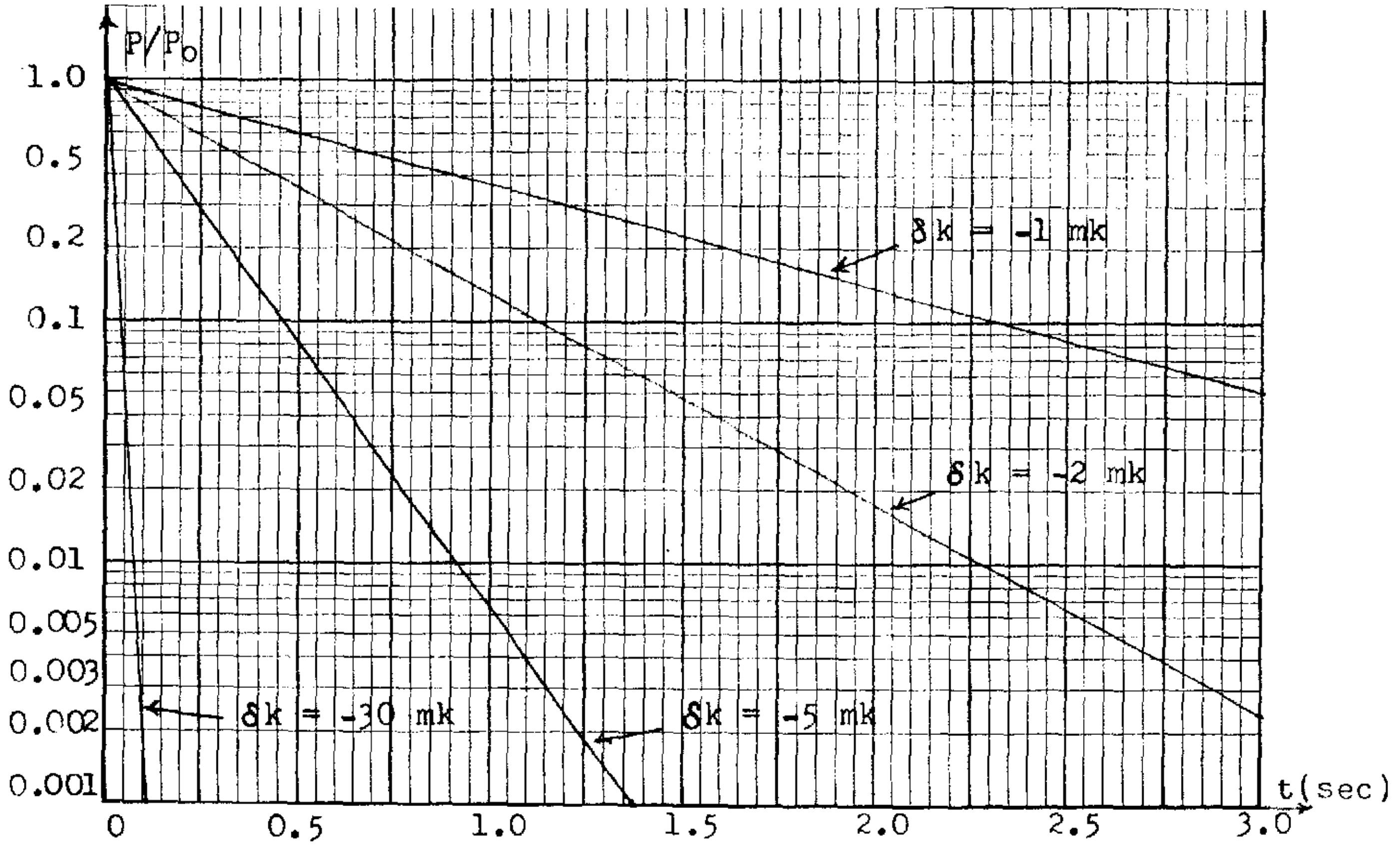


Fig. 2

It may be seen from Fig. 1 that the decrease in power would be quite rapid, even though the negative reactivity is only 1 or 2 mk. When the value of δk is -30 mk, (and this amount of negative reactivity or more is usually available), the power would decrease to one-thousandth of full power in only 0.2 seconds, if all the neutrons were prompt neutrons.

Effect of Delayed Neutrons

As was explained, in a previous lesson, the delayed neutrons form only about 0.75% of all the neutrons formed as a result of fission. However, the delayed neutrons are emitted by fission products, or their daughters. The nuclei which emit these delayed neutrons have half-lives ranging from a fraction of a second to 55.6 sec. Consequently, the production of these delayed neutrons may not occur for several seconds after the U-235 nucleus fissions. Due to this delay, the delayed neutrons are of much greater importance, in reactor regulation, than their numbers would suggest.

Even though they only form 0.75% of all the neutrons produced, they cause a sustantial increase in the average lifetime of all the prompt and delayed neutrons combined.

The average lifetime for prompt neutron alone = 0.001 sec

The average lifetime of prompt and delayed neutrons = 0.1 sec

This is an increase of 100 times in the value of $\mathcal{L}$.

Now let us see how this increase in $\mathcal{L}$ affects power changes or the rate at which the power changes.

With $\mathcal{L}$ = 0.1 sec, and $\delta k$ = +0.5 mk, the reactor period would be 200 sec, and it takes 139 sec for the power to double.

If $\delta k$ = +6 mk, T = 16.7 sec and the power increases by a factor of only 1.18 in 1 sec

However, it must be remembered that, if the reactivity is suddenly increased by a certain positive amount, it takes a tenth to a fifth of a second for the delayed neutrons to become effective. During this initial fraction of a second, the power increases as shown in Fig. 1.

Fig. 3 shows how the power increases, both during this initial period and after the delayed neutrons start playing a part. A logarithmic scale has again been used to cover a wider range in power. The dotted lines show the power increases that would have occurred had all the neutrons been prompt neutrons. The comparison between the dotted line and the corresponding continuous graph shows very clearly how much of an effect the delayed neutrons have on power increases.

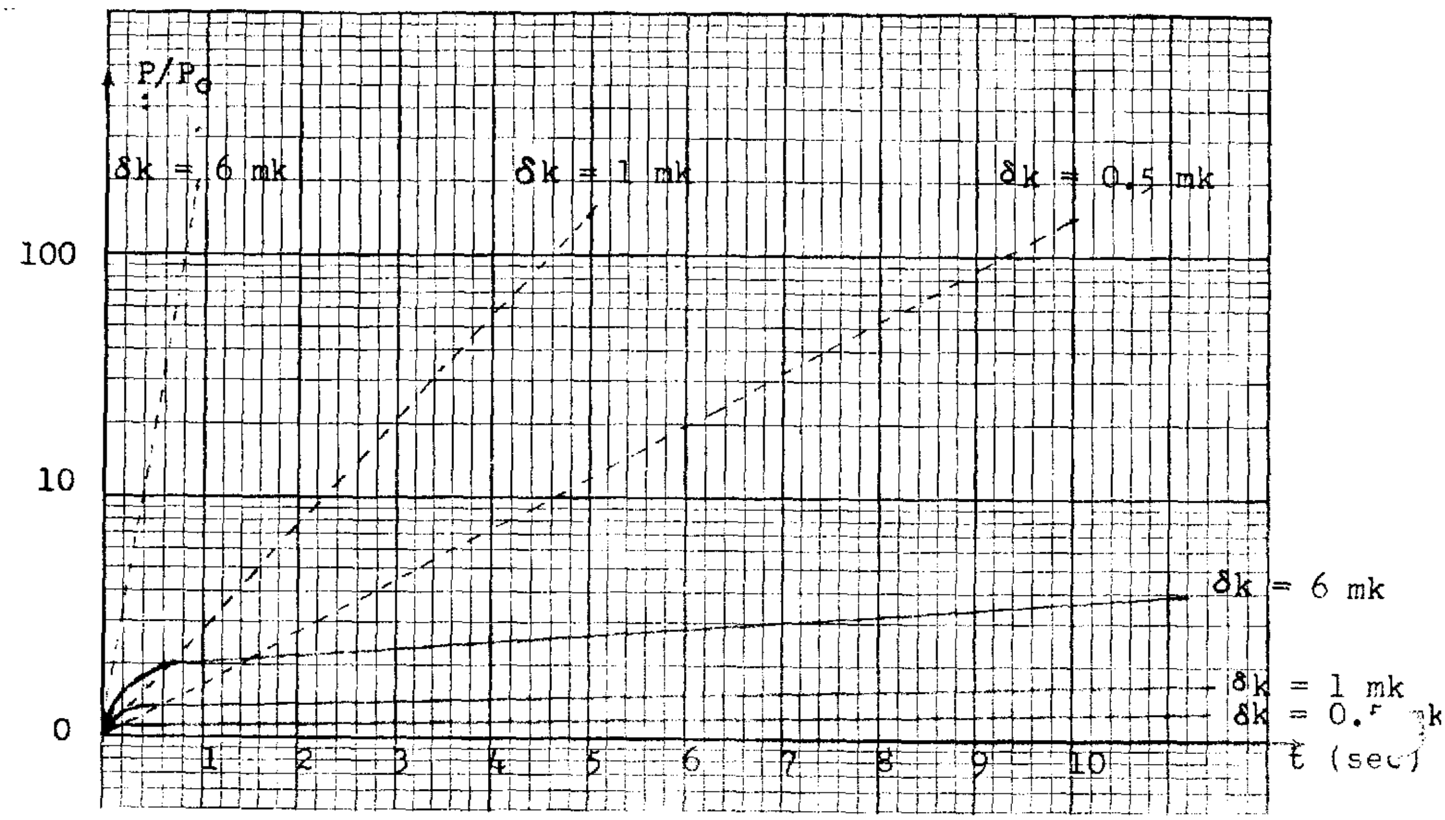


Fig. 3

The effect of the delayed neutrons is felt soon enough for the power increase not to be excessive before the regulating and protective systems can respond. In other words, the delayed neutrons make reactor regulation and protection a practical reality.

What of the power decrease when $\delta k$ is negative. Again, initially, the decrease in power is due entirely to the decrease in prompt neutrons and the power decrease tends to follow the graphs of Fig. 2. However, after the initial rapid decrease in power, the delayed neutrons become the deciding factor. Eventually, the power decrease is governed entirely by the delayed neutrons with the longer half-life (ie, the neutrons emitted by the nuclei with the 55.6 sec half-life).

The graphs in Fig. 4 show how the power decreases for various negative values of $\delta k$. Again, for comparison, the dotted lines show the way the power would have decreased with prompt neutrons alone. Several important facts can be observed from an examination of Fig. 4: -

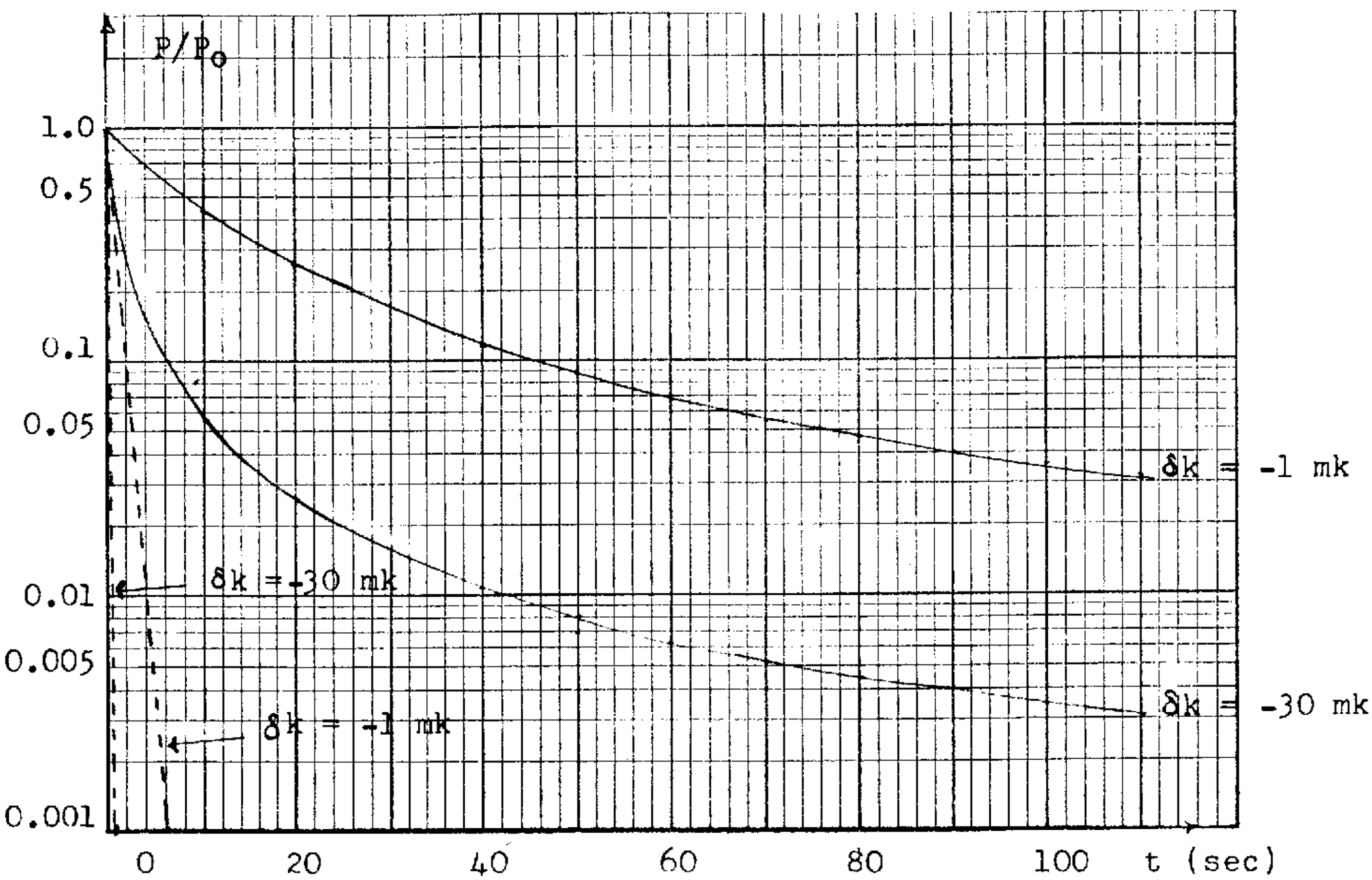


Fig. 4

(1) The delayed neutrons cause a considerable slowing down in power reduction.

(2) A substantial amount of negative reactivity is required to cause an initial rapid decrease in power before the delayed neutrons slow down the power reduction, eg, with $\delta k = -30$ mk the power drops to one-tenth of its initial value in 4 sec but it takes 40 sec for it to decrease by a further factor of 10. Therefore, if the reactor protective system is to cause an initial fast reduction in power, it must be able to introduce large negative reactivities quickly into the reactor.

(3) Since the final power decrease is governed by the delayed neutrons with a half-life of 55.6 sec it takes about 30 minutes for the power to be reduced 10 decades.

eg, from 100 Megawatts to one-hundredth of a watt

<u>Note 1</u>: The power referred to in this lesson is neutron power or the power obtained directly from fission. The thermal power in a reactor can be partially produced from decay of fission products and, as will be seen later, this affects the decrease in the total thermal power.

<u>Note 2</u>: The graphs shown assume that the positive or negative reactivity is introduced in one package in a fraction of a second (this is known as a "STEP" change in reactivity). In practice, it would take a finite time for such a reactivity change to take place.

## ASSIGNMENT

1. If all the neutrons, in a reactor, were prompt neutrons, with a lifetime of 0.001 sec: -

    (a) calculate the reactor period and the power increase in 1 second if $\delta k = 1$ mk and if $\delta k = 6$ mk.

    (b) explain how these power increases would affect reactor regulation and protection.

2. What affects do the delayed neutrons have on: -

    (a) neutron lifetime?

    (b) reactor period?

    (c) reactor regulation?

3. (a) What effects do the delayed neutrons have on the reduction of power when the reactivity is negative?

   (b) Why must a protective system be able to introduce large negative reactivities quickly into the reactor?

A. Williams

Nuclear Theory - Course 227

# EFFECT OF PHOTONEUTRONS ON REACTOR POWER CHANGES

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We have seen how delayed neutrons make reactor regulation possible and how they also slow down power decreases. Now prompt and delayed neutrons are produced directly or indirectly as a result of fission. Other neutrons are also produced in a reactor using heavy water. These are the photoneutrons produced when gamma rays are absorbed in deuterium nuclei.

Photoneutrons are peculiar to reactors using heavy water as a moderator or as a heat transport fluid. They are produced when gamma rays interact with the deuterium nuclei according to the equation: -

$$_1H^2 + \gamma = {}_1H^1 + {}_0n^1$$

Where do the gamma rays come from and when are these photoneutrons produced?

Gamma rays are emitted by radioactive nuclei in the heat transport system, the moderator system and the fuel. So photoneutrons are being produced while the reactor is operating and while these radioactive nuclei are being produced. However, they form a very insignificant part of the total neutron population during reactor operation and cannot really be said to contribute to reactor power.

What of the photoneutrons when the reactor is shut down?

The radioactive nuclei in the moderator and heat transport system decay fairly rapidly and would not exist for long as gamma ray sources. However, some of the fission products, formed in the fuel, have very long half-lives and would remain, as sources of gamma rays, for a long time after the reactor is shut down. Therefore, if heavy water is still present in the reactor core after shutdown, photoneutrons will be produced even though prompt and delayed neutrons are not being produced.

In the Canadian power reactor systems, the heat transport fluid and moderator are both heavy water. The heat transport fluid remains in the reactor during shutdown, but the moderator is usually drained from the reactor. Therefore, during reactor shutdown, a very small photoneutron population exists all the time. When moderator is introduced into the core, prior to starting up the reactor, the photoneutron population immediately increases to a value which depends on the duration of the shut down. Thus a neutron source exists even before the reactor goes critical unless, of course, the reactor has been shutdown for longer than about four months.

Of what significance is this photoneutron source and how does it contribute to power changes? There are two aspects that are important: -

(1) Effect on power reduction

It was indicated, in the previous lesson, that, when negative reactivity was introduced into the reactor, ie, $\delta k$ is given a negative value, the rate of decrease of power is very quickly determined by the delayed neutrons and finally by the delayed neutron group with the 55.6 second half-life. However, here we now have another group of neutrons which decrease more slowly than the delayed neutrons. These photoneutrons, in fact, decrease at the same rate as the fission products decay.

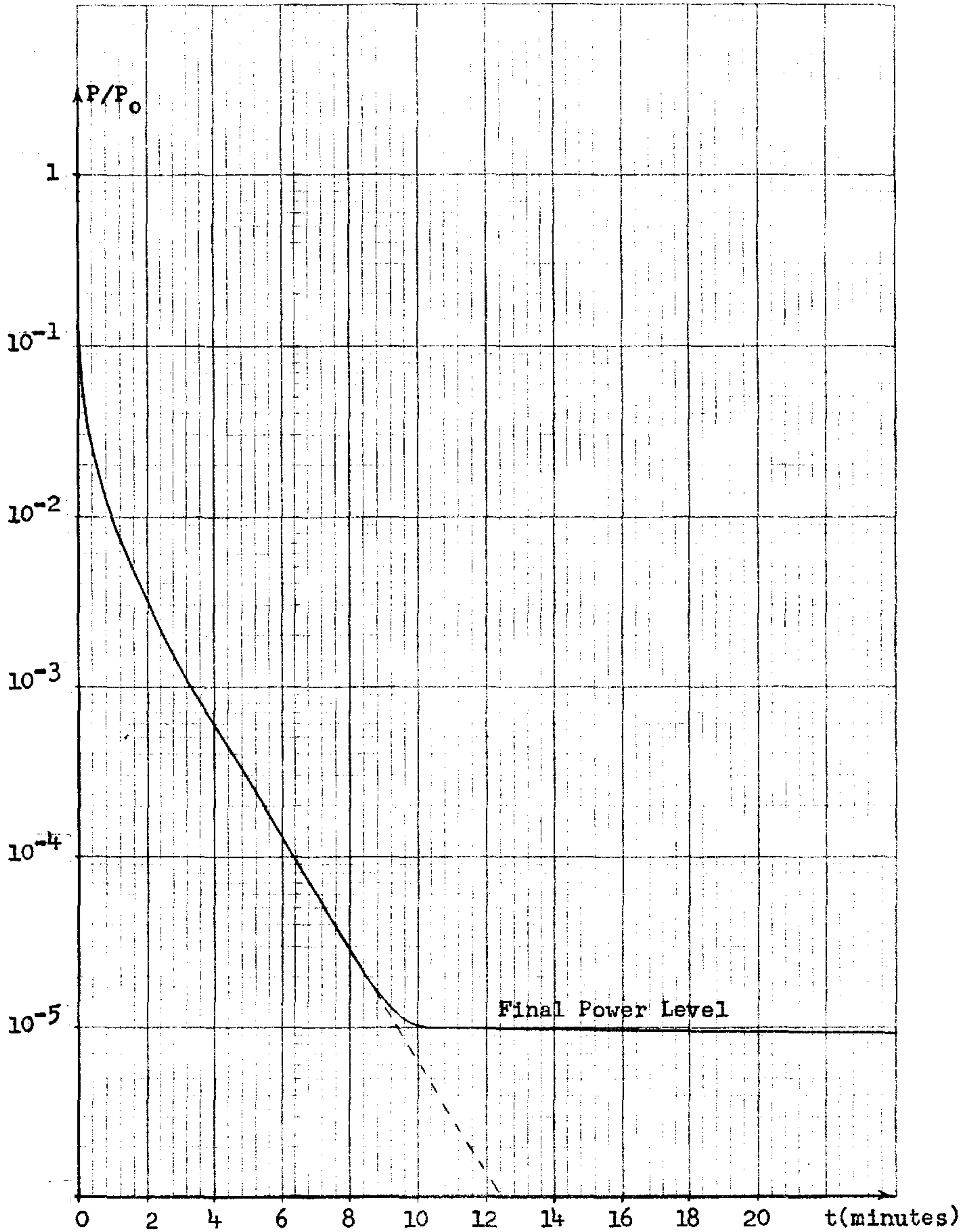
Therefore, the neutron power will decrease, as indicated in Fig. 4 of the previous lesson, until the delayed neutrons have gone and the power will then level out at a value determined by the photoneutron density.

Suppose, for instance that the photoneutron source strength is 30 watts and $\delta k = -30$ mk. Then the final neutron power level would be around $10^{-5}$ of full power ie, the final value of P/Po is $10^{-5}$. The power would therefore decrease as shown in Fig. 1, with the final power level decreasing very slowly as the fission products decay.

The dotted line shows how the power would have continued to decrease without the photoneutrons.

It must be remembered that the graph refers to neutron power as indicated by the neutron measuring instruments. The thermal power does not decrease in this manner because heat is produced by fission product decay. Therefore, the photoneutrons do not affect the thermal power decrease. However, they do not affect the neutron power indications which are obtained on the instruments. The range of neutron power which has to be measured is from 100% down to about $10^{-6}$ full power, a range of 6 decades. Logarithmic instruments are available which will cover this range. Without photoneutrons, the neutron power range would be far wider than any linear instrument could cover, and highly sensitive instruments would be required when starting up each time. These would have to be removed once higher powers are reached to prevent shortening their useful lives.

(2) Effect on reactor is started up for the first time the only sources of neutrons available are the spontaneous fissions, occuring very infrequently. The first time a reactor is started up a source of neutrons must be placed in the core until the multiplication of neutrons is high enough to give a reading on the instruments. Even so, especially sensitive neutron detectors must be used inside the reactor and these must be removed when the power is high enough to give a reading on the regular instruments. During the first start-up, the reactor must be on manual control, ie, regulated by hand instead of automatically by instruments. If photoneutrons were not present, each start-up would involve the same tedious process as the first one. However, if there is a photoneutron source in the reactor, special detectors and neutron sources are not required. Also the power level soon reaches the point where the reactor can be placed on automatic control.



P/P_o
1
10^-1
10^-2
10^-3
10^-4
10^-5
Final Power Level
0
2
4
6
8
10
12
14
16
18
20
t(minutes)


Fig. 1

## ASSIGNMENT

1. (a) How are photoneutrons produced in a reactor?

   (b) Why are they not produced in a reactor in which graphite or light water is used as a moderator?

   (c) Why are the photoneutrons still produced when the reactor is shut down and the prompt and delayed neutrons are no longer produced?

2. How does this photoneutron production affect: -

   (a) the power produced in a reactor?

   (b) the way in which the <u>thermal</u> power decreases after shutdown?

   (c) the way in which the <u>neutron</u> power decreases after shutdown?

3. How is the effect in 2(c) above advantageous as far as the neutron instrumentation is concerned?

4. (a) What are the advantages, during reactor start-up, which result from photoneutron production?

   (b) Why do these advantages not exist when the reactor is started up for the first time?

A. Williams

Nuclear Theory - Course 227

# METHODS OF REACTOR CONTROL

---

We have seen, from the previous lessons, that a reactor is critical, if the chain reaction is just being maintained, the multiplication factor, k, is unity and the reactivity, $\delta k$, is zero. We have also seen that the reactor power will increase if the neutron losses are reduced so that k becomes greater than unity, $\delta k$ becomes positive and the reactor is supercritical. Reactor power is decreased if the neutron losses are increased so that k becomes less than one, $\delta k$ becomes negative and the reactor is subcritical.

In this lesson, then, we will discuss methods by which neutron losses can be increased or decreased to make k negative or positive.

## General Requirements of a Control System

Any method or system used for controlling a reactor must be capable of: -

(1) keeping the value of $k=1$ and $k=0$ during steady power operation and therefore allowing for changes that occur in $\delta k$ due to burning up fuel or other causes

(2) allowing $\delta k$ to become negative to decrease the power and to become positive for increases of power and

(3) decreasing k sufficiently to give $\delta k$ a large negative value for rapid shutdown of the reactor under any circumstances which might prove hazardous to personnel or equipment.

In view of these general requirements the control system must perform two functions which are: -

(1) REGULATION which involves small changes in reactivity either to maintain the power at some predetermined level or to change the power as may be required. The regulating system is the means by which the multiplication factor, k, is adjusted and controlled to shutdown or start up the reactor or to keep the reactor operating at some desired power level.

(2) PROTECTION which provides the automatic rapid shutdown of the reactor should some dangerous condition develop. This rapid shutdown is known as TRIPPING the reactor and the protective system is the means by which this is achieved.

Methods of control

It is not intended, at this stage, to discuss the hardware or components of regulating and protective systems, but rather the principles on which they operate. There are several basic methods of reducing neutron losses to increase k and the methods of control will be discussed under these headings. These are: -

(1) Decreasing neutron absorption in materials other than U-235. This increases the number of neutrons available to cause fission and increases the value of k. The reverse process of increasing neutron absorption in nuclei other than U-235 decreases the value of k.

This method of control is achieved by inserting absorbers of neutrons into the reactor core or by withdrawing absorbers out of the core. Such control methods are illustrated in Figs. 1 and 2.

Fig. 1 shows the conventional method of inserting the absorber into the reactor core in the form of control rods, which are normally made of either boron or cadmium.

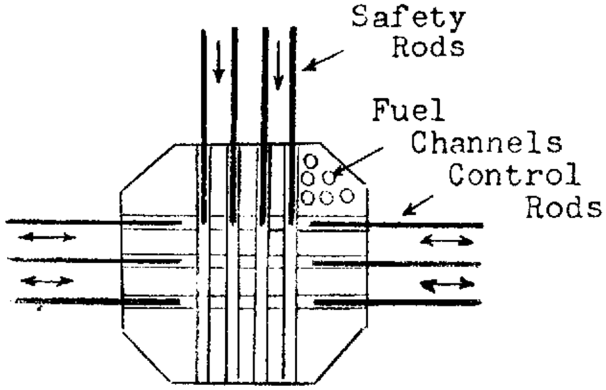



Fig. 1

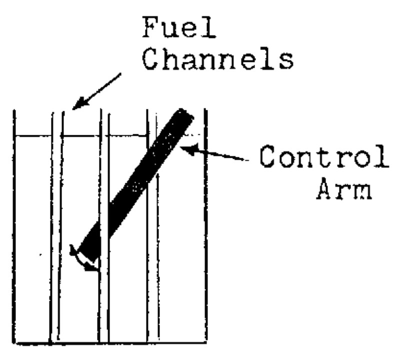



Fig. 2

When the rods move further into the core, the reactivity is reduced. When the rods move further out the reactivity increases. In Fig. 2 the absorber is in the form of a control arm which is pivoted at the top end and moves in and out of the core in much the same way as a railway signal arm.

Rapid reduction of reactivity is achieved by quick insertion of safety or shutdown rods, which normally drop vertically into the core. These are also made of cadmium or boron. The control rods frequently move in and out along horizontal channels and provide the regulation. The safety rods move along vertical channels so that they drop rapidly under gravity and they provide the protection.

(2) The value of k can also be increased by increasing the amount of fuel so that the neutron absorption in fuel nuclei is increased. When the amount of fuel is decreased the value of k is reduced. This is not a widely used method for reactor control but it is worth nothing that it is used in Canadian power reactors when additional reactivity is required to overcome poison build-up.

The method is illustrated in Fig. 3. The fuel rod, known as a BOOSTER ROD, is inserted in the reactor along a horizontal channel as shown or along a vertical channel.

The rod, which is normally made of enriched uranium, is only inserted when additional reactivity is required. When it is not required the rod is withdrawn from the core and the U-235 nuclei are therefore conserved.

Continuous "on-load" refuelling of the Canadian power reactors, is also a method of controlling reactivity by fuel insertion since it compensates for loss in reactivity due to burnup of U-235.

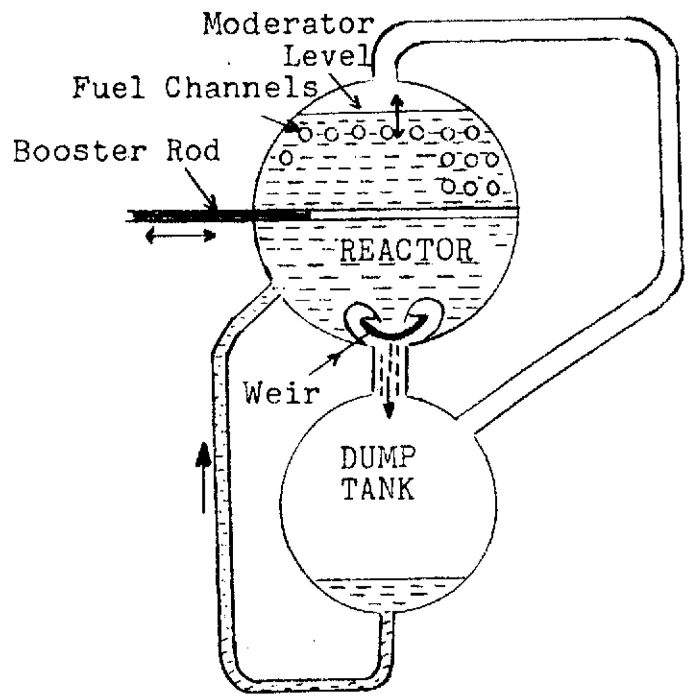



Fig. 3

(3) Decreasing neutron leakage from the reactor will provide more neutrons for fission. Conversely, if neutron leakage is increased the value of k decreases. The neutron leakage can be controlled in one of two ways: -

(a) By addition or removal of reflector. The reflector prevents neutron leakage by reflecting neutrons back into the core. If the reflector thickness is reduced or part of the reflector is removed, neutron absorption is increased and the reactivity decreases.

Fig. 3 illustrates how this method is used in Canadian power reactors. Heavy water moderator continually circulates from the dump tank to the reactor vessel, overflows over the weir and flows back into the dump tank. When it is required to fill the reactor vessel, a pressure differential is established between the dump tank and the top of the reactor vessel. This reduces the flow over the weir and the vessel fills up to the level required.

The outer regions of the moderator acts as a reflector and so the thickness of reflector above the core can be changed by raising or lowering the moderator level. This is used as a means of regulation.

If the protective system causes a reactor trip, the pressure in the dump tank becomes equal to that in the reactor vessel and the reactor vessel is emptied of moderator in a few seconds. This is called "dumping the moderator" and is the means used for protection at NPD.

(b) By increasing or decreasing the core size. The easiest way to do this is by addition or removal of moderator. This method, which is used for coarse control on NRX at AECL, is illustrated in Fig. 4. As the moderator level rises, more and more fuel is covered and the core size increased. This decreases the surface area relative to the volume and a smaller fraction of the neutrons produced escape.

Here again the moderator can be emptied rapidly for a quick shutdown.

Although each method of control has been discussed separately, two or more of them may be used on any one reactor.

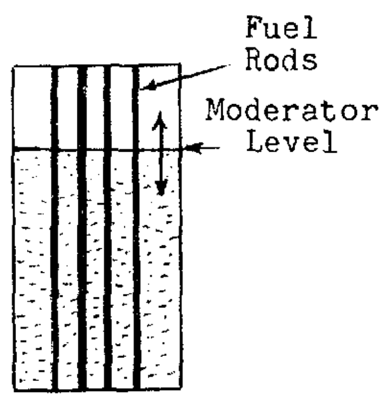



Fig. 4

For instance, on NPD, the reflector thickness variation is used for fine control, continuous refuelling is used for day to day reactivity control and a booster rod is used when additional reactivity is required.

In Douglas Point the above three methods are used and in addition, control rods are used to prevent power fluctuations and cadmium sulphate is used as a neutron absorber in the moderator to enable the reactor to operate with a full tank even when the fuel is new.

## ASSIGNMENT

1. What are the three general requirements of a reactor control system?

2. Explain the difference between the regulating and protection functions of a control system.

3. Explain the three basic methods by which neutron losses can be decreased or neutron utilization increased, and describe, briefly, the control methods based on these.

A. Williams

# Nuclear Theory - Course 227

## STARTING UP A REACTOR AND INCREASING POWER

---

There are, basically, two stages involved in bringing a reactor up to power and these stages are: -

(1) The approach to critical during which the value of k is increased until the reactor becomes critical.

(2) Increasing the power until the operating power level is reached.

### The Approach to Critical

Reactors are brought critical in many different ways: -

(1) By loading fuel into the reactor until a chain reaction is sustained. This is the method that would have to be used in a graphite moderated reactor where the moderator level cannot be changed.

(2) By withdrawing control rods until a chain reaction is sustained. All the fuel would have been loaded into the reactor beforehand and the core size is fixed.

(3) By increasing the size of the core. This is very conveniently done, in a reactor with moderator level control, by simply raising the moderator level to cover more and more fuel channels. Our discussions will be confined to this method.

During the approach to critical, the value of k is less than one and a chain reaction cannot be maintained. A source of neutrons must exist in the reactor if fissions are to occur at all. With fresh fuel in the core, the only source of neutrons are the spontaneous fissions that occur so infrequently. After the reactor has been operating for some time, fission products are produced in the fuel and these emit gamma rays. The gamma rays, in turn, produce photoneutrons in heavy water. It is, therefore, convenient to discuss the approach to critical under two separate headings: -

(1) <u>The First Approach to Critical</u>

This is the approach to critical when no photoneutron sources exist in the reactor. It is termed the "first approach to critical" because it is the approach that would have to be adopted when the reactor is loaded with fuel for the first time, ie, a fresh core. However, the same approach would have to be used if the reactor had been shut down for a long period of time and the fission products had decayed to such an extent that the photo-neutron sources had disappeared. The following problems exist during the first approach to critical: -

(a) It is not known what the moderator level will be when the reactor goes critical, ie, it is not known how many fuel channels have to be covered, although this can be estimated by calculations. This means that the moderator level has to be raised in small steps and held at each level until it can be determined whether or not the reactor is critical.

(b) The instruments which measure neutron power have not been calibrated and cannot be relied upon for measurements or for reactor regulation. The automatic regulating system cannot, therefore, be used and the approach has to be made under manual control, ie, the moderator level is raised by manipulating the controls by hand.

(c) The instruments, normally used to measure neutron power, are not nearly sensitive enough to measure the neutron densities that are produced by spontaneous fissions. Very sensitive neutron detectors, such as fission chambers, are, therefore required which are inserted right into the core.

(d) Even with such sensitive detectors, the initial low neutron densities will not register and so a neutron source has to be inserted into the core to bring the instruments on scale.

With these problems in mind, the first approach to critical is made as follows:

The reactor is provided with a vertical access tube, down which the fission chambers are lowered, as shown in Fig. 1, so that they lie in the lower part of the core. Three such detectors are used and they are connected to counters rather than power instruments. Thus the neutron power is measured from the count rate on three separate detectors.

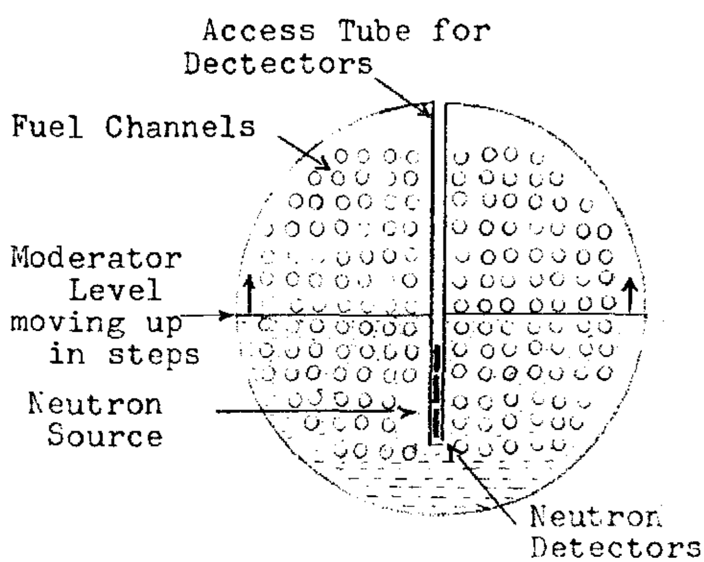



Fig. 1

A neutron source is also introduced into the core, again down some tube. The moderator level is now raised in small steps. Fig. 2 shows how the power or count rate increases for different values of k. When k is 0.5 the count rate levels out rapidly, but as k increases, it takes longer and longer for the count rate to level out. The time to level out must be allowed for each time the moderator is raised.

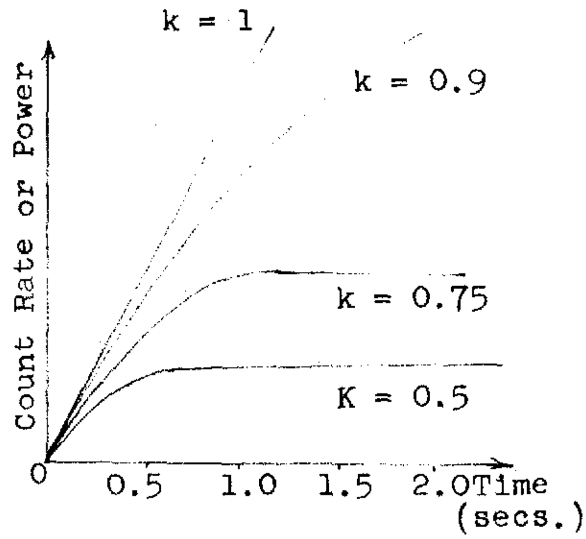



Fig. 2

As k increases, the neutron source increases and the count rate goes up. Thus, eventually the neutron source can be removed. Also the fission chambers have to be moved upwards along the access tube to keep the count rate on scale. When k becomes equal to 1.0 and the reactor is critical, the power and count rate do not level out, but continue to increase. This is how it is known when the reactor is critical.

It is, however, desirable to be able to predict what the moderator level will be when the reactor is critical. This prediction is obtained by noting the count rate at each moderator level and then plotting the reciprocal of the count rate against a function which depends on the moderator height. When the points are joined, they are found to be on a straight line which approaches the value of this function when the reactor is critical.

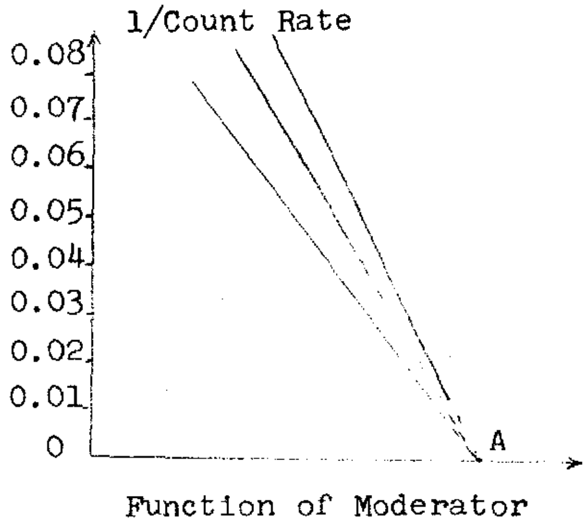



Fig. 3

Thus, in Fig. 3, the three lines, obtained from the three fission chambers approach the point A. A is the value of the function when the reactor is critical. So, from the value at A, we can calculate the critical height.

When the critical height is reached, or just exceeded, the power is allowed to rise very slowly until the normal instruments come on scale and can be calibrated. The automatic regulating system can then be adjusted and put into service.

### (2) Subsequent Approaches to Critical

If the reactor has been operated until fission products have accumulated in the fuel, there will be photoneutron sources in the core when the reactor is shut down. These photoneutron sources are large enough to give a reading on the lower range of the most sensitive neutron power instrument. Therefore, for a startup after a reactor shutdown, we do not need a neutron source or special neutron counters.

However, the neutron power reading is still too low to allow the automatic regulating system to operate satisfactorily, unless, of course, the shutdown was very brief. So the approach to critical may still be made, initially, on manual control, using the normal neutron power instruments.

The moderator level is allowed to rise slowly, but steadily until the neutron power is high enough to allow the automatic regulating system to operate. The power level will then be about 0.01% of full power and when this level is reached, a switch over is made from manual to automatic control.

Subsequent approaches to critical are less hazardous than the first approach, because the critical moderator height is known, the instruments are calibrated and more reliable and poison levels in the fuel are at a much higher level, thus limiting the possible increases in reactivity.

## Raising Reactor Power

Once the reactor is critical, it may be kept at any power level by adjusting the moderator level to keep $k = 1$ or $\delta k = 0$. If the power has to be increased, the moderator level is raised to make k just greater than unity, or $\delta k$ slightly positive. Fig. 4,overleaf, shows such a raise in moderator level at A.

The figure also shows the corresponding changes in k and $\delta k$ and the resulting exponential increase in power. The reactivity change possible is usually limited by design so that the reactor period during the power increase is long. There is also a reactor trip if this period becomes too small, ie, the rate of increase of power is too high.

At B, the required power level has been reached and the moderation level is returned to the point where $k = 1$ and $\delta k = 0$. This is somewhat different from the control method in conventional power plants.

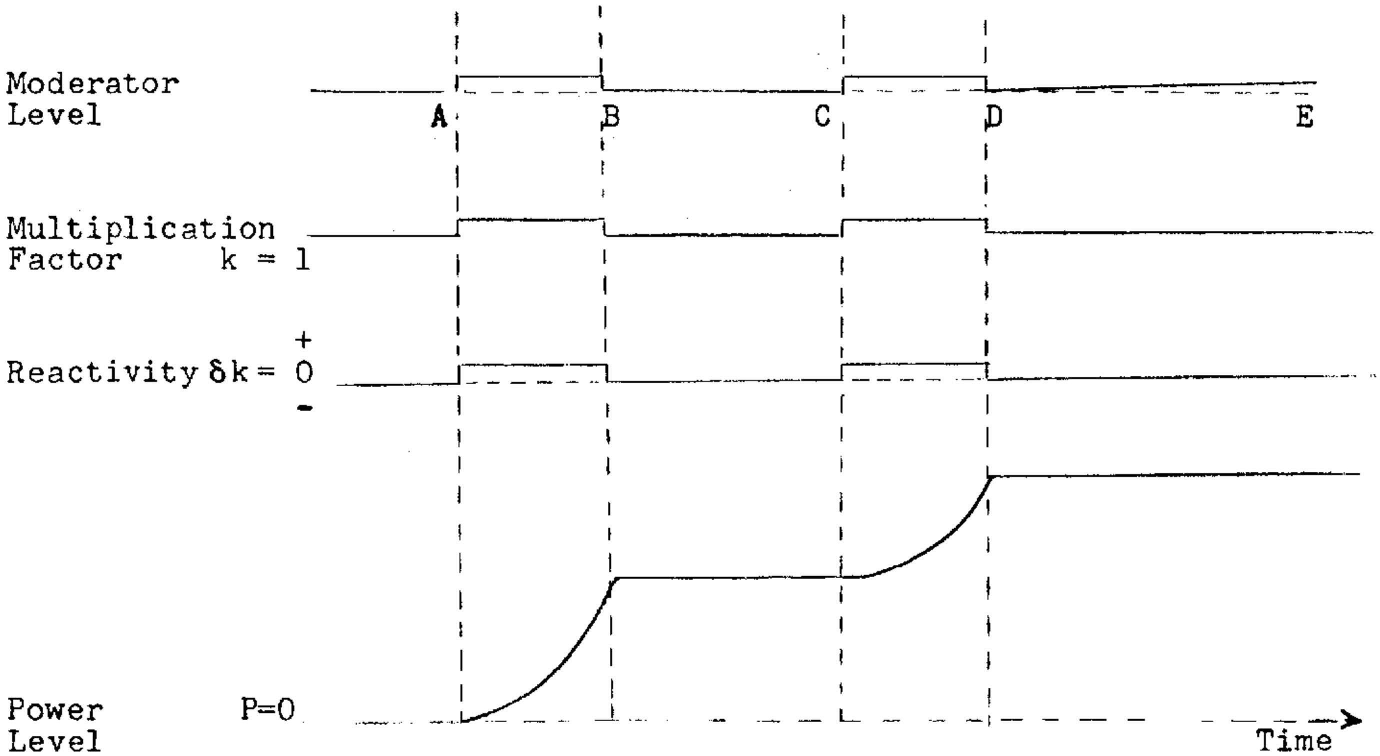



Fig. 4

In a conventional plant, when you want to increase the thermal power, you boost the firing rate by opening the fuel valve wider and increasing the air flow. The valve is then left open at the new setting. However, in nuclear plants, the moderator level is only raised while the power is being raised by increasing neutron density. When the correct power level is reached, the moderator is returned to the old level to prevent further increase in neutron density.

Another increase in power is shown at C which is completed at D. These moderator level adjustments can be made by hand, but they are normally made by the automatic regulating system, as a result of a demand for higher power by the system.

From D to E we see a gradual rise in moderator level that would be necessary to compensate for fuel burn-up or build up of poisons. Note that k is kept at a value of 1.0 and $\delta k = 0$.

## ASSIGNMENT

1. What are the two stages involved in bringing a reactor up to power?

2. (a) Why is the first approach to critical different from any subsequent approach to critical?

   (b) Give three reasons why the first approach to critical is more hazardous than any subsequent approach.

   (c) What two additional items of equipment are required during the first approach to critical which are not required during subsequent approaches?

3. How is it known when the reactor is critical?

4. (a) After the critical moderator level is reached, how is the reactor power raised and then maintained at the required level?

   (b) In what way does this differ from the way in which thermal power is increased in, say, an oil fired boiler?

A. Williams

Nuclear Theory - Course 227

# DECREASING POWER AND SHUTTING DOWN A REACTOR

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As was mentioned in a previous lesson, the control system must be able to lower the reactivity sufficiently to cause the power to decrease and it must also be able to rapidly decrease the reactivity by large amounts to lower the power quickly when the need arises.

The two functions are controlled by two separate systems. The regulating system lowers the reactivity by small amounts for power decreases and the protective system decreases the reactivity by large amounts or trips the reactor. Both of these aspects of power changes will, therefore be considered, separately, in this lesson.

## Lowering Reactor Power

In a reactor, using moderator level control, the value of the multiplication factor, k, is reduced by lowering the moderator level. If the reactor is operating at steady power the lowering of moderator level causes the reactivity, k, to become negative and the power starts to decrease. Lowering or decreasing reactor power is, therefore, accomplished by the reverse of the method of raising reactor power, described in the previous lesson.

Fig. 1 illustrates how this is done. The reactor is operating at steady power up to the point A. At A it is decided to reduce power and the moderator is lowered to make k slightly less than unity and k slightly negative. The reactor power decreases exponentially until the required power level is reached at B. The moderator level is then raised back to its original level so that the reactor is again just critical. The reactor will then continue to operate at the lower power level.

At C is shown the beginning of a reactor shutdown controlled by the regulating system. The moderator level drops slowly and the reactor vessel takes perhaps seven or eight minutes to empty. The reactivity decreases slowly as may be seen in Fig. 2. The reduction in neutron power is not rapid compared to what it would have been if the total loss of reactivity has occurred in a few seconds.

It must be remembered, again, that we are talking about neutron power and not thermal power. There is a considerable delay in thermal power reduction, behind the neutron power reduction, because of the heat released by the fission products. This will be discussed later in the lesson.

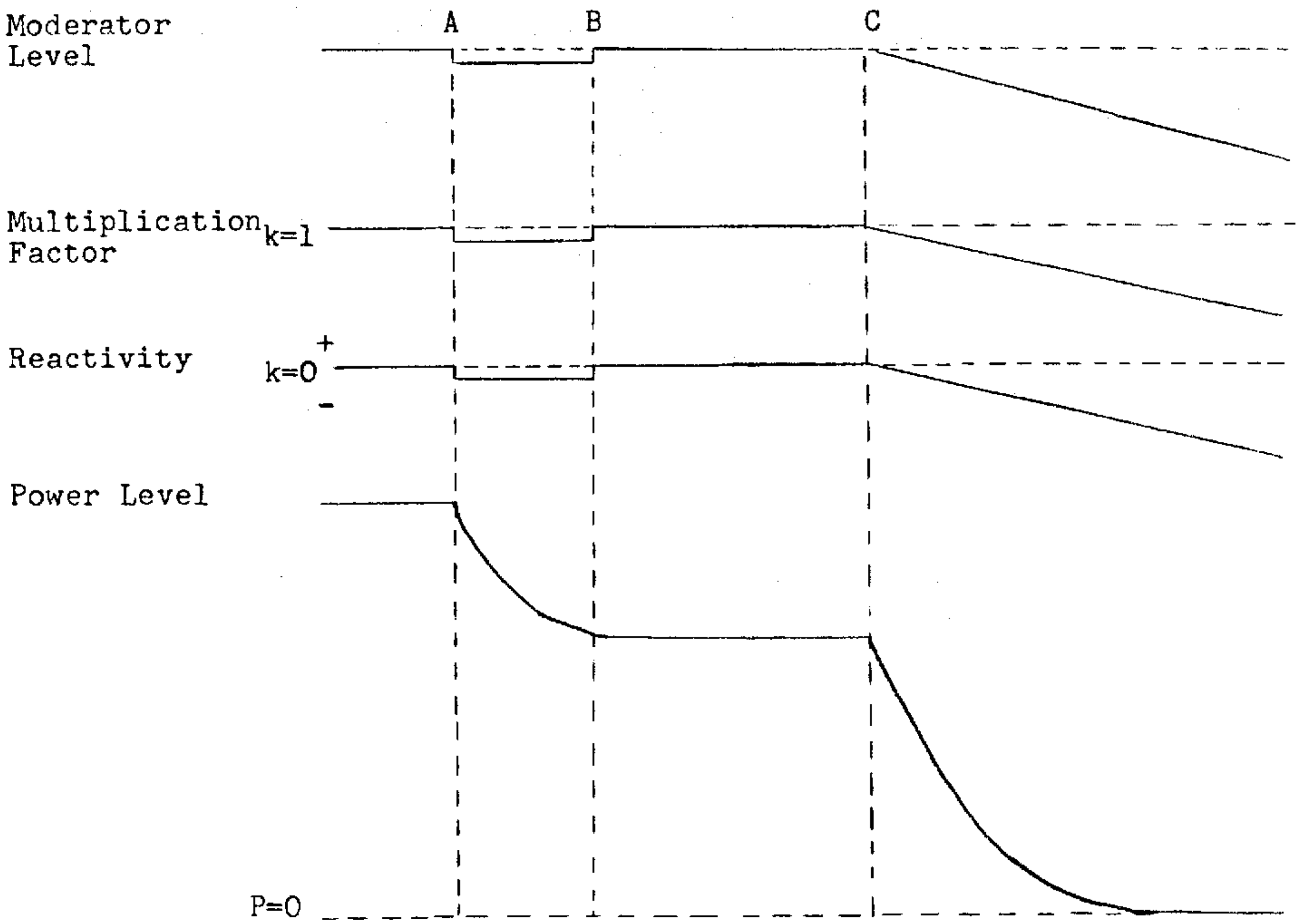



Fig. 1

Reactor Shutdown Due to a Trip

When a reactor trip occurs there is an immediate equalization of pressure between the dump tank and the top of the reactor vessel. The moderator level falls rapidly and the whole of the reactor vessel is emptied in, perhaps, 10 secs. This is called DUMPING the moderator. The loss of reactivity during a trip is shown in Fig. 2 (a) whereas the loss of reactivity during a shutdown controlled by the regulating system is shown in. Fig. 2(b) for comparison.

As may be seen, during a controlled shutdown it takes over two minutes for the reactivity to decrease to -70 mk whereas it takes only 5 seconds during a moderator dump following a trip.

During a trip the reduction in neutron power will be as shown in Fig. 3.

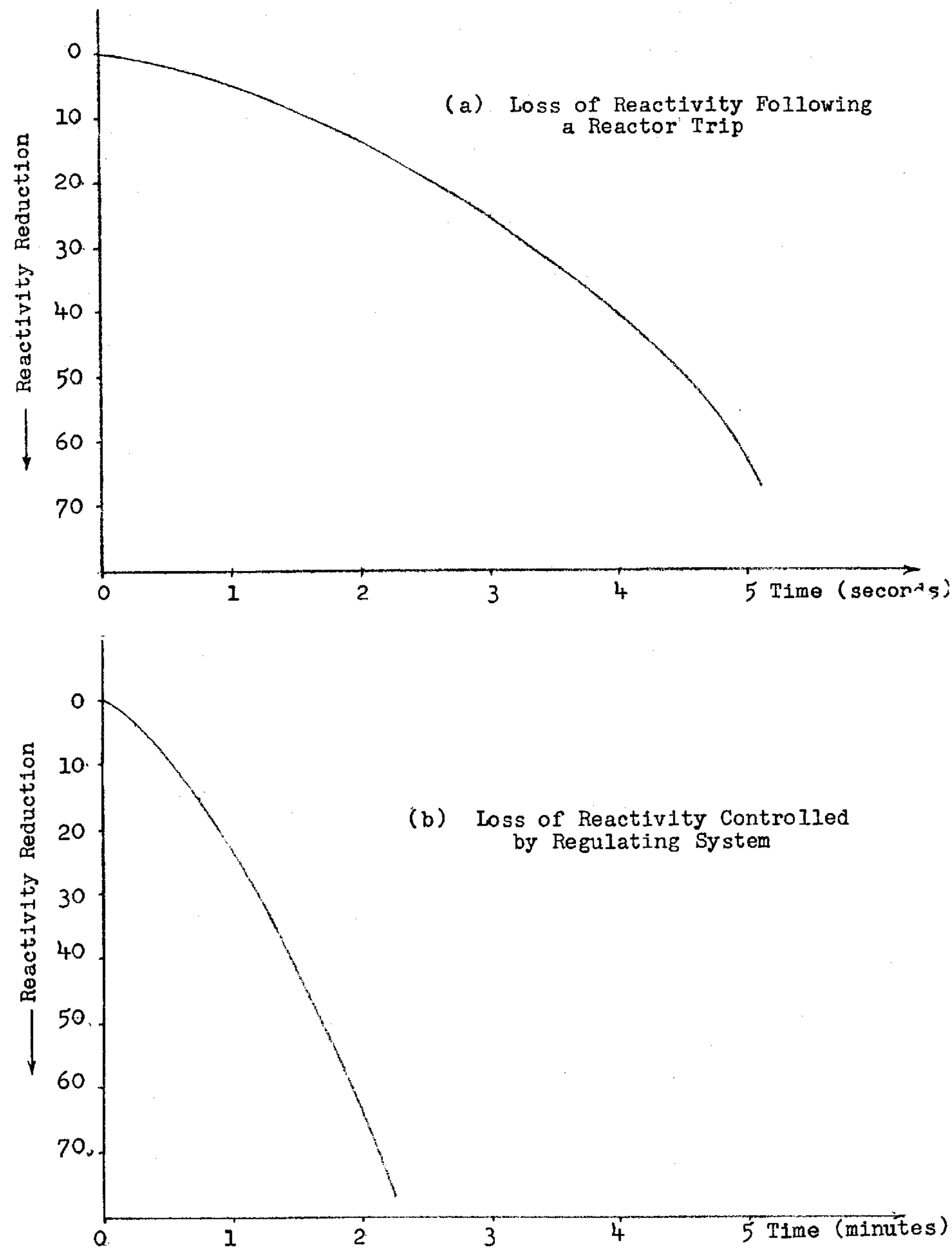
Reduction of Total or Thermal Power

As has already been mentioned, the total or thermal power in a reactor includes both the neutron or fission power and the power released by the fission products in the form of heat. The neutron power is produced at fission whereas the fission product power is released as a result of the decay of the fission products.

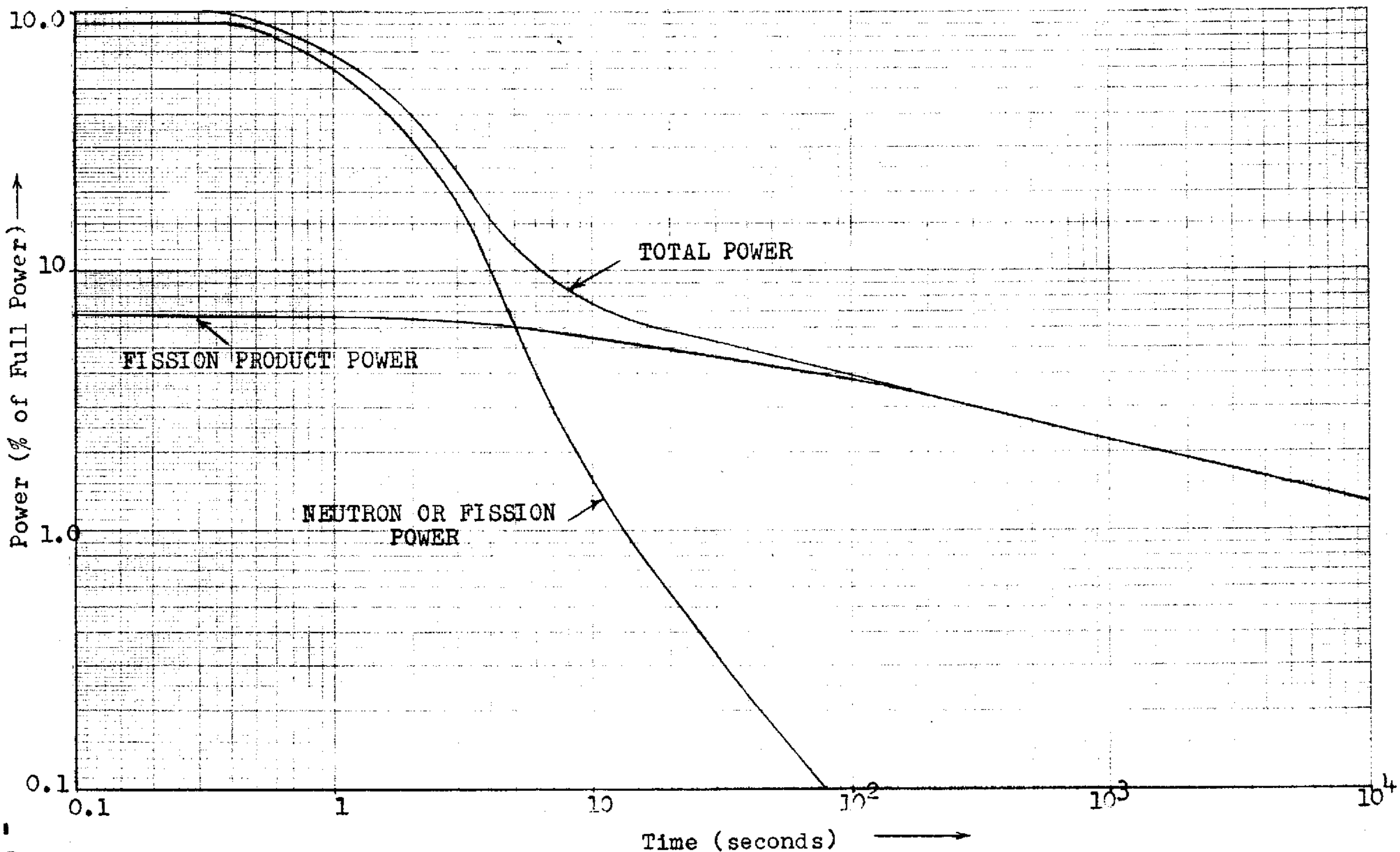
From the point of view of reactor control and reactor safety, the neutron power is the important factor. However, since the thermal power decreases much more slowly, decrease in the total thermal power is of great importance as far as the heat transport system is concerned.

Fig. 3 shows how the neutron power, the fission product power and the total power, (ie, the sum of the two), decrease after a reactor trip. Logarithmic scales have been used for the power and time scales to allow a greater range to be covered in both cases.

Note that the total power decreases to 6% of full power in just over 10 seconds but decreases slowly after this. This explains why there may be a flywheel on the heat transport pumps. If the pump trips, the reactor trips and the flywheel gives the pump a rundown time to allow for this rapid decrease in reactor power ie, the pump still maintains some circulation during the initial power decrease. Thermosyphon flow can subsequently be used to remove heat from the fuel. The graph also shows why some cooling of the fuel must be maintained all the time and, if the main heat transport system has to be isolated, a standby cooling system must be provided.


(a) Loss of Reactivity Following a Reactor Trip
Reactivity Reduction
0
10
20
30
40
50
60
70
0
1
2
3
4
5 Time (seconds)
(b) Loss of Reactivity Controlled by Regulating System
Reactivity Reduction
0
10
20
30
40
50
60
70
0
1
2
3
4
5 Time (minutes)

<u>Fig. 2</u>



TOTAL POWER
FISSION PRODUCT POWER
NEUTRON OR FISSION POWER
Power (% of Full Power)
Time (seconds)
10.0
10
1.0
0.1
0.1
1
10
$10^2$
$10^3$
$10^4$


227.00-12

## ASSIGNMENT

1. How is reactor power reduced, by the regulating system, from one power level to another?

2. What takes place during a reactor shutdown controlled by the regulating system?

3. How does a reactor trip differ from a controlled reactor shutdown?

4. (a) Why does thermal power decrease less rapidly than neutron power?

   (b) Name two important consequences of the way the total or thermal power decreases.

A. Williams

# Nuclear Theory - Course 227

## XENON POISON AND ITS EFFECTS ON REACTOR OPERATION

During the discussions on reactor control and reactivity changes, it was mentioned that the methods used for control would have to compensate for the gradual decrease in reactivity that occurs in a reactor. Such a continual decrease in reactivity may be due to a number of factors, the most important of which are: -

(1) using up or burning the U-235

(2) build up of fission products in the fuel.

The gradual decrease in reactivity due to these effects is compensated for by on-power refuelling, ie, regular replacement of used fuel with new fuel to maintain the amount of U-235 in the reactor and limit the amount of fission products. Some fission products, on the other hand, are strong absorbers of neutrons and require special compensation. Materials which absorb neutrons and therefore leave fewer neutrons to cause fissions are called POISONS. These fission products are therefore sometimes called fission product poisons.

In this lesson, we will discuss how these fission product poisons build-up in the reactor and how the resulting decrease in reactivity is counteracted.

### The Build-up of Fission Product Poisons

Many fission products are strong absorbers of neutrons but the most important, by far, is Xenon-135. It is so much more important that the study and considerations of Xenon Poison are normally dealt with separately from other fission products. We will first consider the growth of Xe-135 in the reactor.

Xenon-135 is produced, in the fuel, in two ways: -

(1) directly as a fission product

(2) indirectly as a daughter of Iodine-135 which is, in turn, produced as a fission product or from the decay of the fission product Tellurium-135. The decay chain would be: -

$$Te^{135} \xrightarrow{\beta^-} I^{135} \xrightarrow{\beta^-} Xe^{135}$$

The half life of Te-135 is much less than either I-135 or Xe-135 and is therefore ignored in the remainder of the lesson when time factors are considered.

About 95% of Xe-135 is formed as a result of Iodine decay and only 5% of it is formed as a fission product.

If Xe-135 was continually produced, in the fuel, in this way and none of it was removed, its concentration would continually increase. Eventually, its absorption of neutrons would prevent the chain reaction from being maintained, however large the reactor might be. Fortunately, Xe-135 is also removed in two ways: -

(1) by decaying to Caesium-135, (which is not regarded as a poison), according to the reaction: -

$$Xe^{135} \xrightarrow{\beta^-} Cs^{135}$$

(2) by capturing thermal neutrons and changing to Xe-136 (which is again less of a problem).

$$Xe^{135} + n^1 = Xe^{136}$$

Here, then we have Xe-135 being produced in two ways and being removed in two ways. To understand what happens, let us compare the Xenon build-up to a can being filled with water. Fig. 1 shows the arrangement for filling the can. The build-up of Xe-135 is represented by the rise in the level of the water in Can A. Can A is filled directly by the small line on the left, which represents the direct formation of Xe-135 as a fission product.

Can A also receives water from Can B. The line passing water from Can B to Can A represents the formation of Xe-135 from I-135. So the rise in the water level in Can B represents the increase in I-135 in the reactor while the line filling Can B represents the production of I-135 from the decay of tellurium.

Water leaks out of Can A through two lines which represent the loss of Xe-135 by decay and by neutron capture.

Suppose that, to start with, both cans are empty. This condition represents the fuel with no iodine or Xenon in it. The reactor is now started up and is operating at steady power. Right away Te-135 is formed, which almost immediately starts decaying into I-135 and Can B starts to fill up.

Water also starts flowing directly into Can A through the small line since a little Xe-135 is being produced as a fission product. After a slight delay, water starts running from Can B into Can A, but the water level in B rises because water is running in faster than it runs out, ie, I-135 is being formed faster than it decays to Xe-135. As the level in Can B rises, the water runs out faster because of the greater head of water, until, eventually, the water runs out of B as fast as it runs into it. From this point on, the level in B remains steady or the I-135 has reached an equilibrium level. It decays at the same rate, as it is formed, and its concentration in the fuel does not therefore increase or decrease.

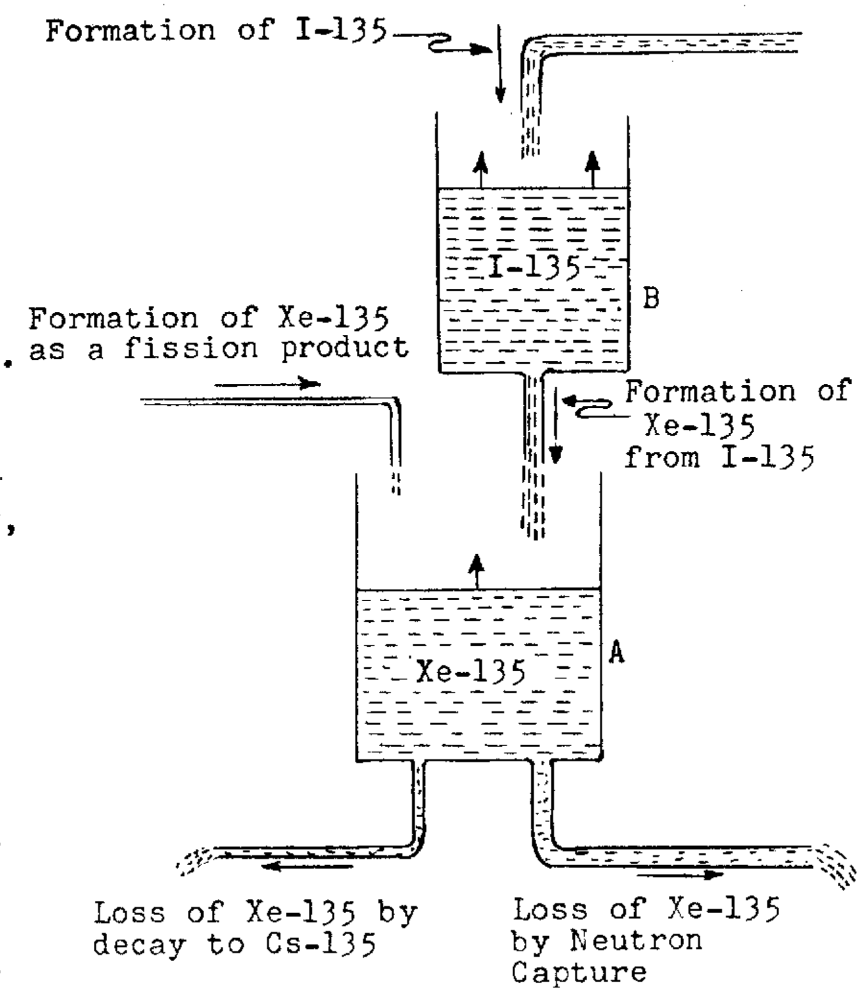



Fig. 1

Meanwhile, Can A is being filled from Can B and by the small direct line. Soon water starts leaking out of Can A through the Xe decay line and the neutron capture line. However, water flows into Can A faster than it leaks out and the water level in A rises, ie, the Xe-135 concentration is increasing. But the higher the level in A, the faster the flow out of A and the slower A fills up. Eventually, the level is such that the water flows out as fast as it flows in and the level in A then remains steady. What we are saying, then, is that the Xe-135 concentration in the fuel increases until the removal of Xenon, by decay and neutron capture increases to the point where it exactly balances the production of Xenon, as a fission product and by decay of I-135. From then on, the Xe-135 concentration remains constant.

The build up of Xe-135 is shown graphically in Fig. 2.

It must be remembered that the final equilibrium value of Xe-135 depends upon the neutron flux and hence on the power at which a reactor is operating. The rate of buildup to equilibrium is almost independent of flux over the range of interest. It takes several days to reach a true equilibrium due to the exponential shape of the curve but buildup during the early part of the curve is fairly rapid. In a typical power reactor the Xe-135 reaches 50% of equilibrium in 10 hrs and 90% of equilibrium in 24 hrs of steady power operation.

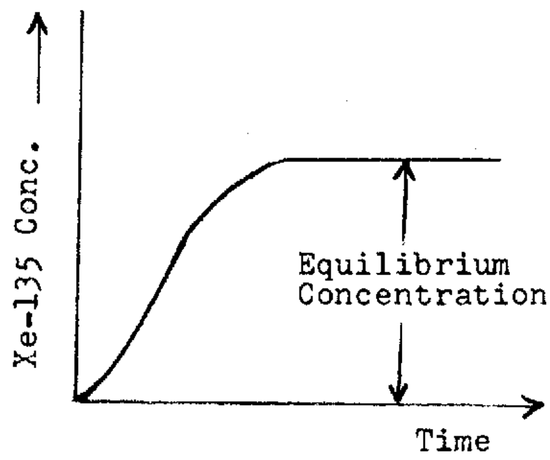



Fig. 2

## Consequences of Xenon Poison Buildup

Xenon 135, and other fission products, are poisons which absorb neutrons and therefore, reduce the multiplication factor k, and reactivity, $\delta k$. As Xe-135 increases from zero to equilibrium concentration, it reduces the reactivity by 20, 25 or even 30 mk. That is, it introduces this amount of negative reactivity into the reactor, just as surely as if boron or cadmium rods has been inserted into the reactor. There are two important consequences of this loss of reactivity: -

(1) The reactor must be big enough so that, as the Xe-135 increases and the reactivity decreases, the regulating system must be able to increase the reactivity by the same amount. That is, enough reactivity must be built into the reactor to balance the loss in reactivity due to equilibrium Xenon poison. This loss in reactivity, due to equilibrium Xenon, is known as the EQUILIBRIUM XENON REACTIVITY LOAD.

With a reactor which uses moderator level for coarse reactivity control, this means that when the Xenon concentration is zero, the critical moderator level will be much lower than it will be at equilibrium Xenon concentration, so that the moderator can rise to compensate for the Xenon load buildup.

(2) With the moderator level low, before the Xenon buildup starts, some fuel may not be covered by the moderator and reactor operating power may be limited to well below full power. In fact the reactor may not be able to operate at full power for 60 or 70 hours after start-up, ie, until the equilibrium Xenon concentration is reached at various power levels which keeps the moderator level rising.

This is obviously a disadvantage and has to be overcome if possible. To overcome this disadvantage, what is done is to introduce a neutron absorber, like cadmium sulphate, into the moderator so that the reactor has to operate full of moderator to overcome the loss in reactivity due to the cadmium sulphate. As the Xenon load increases, the cadmium sulphate is removed and the moderator remains at a high level. There is, therefore, no loss of production of power.

<u>Xenon Build-up After Reactor Shutdown</u>

Let us now suppose that the reactor trips and is shut down. What happens to the Xenon concentration? Going back to Fig. 1, we see that the formation of I-135 stops and so does the direct formation of Xe-135 as a fission product. Thus, Can B is no longer being filled and nor is Can A directly. However, there is water already in Can B (ie, there is I-135 already formed in the fuel), and Can B continues to empty into Can A.

What of the leakage from Can A? Since the reactor is shut down, there are no neutrons to be captured in Xe-135 and there is no loss of Xe-135 by neutron capture. Xe-135 continues to decay to Cs-135 and so there is loss from Can A through the smaller pipe only. The situation, as it now exists, is shown in Fig. 3.

Initially, the flow into Can A is faster than the leakage flow out, and the level starts to rise above the previous equilibrium, ie, Xe-135 is being produced faster than it decays and, immediately after shutdown, the Xe-135 concentration starts to increase above the equilibrium concentration during reactor operation.

Eventually, all the I-135 decays and the Xenon concentration reaches a maximum concentration, shown by line CD in Fig. 3. Loss of Xenon continues to occur by decay only and the Xenon concentration falls, exponentially, from the maximum value, ie, the level in A falls from its value at CD.

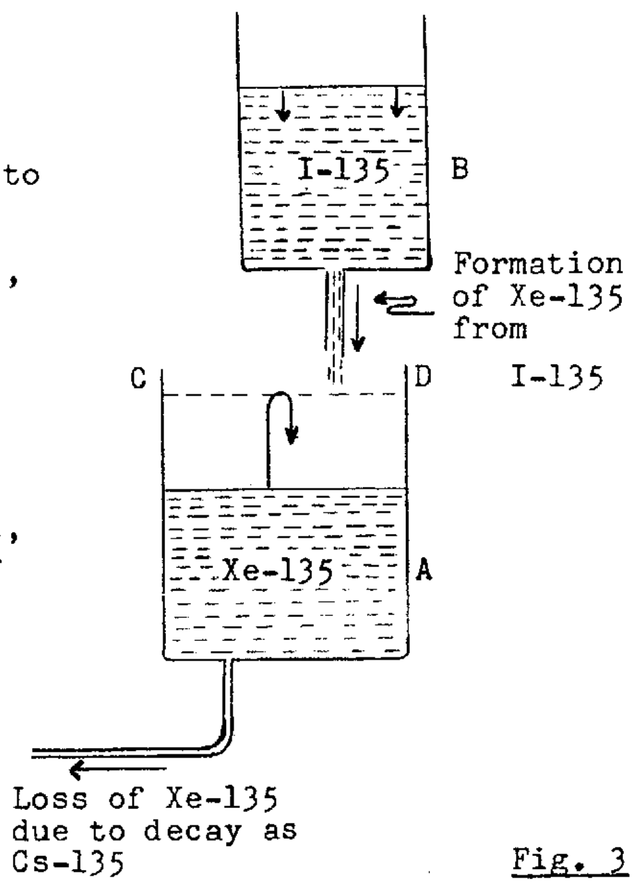



Fig. 3

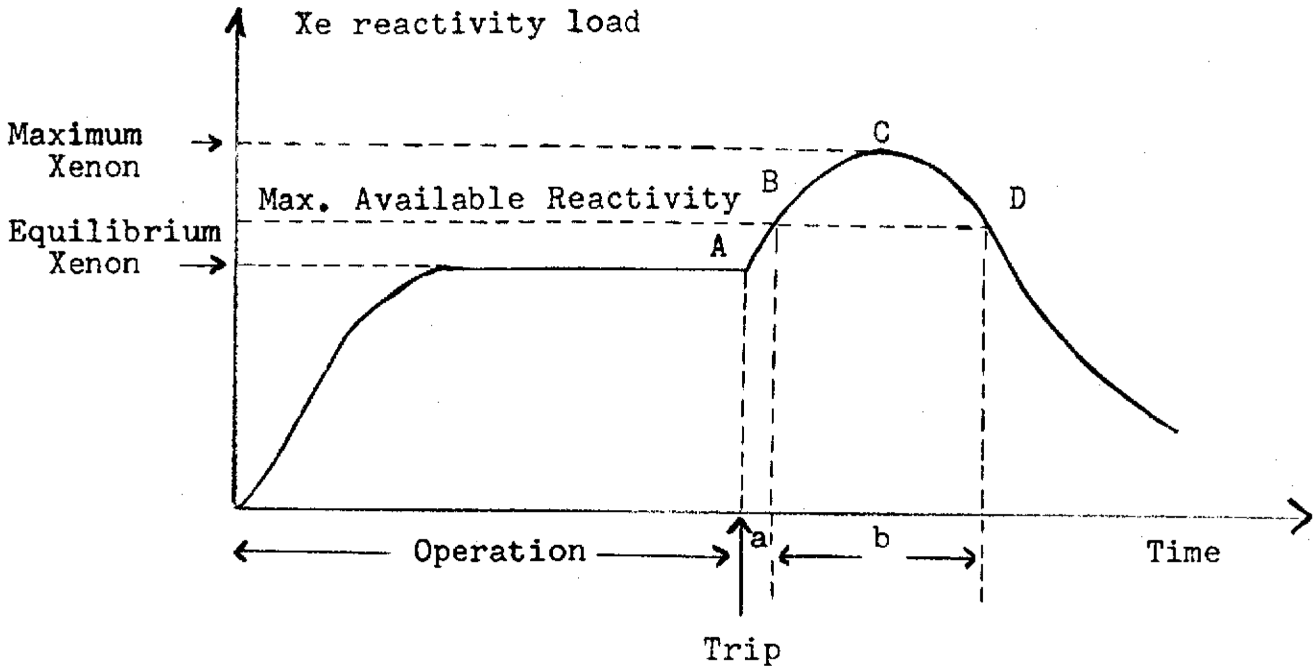



Fig. 4

Fig. 4 shows the increase of Xe-135 during reactor operation, to its equilibrium level and its increase above this equilibrium level following a reactor trip at A. The maximum Xenon concentration occurs at C. Such a build-up of Xenon, following a reactor trip, is known as the XENON TRANSIENT. The maximum or peak Xenon load depends on the power level, prior to the trip, but it may be double the equilibrium Xenon load, ie, 45 to 50 mk or higher.

Although there is enough reactivity to overcome the equilibrium load, it is not feasible to provide enough additional reactivity, in the reactor to overcome the peak load of the Xenon transient. However, sufficient reactivity may be available to overcome the Xenon load up to point B. This makes it possible to start up the reactor again between A and B, but if the reactor power is not restored, before the Xenon reaches B, then the reactor can not be started up until the Xenon comes back down to D.

The time period, 'a', during which there is enough reactivity available to overcome the Xenon poison and to make reactor start-up possible, is known as the POISON OVERRIDE TIME or the TIME TO POISON.

The time period, 'b', during which there is not enough reactivity to overcome the Xenon poison and during which the reactor cannot be started up, is known as the POISON OUT TIME or the POISON SHUTDOWN TIME.

The time periods, shown in Fig. 4, are all out of proportion. In practice, the poison override time is only 15 or 20 minutes, the time from A to the peak Xenon is about 11 hours and the poison shutdown time 24 to 30 hours.

The poison override time can be extended, up to 30 or 40 minutes by providing additional reactivity, when it is required, by inserting additional fuel, into the reactor, in the form of an enriched booster.

## ASSIGNMENT

1. Why are fission products, such as Xenon-135, referred to as poisons?

2. (a) In what two ways is Xe-135 produced in the fuel?

   (b) In what two ways is Xe-135 removed?

3. What is meant by equilibrium Xenon concentration and when is this concentration in the fuel reached?

4. State two consequences of the build up of Xenon poison during reactor operation.

5. Explain the Xenon transient that occurs after a reactor trip.

6. Explain the terms "Poison override time" and "Poison shutdown time".

7. How can the poison override time be extended without increasing the core size?

A. Williams

# Nuclear Theory - Course 227

## EXAMPLES OF PRACTICAL REACTOR BEHAVIOUR

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In the previous lessons, changes in reactor power and build-up of Xenon poison were discussed as two separate considerations. However, Xenon concentration changes when a change occurs in reactor power and so the one consideration cannot be separated from the other. The way the reactor behaves in practice will be determined by both the load or power changes required and by the change in Xenon concentration that results from the power change.

Another factor which is involved, and which may partially or completely mask the effect of the change in Xenon, for very small power changes, is the loss of reactivity due to fuel burnup. The moderator level rises to compensate for the decrease in reactivity due to burnup of U-235 and plutonium. These practical examples of reactor behaviour are intended to illustrate how the response of the reactor is determined by all the factors involved. There are two categories of examples that will be given: -

(1) Changes in moderator level as full power is approached in steps while Xenon build-up takes place.

(2) Moderator level response to load changes.

### Effect of Xenon Delay on the Approach to Full Power

As was seen in the previous lesson, when the reactor is started up following a shutdown, the Xenon concentration is zero. This will result in a much lower critical moderator level than when equilibrium Xenon concentration has been established. With such a low critical moderator level, the thermal neutron flux distribution is such that the reactor operating power is limited to a value well below full power. This delay in being able to operate at full power is known as the XENON DELAY. It represents a loss in energy production which is due entirely to a limitation on operating power because of low Xenon concentration.

It is usually possible to bring the reactor power up to say, 50% of full power. The power is held at this value until the Xenon buildup causes the moderator level to rise and permit the power to be raised further. Therefore, the power is raised in small steps as the moderator level continues to rise as the Xenon concentration continues to increase. The increases in power and the changes in moderator level that occur are shown in Fig. 1.

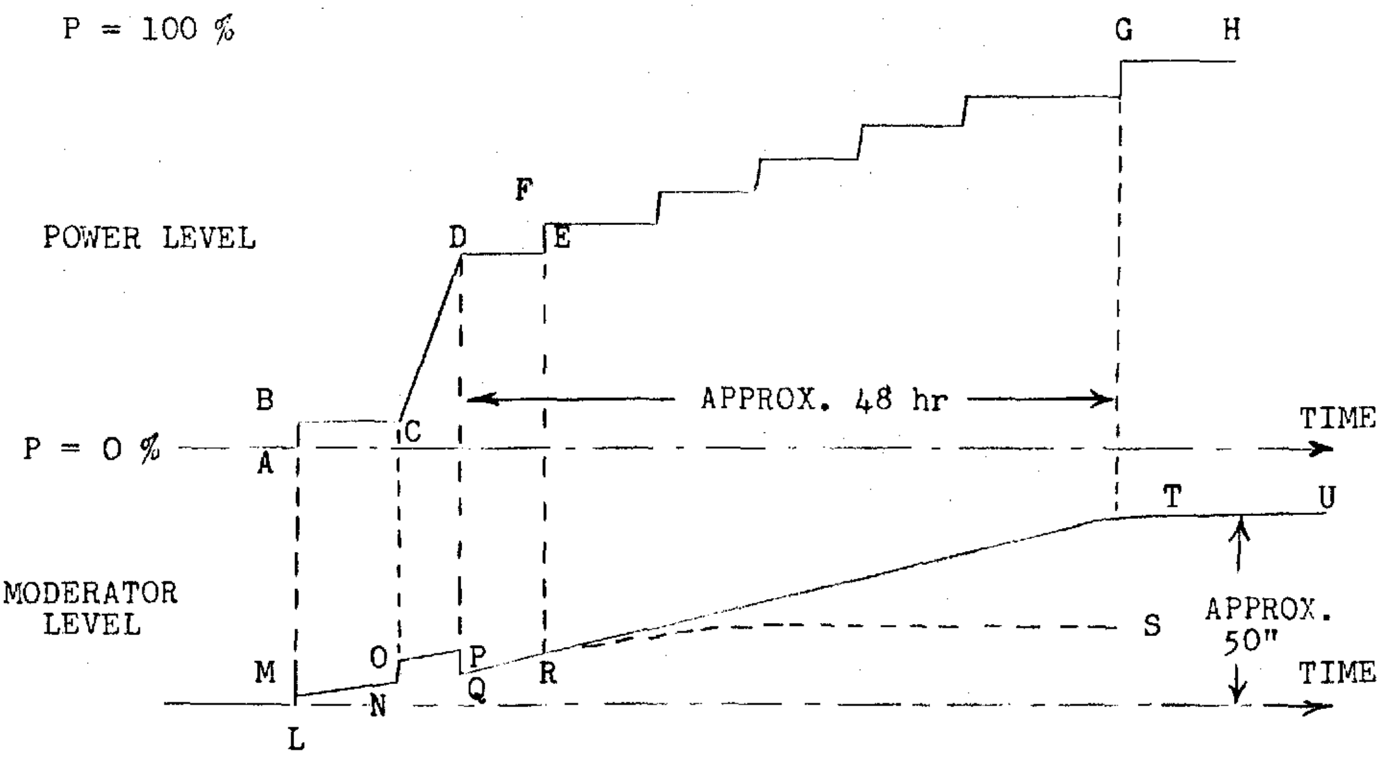


Fig. 1

At L the moderator level rises to M long enough for the power increase A B to take place. The level then returns almost to the previous critical level (plus power coefficient) but rises to N due to the loss of reactivity associated with the heat transport system negative temperature coefficient.

The load is increased at C and the reactor power rises to about 50% of full power. The moderator level rises and remains above critical along O P during the power increase. When the power is levelled off at D the moderator level drops back to the critical level at Q. If the power remained at D the moderator would follow the curve QRS as the Xenon builds up to equilibrium.

In order to reach full power as quickly as possible the power is raised in steps such as E F whenever the moderator level and hence the permissible power is high enough to allow it. This keeps the Xenon buildup continuing until it reaches its equilibrium T U slightly after the power reaches 100% at G H. The fine structure on the moderator curve associated with each individual power step has been ignored for simplicity.

## Moderator Level Response to Load Changes

### (1) Minor Load Changes

A minor change in load may occur due to some change in setting in the regulating system which changes the steam pressure by a few psi.

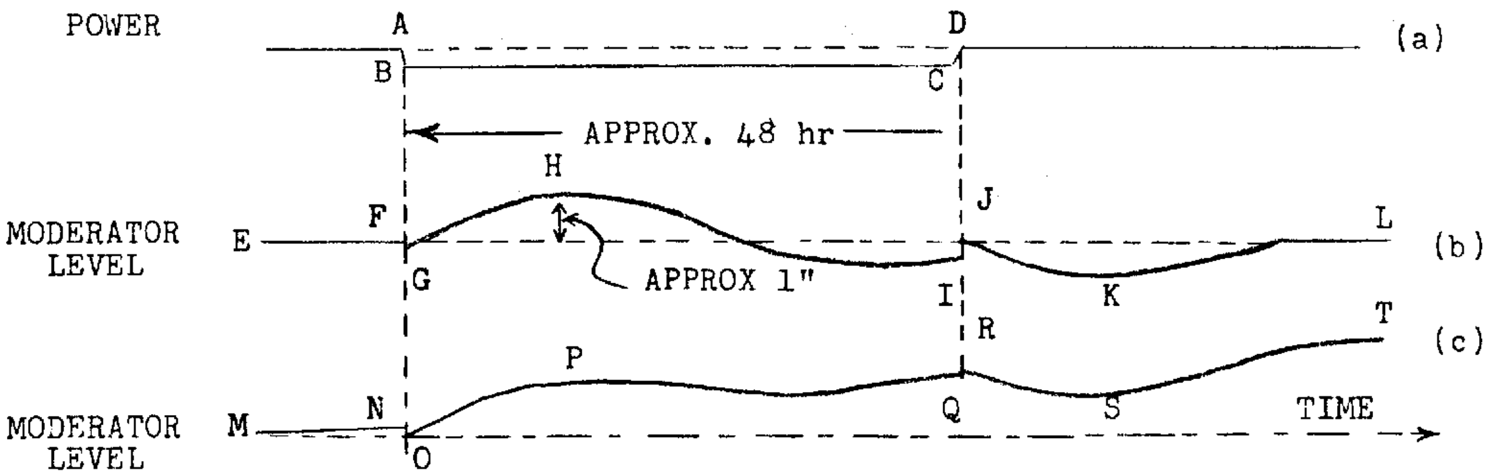


Fig. 2

Fig. 2(a) shows reactor changes following such a minor load change and restoration of full load. Fig. 2(b) shows the response of the moderator level when fuel burnup is ignored and Fig. 2(c) shows the moderator level response when the effect of fuel burnup is not to be ignored.

The reactor power remains steady up to A, when a small decrease in power, AB, occurs due to a load change. Ignoring fuel burnup the level remains constant along E F when a temporary drop occurs to G to provide the necessary negative reactivity. When fuel burnup is taken into account the level rises from M to N and then drops to O. In both cases a rise will then occur due to the Xenon transient. The Xenon will eventually come to an equilibrium which is lower than the original amount and the moderator level is steady at I. In curve (c) the fuel burnup effect may be large enough so that there is no decrease in level from P to Q.

When the power is restored from C to D there is a transient rise in moderator level to J (or R) to supply the necessary positive reactivity during the power change. The level then drops back to critical and a downward Xenon transient starts. This transient is caused by an increase in Xenon burnout which temporarily makes Xenon destruction (burnout and decay) greater than Xenon production. The production rate builds up, however, and the level eventually returns to L or T which corresponds to the original critical level F (or N plus fuel burnup effect in curve (c).

(2) <u>Larger Load Changes</u>

This is the load change that takes place when, for instance, a turbine emergency stop valve is tested by closing it. The reactor power change that occurs may be as much as 15% and the resulting Xenon change is, therefore, much harsher, than in case (a) above.

Fig. 3 shows how the moderator level would respond to such a reduction in power followed by a restoration of full power.

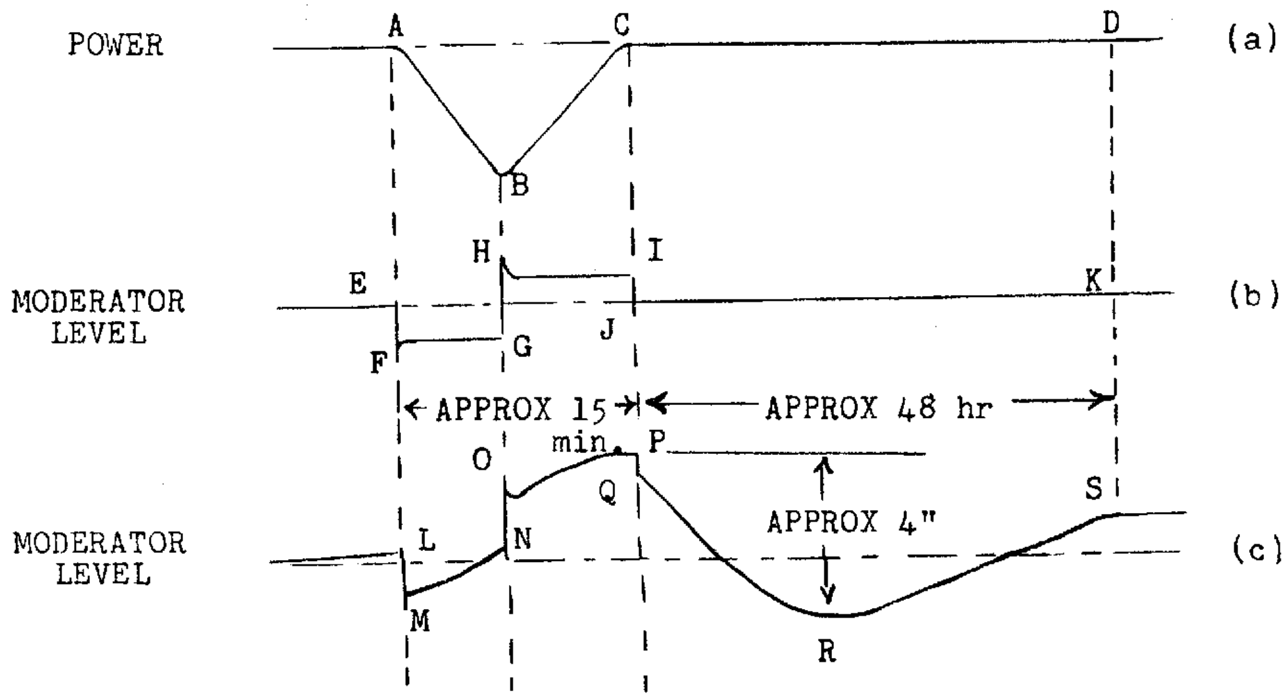


Fig. 3

The load decrease starts at A and continues to B where one stop valve would be fully shut. The load then returns as the valve is opened from B to C. Fig. 3(b) shows the moderator level if Xenon and fuel burnup are ignored. The level decreases from E to F with some overshoot but remains subcritical until G. Positive reactivity is then required and the level rises to H (again with some overshoot) and remains high enough to keep the reactor supercritical while the power is increasing. The level returns from I to J, the original critical level when the power stops rising.

In Fig. 3(c) the effects of fuel burnup and Xenon have been included to show what actually happens. The shape of the curve from L to Q is modified from (b) mainly by Xenon buildup which is most pronounced when the power is lowest (at B in (a)). The transient QRS is due to Xenon burnout and is similar to the transient in Fig. 2. The final level S is higher than L due to fuel burnup.

## ASSIGNMENT

1. (a) What is meant by "Xenon Delay"?

   (b) What is the approximate value of the maxium initial reactor power after startup because of Xenon delay?

   (c) Why is it possible to raise power later?

2. In what manner is full power subsequently achieved and what determines the way in which full power is reached?

3. Describe, with the aid of a diagram, the changes in moderator that take place, following a minor load reduction if:

   (a) the effect of fuel burnup is zero

   (b) the effect of fuel burnup is substantial.

4. If the load reduction is substantially larger than in question 3, and full power is not restored, what is likely to be the result, particularly if the initial moderator level is high?

A. Williams