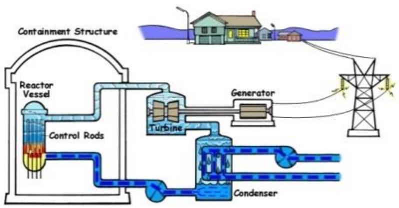
Thermal Power Stations II



Faculty of Engineering
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Lecture (6)
on
Nuclear Reactors Design and Operation Considerations

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Multiplication Factor

- A chain reaction is possible if in each fission reaction, at least one neutron is produced. The ratio of the number of neutrons in any one generation to the number of neutrons in the preceding generation is called multiplication factor k.
- If **k** is less than 1, the number of fissions rapidly decreases and the process dies down.
- If **k** is greater than 1, the number of neutrons increases with every fission and the result could be a nuclear explosion as occurs in an atomic bomb.

Multiplication Factor

- Therefore, it is necessary to keep the value of k as unity.
- To overcome the neutron loss due to leakage, capture in canning, control rods etc., k should be kept slightly greater than 1 : around 1.04.
- Maintaining k at the exact value is the most difficult problem in reactor control.

Critical Size

- If the core assembly of a reactor is infinitely large, there would be no loss of neutrons due to leakage. The value of the multiplication factor, for a reactor having a core of infinite dimensions, is referred to as k^{∞}.
- If the core is made very small, the neutron leakage would be very great, multiplication factor will be less than 1 and the chain reaction would die down.
- Therefore, the reactor has to be of a certain minimum size for chain reaction to continue. This size is referred to as the critical size of the reactor.

Critical Size

- A reactor is built up to critical size in stages. Fuel elements are placed in the core and the neutron production and absorption is watched. The loading is increased till the net rate of production of neutrons remains at a constant value. A great precaution is necessary to ensure that the critical size is not exceeded.

Neutron Slowing Down Process

As mentioned above, it is necessary to slow down the fast neutrons by using a moderator.

The energy loss in a neutron collision with moderator material follows the laws of mechanics. Let there be an elastic collision between a neutron ($mass = 1$) and another nucleus ($mass = A$).

If v_1 and v_2 are the velocities of neutron before and after collision and V is the velocity of the other nucleus after collision, then by law of conservation of energy

$$\frac{1}{2} v_1^2 = \frac{1}{2} v_2^2 + \frac{1}{2} A V^2 \quad \cdots\cdots (1)$$

Neutron Slowing Down Process

and by law of conservation of momentum

$$v_1 = v_2 + AV \quad \cdots\cdots\cdots (2)$$

From Eqs. (1) and (2) we get

$$v_2 = v_1 \frac{A-1}{A+1} \quad \cdots\cdots\cdots (3)$$

The ratio of energy after collision (E) to the energy before collision (E_0) is

$$\frac{E}{E_0} = \frac{0.5\, v_2^{\,2}}{0.5\, v_1^{\,2}} = \left(\frac{A-1}{A+1}\right)^2 \quad \cdots\cdots\cdots (4)$$

If A = 1 i.e., a neutron collides with a hydrogen nucleus, energy after collision is zero.

If A = 12 i.e., neutron-carbon collision, the ratio of energies is 0.72 i.e., loss of energy is 0.28 times initial energy.

Neutron Slowing Down Process

If A is very large i.e. for collision between neutron and very heavy elements, the ratio of energies is always 1 and the loss of energy is almost zero.

This shows that only elements at the top of periodic table or compounds with small molecular weight are suitable as moderator materials.

Out of the elements having small atomic mass, gases are unsuitable on account of their low density and the consequent small number of collisions. Helium and Beryllium are costly. Boron and lithium have high neutron absorption tendency. Heavy water is an ideal moderator material and is used in many reactors in spite of its high cost. Carbon is cheap and satisfactory and can be readily obtained in desired purity. It is used in many reactors. The moderator and the fuel can either be intimately mixed or the fuel may be scattered throughout the moderator in discrete lumps. These two arrangements are known as homogenous and heterogeneous arrangements respectively.

Neutron Slowing Down Process

Eq. (3) is for head on collision between a neutron and another nucleus. In actual practice the collisions take place at all angles from 0 to 180°. By averaging over all angles the average fractional loss of energy per collision can be calculated.

A more useful quantity is the average logarithmic decrease in energy per collision (symbol ξ). Averaging the fractional energy loss expression we get $\xi = ln\ (E/E_0)$

$$\xi = 1 - \frac{(A-1)^2}{2A}\, ln\left(\frac{A+1}{A-1}\right) \quad \cdots\cdots (5)$$

For A> 10, an approximate expression for ξ is

$$\xi = 2/(A + 2/3) \quad \cdots\cdots (6)$$

Example (1)

Example 1. A reactor uses carbon as moderator. If the initial energy of neutron is 3 MeV (*a*) find the ratio of energies per collision (*b*) the number of collisions needed to reduce the energy of neutron to 0.1 eV.

Solution. For carbon A = 12

Using Eq. (6)

$$\xi = 2/(12 + 2/3) = 0.158$$

(*a*) $$\xi = ln\ (E/E_0)$$

or $$\frac{E}{E_0} = e^{\xi} = e^{0.158} = 1.171$$

Ratio of energies per collision $= 1.171$

(*b*) Ratio of initial to final energies $= \dfrac{3 \times 10^6}{0.1} = 3 \times 10^7$

Logarithmic decrement in energy $= ln\ (3 \times 10^7) = 17.2167$

Number of collisions $= \dfrac{17.2167}{0.158} = 109$

Fissile and Fertile Materials

The materials which are known to undergo neutron fission are U^{235}, U^{233} and Pu^{239}. These are fissile materials. Out of these U^{235} is the only one occurring in nature (as 0.72% of the natural uranium) and has served as the basis of nuclear energy programme.

U^{238} and Th^{232} are not fissionable. However these two materials can be converted into Pu^{239} and U^{233} respectively. These two materials (*i.e.* U^{238} and Th^{232}) are known as fertile materials.

Each fission process of U^{235} produces about 2.5 neutrons per fission. Only one of these neutrons is needed to sustain the chain reaction. The excess fission neutrons can be used to produce new fuel atoms by activating the fertile isotopes. An important parameter, in this conversion process, is the parameter *r* defined as

$$r = \frac{\text{number of fertile atoms consumed (i.e. number of new fuel atoms formed)}}{\text{number of original fuel atoms consumed in the fission and radioactive capture processes}} \quad \ldots (7)$$

If the parameter '*r*' is equal to or greater than 1, the reactor is known as breeder reactor and then the parameter '*r*' is known as the breeding ratio. If the parameter '*r*' is non-zero but less than 1 the reactor is a converter reactor and *r* is the conversion ratio. Thus a breeder reactor produces more fuel than it burns.

Fissile and Fertile Materials

Type of reactor	Conversion ratio
BWR, PWR and SGR	1
Aqueous thorium breeder	1.2
Fast breeder reactor	1.6

Nuclear Fuel Performance-Burnup and Specific Burnup

The total energy released in fission by a given amount of nuclear fuel is called the fuel *burnup* and is measured in megawatt days (MWd). The fission energy released per unit mass of the fuel is termed the *specific burnup* of the fuel and is usually expressed in megawatt days per metric ton or per kilogram (that is, MWd/t or MWd/kg) of the heavy metal originally contained in the fuel. According to the discussion in Section , the fissioning of 1.05 g of ^{235}U yields 1 MWd. Thus, it follows that the maximum theoretical burnup for this fuel is

$$\frac{1\ \text{MWd}}{1.05\ \text{g}} \times \frac{10^6\ \text{g}}{t} = 950,000\ \text{MWd/t}$$

or 950 MWd/kg.

Nuclear Fuel Performance-Burnup and Specific Burnup

Fuel performance is also described in terms of *fractional burnup* (often expressed in %), denoted as β and defined as the ratio of the number of fissions in a specified mass of fuel to the total number of heavy atoms originally in the fuel. That is,

$$\beta = \frac{\text{number of fissions}}{\text{initial number of heavy atoms}}. \qquad (8)$$

Since the fission of all fuel atoms ($\beta = 1$) yields 950,000 MWd/t and the specific burnup at any time is clearly proportional to β, it follows that, for ^{235}U,

$$\text{specific burnup} = 950,000\beta \text{MWd/t}. \qquad (9)$$

The maximum specific burnup that can be obtained from a given reactor fuel depends on a number of factors.

Nuclear Resource Utilization

In comparing the merits of different types of reactors and their associated fuel cycles, an important consideration is the efficiency with which each is capable of utilizing natural uranium or thorium resources. Thus, in general, different reactor systems require different quantities of uranium ore or thorium to produce the same amount of electrical energy. Since most countries lack abundant indigenous uranium resources, uranium utilization is often a key factor in a national decision to adopt a particular type of reactor and fuel cycle.

Nuclear Resource Utilization

The nuclear resource utilization, U , is defined quantitatively as the ratio of the amount of fuel that fissions in a given nuclear system to the amount of natural uranium or thorium input required to provide those fissions-that is,

$$U = \frac{\text{fuel fissioned}}{\text{resource input}}.$$

In short, U is the fraction of the naturally occurring resource that can be utilized in a reactor fuel cycle for the purpose of generating electric power.

Nuclear Resource Utilization

As a first example, consider a nuclear reactor operating on a once-through cycle. This reactor is fueled directly with natural uranium dioxide, and the maximum specific burnup of the fuel is about 7,500 MWd per metric ton of uranium.

Let F be the mass of fuel that fissions out of a total uranium fuel load, L. Then U = F / L, which is also equal to the burn up, B, of the fuel . According to Eq. (3 .57), the fissioning of approximately 1 gram of fissile material releases 1 MWd.

Thus, it follows that

$U = 7,500 \text{ g/t} = 7,500 \text{ g}/ 10^6 \text{ g} = 0.0075$.

Thus, only .75% of the natural uranium fuel introduced into a reactor eventually undergoes fission on the once-through cycle.

Thank you