PART IV

27
Introduction to Nuclear Engineering

27.1. INTRODUCTION

Cheap and abundant power is essential to the modern world in coming years. The rapid increase in industry and living standard of the people creates the pressure on conventional sources of power (coal, oil and gas). It is now obvious that these sources will soon be unable to meet the demands of increasingly power-hungry world.

Against this background, the atom, at first regarded as a symbol of destruction, offers an important source of prosperity. Man has now learned to draw the colossal power locked in the heart of the atom and use it for the benefit of humanity. Today the atom is considered as an almost limitless source of power which has opened a new horizon to the mankind.

One of the outstanding facts about nuclear power is the large amount of energy that can be released from a small mass of active material. Complete fission of one kg of uranium contains the energy equivalent to 3100 tons of coal or 1700 tons of oil.

The total amount of uranium and thorium in the earth's crust, to a depth of 5 kilometres, is estimated approximately 10^{12} tonnes. It is further estimated that the economically extractable ores are nearly 25 million tons of uranium and one million ton of thorium. The colossal power hidden in uranium and thorium represents a significant contribution to the world power resources.

The nuclear power is not only available in abundance but it is cheaper than the power generated by conventional sources. It has been estimated that in Britain, the nuclear energy was 30% cheaper in 1970 than the energy derived from fossil fuels while by 1980, it was cheaper by 50% than conventional resources.

The factors which are in favour of nuclear energy are, it is practically independent of geographical factors, no combustion products and it is clean source of power which does not contribute to air pollution. Further, it does not require fuel transportation networks and large storage facilities.

It is estimated that by the end of 1975, the number of power reactors were 283 in 21 countries generating 130,000 MW power. India has also launched the ambitious nuclear power programme. Of course compared with USA, Russia and Britain ; the nuclear energy programme in India is as yet a modest one.

The Tarapur, Rana-Pratap-Sagar and Kalpakkam nuclear power plants will be capable of producing 1200 MW of power and it is proposed to double the output by expansion of these plants in a short period. Several states (Gujarat, Rajasthan and U.P.) especially in areas where thermal and hydro-plants cannot be set up due to long distances of coal mining regions or lack of suitable sites for hydro-electric generation are looking forward to the possibility of setting up atomic energy stations.

Presently, India has 5 operating nuclear power stations. The present generating capacity is 1330 MWe which is 3% of the total generating capacity of the country. Power plant of 235 MWe capacity working on the cycle of Pressurised Heavy Water Reactor (PHWR) already went critical at Narora in U.P. The Nuclear power plant at Kakrapar in Gujarat, the I-set already went critical and II-set is at advanced stage of construction.

It is also estimated that another 12 reactors of 235 MWe each and ten reactors of 500 MWe each would be set up by 2000 A.D. in the country. This would raise the country's nuclear generating capacity to 10,000 MWe and would constitute 10% of the national generating capacity. The first two reactors in this group will be set up at Kaiga in Karnataka and another two would be added to the present site at Kota, Rajasthan. The work of unit II, III and IV at Kaiga is suspended because of collapse of concrete structure dome of 42.6 m in diameter in Unit-I. The commissioning of this unit is indefinitely postponed which was scheduled earlier by the end of 1996.

The second generation reactors after PHWRs would be FBRs using Pu^{239}–U^{238} fuels. Since, the fast reactors are capable of breeding fissile fuel from fertile, they can help to grow nuclear energy generation in the country much faster with 10,000 MWe PHWRs by 2000, about 4000 kg of Pu^{239} will be produced each year. Among the combinations of fissile-fertile materials, Pu–U offers best breeding ratio at present. On this basis, 350,000 MWe can be attained by the middle of 21st century.

Work on fast breeder reactor (15 MWe) was initiated in 1970 at Kalpakkam. The effects are also started in collaboration with France which is based on the design of Rapsodie. This experience will be useful in designing 500 MWe PFBR which may become operational by 2000 A.D.

The third generation reactors would be based on thorium breeding. While the doubling time for these reactors is much longer, this may be compensated by large resources of thorium. Thorium can be breeded to U^{233} which then can be used as fuel in thermal reactors. The work on this type of FBR is yet to be started but developing country like India has to start R & D earliest for self-dependence in future.

27.2. WHY NUCLEAR POWER FOR DEVELOPING COUNTRIES

The adoption of nuclear energy for the generation of power is inevitable to the nations where other sources of generation (fossil and hydel) are inadequate and scarce like in Western Europe. The U.K. is an outstanding example of this where presentlythe major source of power generation is nuclear energy.

The purpose of this article is not to discuss the importance of the nuclear power for a particular country but to find out the economical adoption of nuclear power compared with the other resources available in the developing countries.

The economic advantage of nuclear power can be realised only if one can ensure a guaranteed base load of about 75%. The number of electro-chemical processes (fertiliser plants), desalination of water and use of electricity for pumping water from tube wells assure a constant base load. Therefore, such type of power requirements must be developed before the adoption of nuclear power in the country.

It is meaningless for a country to talk of reliance solely on hydel, or thermal or nuclear power. It is well known that the least cost solution involves a grid in which there is a mix of all three types of generating units. This arises from the special feature of hydro-power which is best for peak load and nuclear power which is best for base load. The cost of generation by nuclear power is least when the load factor is maximum. The cost of hydro is lower than nuclear when it is used as peak load station. The overall cost is minimum when there is proper mix of hydro and nuclear. Then the next problem arises what mix is best for most economical operation of the national grid.

The nuclear power plants should be undertaken by the developing countries only when there are valid reasons to choose nuclear over conventional generations. The economical and feasible hydrosites should be developed first where available. After that, some conventional plants should be built to enlarge the installed capacity. Once the grid capacity reaches to 1500 MW, the feasibility of adding nuclear plants should be evaluated. The first nuclear plant to be added to the grid becomes economical only if its unit capacity lies in the range of 400 to 600 MW.

There are outstanding benefits and challenging problems in adopting the nuclear generation before the developing countries and few of them are listed below :

1. Despite the higher initial capital cost of a nuclear plant, the lower fuel cost permits conservation of foreign exchange in the power sector and allows this saving to be used on other projects.

2. It upgrades the local industry through the use of less expensive electric energy output.

3. It minimises the ecological effects of power generation.

4. It develops the national scientific capability via national nuclear research establishments.

5. It improves the "way of life" and makes the people free from burdensome tasks which can be easily performed by electrical energy.

6. In some cases, siting of nuclear power station in remote area would open a new field to general industrial development which would otherwise not have been possible if conventional fuel sources were to be transported over a long distance.

The major challenges faced by developing countries for adopting nuclear power are :

1. Finding the large sources of capital required as Rs. 25000/kW for a nuclear unit in the range of 400 to 1000 MW.

2. Trained technologically competent manpower—as managers, engineers and operators.

3. Developed grid capacity to accommodate a nuclear plant of sufficient size to be capable of producing power at lower costs.

4. Limiting any single additional unit to approximately 15% of the existing grid capacity when the nuclear plant starts operating.

But these challenges and a chance of taking a quantum step in adopting nuclear generation offer enormous opportunities for the industrialised nations to assist the developing nations. This assistance in building up the basic low cost reliable electric energy will help for the industrialization and developmnt of a developing nation.

It is predicted that Western Europe which has negligible reserves of fossil fuels will concentrate increasingly on nuclear power and in next thirty years, will install 1000 to 1500 GW of nuclear power to reach an annual rate of 200 GW installed at the end of current century. Where India's situation is concerned, all types of power generation, hydro, thermal and nuclear, will increase but rate on increase in nuclear power will be much rapid than other power plants. The high ash and sulphur content coal sources of the nation has already restricted the use of coal for power generation. Unfortunately the hydro-potentials are not uniformly distributed over the whole soil of the country. The vast resources of nuclear fuel in the country have opened a new field for the power generation as mentioned earlier. It is also necessary to use these nuclear resources to sustain the stress of power demand of the country in future.

Late Dr. Bhaba gave the following estimates of power growth in India and percentage contribution by nuclear power in 1964 at the Third International Atomic Conference for Peace in Geneva.

Year	1966	1971	1976	1981	1986
Total power	1,000	24,000	38,000	60,000	90,000
Nuclear power	—	1,180	3,000	10,000	20,000

According to K.L. Vij, chief of power economy committee, Government of India, the limited solid and petroleum fuel resources in the country virtually dictate the use of nuclear power on a long term basis. According to Dr. Kraemer of Germany, heavy water reactors would give best starting conditions for the system now planned for India. India has already started the nuclear programme and established three nuclear power stations, Tarapur (400 MW), using enriched uranium as fuel and boiling water reactor, Rana Pratap Sagar (400 MW) Candu type reactor, and Kalpakkam (400 MW) Candu type reactor. Two nuclear reactors each of 600 MW capacity in Western U.P. and a reactor of 1200 MW capacity at Kutch in Saurashtra are already suggested by the Government of India considering all economical potentialities and development required in those regions. It is also estimated that the nuclear capacity of India will be 15,000 MW out of 38,100 MW expected by the end of 1980 which will share 35% of the national power need. No doubt, with the development of fast breeder reactors, India has a better future as thorium resources in nation are considerably large and which can be used more economically for power generation.

A developing country generally faces many problems while initiating nuclear power programme in view of various limitations involving technical, economical and national considerations. The most important bottleneck in India's nuclear programme today is the shortage of trained scientists and engineers with experience to take over independent responsibility for design, construction and operation of nuclear power plants. India presently has about 2500 trained scientists (in 1975) and engineers but if squarely responsibility is to be shouldered with indigenous materials and design to build atomic power plants for future, India has to pass a long way to train more number of people to shoulder this responsibility.

27.3. ATOMIC NUCLEI

The atom of which matter is composed can be visualised as tiny solar system. An atom consists of a positively charged nucleus surrounded by a number of negatively charged electrons as shown in Fig. 27.1. The nucleus contains two types of particles *e.g.* protons and neutrons both being referred as nucleons. Each proton carries unit positive charge and neutrons are uncharged particles. Each electron carries a unit negative electrical charge. The number of electrons on the orbits is equal to the number of protons in the nucleus, therefore, a normal atom is uncharged electrically. The electric charge completely determines the chemical properties of the corresponding atom. Any addition of electron to the neutral atom makes the atom negatively charged. Similarly, any subtraction of electron will make it positively charged. Such atom is known as ion and the process of charging the atom is known as ionization.

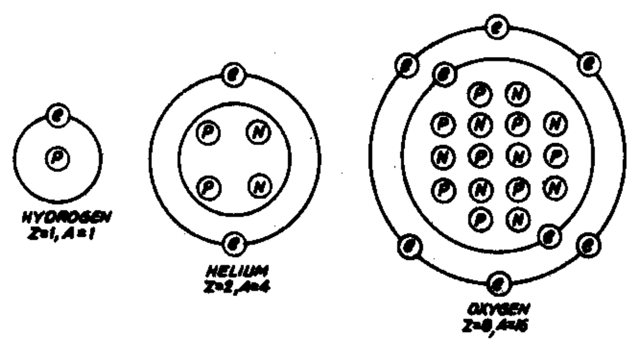


Fig. 27.1. Atomic Structure of some elements.

Chemical energy which is obtained by the combustion of coal and oil, results in an arrangement of atoms due to redistribution of electrons. On the other hand, atomic energy is a consequence of the redistribution of particles with the atomic nuclei. Therefore, the nuclear power engineering is specially connected with the variations of nucleons in nucleus.

27.4. ATOMIC NUMBER AND MASS NUMBER

The number of protons in a given atom (number of positive charges) is known as atomic number. It is usually represented by a symbol Z. The atomic number of hydrogen is 1, of He is 2 and of Li is 3 and of Uranium is 92, the element with highest atomic weight existing in nature. A number of heavier elements have been made artificially. Plutonium having an atomic number 94 is important in connection with the release of nuclear energy.

The total number of protons and neutrons in an atomic nucleus is called "mass number" and is denoted by a symbol A. Therefore, number of neutrons in a given atomic nucleus is $(A - Z)$. Since both the neutron and proton have masses close to unity on the atomic mass scale, it is evident that the mass number is the integer nearest to the atomic weight of the species under consideration.

27.5. ISOTOPES

The atomic number of the element determines the chemical nature of element. This is because the chemical properties depend in the electrons on the orbits and the number of electrons in an atom is equal to atomic number. Consequently, atoms with nuclei containing the same number of protons, *i.e.* same atomic number but with different number of neutrons, *i.e.* with different mass numbers, are chemically identical although they exhibit marked differences of nuclear stability. Such species having the same atomic number but different mass numbers are called "Isotopes". The isotope may be stable or radioactive.

Many elements present in nature exhibit two or more stable isotopes which are virtually indistinguishable chemically because they all occupy the same position in periodic table although their mass numbers and atomic weights are different. Nearly 280 stable isotopes and 50 unstable isotopes are identified as occurring naturally. Some 700 or more unstable isotopes have been obtained artificially by various nuclear reactions.

In order to distinguish among the different isotopes of a given element, it is usual to indicate the mass number together with the symbol of the element. The uranium element which is most important for the release of nuclear energy exists in three forms in nature having mass number 233, 235 and 238. It is seen that uranium 238 is most abundant isotope found in natural ore where the percentage of other isotopes are very less. The natural uranium contains 0.006% of $_{92}U^{233}$, 0.714% of $_{92}U^{235}$ and remaining 99.28% of $_{92}U^{238}$.

The isotope of any element say 'uranium' is represented as $_{Z}U^{A}$ or uranium 238 is represented as $_{92}U^{238}$. The number 92 represents 92 protons and number (238 – 92) = 146 represents number of neutrons where 238 is mass number.

27.6. ATOMIC MASS UNIT

In many calculations involving mass energy equivalence in processes occurring within the interior of atoms, it is convenient to express masses in terms of atomic mass (amu). On the atomic mass scale, the mass of any single particle, *e.g.* an electron or an atom, in amu is numerically equal to its atomic weight. For example, the mass of electron is 0.00055 amu and that of proton is 1.007587 amu. The mass of a particle in grams can be obtained by dividing its mass on the atomic mass scale by Avogadro constant *i.e.* 6.0225×10^{23}

$$1 \text{ amu} = \frac{1}{6.0225 \times 10^{23}} \text{ grams} = 1.6604 \times 10^{-24} \text{ grams.}$$

The masses of the electron, proton, neutron and hydrogen atom on the atomic mass scale are given as follows :

$$\begin{aligned} \text{mass of electron} &= 0.00055 \text{ amu} \\ \text{mass of proton} &= 1.00758 \text{ amu} \\ \text{mass of neutron} &= 1.00897 \text{ amu} \\ \text{mass of hydrogen atoms} &= \text{mass of proton} + \text{mass of electron} \\ &= 1.00758 + 0.00055 = 1.00813 \text{ amu.} \end{aligned}$$

The neutron which is of fundamental importance in connection with the release of nuclear energy is electrically neutral and carries no charge. The mass of neutron is somewhat greater than that of a proton and even of hydrogen atom.

27.7. RADIOACTIVITY AND RADIOACTIVE CHANGE

The naturally occurring elements of highest atomic weight, such as thorium, radium and uranium consist of unstable isotopes. These elements undergo spontaneous change referred as radioactive change. The radioactive change is accompanied by the emission of α-particles, β-particles or γ-particles or simultaneously two or three from the atomic nucleus. After a number of stages of disintegration, an atomic specie with a stable nucleus is formed. The natural phenomenon of emitting radioactive waves (α, β and γ-rays) by the unstable element is known as radioactivity of the element.

It has been observed that rate of radio-radiations from unit mass is fixed and cannot be changed by any method. Therefore, the quantity of radiation per unit time for any isotope can be determined easily. The different types of radioactive radiations are discussed below.

1. **α-Radiation.** The α-particle contains helium nuclei which contains two neutrons and two protons ($_2He^4$). Definite amount of energy is required to emit α-particle from a particular isotope. The energy of 4.3 MeV is required to emit α-particle from $_{92}U^{238}$.

$$\therefore \qquad _{92}U^{238} \rightarrow {}_{90}Th^{234} + {}_2He^4.$$

The decay of plutonium also emits α-particles

$$\therefore \qquad _{94}Pu^{239} \rightarrow {}_{92}U^{235} + {}_2He^4.$$

It is obvious from the above two equations that the α-emission from isotopes produces new isotope whose number is reduced by 4 and nuclear change by two units.

2. **β-Radiation.** When the number of neutrons and protons in the nucleus of given isotope are such that ratio of neutron to proton lies outside the stability range for that mass number, the nucleus will undergo spontaneous change in the direction of increased stability. The neutron will be spontaneously converted into a proton and at the same time a negative electron. A negative β-particle will be expelled out as given by the following equation :

	Neutron $\rightarrow$	Proton +	Negative β-particle
charge	0	+ 1	− 1
mass	1	1	0.

This can be represented as

$n \rightarrow p + {}_{-1}e^0$ where ${}_{-1}e^0$ represents negatively charged electron and n and p are neutron and proton.

$\therefore \quad n \rightarrow p + \beta^-$.

The result of the above change is to replace a neutron by a proton so that the atomic number of the product element is increased by one although its mass number is unchanged. In the new element formed after the above-mentioned change, the neutron to proton ratio will be less than neutron to proton ratio in nucleus of the parent element, because change of neutron to proton means decrease in number of neutrons and increase in number of protons. After one, two or more stages of disintegration, a stable element will be formed.

A disintegration of a particular isotope emitting β-particles (–vely charged electron) is given by the following equation :

$${}_{56}Ba^{140} \rightarrow {}_{57}La^{140} + {}_{-1}e^0 \rightarrow {}_{58}Ce^{140} + {}_{-1}e^0 \rightarrow {}_{58}Ce^{140} + \beta^- \text{ where } {}_{58}Ce^{140} \text{ is stable.}$$

or

$${}_{57}La^{140} \rightarrow {}_{58}Ce^{140} + \beta^-$$

A nucleus will also be unstable if the number of neutrons is too small or the number of protons is too large. Now a proton will be converted into a neutron and a positive electron known as positron will be emitted out. The change is shown by the following equation :

	Proton $\rightarrow$	Neutron +	Positive β charge
charge	+ 1	0	+ 1
mass	1	1	0.

This can be represented as

$p \rightarrow n + {}_1e^0$ where ${}_1e^0$ represents positively charged electron

$\therefore \quad p \rightarrow n + \beta^+$.

The product nucleus will then have an atomic number one unit lower than its parent, although the mass number will be same. The daughter nucleus may still be unstable and it will be again radioactive. After one or more stages of positive β decay, a stable nucleus having a neutron to proton ratio within the stability range will be formed.

This type of disintegration for a particular isotope emitting β-particle (+vely charged electron) is given by the following equation :

${}_7N^{13} \rightarrow {}_6C^{13} + {}_1e^0$ where ${}_6C^{13}$ (stable isotope)

or ${}_7N^{13} \rightarrow {}_6C^{13} + \beta^+$.

The stability range for different types of isotopes is shown in Fig. 27.2.

3. **γ-Radiation.** When the nucleus remains in excited state (if the nucleus absorbs neutron) it suffers radioactive decay. The product nucleus starts radiating energy in the form of electromagnetic waves and

tries to be stable within a very short period (10^{-15} sec). The radiated energy in the form of electro-magnetic waves is known as γ-radiation. These rays are similar in character to X-rays, they are highly penetrating and have wavelengths in the range of 10^{-8} to 10^{-11} cm or less. The disintegration of CO^{60} isotope emits γ rays. Decay of CO^{60} isotope results in the excited nucleus of Ni^{60} isotope which also emits γ-rays instantaneously.

4. **Neutron emission.** It the nucleus of a radioactive isotope remains in highly excited state, the emission of neutron from the nucleus takes place. In this case, the mass number of the daughter isotope is reduced by unity as given by following equation :

$${}_{53}I^{137} \rightarrow ({}_{-1}e^{0} + {}_{54}I^{137}) \rightarrow ({}_{0}n^{1} + {}_{53}I^{136}) \text{ (stable).}$$

27.8. RATE OF RADIOACTIVE DECAY

The radioactive emission of the isotopes in the form of α, β and γ radiations is not an instantaneous process. The emission process follows certain law with respect to time and the parent isotope tries to be stable. The time during which the radioactive isotope becomes stable is known as decay period of the isotope.

The rate of decay at any instant is always directly proportional to the number of radioactive species of an isotope present at that instant.

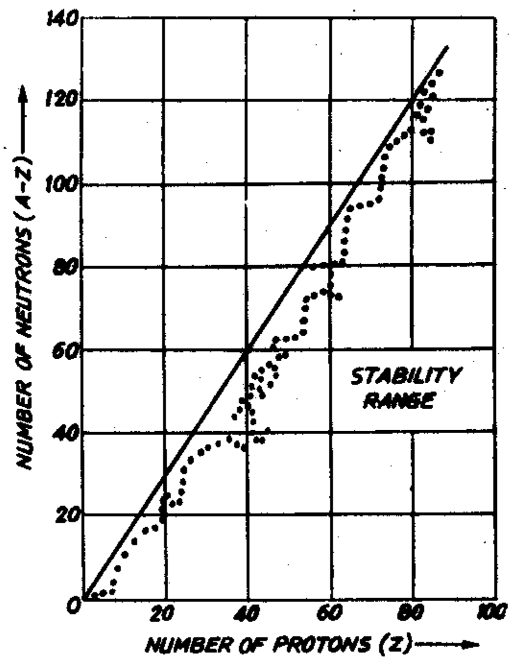


Fig. 27.2. Number of neutrons and protons in stable range.

If N is the number of radioactive species present in radioactive isotope at any instant (time) t, the decay rate is given by

$$\frac{dN}{dt} = -\lambda N$$

where λ is known as decay constant.

$$\therefore \quad \frac{dN}{N} = -\lambda dt$$

Integrating the above equation, we get

$$\log_e N = -\lambda t + C \text{ where } C \text{ is constant of integration.}$$

$$\therefore \quad N = C \cdot e^{-\lambda t}$$

If $N = N_0$ at $t = 0$, then $C = N_0$.

$$\therefore \quad N = N_0 \, e^{-\lambda t} \qquad ...(27.1)$$

The radioactive nature of different isotopes is measured by a time known as 'Half life time'. ***The half life time of a particular isotope is defined as the time required for the number of active nuclei to decay to half of its initial number.***

If $N = N_0/2$, the corresponding time required to bring the number of active particles from N_0 to $N_0/2$ is known as half life time and it is given by,

$$\frac{N_0}{2} = N_0 \, (e)^{\lambda t_f}$$

where t_f is half life time.

$$\therefore \quad t_f = \frac{\log_e (2)}{\lambda} = \frac{0.6931}{\lambda} \qquad ...(27.2)$$

The half life time is inversely proportional to the decay constant. The half life times of known radioactive isotopes range from a small fraction of a second to billions of years. The half life times of few radioactive isotopes are listed below :

Isotope	U^{228}	U^{239}	Np^{239}	Pu^{239}	Th^{232}	Th^{233}	Pa^{233}	U^{233}
Half life time	4.5×10^9 years	23.5 minutes	2.3 days	2.4×10^4 years	1.4×10^{10} years	22 minutes	27.4 days	1.6×10^{15} years

Measuring unit of radioactivity. The common unit of radioactivity is '*Curie*', a unit originally defined in terms of the rate of decay of radium which is 3.7×10^{10} disintegrations per second. This unit is considerably large for practical purposes, therefore, units of millicurie and microcurie are used. The National Bureaus of Standards recommends a unit known as '*Rutherford*' which is defined as 10^6 disintegrations per second. This unit is more desirable and is used to a greater extent.

Average or Mean Life. It is the average of total time for which radioactive nuclei has distintegrated for several half lives. This is obtained by taking the sum of the decay time of the radioactive nuclei and dividing by the number of initial radio-active isotopes.

$$\therefore \quad t_m = \frac{-\int_0^\infty N \,.\, dt}{N_0} = \frac{-\int_0^\infty N_0\, e^{-\lambda t} \,.\, dt}{N_0} = \frac{1}{\lambda} \qquad \text{...(27.3)}$$

but half life time is given by $t_f = \dfrac{0.6931}{\lambda}$.

This shows that the mean life time is more than the half life time of the radioactive nuclei and relation between the two is given by

$$\frac{t_m}{t_f} = \frac{1}{0.6931} = 1.445 \qquad \text{...(27.4)}$$

27.9. MASS ENERGY EQUIVALENCE

Einstein has given a theoretical relationship between the mass and energy. The theory proposed by Einstein states the mass of the moving body varies with its velocity and it is given by

$$m = \frac{m_0}{\sqrt{1 - \left(\frac{v}{c}\right)^2}} \qquad \text{...(27.5)}$$

where m_0 = mass when body is at rest

v = velocity of the moving body

and c = velocity of light.

$$\therefore \quad m = m_0 \left[1 - \left(\frac{v}{c}\right)^2\right]^{-1/2}$$

Expanding the right-hand equation using Binomial theorem and neglecting all other terms except first, we get

$$m = m_0 + \frac{\frac{1}{2} m_0 v^2}{c^2} = m_0 + \frac{E_k}{c^2}$$

where E_k is kinetic energy of moving body.

$$\therefore \quad \Delta m = (m - m_0) = \frac{E_k}{c^2}$$

$$\therefore \quad E_k = \Delta m . c^2 \qquad \text{...(27.6)}$$

It would appear from the above equation that there is definite relationship between the mass and energy. Therefore, in general the relation between mass and energy can be expressed as

$$E = mc^2$$

where m is the mass equivalence of energy.

This equation is often referred as Einstein mass energy equation and is a fundamental to the subject of atomic energy.

Substituting the value of

$$c = 2.998 \times 10^{10} \text{ cm/sec.}$$

and mass m in grams in the equation,

$$E = m \text{ (grams)} \times (2.998 \times 10^{10})^2 = m \text{ (grams)} \times 8.989 \times 10^{20} \text{ ergs.} \qquad (27.7)$$

In nuclear engineering, the energy is generally represented by electron volt (eV) or million of electron volt (MeV).

An electron volt is the amount of energy required to move an electron charge a distance corresponding to a change in voltage from 0 to 1 volt. The value of one electron volt in terms of ergs is given as

$$1 \text{ eV} = 1.602 \times 10^{-12} \text{ ergs.}$$

$$1 \text{ MeV} = 10^6 \text{ eV} = 1.602 \times 10^{-6} \text{ ergs}$$

Substituting this value in equation (27.7)

$$E = m \text{ (grams)} \times \frac{8.989 \times 10^{20}}{1.602 \times 10^{-6}} \text{ MeV} = m\text{(grams)} \times 561.1 \times 10^{24} \text{ MeV} \quad ...(27.8)$$

If the mass m in amu is substituted in equation (27.8), then

$$E = m \text{ (amu)} \times 1.6605 \times 10^{-24} \times 561.1 \times 10^{24} \text{ MeV.}$$

$$= m \text{ (amu)} \times 931.4 \text{ MeV.} \qquad ...(27.9)$$

It is obvious that one atomic mass unit is equivalent to 931.4 Mev energy.

The energy deficiency which binds a valence electron to an atom is of the order of few electron volts whereas the energy deficiency which binds nucleons (neutrons and protons) together to form an atomic nucleus is of the order of millions of electron volts per nucleon. Consequently, the energy released per atom in burning fossil fuel is of the order of a few electron volts whereas the energy released in nuclear interactions is measured in millions of electron volts. This is obvious from the following chemical and nuclear reactions.

Chemical :

$$2H_2 + O_2 = 2H_2 + 5.92 \text{ eV}$$

$$C + O_2 = CO_2 + 4.17 \text{ eV}$$

Nuclear (Fission)

$$U^{235} + n \longrightarrow \text{stable nuclei} + x \,.\, n + 1.65 \times 10^8 \text{ eV.}$$

Usefulness of Einstein Theory. Consider a nuclear reaction

$$a + b \longrightarrow c + d + Q$$

where a, b, c and d are nuclear particles and Q is the amount of energy released by the reaction. If mass of $(a + b) >$ mass of $(c + d)$, then the above reaction is possible as per Einstein theory.

If mass of $(c + d) >$ mass of $(a + b)$ then the value of Q becomes negative and this indicates that such reaction does not occur under natural circumstances.

Consider a neutron is spontaneously decaying into a proton and an electron

$${}_0n^1 \longrightarrow {}_1H^1 + {}_{+1}e^0 + Q$$

is possible as the mass of neutron (1.00897 amu) is greater than the mass of hydrogen (1.00813 amu).

But the reaction of hydrogen atom becoming a neutron is impossible as

$${}_1H^1 + {}_{+1}e^0 \longrightarrow {}_0n^1 + Q.$$

Einstein's law is useful in two ways :

1. It gives necessary condition for a nuclear reaction to take place.
2. It allows to calculate exactly the energy released in a nuclear reaction.

Consider the decay reaction of cobalt isotope as given below :

$${}_{27}Co^{60} \longrightarrow {}_{28}Ni^{60} + {}_{+1}e^0 + Q$$

$\therefore$ Q = (mass of cobalt isotope − mass of nickel isotope) × 931.4 MeV
= (59.9525 − 59.9495) × 931.4 = 2.81 MeV.

27.10. BINDING ENERGY

Atomic nuclei consists of two kinds of primary particles, called protons and neutrons as mentioned earlier. Both protons and neutrons can be obtained in free states and their individual properties can thus be studied.

It might be thought that a nuclei packed with positively charged protons would fly apart because of electrostatic repulsion of the charges. It is obvious that the stability of the atomic nuclei is related to the presence of neutrons in addition to protons.

It is accepted fact that attractive forces among neutron-neutron ; proton-proton and neutron-proton exist even though little is known about the nature of these forces.

Consider an isotope whose atomic number is Z and mass number is A. The total mass of the nuclei of this isotope

= mass of Z protons + mass of Z electrons + mass of $(A - Z)$ neutrons

$$= Zm_p + Zm_e + (A - Z)m_n = Z(m_p + m_e) + (A - Z)m_n \doteq Zm_h + (A - Z)m_n$$

where m_p, m_e and m_n are the masses of proton, electron and neutron respectively. m_h is the mass of hydrogen which is equal to the mass of proton and electron $(m_p + m_e)$.

The actual mass of the nuclei (M) calculated experimentally is always less than the mass of nuclei theoretically determined by the above expression. The difference between the theoretical mass and experimental mass M is known as "Mass Defect" and it is given by

Mass defect $= [\{Zm_h + (A - Z)m_n\} - M]$ ...(27.10)

This mass defect represents the mass which would appear in the form of energy and it is used to bind the neutrons and protons together in the nuclei. The energy equivalence of the mass defect is taken as measure of binding energy of a particular nuclei because the same amount of energy would have to be supplied to the nuclei in order to break it up into its constituents. The binding energy of a nuclei is given by

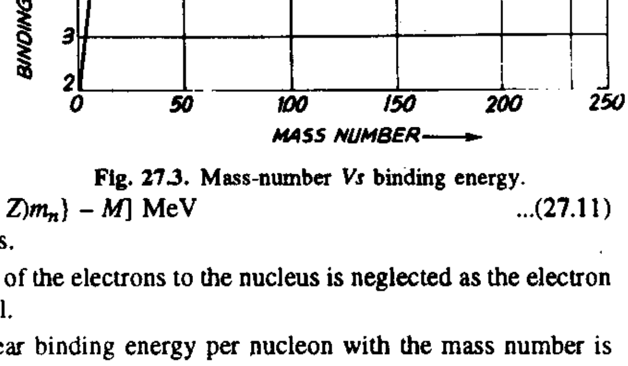


Fig. 27.3. Mass-number Vs binding energy.

$$E_b = 931.4\ [\{Z_{mh} + (A - Z)m_n\} - M]\ \text{MeV} \qquad ...(27.11)$$

where the masses are given in atomic mass units.

In the above calculation, the binding energy of the electrons to the nucleus is neglected as the electron binding energy is very small fraction of the total.

A curve representing the variation of nuclear binding energy per nucleon with the mass number is shown in Fig. 27.3. The figure shows that the binding energy increases with an increase in mass number and reaches the peak of 8.8 MeV at 60 mass number. After this the binding energy falls as the mass number increases. The binding energy value reaches to 7.6 MeV at the mass number 238 which represents Uranium nuclei.

27.11. RELEASE OF ENERGY BY NUCLEAR REACTION

The energy can be released in two ways :

1. By combining light nuclei like H_2 and He. This process of releasing energy is known as **'Fusion'**.

2. By breaking up heavy nuclei into nuclei of intermediate atomic number. This process of releasing energy is known as **Fission.**

The result of fission is an increase in binding energy per nucleon. The change in the binding energy develops kinetic energy and heat.

27.12. TYPES OF NUCLEAR REACTIONS

The change in the mass of particles during nuclear reactions represents the release or absorption of energy. If the total mass after the reaction is reduced, the reaction releases the energy and an increase in mass causes the absorption of energy.

The nuclear reactions which take place are four in nature :

1. **Inelastic Scattering.** When a neutron undergoes inelastic scattering, it is first captured by the target to form a compound nucleus, then a neutron of lower kinetic energy is expelled from the target nucleus leaving the nucleus in excited state. Thus in inelastic scattering process, part of kinetic energy of the neutron is converted into internal or excitation energy of the target nucleus. The excitation energy is subsequently emitted in the form of γ-radiations.

2. **Elastic Scattering.** The situation of elastic scattering is different from inelastic scattering. In elastic scattering, the kinetic energy of the neutron is conserved. When the neutron strikes the target nucleus, it imparts the part of kinetic energy to the target nucleus and its original kinetic energy is reduced. The process of elastic scattering is as "billiard ball type collision." In each collision with stationary nucleus, the neutron transfers part of its kinetic energy to the nucleus and its own kinetic energy is reduced. This process of scattering slows down the neutrons. The fraction of energy transferred to the target nucleus per collision depends upon the angle through which neutron is scattered and the mass of the target nucleus.

When the neutron strikes a light nucleus such as H_2, the kinetic energy of the neutron is very much reduced and it is transferred to proton. Here the most of the neutron energy is transferred to proton as both particles have nearly same mass. With single collision of neutron with hydrogen atom, nearly 75% of the neutron energy is transferred to proton. The energy transfer from the neutron to heavier target nucleus per collision is considerably less. When a neutron collides with a carbon nucleus, the kinetic energy transferred to carbon nucleus per collision is hardly 15% of the initial kinetic energy of the neutron. The elastic scattering process plays very important role in the operation of nuclear reactors.

3. **Neutron Capture.** In this process, the colliding neutron is absorbed by the target nucleus and increases its mass number by unity. The target nucleus becomes excited and emits the energy in the form of γ-radiations. The artificial radioactive elements are produced by this method. This type of nuclear reaction is not at all important in nuclear-power engineering.

4. **Fission.** In this type of reaction, the target nucleus absorbs thermalised (slow) neutron and becomes highly excited. Therefore, it splits into two different masses. The product masses will also be in excited state and they try to be stable by emitting neutrons. Such type of reaction is possible only with the heavy nucleus such as $_{92}U^{233}$, $_{93}U^{235}$ and $_{94}Pu^{239}$.

This reaction is known as fission reaction. The nuclei produced after reaction are lighter than original nuclei and having more binding energies. The release of energy is due to an increase of mass defect of the lighter nuclei. This reaction is most important from the nuclear-power engineering.

27.13. INITIATION OF NUCLEAR REACTIONS

To start the nuclear reaction, sufficient energy must be given to the nuclei to overcome the electrostatic force of repulsion. This can be done by following methods :

1. **First method.** In this method, the nuclei of an element are accelerated by means of accelerator and these highly accelerated nuclei are bombarded on the target nuclei. The particles generally used for these purposes are H_2 or He (α-particles) because they have less repulsive forces. This method of initiation of nuclear reaction has no much practical importance as most of the accelerated particles after striking with target nuclei lose their energy in joining the atom and are unable to fulfill the requirements.

2. **Second method.** In this method, the substance is heated to a very high temperature (millions of °C). At this temperature, the motion of the particles due to high thermal energy is sufficient to overcome the electrostatic repulsive forces. This is known as thermo-nuclear reaction. This is not practical for the present time as materials are not available to sustain such high temperatures. Scientists have predicted that the reactors working on this principle (fusion reactors) will be available by the end of this century. This type of reaction takes place in the central core of the sun and is limited to light nuclei such as hydrogen.

3. **Third method.** In this method, the neutrons are used as bombarding particles. The main advantage of neutron is that they are neutral (no charge) and, therefore, they can make the way through the shells of electrons and then through the nucleus even at low energy. This is the practical method used in all modern fission reactors. This initiation of nuclear reaction is known as neutron-induced uranium fission reaction.

27.14. CROSS-SECTION

The cross-section of the target nuclei may be fuel or moderator plays very important part in thermal reactors. The probability of reaction of the impinging particle with the target nucleus is directly dependent on the cross-section area of the target nucleus. A nuclear reaction is more likely to occur if the cross-section of the target nucleus is large.

The concept of cross-section of target nucleus can easily be visualised as cross-section presented by target nucleus to an incident neutron as shown in Fig. 27.4. If the nuclei is regarded as a sphere of radius r, and the incident neutron as point particle, then the cross-sectional area of a target nuclei is

$$\sigma = \pi r^2 \text{ cm}^2 \text{ where } r \text{ is in cm.}$$

As r for different nuclei is of the order of 10^{-12} cm, r^2 is of the order of 10^{-24} cm^2. The basic unit chosen for cross-section (σ) is 10^{-24} cm^2 and it is designated as "Barn".

The projected neutron may collide with target nuclei either by elastic scattering or inelastic scattering or it may be captured by the target nuclei. Another possibility is that the neutron may not interact at all and may escape. This may happen if the gaps between the adjustment nuclei are bigger in sizes than the colliding neutron.

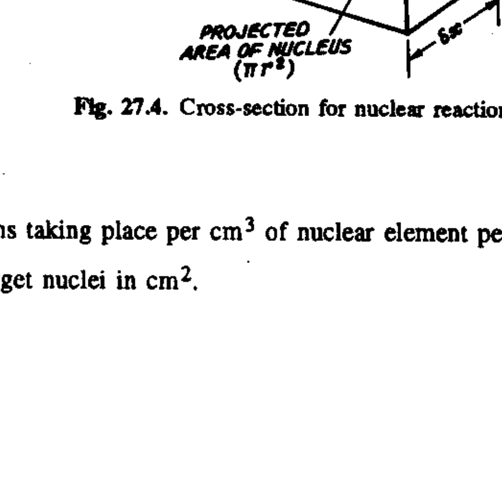


Fig. 27.4. Cross-section for nuclear reactions.

The rate of nuclear reaction in a nuclear element is given by the following expression :

$$\frac{dN}{dt} = -\sigma\phi N \qquad ...(27.12)$$

where $\frac{dN}{dt}$ = Number of reactions taking place per cm^3 of nuclear element per second

σ = cross-section of target nuclei in cm^2.

ϕ = Neutron flux (number of neutrons/cm^2/sec).

N = Number of target nuclei per cu cm of target material.

The negative sign indicates that the number of nuclei for further reaction decreases with time.

As we have seen earlier that the reaction of neutron on target nuclei may be scattering fission or capture. Therefore, the process of inter-reaction has been given various names to the cross-section, *i.e.* capture cross-section, scattering cross-section and fission cross-section. The total cross-section is the sum of the cross-sections mentioned above

$$\therefore \qquad \sigma_t = \sigma_c + \sigma_s + \sigma_f. \qquad ...(27.13)$$

In the above discussion, we have not considered the effect of velocity on the cross-section. The neutrons in a nuclear reactor are present with a large range of velocities and it has been found that the neutron cross-section varies with the velocity of the neutron. The variation generally follows the law

$$\sigma = \text{constant or } \sigma \propto \frac{1}{v}.$$

It is not necessary that all neutrons should follow $1/v$ law. The cross-section variation of neutrons on the basis of energy (velocity) level are classified into three regions.

1. In low range of energy (0.25 eV or less), the cross-section variation follows the law $\sigma \propto (1/v)$ as shown in Fig. 27.5 (*a*) by curve *a*.

2. In some reactions, the cross-section has one resonance peak and for the rest region it follows $1/v$ law as shown in Fig. 27.5 (*b*).

3. In this type of nuclear reaction, the cross-section follows $1/v$ law for narrow range of energies and it forms many peaks in the remaining range of the energy as shown in Fig. 27.5 (*c*). This phenomenon of odd behaviour is known as resonance. The resonance peaks are considered in the design of nuclear reactors.

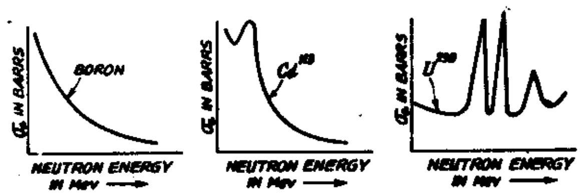


Fig. 27.5. Variation of cross-section with the energy in different types of nuclear reactions.

27.15. NUCLEAR FISSION

When unstable heavy nucleus is bombarded with high energy neutrons, it splits into two fragments more or less of equal mass. This process is known as "Nuclear Fission". The fission fragments formed due to fission are the isotopes which are located in the middle of periodic table. The binding energy per nucleon for the elements in the middle range of periodic table is more than that of heavy nuclei. Therefore, the nuclear fission is always associated with the release of large amount of energy. Further nuclear fission is followed by the emission of several high energy neutrons per neutron bombarded. This is the essential condition for self-sustaining the fission reaction. The fission reaction is shown in Fig. 27.6.

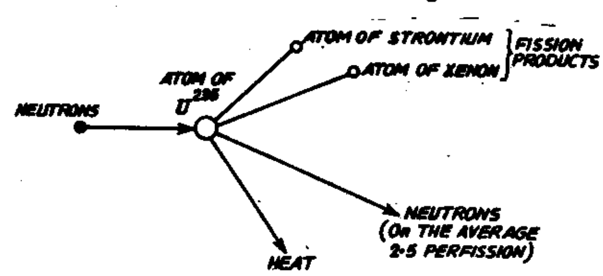


Fig. 27.6.

The self-sustaining nuclear fission reaction associated with release of energy is very important in the power development.

The energy required to break the nucleus must be sufficient to overcome the repulsive forces (nuclear binding energy) between the two fragments. The excitation energy required to split the nucleus is called the critical energy. It is obvious that the critical energy is more than the neutrons binding energy. The nuclei of $_{92}U^{238}$ and $_{90}Th^{222}$ (even-even type) which can be fissioned by neutrons possessing kinetic energy of

1.1 MeV. If the critical energy is less than neutron binding energy, the fission of nucleus is possible with slow neutrons (0.025 eV). Even-odd type nucleus like $_{92}U^{233}$ and $_{96}Pu^{239}$ can be fissioned by slow neutrons.

To sustain the fission processes, the following requirements must be fulfilled :

1. The bombarded neutrons must have sufficient energy to cause fission of another nucleus (to overcome binding energy).

2. The number of neutrons produced must be able to increase the rate of fission as certain loss of neutrons by absorption, and leakage is unavoidable.

3. The fission process must generate the energy.

4. The fission process must be controlled.

The Fission Products. The nuclear fission is accompanied by the release of enormous amount of energy, fission products and variable numbers of neutrons. When the fission of $_{92}U^{235}$ takes place ; all nuclei do not fission in an identical way. It is known fact that the fission of $_{92}U^{235}$ takes place in more than 70 different ways, producing about 140 different nuclear species. The most probable mass numbers for the fission fragments are 95 and 139 and their nuclide constitute about 6% of the total fission nuclides.

A few possible fission reactions of $_{92}U^{235}$ are given below :

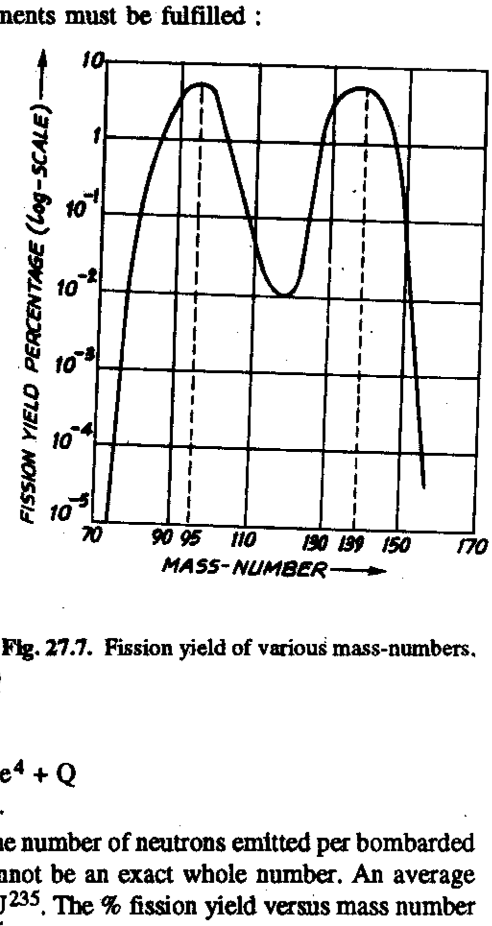


Fig. 27.7. Fission yield of various mass-numbers.

$$_{92}U^{235} + {}_0n^1 \longrightarrow {}_{56}Ba^{141} + {}_{36}Kr^{92} + 3\,{}_0n^1 + Q$$
$$_{92}U^{235} + {}_0n^1 \longrightarrow {}_{57}La^{147} + {}_{35}Br^{87} + 2\,{}_0n^1 + Q$$
$$_{92}U^{235} + {}_0n^1 \longrightarrow {}_{28}Sr^{94} + {}_{54}Xe^{139} + 3\,{}_0n^1 + Q$$
$$_{92}U^{235} + {}_0n^1 \longrightarrow {}_{51}Sb^{134} + {}_{39}Y^{95} + 3\,{}_0n^1 + {}_2He^4 + Q$$
$$_{92}U^{235} + {}_0n^1 \longrightarrow {}_{57}La^{147} + {}_{35}Br^{87} + 2\,{}_0n^1 + Q.$$

Now from the above nuclear reactions, it is obvious that the number of neutrons emitted per bombarded neutron is not integer because average of all the reactions cannot be an exact whole number. An average 2.5 ± 0.1 neutrons are emitted for each neutron absorbed by $_{92}U^{235}$. The % fission yield versus mass number is plotted in Fig. 27.7. The fission yields range is from 10^{-5} to 6%.

The fission products constitute a complex mixture of about 400 different types of radioactive nuclides with decay period varying from a fraction of second to several years.

The Energy Release in Fission. Consider a fission reaction of $_{92}U^{235}$ as given below :

$$_{92}U^{235} + {}_0n^1 \longrightarrow {}_{42}Mo^{95} + {}_{57}La^{139} + 2\,{}_0n^1 + Q$$

The energy released (Q) is given by

$Q = 931.4\ (\Delta m)$ where Δm is the difference in mass after reaction.

Δm = (mass of $_{92}U^{235}$ + mass of one neutron) – (mass of $_{42}Mo^{95}$ + mass of $_{57}L^{139}$ + mass of two neutrons).

$$= (235.124 + 1.00897) - (94.945 + 138.955 + 2 \times 1.00897)$$
$$= 0.215 \text{ amu.}$$

$$\therefore \quad Q = 0.215 \times 931.4 = 200 \text{ MeV.}$$

$$\text{Now } 200 \text{ MeV} = \frac{200}{0.625 \times 10^6 \times 10^7} = 3.21 \times 10^{-11} \text{ joules.}$$

∴ To produce one joule of energy per second (one watt), the number of fissions required

$$= \frac{1}{3.21 \times 10^{-11}} = 3.1 \times 10^{10}$$

27.16. THE FISSION CHAIN REACTION

The essential condition for the practical utilization of nuclear energy is that a self-sustaining chain reaction should be maintained. Once the fission process is initiated in a few nuclei of fissile material, it should be able to continue throughout the remaining material without external influence.

As seen earlier, nearly 2.5 neutrons capable for causing further fission are released in each act of fission and, therefore, self-sustaining chain-reaction is possible. However, account must be taken of the neutrons which can take part in other non-fission reactions. In addition to this, there is inevitable loss of neutrons by leakage through the system. Therefore, the number of neutrons produced must be sufficient to maintain the chain-reaction and to supply the losses (escape, absorption and non-fission reactions). Therefore, the essential condition to maintain the chain reaction is that the fission nucleus must produce at least one secondary neutron to cause fission of other nucleus.

The above-mentioned condition can be expressed in terms of multiplication factor or reproduction factor defined as

$$k = \frac{\text{Number of neutrons of any one generation}}{\text{Number of neutrons of immediately preceding generation}}$$

Since one neutron is required to maintain the chain reaction, the increase in number of neutrons in one generation will be $(k-1)$. If there are n neutrons initially (at the start of reaction) ; the increase in number of neutrons in the preceding generation will be $n(k-1)$. If t_1 is the average time between successive neutron generation, the rate of increase in neutron generation is given by

$$\frac{dn}{dt} = \frac{n(k-1)}{t_1}$$

$$\therefore \quad \frac{dn}{n} = \frac{k-1}{t_1}\,dt.$$

$$\therefore \quad \int \frac{dn}{n} = \int \left(\frac{k-1}{t_1}\right) dt.$$

$$\therefore \quad \log_e n = \left(\frac{k-1}{t_1}\right) t + \log_e C \text{ where } \log_e C \text{ is a constant of integration.}$$

$$n = n_0 \text{ at } t = 0 \quad \therefore \quad C = n_0.$$

$$\therefore \quad \log_e n = \left(\frac{k-1}{t_1}\right) t + \log_e n_0$$

$$\therefore \quad \log_e \left(\frac{n}{n_0}\right) = \left(\frac{k-1}{t_1}\right) t.$$

$$\therefore \quad n = n_0 (e)^{(k-1)t/t_1} \qquad \text{...(27.14)}$$

It is obvious from the above equation that if $k > 1$, the number of neutrons increases exponentially with time.

Suppose in a particular case $k = 1.005$ and $t_1 = 0.001$ sec. (average life time for neutron), the number of neutrons after one second will increase by

$$\frac{n}{n_0} = (e)^{\left(\frac{0.005 \times 1}{0.001}\right)} = e^5 = 150 \text{ times.}$$

If $k < 1$, the chain-reaction cannot be maintained as neutron production decreases in an exponential manner.

The neutrons are lost during the fission leakage, non-fission capture by U^{238} and U^{235} (known as resonance capture) and non-fission capture by moderator, structural material, coolant and fission products.

If say $n_0 = 100$, and $k = 2$

then the fission causes to release 200 fission neutrons in the next succeeding fission. If 100 neutrons out of 200 are lost by the ways mentioned above and 100 are available for next fission then the chain-reaction is possible. If the loss is more than 100, the reaction dies out.

When $k < 1$, the system is known as subcritical (stopping the reactor) and when $k > 1$, the system is known as super-critical (atom-bomb) and when $k = 1$, the system is known as critical and this is the desirable requirement for power reactors.

As already mentioned, the chain reaction dies out if the loss of neutrons per successive generation is greater than the number of neutrons at the beginning of fission. The loss can be reduced by increasing the mass of the fissile material because the number of atoms available to be struck by neutron increases on cube law, whereas the surface from which they can escape increases on square law. There will be a particular size of fissile material for which the neutron production by fission is exactly balanced by neutron leakage and absorption. This is called the "Critical Size" and it is that at which the chain reaction will just be self-sustaining.

Therefore, the size of a neutral uranium reactor is determined by nuclear considerations of criticality and not by the power output. In principle, the power output is independent of its size. Power is determined essentially by the rate at which heat can be removed from the reactor. A small reactor could be designed to produce tremendous amount of power if extremely high heat transfer rate could be achieved.

The production of neutrons directly controls the power level in the reactor. If k is greater than unity, power level will rise and when it attains the required level, k is reduced to unity and it will be maintained at a constant value.

A fission uncontrolled chain reaction is shown in Fig. 27.8 (a) when $k > 1$ and controlled chain reaction is shown in Fig. 27.8 (b) when $k = 1$.

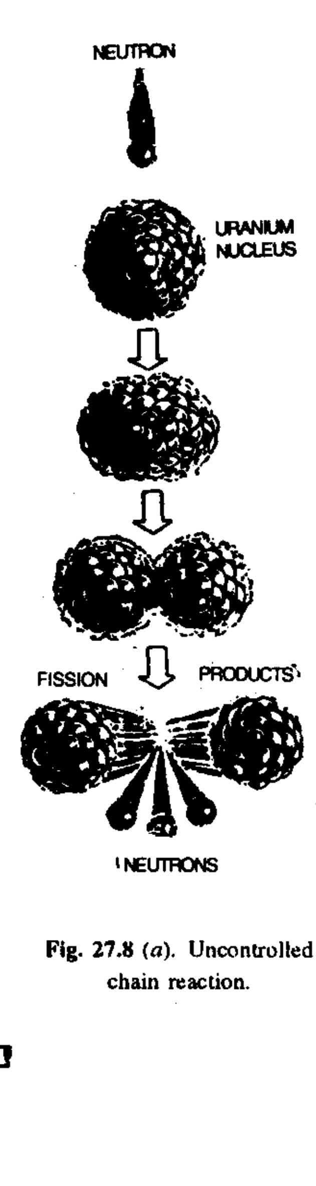


Fig. 27.8 (a). Uncontrolled chain reaction.

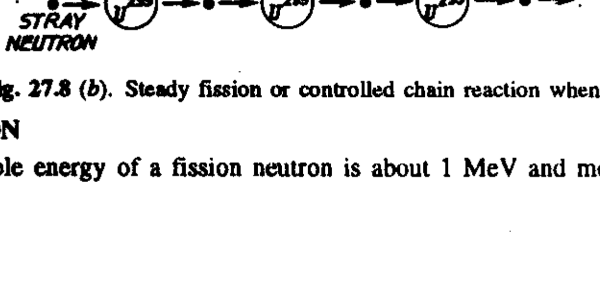


Fig. 27.8 (b). Steady fission or controlled chain reaction when $k = 1$.

27.17. MODERATION

The most probable energy of a fission neutron is about 1 MeV and most of the neutrons coming out of fission possess this energy. The fission threshold energy of $_{92}U^{238}$ is 1 MeV and, therefore, no fission is possible in $_{92}U^{238}$ by neutrons with energies lower than 1 MeV. If neutron coming out of fission with a high energy collides with any other nucleus, it is almost sure to be slowed down below the fission threshold of the $_{92}U^{238}$ without causing it to fission. Therefore, most of the fissioning process in nuclear reactor is by $_{92}U^{235}$, although a few of high energy neutrons may cause fission in $_{92}U^{238}$. The fission cross-section of $_{92}U^{235}$ is much higher in the thermal energy range and this is main reason of using $_{92}U^{235}$ in thermal reactors.

The better fission with much lower energy neutrons is possible in $_{92}U^{235}$ as its fission cross-section is higher in lower energy range. The possibility of interaction with $_{92}U^{235}$ nuclei increases and thus the reproduction factor is maintained above unity. Therefore, it is necessary to slow down the neutrons from fission energies (2 MeV) to thermal energies (0.25 eV). The process which does this is known as moderation.

Neutrons lose most of the energy when they collide with nuclei of low mass and least energy when they collide with nuclei of high mass. Therefore, presence of high mass in the reactor will require many collisions for neutrons to slow down from fission energies to thermal energies. The fission neutrons produced at high energy level (2 MeV) must be slowed down to thermal energy level (0.25 eV.) by elastic collisions without excessive absorption by any of the materials within the reactor in order to maintain a chain reaction.

In slowing down from 2 MeV to 0.25 eV, the neutron must pass through those energies at which absorption cross-section for $_{92}U^{238}$ is extremely high, that is the energies at which the resonance peaks occur as shown in Fig. 27.5 (*a*). It is obvious from the figure that $_{92}U^{238}$ is "eager" to capture neutrons in the energy range of 5 to 1000 eV. Therefore, we must see that (arrangement of fuel rods should be made) when the neutrons lose energy, they should pass through danger zone (5 to 1000 eV) as rapidly as possible without encountering any $_{92}U^{238}$ nuclei.

Thermalisation of Fast Neutrons. In elastic collisions, the kinetic energy and momentum are conserved. Any kinetic energy lost by neutron is gained by the nucleus that has been struck.

Consider a collision between a neutron of mass '*m*' and a velocity of v_1/sec with a target nucleus of mass *M* which is at rest. If the velocity of neutron is reduced from v_1 to v_2 after collision and the velocity of target nucleus is increased from 0 to *V* then according to Newton's law of conservation of energy and conservation of momentum, we can write

$$\frac{1}{2} m v_1^2 = \frac{1}{2} m v_2^2 + \frac{1}{2} M V^2 \qquad ...(27.15)$$

and

$$m v_1 = m v_2 + M V \qquad ...(27.16)$$

From the above two equations, we can write

$$\frac{m}{M}(v_1^2 - v_2^2) = V^2 \qquad ...(27.17)$$

$$\frac{m}{M}(v_1 - v_2) = V$$

$$\therefore \quad \left(\frac{m}{M}\right)^2 (v_1 - v_2)^2 = V^2 \qquad ...(27.18)$$

Dividing equation (27.18) by (27.17), we get

$$\frac{\left(\frac{m}{M}\right)^2 (v_1 - v_2)^2}{\frac{m}{M}(v_1^2 - v_2^2)} = 1.$$

$$\frac{m}{M} \cdot \frac{(v_1 - v_2)}{v_1 + v_2} = 1.$$

$$\frac{(v_1 - v_2)}{(v_1 + v_2')} = \frac{M}{m}.$$

Adding 1 once and subtracting 1 from the above equation, we get

$$\frac{2v_1}{v_1 + v_2} = \frac{M + m}{m} \qquad ...(27.19)$$

and

$$\frac{-2v_2}{v_1 + v_2} = \frac{M - m}{m} \qquad ...(27.20)$$

Dividing the equation (27.20) by (27.19) :

$$-\frac{v_2}{v_1} = \frac{M - m}{M + m}$$

$$\therefore \quad \frac{v_2^2}{v_1^2} = \left(\frac{M - m}{M + m}\right)^2$$

$$\therefore \quad \frac{\frac{1}{2} m v_2^2}{\frac{1}{2} m v_1^2} = \left(\frac{M - m}{M + m}\right)^2$$

$$\therefore \quad \frac{E_2}{E_1} = \frac{\frac{1}{2} m v_2^2}{\frac{1}{2} m v_1^2} = \left(\frac{M - m}{M + m}\right)^2$$

$$\therefore \quad \frac{E_2}{E_1} = \left(\frac{M - 1}{M + 1}\right)^2 \text{ as } m \approx 1 \text{ amu.} \qquad ...(27.21)$$

If the neutron collides with hydrogen nucleus for which

$$M \approx 1.$$

$$\therefore \quad \frac{E_2}{E_1} = 0$$

$$\therefore \quad E_2 = 0.$$

If the neutron collides with $_{92}U^{238}$ for which

$$M = 238.$$

$$\therefore \quad \frac{E_2}{E_1} = \left(\frac{237}{238}\right)^2 \approx 1$$

and there is practically no reduction in neutron energy.

It is obvious from the above analysis that if the reactor contains only U^{235} or U^{238}, it would require more than 2100 collisions to slow down the neutron from fission energy level (2 MeV) to thermal energy level (0.025 eV) necessary to continue the chain reaction. The energy lost by neutron in each collision would be very small and it would require many collisions as mentioned above for them to slow down, through the region of resonance peak of $_{92}U^{238}$. This would result in the capture of neutrons by $_{92}U^{238}$ neutron would be available to cause thermal fission.

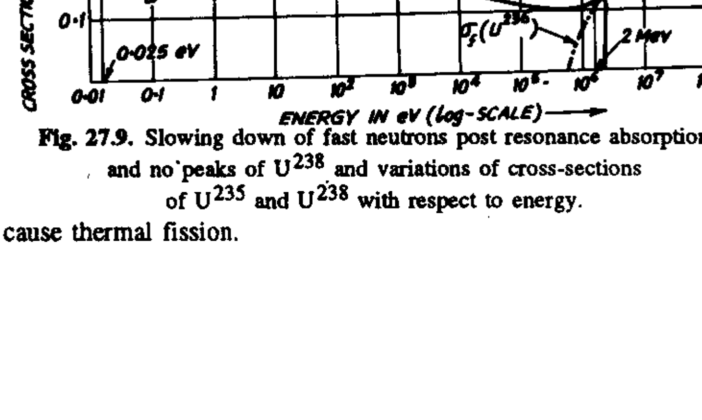


Fig. 27.9. Slowing down of fast neutrons post resonance absorption and no peaks of U^{238} and variations of cross-sections of U^{235} and U^{238} with respect to energy.

If on the other hand, a lighter material is used as moderator, fewer collisions would be required and the chances of slowing down the neutron to the thermal range would be greatly increased.

Therefore, it is obvious that the light materials like Hydrogen (H_2), Deuterium (Isotope of hydrogen), Helium (He), Lithium (Li), Beryllium (Be), Boron (Bo), Carbon (C), Nitrogen (N_2) and Oxygen (O_2) can be used as better moderators. The gases like H_2, He, N_2 and O_2 are not suitable as they have low densities and chances of collisions are less. The Hydrogen is available in ordinary water and heavy water, therefore, these (H_2O, D_2O) can be used as better moderators. Ordinary water is cheap in first cost but it has very high neutron absorption, therefore, it is suitable only with enriched uranium. Heavy water has low neutron absorption, therefore, it is ideal moderator but it is very costly (Rs. 1400 per kg).

The high slowing down nature of the moderator is not the only required property of the good moderator but it must possess high probability to scatter the neutrons (high scattering cross-section). The slowing down power is the product of log-decrement (higher decreasing fission cross-section with increasing energy of neutron as shown on Fig. (27.9) and scattering cross-section). The other important property of a good moderator is low absorption of neutrons and this is taken into account by a factor known as moderation ratio. The slowing down power and moderation ratio for few moderators are given below :

Moderator	H_2O	D_2O	Be	C
Slowin-down power (cm^{-1})	1.53	0.37	0.176	0.064
Moderating ratio	72	12000	159	170
Number of collisions needed for thermalization of 2 MeV Neutrons.	18 for H_2	25	86	114

It is obvious from the above table that the heavy water is far superior than ordinary water as moderator. This is because the absorption cross-section of D_2O is much lower and the heavy water is much more capable of slowing down neutrons without capturing them.

For moderation purposes, the uranium is not distributed uniformly with the moderator throughout the body of the pile but placed in the form of individual blocks separated by a moderator.

Given such an arrangement, the bulk of fast neutrons produced in fission will be slowed down to the energies less than 5 eV at a distance from the uranium pile and these neutrons will pass through the danger zone of energy of the ${}_{92}U^{238}$ and will be absorbed by ${}_{92}U^{235}$ which will cause the fission. Thus the most advantageous arrangement is not the mixing of uranium and moderator but the distribution of the uranium and moderator alternately in the form of lattice.

It can be concluded from the above discussion that the reproduction factor (K) can be maintained unity or more to sustain the chain reaction by one of the following methods :

1. If the ${}_{92}U^{235}$ percentage in uranium content is increased from 0.7% to 10%, then the number of neutrons captured by ${}_{92}U^{238}$ are reduced and probability of fission is increased. In this process, fast and slow neutrons are used to maintain the chain reaction.

The enrichment of ${}_{92}U^{235}$ is not possible by chemical means as ${}_{92}U^{235}$ cannot be separated from ${}_{92}U^{238}$ by chemical process. The gaseous diffusion process is used to enrich the ${}_{92}U^{235}$ which is very costly.

2. In this method, the fast neutrons are thermalised as mentioned earlier and the thermalised neutrons are capable to maintain the chain reaction with the neutral uranium having 0.7% fissible ${}_{92}U^{235}$. In this method, the danger of absorption of emitted neutrons by ${}_{92}U^{238}$ is reduced avoiding the resonance range of ${}_{92}U^{238}$ as mentioned earlier.

This discussion further indicates that with the use of one type of fuel, the use of particular moderator is essential. If the natural uranium is used as fuel, the heavy water must be used as moderator as the absorption of D_2O is very low compared with H_2O and its moderation ratio is very high.

The ordinary water can be used as moderator when enriched uranium is used as fuel. In this case even the moderation ratio of H_2O is low, the chain reaction is maintained by fast as well as slow neutrons, therefore, the effect of low moderation ratio is not very significant in this case.

27.18. FERTILE MATERIALS AND BREEDING

There are materials like $_{92}U^{238}$ and $_{90}Th^{232}$ which are not fissible but can be converted into fissile materials by the bombardment of neutrons. Such materials are known as fertile materials.

The Uranium 238 absorbs some of the neutrons and converts into $_{94}Pu^{239}$ which is a fissile material. The produced $_{94}Pu^{239}$ can be used as fissile material for power production if separated. The nuclear reactions during the conversion of $_{92}U^{238}$ to $_{94}Pu^{239}$ are listed below :

$$_{92}U^{238} + {_0n^1} \longrightarrow {_{92}U^{239}} + \gamma \text{ (half life of } _{92}U^{239} \text{ is 23.5 minutes)}$$

$$_{92}U^{239} \longrightarrow {_{93}Np^{239}} + \bar{\beta} \text{ (half life of } _{93}N^{239} \text{ is 2.3 days)}$$

$$_{93}Np^{239} \longrightarrow {_{94}Pu^{239}} + \bar{\beta} \text{ (half life of } _{94}Pu^{239} \text{ is 24000 years).}$$

The conversion of $_{90}Th^{232}$ into $_{92}U^{233}$ is brought about by surrounding a reactor core with $_{90}Th^{232}$ blanket and arranging for as many neutrons as possible to escape from the core and fall on the blanket. The nuclear reactions during the conversion of $_{90}Th^{232}$ to $_{92}U^{233}$ are given below :

$$_{90}Th^{232} + {_0n^1} \longrightarrow {_{90}Th^{233}} + \gamma \text{ (half life of } _{90}Th^{233} \text{ is 23.3 minutes).}$$

$$_{90}Th^{233} \longrightarrow {_{91}Pa^{233}} + \bar{\beta} \text{ (half life of } _{91}Pa^{233} \text{ is 27.4 days).}$$

$$_{91}Pa^{233} \longrightarrow {_{92}U^{233}} + \bar{\beta} \text{ (half life of } _{92}U^{233} \text{ is } 1.6 \times 10^5 \text{ years).}$$

There is surety of having more than one neutron per fission left over for conversion of fertile $_{90}Th^{232}$ to fissile $_{92}U^{233}$. This would result in the production of more fissile atoms from fertile atoms than fissile atoms consumed in reactor and effect is net gain of fissile material.

This process of converting more fertile material into fissile material in a reactor is known "Breeding". Its importance lies in consequence that all available fertile material can be converted into fissile material.

The fissile products ($_{94}Pu^{239}$ and $_{92}U^{233}$) produced from fertile materials and the fertile materials themselves, are long lived and can be stored without any loss.

EXERCISES

27.1. Distinguish between Atomic number and mass number.

27.2. What is amu ? What is its importance in nuclear physics ?

27.3. What do you understand by natural radioactivity ? Describe the different radioactive emissions and their effects on parent atom.

27.4. Define "half life', "mean life" and decay constant. What are their significances in nuclear engineering ? The half life of a radioactive element is 3.82 days, find its decay constant. What percentage of radioactive atoms originally present will decay in 30 days ?

27.5. What do you understand by terms "binding energy" and "mass defect' ? How they are related to each other ?

27.6. State the law of mass energy equivalence and calculate the energy in kW likely to be produced by one gram of matter taking light velocity as 3×10^8 m/sec.

27.7. What are the different types of nuclear reactions that take place ? Explain the significance of each in nuclear power generation.

27.8. Explain clearly the difference between "fast neutrons" and "thermal neutrons". Explain clearly why thermal neutrons can cause fission of $_{92}U^{235}$ but not of $_{92}U^{238}$.

27.9. Explain the phenomenon of resonance absorption. Why is it so important in reactor physics ?

27.10. What do you understand by nuclear fission ? What are the essential requirements to cause nuclear fission ?

27.11. What is chain reaction ? How it is maintained ? What is the difference between controlled and uncontrolled chain reaction ? Explain with neat sketches and with examples.

27.12. What is reproduction factor ? It is said that the reproduction factor of $_{92}U^{235}$ fission is 2.5. Justify the statement. Explain the terms critical, sub-critical and super-critical used in reactor physics.

27.13. What do you understand by moderation ? Why it is essential ?

27.14. What different moderators are used in practice ? What different properties make them suitable moderators ?

27.15. Explain clearly why thermal neutrons can cause fission of $_{92}U^{235}$ but not of $_{92}U^{238}$.

27.16. What do you understand by fertile material and breeding ? What is the importance of breeding in power engineering ?

28

Nuclear Reactors

28.1. INTRODUCTION

Mankind is searching tirelessly for new power sources as the conventional sources of power generation are limited. Moreover, the world demand for electric power is doubling after every decade due to booming increase in population. It is estimated that the present fossile fuels will not last more than few decades to supply the rapid increasing demand of electric power.

The discovery of the fission of uranium was of tremendous importance as it opened the prospects of using the energy stored in the atomic nucleus for the production of electric power. Presently, the nuclear energy enormously enlarged the world's power sources. The present estimates reveal that fissionable uranium alone contains far more energy than that all the world's reserves of coal and petroleum put together. This figure further increases several fold if one takes into account the possibility of obtaining nuclear fuel from thorium.

An unique feature of nuclear energy is an exceptionally high degree of concentration which exceeds by millions of times the concentration of energy in the conventional fossile fuels. For example, the energy of one kg of uranium is equivalent to about 20×10^6 kW-hrs of heat energy or burning of 2000 tons of high grade coal.

The first nuclear reactor was built under the strict security arrangements in the squash court under the University of Chicago's Stagg Field Stadium by Fermi and his collaborators. The Chicago pile diverged on 2nd December 1942 which was an epoch making event in the history of power development. A number of reactors were built following the success of the Chicago pile which was used for the production of plutonium fuel for military purposes.

World's first atomic power plant of 5 MW capacity for industrial purposes was commissioned in U.S.S.R. on June 27, 1954 and this date is regarded as the birth of new source of power generation. The next nuclear Calder Hall Power Plant in Cumberland was commissioned in October 1956 to feed the electricity to the national grid of U.S.A. Many nuclear power stations are now operational throughout the world and it is estimated that the major source of power generation by the end of twentieth century would be nuclear.

The nuclear power generation in India has just started and is steadily increasing. As mentioned earlier, among the developing countries, India is progressing well in accepting the nuclear energy for electric power production. As nuclear resources are concerned, India is one of the richest countries in the world. The total uranium in India is nearly 32000 tonnes and thorium is 500,000 tonnes which would be sufficient for thousands of years to fulfill the power needs of the country provided thorium is used for power generation in future.

28.2. GENERAL COMPONENTS OF NUCLEAR REACTOR

The nuclear reactor may be regarded as a substitute for the boiler fire box of steam plant or combustion chamber of a gas-turbine plant. The heat produced in the nuclear power plant is by fission whereas in steam and gas turbine plants, the heat is produced by combustion. The other cycle of operation and components required are exactly same either as steam plant if steam is generated by using the heat of fission or a gas turbine plant (closed or open type) if gas is heated by using the heat of fission. The steam or the gas may be used as working fluid in nuclear power plant. The nuclear power plant may be of steam driven turbine or gas driven turbine as per the choice of the fluid.

The general arrangement of Nuclear power plant with essential components using steam as working fluid is shown in Fig. 28.1.

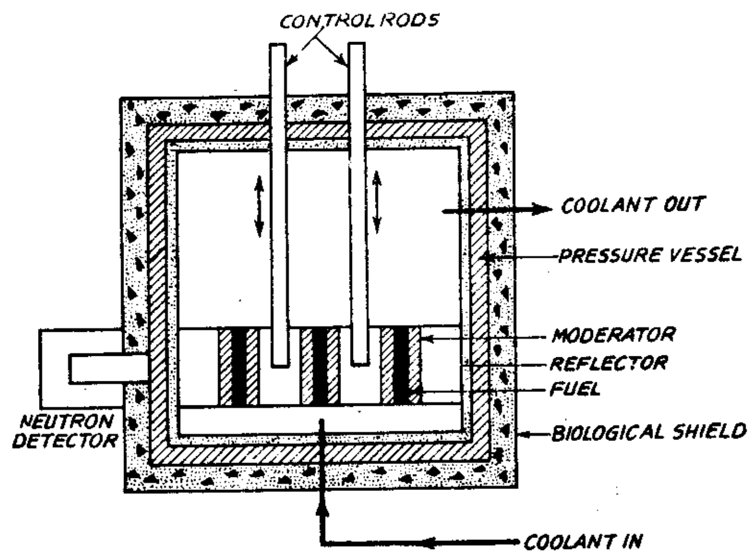


Fig. 28.1. Components of Nuclear Reactor.

1. Fuel. The nuclear fuels which are generally used in reactors are ${}_{92}U^{235}$, ${}_{94}Pu^{239}$ and ${}_{92}U^{233}$. Among the three, the ${}_{92}U^{235}$ is naturally available upto 0.7% in the uranium ore and the remaining is ${}_{92}U^{238}$. The other two fuels ${}_{94}Pu^{239}$ and ${}_{92}U^{233}$ are formed in the nuclear reactors during fission process from ${}_{92}U^{238}$ and ${}_{90}Th^{232}$ due to the absorption of neutron without fission.

The fuel is shaped and located in the reactor in such a manner that the heat production within the reactor is uniform. The fuel elements are designed taking into account the heat transfer, corrosion and structural strength.

In homogeneous reactors, the fuel and moderator are mixed to form a uniform mixture *i.e.* uranium and carbon and then it is used in the form of rods or plates in the reactor core. In heterogeneous reactors, the fuel is used in the form of rods or plates and moderator surrounds the fuel elements. This arrangement is commonly used in most of the reactors. The fuel rods are clad with aluminium, stainless steel or zirconium to prevent the oxidation of uranium.

2. Moderator. The moderator is a material which reduces the kinetic energy of fast neutron (1 MeV or 13200 km/sec) to slow neutron (0.25 eV or 2200 m/sec) and this is done in a fraction of second. The fission chain-reaction in the nuclear reactor is maintained due to slow neutrons when the ordinary uranium is used as fuel. The function of the moderator is to increase the probability of reaction. The slowing down of neutrons is effectively done by the light elements as H_2, D_2, N_2, O_2, C and Be as mentioned earlier. The graphite, heavy water or beryllium can be used as moderator with natural uranium. The ordinary water is used as moderator only with the enriched uranium for the reason mentioned earlier.

The desirable properties of a good-moderator are listed below :

1. It must be as light as possible, as slowing down action is more effective in elastic collision with light elements.

2. The moderator must be able to slow down the neutrons earliest possible but it must not absorb them (low absorption cross-section).

3. It must have resistance to corrosion as it has to work under high pressure and high temperature.

4. It must have good mechanability if the moderator is used in solid form.

5. The moderator must have high melting point if it is solid.

6. It should not be decomposed due to the nuclear radiation as well as it must have high chemical stability.

7. It must also have good conductivity as it is one of the essential properties for better heat transfer in the reactor core.

8. The material selected must be cheap and must be available in abundance and in pure from.

3. Reflector. It is always necessary to conserve the neutrons as much as possible in order to reduce the consumption of fissile material and to keep the size of the reactor small. The neutrons which are released in fission process can be absorbed by the fuel itself, moderator, coolant or structural materials. Some neutrons may escape from the core without absorption and will be lost for ever. To reduce the loss of escape, the reactor inner surface is surrounded by a material which reflects the escaping neutrons back into the core. This material is called the reflector.

The required properties of a good reflector are low absorption and high reflection for neutrons, high resistance to oxidation and irradiation as well as high radiation stability.

Many times the material used as moderator is also used as reflector because the moderating materials have good reflecting characteristics. The H_2O, D_2O and carbon are also used as reflectors. The amount of fissionable material required can be reduced with the use of good reflector.

It is necessary to provide some method of cooling the reflector as it gets heated due to collision of neutrons with its atoms.

4. Coolant. The main purpose of the coolant in the reactor is to transfer the heat produced inside the reactor. The same heat carried by the coolant is used in the heat exchanger for further utilization in the power generation.

Some of the desirable properties of a good coolant are listed below :

1. It must not absorb the neutrons.
2. It must have high chemical and radiation stability.
3. It must be non-corrosive.
4. It must have high boiling point (if liquid) and low melting point (if solid).
5. It must be non-oxidising and non-toxic.

The above-mentioned properties are essential to keep the reactor core in safe condition as well as for the better functioning of the coolant.

6. It must also have high density, low viscosity, high conductivity and high specific heat. These properties are essential for better heat transfer and low pumping power.

The water, heavy water, gas (He, CO_2), a metal in liquid form (Na) and an organic liquid are used as coolants.

The coolant not only carries large amounts of heat from the core but also keeps the fuel assemblies at a safe temperature to avoid their melting and destruction.

5. Control Rods. Some type of control is absolutely necessary to fulfill the following functions :

(*a*) To start the nuclear chain reaction when the reactor is started from cold.

(*b*) The chain reaction should be maintained at steady state condition (controlled chain reaction) at the required level.

(*c*) To shut down the reactor automatically under emergency condition, *i.e.* a pump circulating the coolant through the reactor fails.

The control is necessary to prevent the melting of fuel rods, disintegration of coolant and destruction of reactor as the amount of energy released is enormous.

The control of the chain reaction is possible either by removing the fuel rods or the neutrons continuing the chain reaction from the core of the reactor. It is always easy to get rid of the neutrons. It is generally done by inserting the control rods into the fuel tubes as shown in Fig. 28.1.

The materials used for control rods must have very high absorption capacity for neutrons. The common materials used for control rods are cadmium, boron or hafnium. The action of control rods can be very well compared with the action of the blotting paper which absorbs extra ink without spreading.

6. Shielding. The reactor is a source of intense radioactivity as mentioned earlier and these radiations are very harmful to the human life. The common radiations from the reactor are α-rays, β-rays, γ-rays and fast neutrons. To prevent the effects of these radiations on the human life, it is necessary to absorb them before emitting to the atmosphere.

Neutrons, γ-rays and all other radiations are effectively absorbed by the concrete and steel. The inner lining of the core is made of 50 to 60 cm thick steel plate and it is further thickened by few metres using concrete. The lining of steel plate absorbs these energies and becomes heated but prevents the adjacent wall of reactor vessel from becoming heated. The thermal shield (steel plate) is cooled by the circulation of water.

7. Reactor vessel. The reactor vessel encloses the reactor core, reflector and shield. It also provides the entrance and exit passages for directing the flow of coolant. The reactor vessel has to withstand the pressure as high as 200 bar or above. The holes at the top of the vessel are provided to insert the control rods. The reactor core (fuel and moderator assembly) is generally placed at the bottom of the vessel.

28.3. GENERAL PROBLEMS OF REACTOR OPERATION

During construction and operation of nuclear reactors, some special problems are to be faced which never occur with other types of power plants. Few of them are listed below :

1. The uranium is so abrasive that special cutting compounds are required to keep it below the ignition point while being machined to shape.

2. Another major problem faced with uranium fuel is, the uranium behaves like magnesium. If heated to a high temperature (to supply steam or gas at high temperature), it bursts into flame and if brought into contact with water they rapidly hydrolise, therefore, some form of protection has to be given to these materials.

3. Another major problem faced by the operators is the disposal of burnt products which are highly radioactive materials giving out large quantities of γ-rays. The γ-rays destroy all the living matters through which they pass. It is estimated that the waste products of 400 MW power station would be equivalent to something like 100 tons of radium daily and if this material is distributed uniformly over Great Britain, then the products would kill everyone within an area of about 250 square kilometres. The problem of disposal of nuclear ashes still remains unsolved. The safe disposal of waste poses a costly problem.

4. The charging of fuel in the reactor and removal of waste after few years involves special methods. Some of the fission fragments acquire high intensity radioctivity with a half life of number of years. The accumulated fission products absorb the neutrons after certain proportion of fuel has fissioned and chain reaction cannot be supported. These spent fuel assemblies must be replaced with fresh fuel. The radioactivity makes essential to perform this operation remotely to protect the operators.

5. As it is mentioned earlier, the fertile material (U^{238}) converts into fissile material (Pu^{239}) during reactor operation which can be used again as a fuel if separated from the fission products. This is necessary to recover the fissionable fuel. The separation of Pu^{239} from fission products involves some 30 chemical reactions and many other operations. Before taking the fission products to the chemical plants, it is necessary to keep them in deep pool of water for few weeks where the waste disposal is cooled as short-lived active isotopes dissipate enough heat energy. The special arrangement for the separation of fission fuel from waste products is extremely costly.

6. Fissioning of fuel deforms the fuel assembly by wrapping and hardening. This must not proceed to a point that will prevent the removing of the assembly from the reactor core. The neutrons displacement

in the materials used for cladding and supporting also weakens them and poses a major problem during operation.

7. The gas-cooled reactor still poses a difficult problem than water-cooled or metal-cooled reactors due to its low heat carrying capacity.

8. It is estimated that nearly 10^4 cu-m of air per minute at a speed of 20 to 25 m/sec in the main duct and hundreds of metres per second through the fuel assemblies is required to be circulated for a plant of 30 MW capacity. The power required for the blower to supply such large quantity of gas or air is 5000 kW.

9. One uncommon difficulty has been experienced in the lubrication. It has been observed that the effect of nuclear radiation on the lubricants and greases is to reduce the viscosity and penetration. Sometimes with heavy radiations, the solidification of liquid lubricant is also possible. The formed product also suffers a reduction in oxidation stability. Therefore, the lubricants used in nuclear power plant should be able to resist or withstnd the effects of nuclear radiation and maintain the lubrication properties. Many lubricants like synthetic aromatics can withstand the radiation effects upto 100 Mrad doses. The actual radiation levels are much lower than this, but in the event of any failure ; it may exceed the safe limit.

Generally turbine oil is capable of withstanding a high degree of irradiation before its performance characteristics deteriorate. The viscosity and oxidation characteristics of turbine oil do not deteriorate significantly until a dosage of atleast 100 Mrad is reached and this also applies to other characteristics such as corrosion, wear and load carrying properties. In some instances, it is possible to extend the lubricating property of oils upto 500—1000 Mrad doses by the use of anti-rads as "free radical scavengers" which prevent the formation of unsaturated compounds during reaction and aromatic compounds which absorb high energy radiation and presents crosslinking.

28.4. DIFFERENT TYPES OF REACTORS

The nuclear reactors are classified on the following basis :

1. On the basis of Neutron Energy

(*a*) **Fast Reactors.** In these reactors, the fission is effected by fast neutrons without any use of moderators.

(*b*) **Thermal Reactors.** In these reactors, the fast neutrons are slowed with the use of moderators. The slow neutrons are absorbed by the fissionable fuel and chain reaction is maintained. The moderator is most essential component in these reactors.

2. On the basis of Fuel used

(*a*) **Natural Fuel.** In this reactor, the natural uranium is used as fuel and generally heavy water or graphite is used as moderator.

(*b*) **Enriched Uranium.** In this reactor, the uranium used contains 5 to 10% U^{235} and ordinary water can be used as moderator.

3. On the basis of Moderator used

(*a*) Water moderated (*b*) Heavy water moderated (*c*) Graphite moderated (*d*) Beryllium moderated.

4. On the basis of Coolant Used. (*a*) Water cooled reactors (ordinary or heavy) (*b*) Gas cooled reactors (*c*) Liquid metal cooled reactors (*d*) Organic liquid cooled reactors.

Few of the reactors which are commonly used for power generation are discussed below.

28.4.1. Pressurised Water Reactor (PWR). The arrangement of pressurised water reactor is shown in Fig. 28.2. In its simplest form, a pressurised water reactor is a light water cooled and moderated reactor. It uses enriched uranium as fuel.

The pressurising tank included in the circuit maintains the constant pressure in the circuit throughout the load range. Electric heating coil in the pressuriser boils the water to form the steam which is collected in the dome as shown in figure and pressurises the entire coolant circuit before starting reactor. To reduce the pressure, water spray is used to condense the steam.

The fuel which is generally used is UO_2. The uranium oxide is highly resistant to irradiation damage and is very well adopted to the high burn-ups. It is also highly resistant to corrosion by high pressure water in the event of a break-up in the fuel cladding.

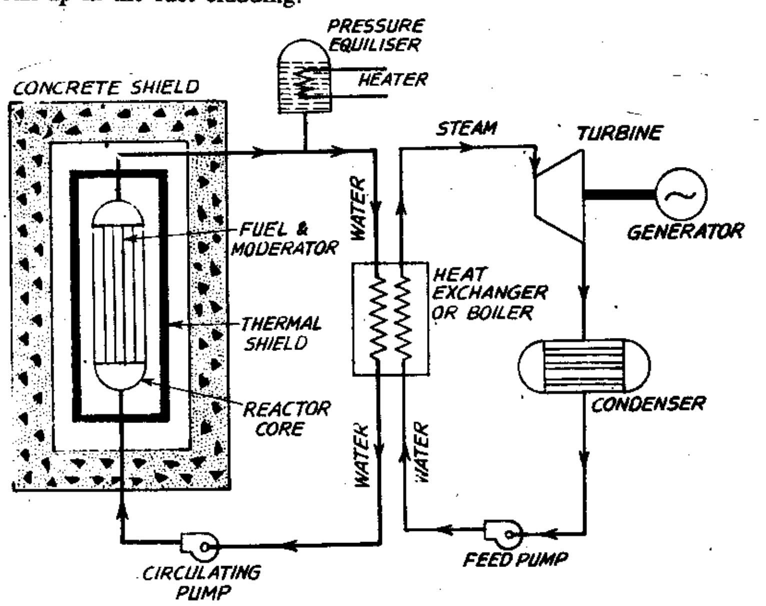


Fig. 28.2. Outline of pressurised water reactor nuclear power plant.

The water in the primary circuit gets heated by absorbing the fission energy in the reactor core and same energy is given in the heat exchanger to generate the steam. The water coming out of the heat exchanger is circulated by the pump to maintain the pressure in the circuit in the range of 100 to 130 bar.

The water becomes radioactive in passing through the reactor, therefore, the entire primary circuit including steam generator (heat exchanger) must be shielded to protect the operating persons. The radioactive coolant does not make the steam radioactive in the boiler. Beznau I is the first commercial nuclear station in Switzerland working on PWR.

The advantages and disadvantages of this type of reactor are listed below :

Advantages. 1. The water which is used as coolant, moderator and reflector is cheap in first cost and available in plenty.

2. The reactor is compact and higher power density (65 kW per litre) is the most distinguished feature of modern PWR cores. Fuel loading of 190 tons can be made in modern PWR.

3. The desirable characteristics of modern PWR are requirement of small number of control rods. The reactivity changes associated with fuel depletion and formation of fission products are compensated by neutron absorber and hardly less than 60 control rods are required in 1000 MW plant of this type.

4. The separation of secondary circuit from primary provides the capability to optimise the turbine cycle for the purpose of low heat rate supply. Normal turbine maintenance technique can be used as the steam is not contaminated by radiation.

5. It provides complete freedom to inspect and maintain the turbine, feed heaters and condenser during operation.

6. PWR allows to reduce the fuel cost extracting more energy per unit weight of fuel as PWR is ideally suited to the utilization of fuel designed for higher burn-ups.

7. The high negative temperature coefficient of a pressurised water reactor contributes greatly to its safe and stable operation and the safe regulation of the reactor is possible. The control rods normally need not be used for the load change. The rods are generally used only during start-up and to adjust for slow reactivity changes.

8. Another desirable characteristic of this reactor is power demand coefficient ; that is, when more power is demanded, the reactor responds to supply the same. The negative temperature coefficient makes this almost automatic. This positive power demand coefficient is not a characteristic of the other reactors. On the other hand, the power demand coefficient is negative for boiling water reactor.

9. As the enriched fuel is used, the reactor can be made more compact in size.

Disadvantages. 1. High primary circuit pressure requires strong pressure vessel and so high capital cost.

2. The thermodynamic efficiency of this plant is as low as 20% due to low pressure (60—70 bar) in the secondary circuit.

3. The corrosion problems are more severe as the corrosion is accelerated in the presence of high pressure, high temperature water. Therefore, use of stainless steel for vessel and cladding is necessary which further increases the cost of the plant.

4. It is necessary to shut down the reactor for fuel charging which requires a couple of months' time.

5. Fuel suffers radiation damage and, therefore, its reprocessing is difficulty.

Presently Westinghouse is offering PWR plants upto 1000 MW capacity and it is predicted that by the end of this decade, the largest unit of 2500 to 3000 MW capacity will be available of this type.

6. When the γ-radiations pass through the pressure vessel material, the heat is released and causes uneven heating. This uneven heating imposes thermal stressees in addition to the pressure stresses and makes the design difficult.

28.4.2. Boiling Water Reactor (BWR). In this type of reactor, enriched uranium is used as fuel and water is used as coolant, moderator and reflector like PWR except the steam is generated in the reactor itself instead of separate steam boiler. The arrangement of the single circuit system is shown in Fig. 28.3.

The majority of the power plants in U.S.A. are of PWR and BWR types as diffusion plants are established for enrichment of fuel during second world war.

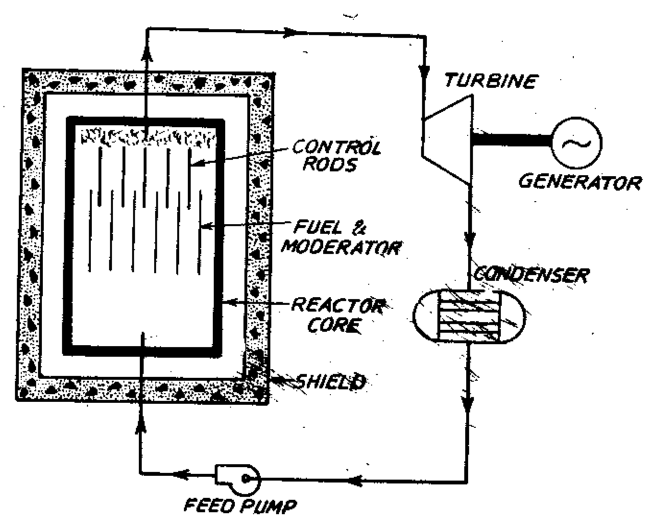


Fig. 28.3. Boiling water reactor power plant with single circuit.

The advantages and disadvantages of this type of reactor are listed below :

Advantages. 1. The pressure inside the reactor vessel is considcrably smaller than PWR as water is allowed to boil inside the reactor. Therefore, the reactor vessel can be much lighter than PWR and reduces the cost of pressure vessel considerably.

2. This reactor does not require steam generator, pressuriser, circulating pump and connecting piping. Therefore, the cost is further reduced.

3. The metal surface temperature is lower than PWR as boiling is allowed inside the reactor. The

temperature of the fuel surface is 240°C to get the steam of 30 bar in BWR whereas the temperature of fuel surface is 320°C and pressure is 115 bar in PWR to supply the steam to the turbine at 30 bar.

4. A boiling water reactor is more stable than PWR and much stable than any other type of reactor as increase in the reactivity increases the steam formation and reduces the hydrogen content per unit volume. The reactivity automatically is reduced as the vapour is not dense enough to moderate the neutrons effectively and reactor goes sub-critical. Therefore, BWR is known as self-controlled reactor.

5. The thermal efficiency of this reactor plant (30%) is considerably higher than PWR plant (20—22%).

Disadvantages. 1. The boiling water reactor cannot meet a sudden increase in power demand because as the power output of the turbine increases, the pressure in the reactor falls and the specific volume increases. The steam bubbles within the reactor would expand, expel the moderator and tend to shut down the reactor. Therefore, unless it is properly designed, the boiling water reactor might have negative power demand coefficient, so that when more power is demanded from the reactor, it may produce less.

2. The steam leaving the reactor is slightly radioactive with an half-life of the order of 15 minutes. Therefore, light shielding of turbine and piping is necessary.

3. The power density of this reactor is nearly 50% (33.6 kW/litre) of PWR, therefore, the size of the vessel will be considerably large compared with PWR.

4. The possibility of "burn-out" of fuel is more in this reactor than PWR as boiling of water on the surface of fuel is allowed. As the surface heat flux is increased beyond 500 kJ/cm^2/hr which is always possible in boiling region, the fuel outer surface is blanked with steam and reducees the heat transfer coefficient drastically and increases the fuel surface temperature above the safe limit causing burn out.

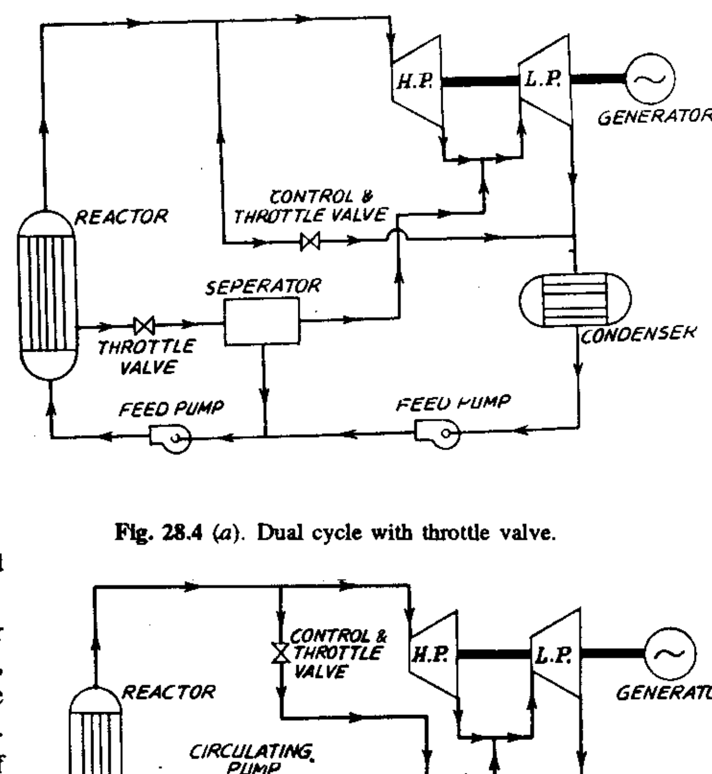


Fig. 28.4 (*a*). Dual cycle with throttle valve.

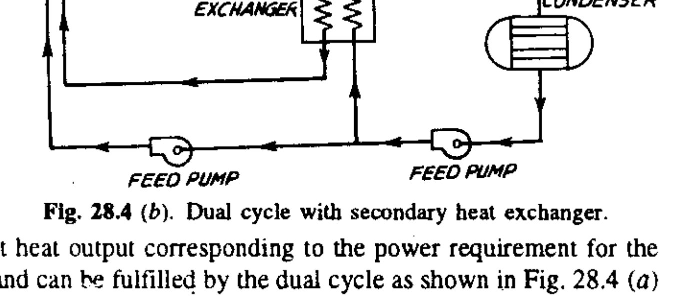


Fig. 28.4 (*b*). Dual cycle with secondary heat exchanger.

The single circuit cycle does not adjust heat output corresponding to the power requirement for the reason mentioned above. The change in demand can be fulfilled by the dual cycle as shown in Fig. 28.4 (*a*) or Fig. 28.4 (*b*).

In this arrangement, the high pressure steam from the reactor is directly supplied to the H.P. turbine and the low pressure steam formed either by throttling the high temperature water or generating the steam with the use of high temperature water, is fed to the L.P. turbine. This dual supply arrangement is used to govern the speed of the turbine by low pressure steam. This enables to adjust the increased power demand without shutting down the reactor which is common in single circuit.

28.4.3. Heavy Water Cooled and Moderated CANDU (CANadian Deuterium Uranium) Type Reactor. These reactors are more economical to those nations which do not produce enriched uranium as the enrichment of uranium is very costly. In this type of reactor, the natural uranium (0.7% U^{235}) is used as fuel and heavy water as moderator.

This type of reactor was first designed and developed in Canada. The first heavy water reactor in Canada using heavy water as coolant and moderator of 200 MW capacity with 29.1% thermal efficiency was established at Douglas (Ontario) known as Douglas power station. The arrangement of the different components of CANDU type reactor is shown in Fig. 28.5.

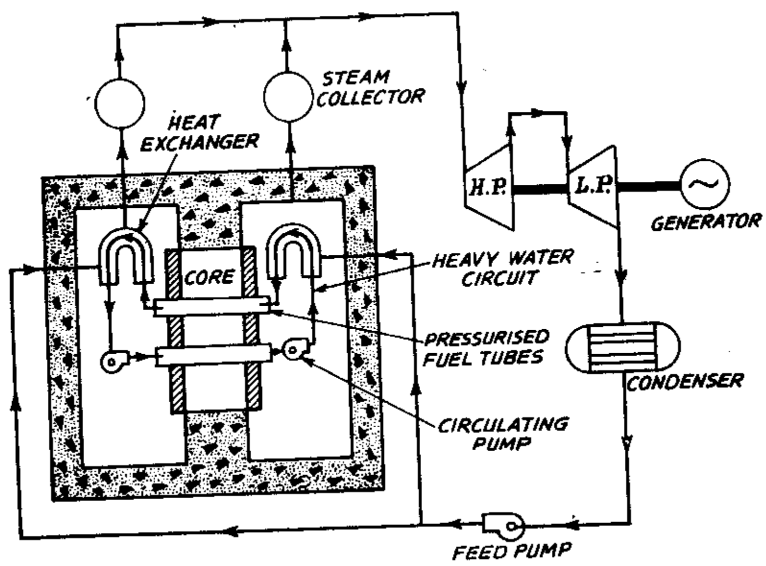


Fig. 28.5. Douglas-point Candu Type heavy water moderated and cooled nuclear reactor power plant.

The coolant heavy water is passed through the fuel pressure tubes and heat-exchanger. The heavy water is circulated in the primary circuit in the same way as with a PWR and the steam is raised in the secondary circuit transferring the heat in the heat exchanger to the ordinary water.

The control of the reactor is achieved by varying the moderator level in the reactor and, therefore, control rods are not required. For rapid shutdown purposes, the moderator can be dumped through a very large area into a tank provided below the reactor.

The advantages and disadvantages of this reactor are listed below :

Advantages. 1. The major advantage of this reactor is that the fuel need not be enriched.

2. The reactor vessel may be built to withstand low pressure compared with PWR and BWR. Only the fuel tubes are designed to withstand high pressure, therefore, the cost of the vessel is less.

3. No control rods are required, therefore, control is much easier than other types.

4. The moderator can be kept at low temperature which increases its effectiveness in slowing-down neutrons.

5. Heavy water being a very good moderator, this type of reactor has higher multiplication factor and low fuel consumption.

6. A shorter period is required for the site construction compared with PWR and BWR.

Disadvantages. 1. The cost of heavy water is extremely high (Rs. 500/kg).

2. The leakage is a major problem as there are two mechanically sealed closures per fuel channel. Canadian designs generally are based on recovering high proportion of heavy water leakage as absolute leak-tightness cannot be assured.

3. Very high standard of design, manufacture inspection and maintenance are required.

4. The power density is considerably low (9.7 kW/litre) compared with PWR and BWR, therefore, the reactor size is extremely large.

Even though CANDU-type reactors look promising in future, light water reactors all over the world proved more efficient than heavy water and in fact only 36 out of 529 power reactors in the world are based on heavy water.

28.4.4. Gas Cooled Reactors. The reactor is cooled by the gas and the heat carried by the gas from the reactor is either used for generating steam in the secondary circuit like PWR or it can be directly used as the working fluid in gas turbine plant.

(*a*) **Indirect circuits.** The carbon dioxide gas is used as primary coolant which in turn generates steam in the secondary circuit as shown in Fig. 28.6 (*a*). The corresponding temperature enthalpy diagram is shown in Fig. 28.6 (*b*).

The inlet temperature of the turbine is fixed by the reactor outlet temperature. The temperature potential causing the heat transfer from gas to steam in the reactor is very small (5 – 15°C). Therefore, it is necessary to balance the power required to circulate the gas at higher velocities against the increased capital costs for extended heat transfer surfaces.

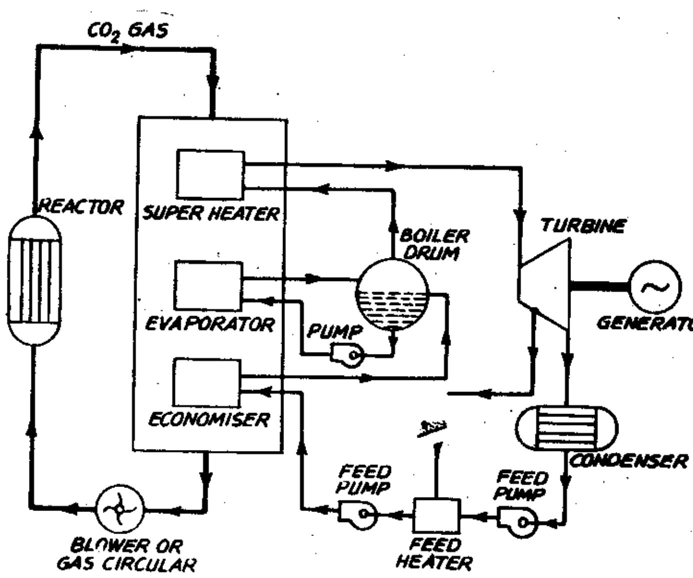


Fig. 28.6 (*a*). Single pressure steam cycle.

The choice of the feed water temperature for a given steam pressure affects the temperature of the gas at the inlet of the reactor. A "pinch point" occurs where the feed water temperature reaches the saturation

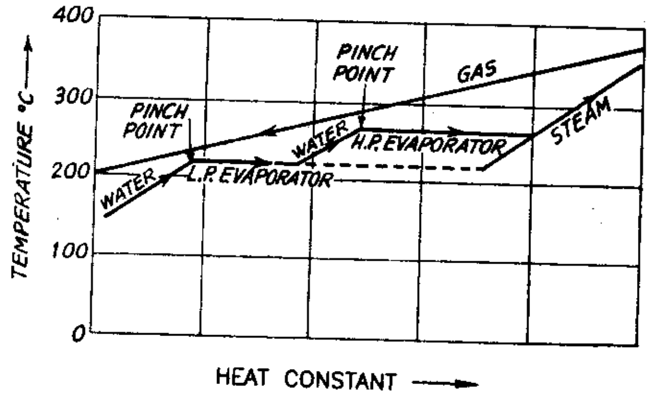


Fig. 28.6 (*b*). Heat diagrams for single pressure steam cycle.

temperature and approaches to within a few degrees of the gas temperature at pinch point. Therefore, the steam pressure in the cycle is limited to the pressure corresponding the saturation pressure at "pinch point". If less heat is added in the economiser and pinch point occurs at lower gas temperature (shown by dotted line), the steam is generated at lower pressure as the evaporation starts at lower saturation temperature.

The Hinkely point power station of 248 MW capacity with 26% efficiency at Somerset in England is an example of this type of power plant. The reactor consists of a spherical pressure vessel made of low carbon steel. The core is a twenty-four sided prism in which the fuel rods are arranged between graphite bricks which act as a moderator and reflector. The reactor is surrounded by a concrete shield about 3 metre thick which acts as a biological shield. The CO_2 gas is used as coolant, graphite as moderator and natural uranium as fuel.

The cycle efficiency is approximately proportional to the area under water-steam line on the temperature-enthalpy diagram. Therefore, the cycle efficiency can be increased by combining a number of steam cycles having different pressures and pinch points. The maximum number is limited to two due to increased complexity and capital cost. It is clear from Fig. 28.6 (*b*) that with dual pressure arrangement, a further "pinch point" is added and substantial quantity of steam is obtained at higher pressure. The quantities of steam generated at high and low pressure are in the proportion of 2 : 1 which is represented by the comparative lengths of the two horizontal lines. The dual pressure steam cycle was adopted for Calder Hall power plant in U.K. These types of reactors are very popular in U.K. as 26 reactors of this type were built by the end of 1971. This type of reactor is generally known with a name as "Magnox".

Advantages and Disadvantages of Gas Cooled Reactor

Advantages. (1) The greatest advantage of this is a simpler fuel processing. There is no problem of cladding the metallic fuel. Uranium carbide and graphite are merely ground together and reprocessed.

(2) The uranium carbide and graphite are able to resist high temperatures and, therefore, the problem of limiting the fuel element temperature is not as serious as in other reactors.

(3) The use of CO_2 as coolant completely eliminates the possibility of explosion in reactor which always presents in water-cooled plants

(4) There is no corrosion problem.

(5) It gives better neutron economy due to low parasitic absorption.

(6) Graphite remains stable under irradiation at high temperatures.

Disadvantages. (1) Power density is very low due to low heat transfer coefficient, therefore, large vessel is required.

(2) The loading of fuel is more elaborate and costly.

(3) The leakage of gas is a major problem if helium is used instead of CO_2.

(4) Coolant circulation requires much power as high as 10 to 20% of plant capacity whereas only 5% is required in water-cooled reactors.

(5) The critical mass is high therefore large amount of fuel loading is initially required.

(6) The control is more complicated due to low negative coefficient as helium does not absorb neutrons.

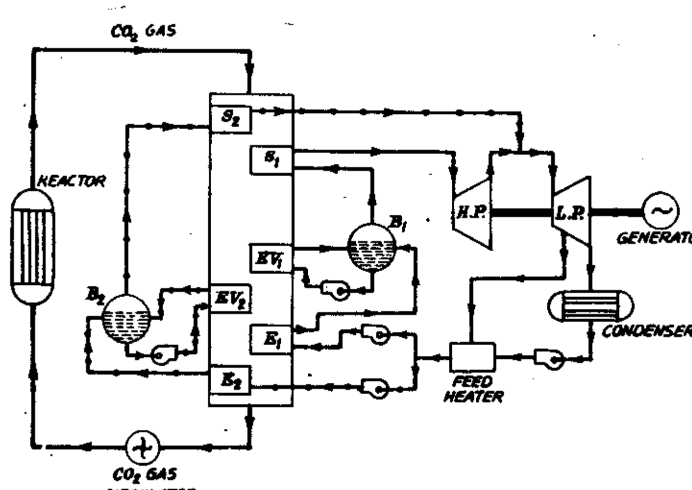
(*b*) **Direct Circuit.** The combination of direct closed-cycle gas turbine with a high temperature sector gives highest thermal efficiency and least impact on the environment compared with any nuclear system yet built.

Several small fossil-fired closed cycle gas turbine plants for the production of electricity have been built in different parts of the world for the advantages mentioned earlier. A major problem faced by the designer in the development of closed cycle gas turbine plants using fossil fuels is the transfer of the heat chemically produced to the working fluid. Another problem equally faced was corrosion from combustion products in fossile-fired operations. However, both these problems can be solved if the closed-cycle gas turbine plant is combined with high temperature reactor as heat source. In this combination, the entire primary circuit is operating in clean and non-corrosive working fluid.

Modern steam turbine power plants provide net efficiency of the order of 41-42% with single reheat cycle when the steam temperature is 540°C. This figure of temperature is considered as upper limit ; even still higher temperatures may give better efficiency because relatively cheap ferritic materials can be used for boiler tubes and steam piping. Exceeding this steam temperature rarely brings economic advantage.

In a helium cooled ractor with steam plant (as described earlier), the temperature of 650—700°C is required to produce a steam at 540°C in the boiler. The only advantage of using still higher temperature gas would be a reduction in boiler size, since the steam temperature and hence plant efficiency are held constant by material limitations. Therefore, the incentive to provide higher gas temperature in gas-cooled reactors using indirect steam cycle is rather limited.

To obtain the similar efficiency as indirect steam cycle, the helium-cooled reactor and helium operated closed cycle gas turbine requires reactor outlet temperature of about 850°C. The material problems in reactor and primary circuit components become more difficult if the higher gas temperature is employed. In fact



1—represents the high pressure circuit.

2—represents the low pressure circuit.

E—Economiser, *EV*—Evaporator.

S—Superheater, *B*—Boiler drum.

Fig. 28.7 (*a*). Dual pressure steam cycle.

in many power plants, the first principle of designing the plant is to produce the electrical energy at the cheapest cost and always highest efficiency does not mean the lowest generation cost. Therefore, the main objective of the power station design is to produce cheap electricity rather than to provide high efficiency at any cost.

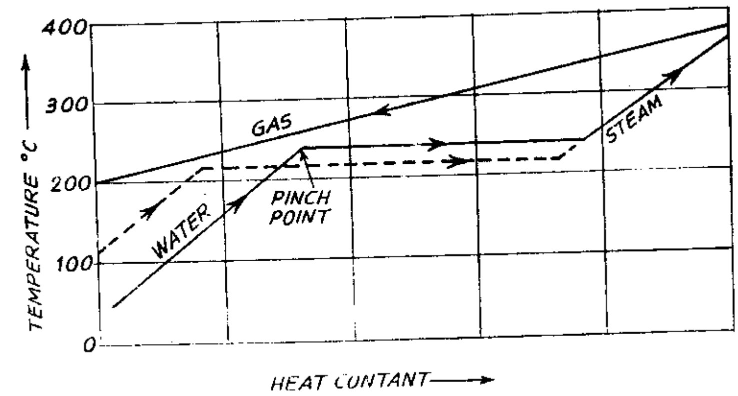


Fig. 28.7 (*b*). Heat diagram for dual pressure cycle.

From the point of view mentioned above, the helium cooled reactor closed cycle gas turbine plant is simpler in construction, low in first cost, and more compact than the corresponding indirect cycle system. Depending on the relationship of capital and operating cost characteristics of a particular reactor, the direct cycle gas turbine plant can produce cheaper power than indirect cycle steam plant even if the efficiency of the direct closed-cycle plant is considerably lower.

The gas turbine power plant is particularly attractive when the operating cost is only slightly affected by station efficiency. In fact, this feature is a particular characteristic of helium cooled, fast breeder reactors. Such type of reactor produces more fuel than it consumes and, therefore, it can be considered as a power plant as well as a source of fuel. Therefore, helium cooled fast reactors do not require high operating efficiencies. The attractiveness of using gas turbine thus rests on the ability to achieve a reduction in capital investment which is sufficient to produce low cost energy.

The direct-cycle gas turbine combined with nuclear reactor offers many advantages over the indirect cycle and a few of them are listed below :

1. More efficient use of high temperature heat, generated in reactor without the temperature degradation that necessarily occurs in indirect steam generation cycle.

2. The capital cost is considerably reduced as number of components of the system is reduced.

3. The power plant is more compact as high density of working fluid can be used in the closed cycle gas turbine plant.

4. The design of the gas turbine offers greater siting flexibility and fewer environmental effects from heat rejection, because it does not depend on the water cooling.

5. Another outstanding advantage is the high potential for further improvement in efficiency and capacity. The present design is based on a helium outlet temperature (out of reactor) of 816°C. It appears feasible that this temperature could be raised to 927°C in near future and this would result in an increase of plant efficiency to 41.2% and 28% increase in power output.

The arrangement of the components of direct closed cycle gas turbine power plant combined with nuclear reactor is shown in Fig. 28.8. Helium cooled reactors combined with direct cycle gas turbines represent a very promising solution for future electric power generation. The detailed developmeent is under way and the results are undoubtedly considered the biggest change in electric power generation since the invention of the steam turbine. It is predicted that many plants of this type would be available by the end of this century.

The first U.S. prototype HTGR of 40 MW capacity using He as coolant and using direct cycle for power generation was established at Peach Buttom in 1967. Two smaller capacity reactors of this type are also operating in England and Germany.

There are certain advantages in using gases like CO_2 and He as coolant instead of water. A few of them are listed below :

1. The gases do not react chemically with the structural material like water.

2. Gas can attain any temperature but the maximum temperature is decided by the permissible working temperature of the fuel. It can also be subjected to any pressure by maintaining the fixed temperature of the gas.

3. They also do not absorb neutrons while passing through the reactor and helps as moderator as well as reflector.

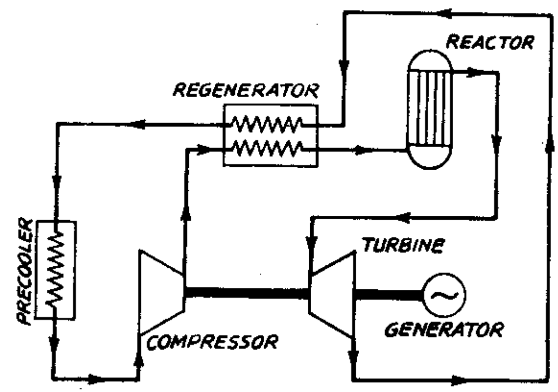


Fig. 28.8. Direct cycle high temperature reactor gas turbine power plant.

4. In water-cooled reactors, the reactivity varies with the amount of water. The reactivity can be increased or decreased with an increase or decrease of water amount. Therefore, the reactor is unstable. If by chance the leak develops in the fuel channel, the reactivity of water will increase causing the reactor to run away. With the use of gas coolant, the reactivity is not a function of gas content, therefore, the reeactor is more stable and inherently safe. Therefore, the leakage of gas will not affect the reactivity.

5. The gas coolant provides best neutron economy.

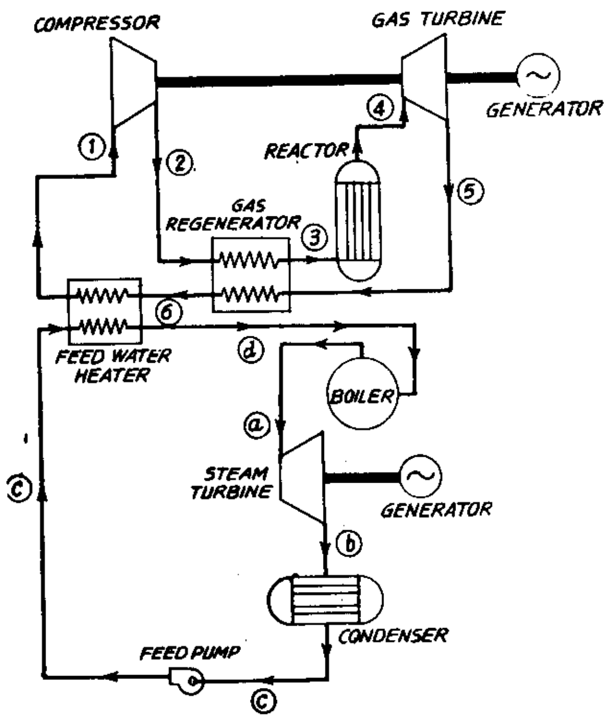
The only drawback of a gas-cooled reactor is poor heat transfer properties of the gases. The pumping power required in gas cooled reactor is as large as 20% of generated power against 5% with water-cooled reactor. Therefore, the overall efficiency of gas cooled reactor is lower than water-cooled reactor. The decrease in efficiency can be compensated with an increase in gas temperature. The overall generating cost of gas-cooled reactor plant is less for the reasons mentioned earlier.

The choice of the gas among carbon dioxide and helium depends upon the cost and thermodynamic properties of the gases. The helium is best suited as it has negligible capture cross-section to thermal neutrons and has high specific heat. But the cost of helium is extremely large compared with CO_2. On the other hand, CO_2 is cheap and its thermal neutron capture cross-section is also satisfactory.

Hydrogen can also be used, as the blower power is reduced to 10% of power required for CO_2 and it has better moderation capacity than CO_2 which further reduces the cost of graphite structure. But its use is objected above a certain temperature because it acts chemically with uranium fuel. The diffusion of hydrogen into the structural steel causes further troubles. Therefore, the hydrogen is not generally used for gas-cooled reactor.

(*c*) **Combined cycle.** A closed cycle helium gas turbine can be combined with a Rankine steam cycle to achieve an appreciable improvement in thermal efficiency. The thermal energy in the hot gases coming out from the regenerator for the helium cycle as described in Fig. 28.8. can be used for heating the feed water in the Rankine cycle. Such combination is thermodynamically advantageous and gives an overall higheer thermal efficiency than that of either the steam or gas cycle operating separately. The improvement in the thermal efficiency is mainly achieved through the partial or total use of waste heat from the gas cycle into a steam cycle.

With this combination, a simple Rankine cycle can be transferred into a more efficient cycle with the use of waste heat of helium in feed water heater. Usually, the steam is extracted from the turbine for feed heating in usual steam cycle.



The efficiency of such plant is as high as 50% is claimed. Such types of plants are yet to be designed and it is expected that these plants will be available on commercial basis in 1985.

It is believed that such combined cycle plant represents a successful and most economic solution for very large capacity power plant (2000 to 2500 MW).

The arrangement of different components of combined cycle is shown in Fig. 28.9 (*a*) and corresponding *T-s* diagram in Fig. 28.9 (*b*).

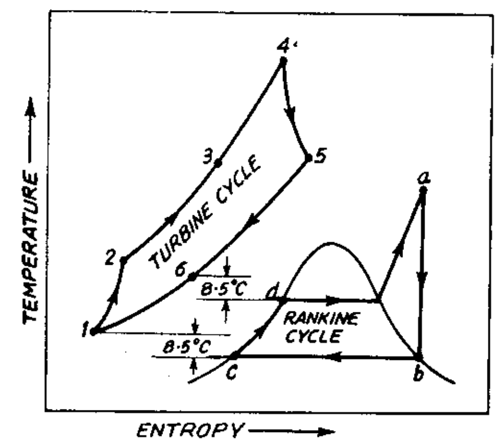


Fig. 28.9 (*a*). Combined-cycle arrangement. **Fig. 28.9** (*b*). *T-s* diagram of combined cycle.

28.4.5. Liquid Metal-Cooled Reactors. High temperatures are always desirable in all thermal plants to increase the efficiency of a power plant. The excellent heat transfer characteristics and heat transfer-capacity of liquid metals make them potentially attractive as reactor coolants. The liquid metal coolant can be circulated through the reactor at moderate pressures and yet have temperature as high as 540°C. Therefore, it is possible to achieve low cost power with metal cooled reactor. The arrangement of the components of a metal-cooled reactor is shown in Fig. 28.10 in which little enriched uranium is used as fuel, graphite as moderator and sodium as coolant.

The common metals which can be used as coolant are sodium and potassium. The eutectic alloy of these metals is also used for convenience, since it exists in liquid form at room temperature. However, sodium is best suited as coolant as it has low absorption cross-section (0.5 to 2 barns), low melting point (98°C), high boiling point (883°C), high specific heat (1.2 kJ/kg/°C), high thermal conductivity (120 times of water) and considerably cheap.

The advantages and disadvantags of metal-cooled reactors are listed below :

Advantages. 1. High temperatures can be achieved in the cycle and that means high thermal efficiency at low cost and low cost power.

2. The sodium as a coolant need not be pressurised.

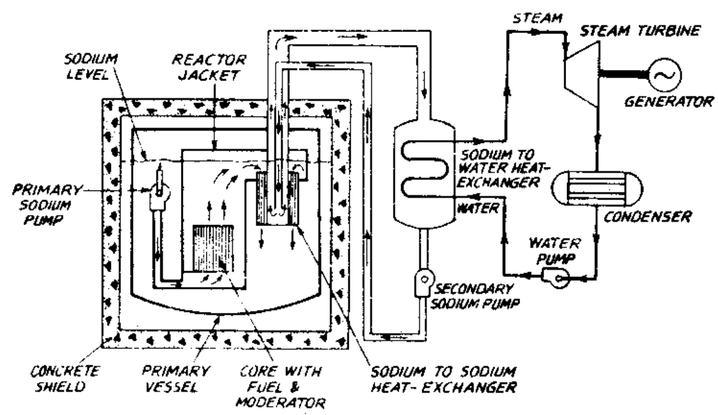


Fig. 28.10. Sodium cooled reactor.

3. The neutron absorption cross-section of sodium is low and, therefore, it is best suited to thermal reactor with slightly enriched fuel.

4. The low cost graphite moderator can be used as it can retain its mechanical strength and purity at high temperatures.

5. In other reactors, the loss of coolant increases the reactivity because of the removal of neutron absorber. In such cases, the reactor must be shut down by safety rod otherwise it may cause severe overheating. The metal-cooled reactor is more stable from safety point of view as the reactivity decreases with the rise in temperature.

6. The reactor size is comparatively small.

Disadvantages. (1) The neutron economy is reduced with an increase in temperature because, the high energy neutrons are subjected to resonance peaks when the moderator is hot and increases the chances of non-fissionable absorption of neutrons.

(2) It is always necessary to keep the graphite and sodium separate as porous graphite may absorb sodium and increase the absorption capacity of the graphite. The penetration of sodium between the layers of graphite can cause mechanical failure, therefore, each block of graphite is provided with a cladding which increases the construction cost.

(3) It is necessary to shield the primary and secondary cooling systems with concrete block as sodium becomes highly radio-active due to neutron bombardment.

(4) The leak of sodium is very dangerous compared with other coolants as it comes out of reactor in highly radioactive state.

(5) A precaution must be taken to see that sodium does not come in contact with water as it becomes highly reactive forming caustic soda with the evolution of heat.

The experimental sodium-cooled reactor plant of 5.7 MW capacity at Santa Surna, California (U.S.A.) is an example of this type.

28.4.6. Organic moderated and Cooled Reactors. Some hydrocarbons, notably, the polyphenyls, have been used as coolants instead of sodium. These substances are powerful moderators as they contain only hydrogen and carbon. Therefore, the use of separate moderator is not necessary. The organic liquids are used as coolant as well as moderator. Organic cooled and moderated reactor is shown in Fig. 28.11.

Organic coolants have the advantage of sodium that the vapour pressure is low at high useful temperatures and no heavy pressure is required in the reactor vessel. The additional merit with organic coolants is non-corrosive property and, therefore, low cost mild steel piping can be used in the system.

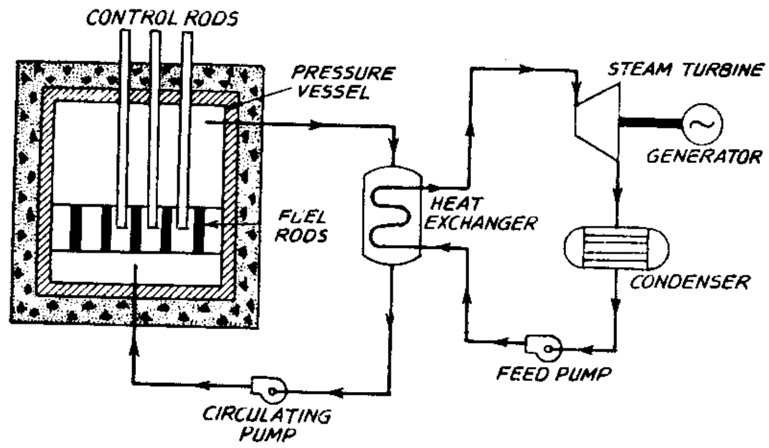


Fig. 28.11. Piqua-Organic cooled and moderated Reactor.

The advantages of this type of reactor are listed below.

Advantages : (1) They have supermoderating properties.

(2) Organic liquid can be used as coolant as well as moderator and, therefore, compact core design is possible.

(3) Core operating pressure is low as it has high temperature at low vapour pressure.

(4) A wide variety of fuels as uranium, uranium alloy, uranium oxide or uranium carbide can be used with organic coolants due to their excellent moderating properties.

(5) Low carbon steel can be used for vessel and piping as organic fluid does not corrode the tubes and, therefore, initial capital investment is low.

(6) The steam at high pressure and temperature can be produced in the heat exchanger as the steam generated in the system is subjected to a very small amount of radioactivity.

Disadvantages : 1. Organic fluids have poorer heat transfer properties than water.

2. Organic fluids suffer damage from radiation and heat, tending to leave slurry deposits, on fuel-can surfaces which further reduces the heat transfer.

3. The organic fluids are inflammable, therefore, special precautions must be taken to avoid the accidents.

The Piqua Nuclear Power Plant of 11.4 MW capacity with 25% overall efficiency near Piqua city, Ohio, in U.S.A. is an example of such power plant. It uses enriched uranium as fuel, a mixture of orthometa terhenyl and paratherphenyl as coolant and moderator and boron as control rods.

Such types of power plants are not yet commercially developed as proper technology is yet to develop. These types of plants have better future in coming years as they offer number of advantages over conventional reactors.

28.5. BREEDER REACTORS

Nuclear fuel burn-up and mechanism of breeding. It is already mentioned earlier that the U^{235} can be fissioned by slow neutrons ejecting an average of 2.5 neutrons to maintain further chain reaction. Among the 2.5 neutrons ejected, one neutron is used to maintain steady-state chain reaction. Among the remaining 1.5 neutrons, 0.9 neutron is captured by U^{238} and 0.6 neutron is absorbed by the coolant, moderator, structural material and the remaining is escaped from the reactor. The absorption of neutron by U^{238} converts U^{238} to Pu^{239} which is a fissile material as mentioned earlier. This (Pu^{239}) is a man-made fuel which can be used for further fission, therefore, it is known as "Secondary Fuel". Similarly, Th^{232} can also be converted into U^{233} which is also secondary fissionable fuel. The secondary fuels Pu^{239} and U^{233} can be fissioned by slow neutrons.

Thus, the burn up of primary nuclear fuel may be compensated to some extent by the production of secondary fuel. The compensation is measured by a factor known as "Conversion Factor". The conversion factor is a ratio of the number of secondary fuel atoms formed to the number of primary fuel atoms consumed. Generally uranium-graphite moderate reactor gives 0.9 conversion ratio. Therefore, the final effect with uranium-graphite reactor is the consumption of fuel (90% is recovered and 10% is consumed).

However, a reactor can be so designed that the conversion ratio will be unity or higher. This can be done by reducing the losses by absorption and capture. A reactor with a conversion ratio above unity is known as "*Breeder*". This fact is of great commercial importance.

It has been experienced that the amount of plutonium formed in the reactor or is less than amount of U^{235} used. There is thus no profit in their operation as breeding is concerned. These reactors are used to produce plutonium because Pu^{239} is easier to separate from U^{238} than in U^{235} as the two forms of uranium are isotopes. Since Pu^{239} is also fissionable, theoretically entire mass of U^{238} can be converted into fissionable Pu^{239}. This type of breeding would bring the cost of fuel to the cost of natural uranium (Rs. 630 per kg) instead of high cost of U^{235} (Rs. 400,000 per kg). In practice, any reactor that uses natural uranium as fuel produces plutonium which either is reused in same reactor or is separated from fission product and can be used elsewhere. But this is an incidental by-product and inefficient process of conversion.

A much larger prospect is offered by breeding fuel when the conversion factor is greater than one. In breeding process, the objective is not only to replace some of the fuel burnt but to produce more fuel than is used which can be separated and sold as a profit item. This attractive possibility has motivated the research in this field throughout the world.

At first glance, it might appear that a nuclear reactor with a breeding ratio equal to unity or higher could operate indefinitely provided the spent fertile material (U^{238} or Th^{232}) is replenished. But this is not so in practice. Because uranium fission products are accumulated during reactor operation and many of these are strong absorbers of neutrons, therefore, the fraction of unproductively absorbed neutrons increases with time and reactivity gradually falls bringing the chain reaction (reactor) to a halt. This circumstance imposes serious limitation on maximum possible fuel burn-up and on continuous reactor operation.

As the reactor continues operating, U^{235} is used up and its concentration reduces. At the same time, neutron absorption by U^{238} results in the production of Pu^{239}. The concentration of Pu^{239} increases continuously at a rate equal to that of the decrease in U^{235} concentration, then it becomes slower because the formed Pu^{239} also takes part in the fission. Finally, the concentration of Pu^{239} reaches a saturation point.

The average number of thermal neutrons emitted with U^{235} as fuel is 2.12. With one being used to maintain the chain reaction only 1.12 remain for breeding. This is not enough for breeding because the margin of 0.12 neutron is not enough to allow for neutron losses. With plutonium as fuel, the situation is even worse as the number of neutrons at thermal energy level is only 1.94. But situation is different if neutrons are used not at their energy level but at their original high velocity energy level. With fast neutrons, the chances of fission are less and chances of neutron capture to change U^{238} to Pu^{239} or Th^{232} or U^{233} are relatively greater as their absorption cross-sections (U^{238} and Th^{232}) are considerably larger at higher neutron energy level. Therefore, the breeding possibility is considerably higher with fast neutrons and, therefore, breeder reactors are necessarily fast reactors. They do not use the moderators to slow down the neutrons to thermal speed. The fuel used must be necessarily enriched uranium or plutonium.

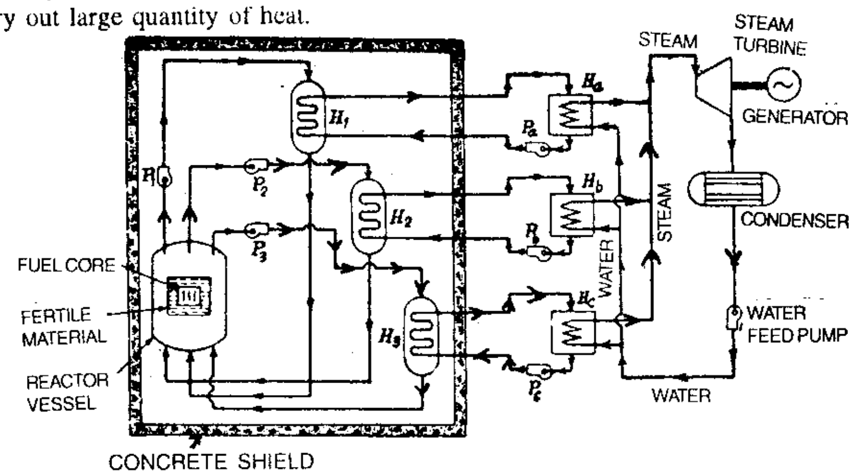
Doubling time of the system. The accumulating rate of fissionable fuel in breeder reactors is expressed in 'doubling time'. It is the time needed for the initial fissionable material to be doubled. The effectiveness of the breeding reactors is best characterised by a term called "System Doubling Time". The typical system doubling times for helium-cooled and sodium-cooled fast breeder reactors are 10 and 20 years respectively.

Fast Breeder Reactor. In fast breeder reactor, an enriched uranium or *plutonium (upto 10%) is kept in the casing without moderator. The casing is surrounded by fairly thick blanket of depleted fertile uranium. The ejected excess-neutrons are absorbed by the fertile blanket and it converts into fissile material. The heat produced in the reactor core is carried by liquid metal. The arrangement of Fermi fast breeder reactor is shown in Fig. 28.12.

The Fermi-fast breeder reactor located at Lagoona Beach, Detroit, Michigan in U.S.A. is first fast breeder reactor of the world. Enriched uranium was used as fissile fuel and depleted uranium as breeding material. The total power output was 94 MW electrical at an efficiency of 31.3%.

The major advantage of fast breeder is at high energies, the structural materials of the reactor do not absorb neutron, therefore, a wide choice of constructional materials is possible.

The major difficulty is to remove the large quantities of heat from the core as the power density is as high as 430 kW per litre of core volume which is 40 times greater than Candu type, 13 times greater than BWR and 200 times greater than gas-cooled reactor. Therefore, special coolants and special arrangements are necessary to carry out large quantity of heat.



P_1, P_2, P_3 → Primary circuit sodium circulating pumps.
P_a, P_b, P_c → Secondary circuit sodium circulating pumps.
H_1, H_2, H_3 → Primary sodium to sodium heat exchanger.
H_a, H_b, H_c → Secondary sodium to water heat exchangers.

Fig. 28.12. Fermi-Fast breeder reactor.

The Thorium Breeder. The breeding of thorium is more important than uranium as India is concerned because she has got large sources of thorium. The breeding of thorium is becoming increasingly important than uranium as it steadily converts to fissionable U^{233} when exposed to neutrons ejected from U^{235} in the reactor. The fissionable U^{233} is more attractive fuel than either uranium or plutonium because it emits 2.31 thermal neutrons on fission against 2.2 for U^{235} and 1.94 for Pu^{239} which gives an ample margin for breeding. Since thorium is efficiently converted into U^{233} by slow thermal neutrons, the prospects of breeding a respectable excess of fuel during power production are excellent.

Such type of breeding reactor is established at Oak Ridge National Laboratory with 10 MW capacity to find out breeding potentialities of thorium loaded reactor.

Choice of coolant for fast breeder Reactor. Three coolants have been considered for fast breeder reactors, liquid metal (Na or Nak), helium (He) and carbon dioxide (CO_2). Particularly sodium has several unique advantages over the other coolants as listed below :

*To use Pu^{239} is more preferable and economical as it works better for breeding by ejecting average 3 fast neutrons instead of 2.5 with U^{233} per neutron consumed.

(1) It has good heat transfer properties at high temperature and low pressure.

(2) It has good emergency cooling characteristics.

(3) It does not react on any of the structural materials used in primary circuits.

(4) It has very low absorption cross-sectional area.

Very recently, the gas-cooled fast breeder reactor concept is being promoted by some designers on the grounds that the sodium has serious disadvantages as chemical reactivity with water and air, highly induced radioactivity and very expensive engineering.

The compact design of secondary heat transfer circuit (heat transfer from sodium to water) is required to avoid the possibility of steam entering into the primary circuit and reacting with radioactive sodium.

The helium offers some special advantages as coolant in fast breeder reactor over sodium as listed below.

(1) The metallurgical and safety problems with the use of helium as coolant are much less severe.

(2) Helium is chemically inert, does not become radioactive, does not change the phase and does not degrade the neutron spectrum, thus leading to high conversion factor and negligible void reactivity coefficient.

(3) Heat transfer characteristics of helium are as superior as sodium and it has been found that the surface heat transfer coefficient of the fuel with helium can be increased significantly (> 2) by artificial roughening of the fuel rod surfaces.

(4) The major advantage of helium-cooled fast breeder reactor is low doubling time (10-12 years) compared with sodium-cooled reactors (20 years).

The only drawback of helium-cooled fast breeder reactor is required pressurization (70 to 80 bar) of the reactor vessel.

It is predicted that helium-cooled fast breeder reactors will be more promising power generators in future.

Advantages and Disadvantages of Breeder Reactor

Advantages : (1) It does not require moderator.

(2) It gives high power density than any other reactor, therefore, small core is sufficient.

(3) High breeding $[(k - 1) > 1]$ is possible.

(4) The operation of the reactor is not limited by Xe poisoning.

(5) High burn-up of fuel is achievable.

6. The parasite absorption of neutrons is low.

Disadvantages : (1) The specific power of the reactor is low.

(2) It requires highly enriched (15%) fuel.

(3) The control is difficult and expensive as neutron flux is high and neutron lifetime is short.

(4) Safety must be provided against melt-down.

(5) The handling of sodium is major problem as it becomes excessively hot and radioactive.

The fast breeder test reactor (FBTR) of 40 MW thermal and 14 MWe attained its criticality on 18 Oct. 1985 at Kalpakkam. India is the first developing country and seventh in the world to operate FBR.

With commissioning of FBR, India has reached the beginning of II-phase of the nuclear power programme for construction of FBRs which will produce more fuel than they consume while supplying electricity to national grid.

Fast Breeder Reactors for India (FBR)

This technology is not adopted by US, UK and Germany as these countries are saturated with energy generation as per capita consumption is as high as 10,000 units against 350 units in India. In these countries, interest in FBR was given an impetus with the Middle East oil crisis in 1970s when both Oil and Uranium prices increased *sevenfold.* Presently, the Scenario is changed. Uranium prices are back to old level, North

Sea oil has become a new source in Europe. In the US, a new gas sources have been discovered. Germany has a new gas pipeline from Russia. Therefore, they have no need to recycle plutonium in the fast breeders. The plutonium is a dual fuel as it is used in nuclear bomb as well as for power generation in FBRs. Therefore there is also the fear of proliferation because FBR produces more plutonium than it consumes.

But, with fast breeder programme like India, one can use the excess plutonium for power generation. That is the reason, we talk about "Double Time" (the time required to build one more reactor using the excess fuel produced from an operating FBR).

The inevitability of FBR in India arises from their resources utilization capacity more than from their growth capacity. By the use of FBRs, the utilization of Uranium can reach 60 to 80% as compared to less than 11% with LWRs and PHWRs as shown in Fig. 28.13 (*a*). In Fig. 28.13 (*b*), the potential of different energy sources in India is indicated. The power from coal has many environmental problems. And therefore, the projected increase in electricity generation which is expected to be between 300 – 500 GW per year by 2020 can be sustained for 50 years by the FBR route only. The U-233 (breeded from Th-232) breeder option provides a power generation of 500 GW per year which can be sustained for 400 years.

The stakes in breeder technology for India are high as it has potential to ensure energy independence for 400 years. If our energy independence in the years after 2020 is going to depend upon fast breeders, we must build the prototype to get the full benefits because it takes minimum two decades to generate the required plutonium.

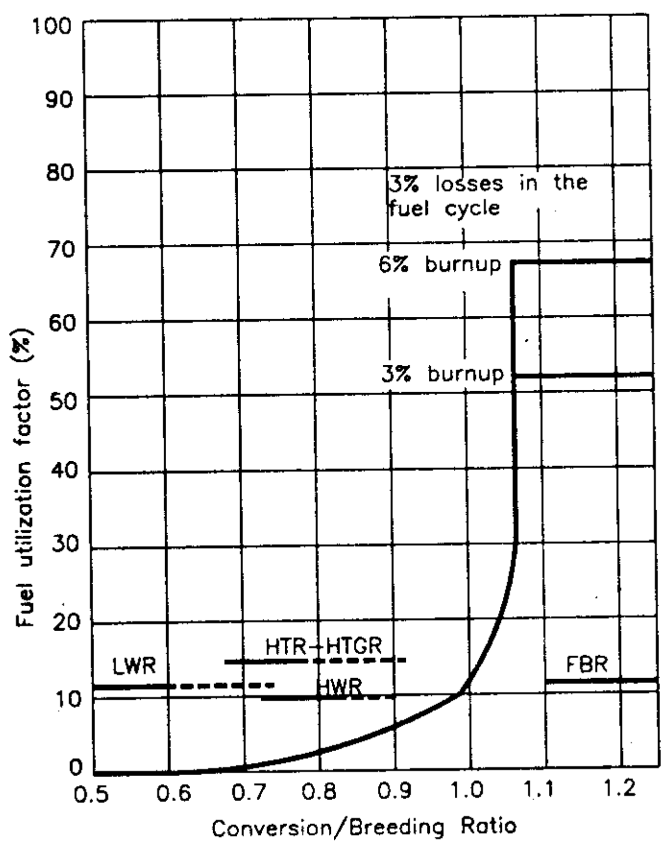


Fig. 28.13 (*a*). Utilisation of Uranium as a function of conversion/breeding ratio.

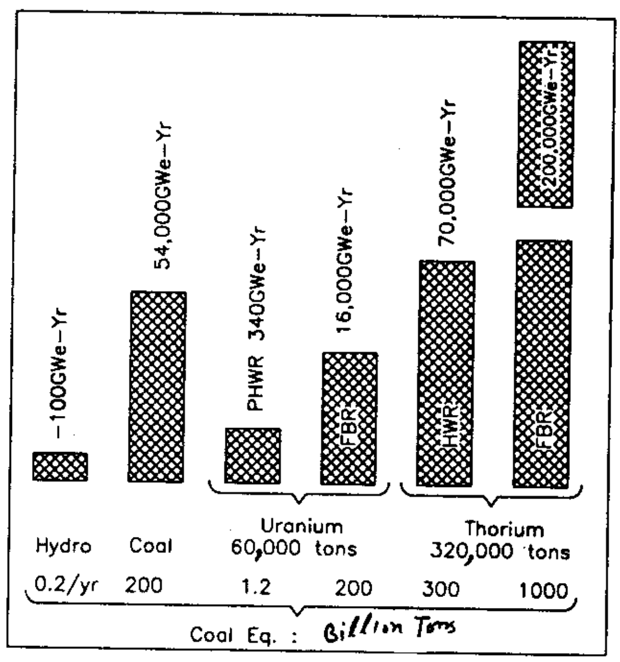


Fig. 28.13 (*b*). Comparison of different energy resources for electricity generation in India.

India needs FBRs today, the rest of the world will need it tomorrow. There is no alternative in future to provide the required power need except the FBRs have to come back. Fig. 28.13 (*c*) shows world fixed energy resources in TW/year and it is clear that a total nuclear generation of 1 TW/year can be sustained without fuel recycling only for 44 years. The sodium cooled FBRs operating on recycling principle only can provide energy to meet the demands that are likely to arise by 2050 and beyond in the whole world.

Also, the question is not whether the world needs FBRs or not, but when ? It is inevitable that as the coal resources in the world get depleted and the environmental concern for CO_2 emission becomes compelling, there will be a return of the nuclear power and as the uranium prices than are bound to go up, one day not very soon into the 21st century, FBRs would become a necessity for the whole world.

In India, 40 MW FBR was commissioned in Oct. 1985. With unique experience in developing 40 MW FBR using carbide fuel and sodium coolent, 500 MWe prototype FBR has been undertaken. It is also planned to start the construction of FBR of 500 MWe in April 2001 and the project will be completed by 2010. After successful commissioning of 500 MWe FBR, it is proposed to construct four 500 MWe FBRs at a suitable site in a phased manner.

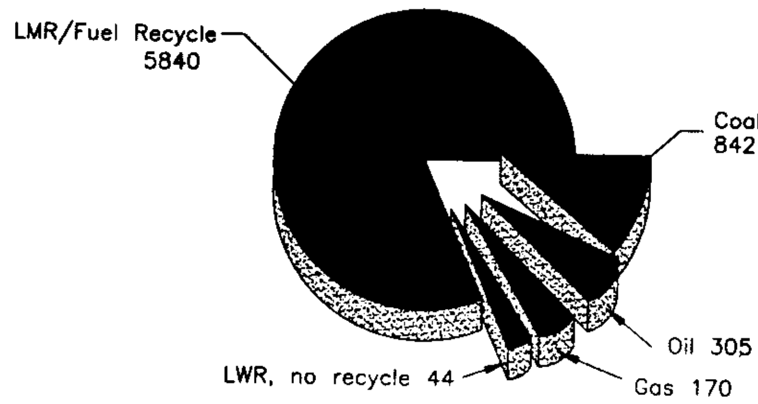


Fig. 28.13 (*c*). World fixed energy resources (TWy).

28.6. REACTOR CONTAINMENT DESIGN

It has been mentioned earlier that there is loss of neutrons due to the surface leakage and the escape loss is directly proportional to the reactor surface.

The number of neutrons produced is directly proportional to the volume of the reactor core. The volume of core increases as the ratio of surface area to volume decreases. Therefore, to minimise the percentage leakage of neutrons from the core, the more number of neutrons are to be produced, *i.e.* the core size must be increased.

The condition required to maintain the chain reaction in the reactor is given by

$$K_e = K \cdot P = 1$$

where K_e = Efficiency multiplication factor

K = Multiplication factor

P = Probability of non-escaping of the neutrons from the reactor.

To maintain the chain reaction, the value of K_e must be equal to unity. The size of the reactor which makes $P \cdot K = 1$ is known as "critical size". It is possible to reduce the critical size by increasing the value of K using enriched uranium. The value of P decreases as K increases. The second method is by using a spherical core as spherical core gives minimum surface to volume ratio. The spherical core needs less fuel than any other type of the core for the same power production. This is the main reason to use spherical type reactor.

28.7. LOCATION OF NUCLEAR POWER PLANT

The detailed investigations for suitable site for nuclear power plant require geological, meteorological, hydrological, topographical, special transport and radiological investigations. It is not the purpose to give details of these investigations here but to list out the factors which control the site selection for the power plant.

1. Availability of Cooling Water. The plant requires huge quantity of cooling water, therefore, it must be located near a river, lake or sea so that ample quantity of water will be available.

2. Transportation Facilities. The power plant should be nearer to the transport facilities as much as possible as heavy equipments are to be transported to the site.

3. Distance from Load Centre. It should be located near the load centre. This will reduce the power loss as well as the cost of long transmission lines. There is wider choice for suitable location of nuclear plant in comparison to hydro-electric stations. The transmission lines are considerably long in case of hydro-stations as their location is only dependent on the availability of water and head. Therefore, there is no choice for hydro-plant as load centre is concerned.

4. Safety. The power plant should be located at a reasonable distance from the populated area to avoid the radiation hazards.

5. Radioactive Waste Disposal Facility. The wastes of the nuclear plant are highly radioactive and therefore sufficient space must be available near the plant for short time storage as well as for long-time burial of the radioactive waste.

6. Foundation Requirement. The site selected for the plant must be suitable for foundation and should be strong enough to support the reactor and auxiliary equipments.

28.8. NUCLEAR POWER STATIONS IN INDIA

A. Tarapur Atomic Power Station. The Tarapur power plant is India's first atomic power plant situated 65 miles north of Mumbai. The site at Tarapur on the Arabian Sea coast was selected Aug. 1960 by the Atomic Energy Commission after detailed investigations of various locations to determine the availability of adequate cooling water, foundation, health and safety conditions, power transmission possibilities, and other facilities as transport and so on. It is one of the largest power stations in Asia.

It consists of two BWRs of equal capacity with total generating capacity of 380 MWe. The choice of the boiling water reactor which uses enriched uranium in preference to the natural uranium reactor has been determined by financial coniderations. The cost quoted by the British for building a natural uranium plant was much higher than the American quotation for enriched uranium plant. The other cause of selecting BWR was the availability of loan from U.S.A. on extremely favourable basis.

Tarapur power plant occupies nearly 1270 acres which includes a township, rail-siding, and air-strip. The cost of the total plant was Rs. 65 crores (on the basis of price line in 1964), out of that Rs. 45.25 crores was financed by U.S. Government on repay basis within 40 years.

The total charge of fuel required to start both reactors was about 80 tonnes. No additional fuel was supplied for the first $2\frac{1}{2}$ years. It is also estimated that certain amount of replacement is necessary periodically after $2\frac{1}{2}$ years. It is also estimated that the average replacement of uranium required is 22 tonnes per year for both reactors which is equivalent to 3-full train load of coal every day. The U.S. government has undertaken to supply fuel requirements covering the whole life of the plant—about 30 years.

The major problem faced by the engineers is the transport of heavy equipments. The weight of each pressure vessel was 200 tons having a diameter 3.6 metres and height 15.9 metres. The turbine generator sets were also equally heavy. A special jetty was built to receive these heavy equipments which were brought directly to the Tarapur site by a special ship. A joint crane was built on the top of the building to lift the equipments and place them inside the building at proper locations.

The thermodynamic cycle used at Tarapur atomic plant is shown in Fig. 28.14. Water boils in the reactor vessel to produce the steam which passes through steam separator and dryer as shown in Fig. 28.14 and then supplied to H.P. turbine. The steam from the turbines is condensed using sea water. The condensate is divided into two streams, one is returned to reactor and other is passed through secondary steam generator as shown in figure. The low pressure steam generated in the secondary generator is admitted to a lower stage in the H.P. turbine.

The dual cycle of primary and secondary steam permits operation of turbine to follow the system load demands automatically over a reasonable load change.

The cooling water (240×10^7 kg per day) for condensers is drawn from Arabian Sea through an intake structure from an open cut channel extending to about 910 metres into the sea. Two discharge

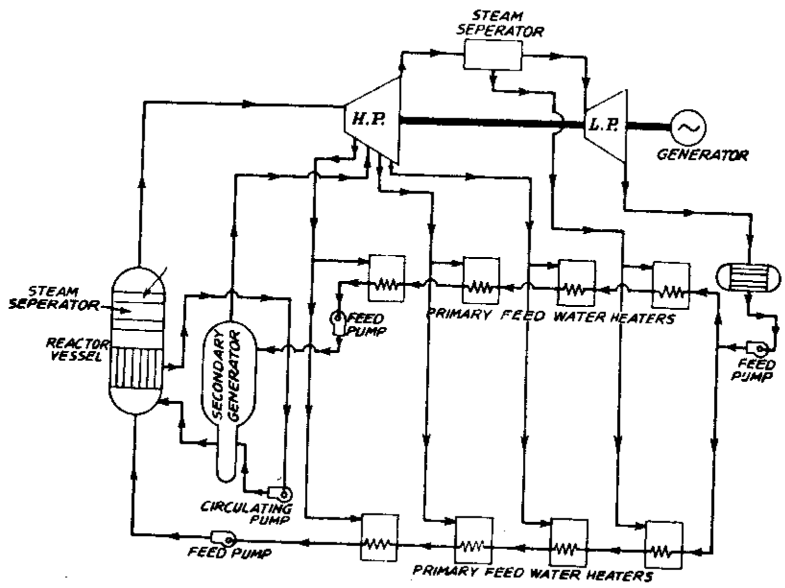


Fig. 28.14. Steam cycle used at Tarapur atomic power plant.

channels are provided, one leading to the north and other to the south. One of them is used at a time according to the current in foreshore towards the north or south for discharging the warm water back to the sea after passing through the condenser. This dual discharge system avoids the mixing of discharge water with the intake water.

The fresh water required for steam generation is drawn from a reservoir located several miles from the station. A special dam and reservoir are built for the purpose.

This plant has completed 25 years of commercial operation. During 1982, it has generated 1239 million units supplying—442 to Maharashtra and remaining 705 to Gujarat.

The Tarapur Atomic Power Station (TAPS), the country's first prestigious nuclear power plant, was built by U.S.A. and began commercial operation on Nov. 1, 1969. It consists of 2 units of 210 MWe (each) boiling water reactors, uses enriched uranium (1.5 to 2.5%) and ordinary water is used as coolant and moderator. Its cost was Rs. 92 crore and supplied electricity at the rate of 63 paise per kWh, probably the lowest for nuclear electricity in the world. It has so far earned Rs. 1098.2 crore and earned net profit of Rs. 211.5 crore.

TAPS, the second oldest boiling water reactor in the world, has outlived all its sister plants and has an enviable safety record. There had been no increase in background radiation level during last 25 years. TAPS is maintaining a very high safety standard compared with any other nuclear facility abroad. The waste generated, either gaseous or liquid, is rendered safe before discharging to the environment.

TAPS has suffered a lot as U.S.A. stopped supplying enriched uranium from 1974 after the Pokhran nuclear blast. Then France came into picture to fill the breach to supply nuclear fuel for 30 years till 1 Nov. 1993.

India is forging ahead for the use of indigenously fabricated mixed oxide (MOX) fuel to keep the plant running beyond 2000 A.D., following the stoppage of fuel supply from France. The MOX is a mixture of oxides of uranium and plutonium, developed by Indian scientists with efforts of 10 years. Plutonium-239 can be extracted from the spent fuel and converted into ceramic oxide and then mixed with natural or depleted uranium oxide. Both have roughly the same amount of U^{235} isotope. This mixture is called MOX and is used in the form of pellets. To produce MOX, India has set up a reprocessing plant at BARC, Mumbai.

In addition to this, facilities for quality control of MOX fuel were also set up in 1994 only as per report of the Department of Atomic Energy.

With bad experience of running TAPS, the Govt. of India decided to develop further Nuclear power plant using pressurised heavy water reactors (PHWRs) which run on natural uranium as fuel and heavy water as moderator. The seven new reactors built after TAPS are based on PHWR and 7 more are being set up.

All other Nuclear power plants described below use PHWR technology.

B. Rana Pratap Sagar Atomic Power Station. This is the second atomic power plant of India which works on CANDU type principle. This power plant consists of two identical units of 200 MW each. The first unit achieved its criticality few years back and second unit was expected to go critical in 1976 but it went critical only in 1980 due to materials delay and technical difficulties. The station is delivering 400 MW power.

The station lies on the right bank of Rana Pratap Sagar lake formed between two dams—Gandhi Sagar dam upstream and Rana Pratap Sagar dam downstream of the Chambal river. The station is 64 km from Kota city and occupies an area of about 8 hectares.

The station is intended to operate as a base load plant. Power will be delivered at 50 cycles/sec to Rajasthan grid. In addition to this, provision has been made to supply 15% of the generated steam to heavy water production plant as process heat which is close to the reactor site.

The station incorporates twin units of CANDU-PHW (Canadian-Deutesium-Uranium, pressurised heavy water cooled) type of reactor system. This reactor is characterised by

(*a*) a horizontal pressure tube reactor,

(*b*) natural uranium as fuel,

(*c*) low pressure heavy water as moderator,

(*d*) high pressure heavy water as coolant, and

(*e*) bidirectional fuelling on power.

This type was selected for two reasons :

(*a*) It uses natural uranium which is available in the country. This is important as country wants to be self-sufficient as there are no facilities to enrich uranium indigenously. The recent bitter experience of Tarapur has strengthened the self-sufficient policy of India.

2. The CANDU type gives high specific power rating because of good neutron economy and this results in more effective utilization of available fuel.

Details of the Power Plant

1. **Reactor.** The reactor is of horizontal pressure tube type fuelled with natural uranium and moderated and cooled by heavy water, separate circuits being used for the coolant and moderator.

Fuel bundles are loaded on power at one end of the pressure tube and discharged at other ; fuelling alternate tubes at opposite ends prevents the reactor from having all new fuels at one end and burned up at the other. Similarly, arranging the coolant inlets and outlets at opposite ends of alternate tubes maintains a more even temperature throughout the reactor.

The reactor is controlled with the help of cobalt absorber assemblies under normal working conditions and moderator level remains uncharged. Safety shut-down is rapidly achieved by dumping the moderator in the dump tank located below the calandria. The moderator dumping rate reduces the reactor power from 100% to 50% in 7 seconds, from 50% to 20% in 13 seconds and from 20% to 10% in further 15 seconds.

Initially both reactors would be fuelled by a total quantity of 112 tons of natural uranium dioxide fuel. Subsequently, they would need only 200 kg of fuel per day.

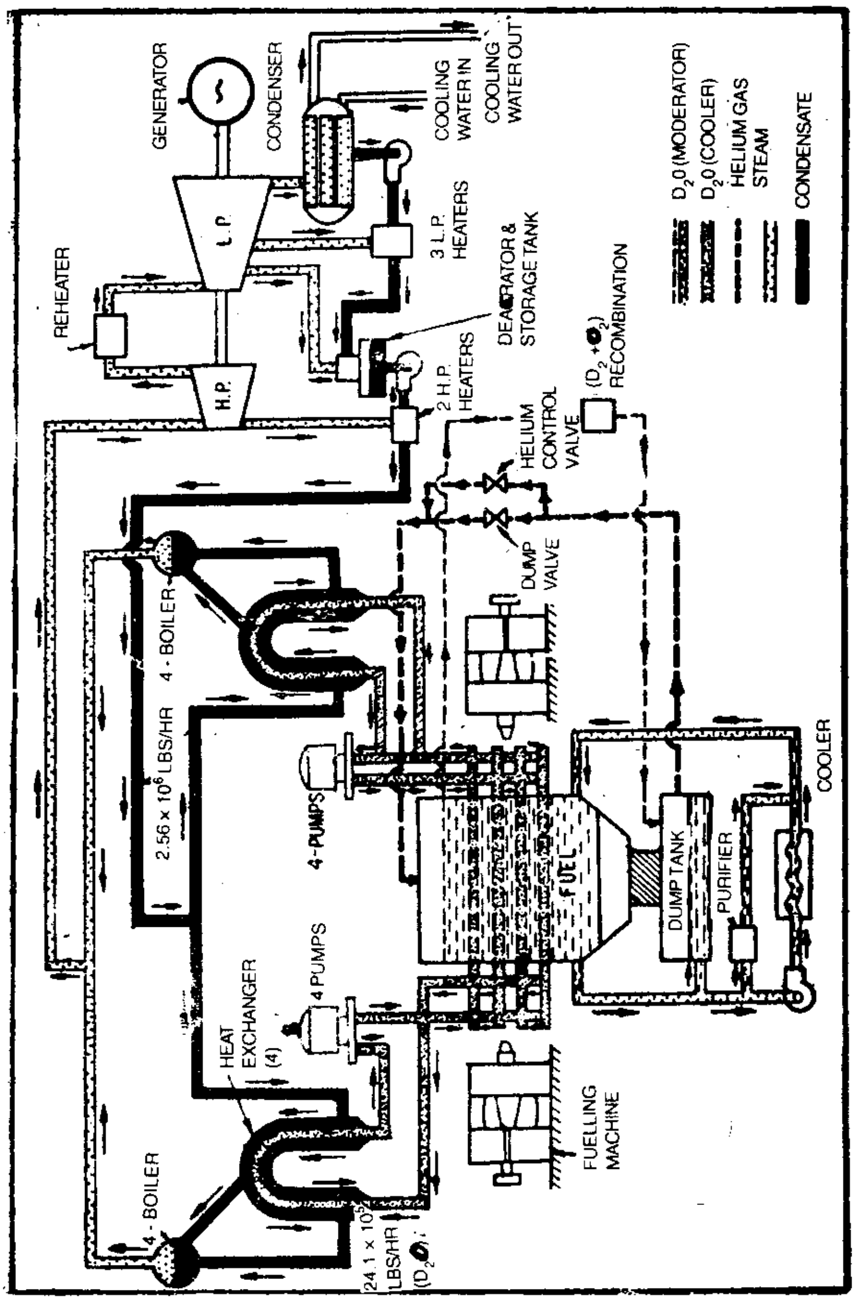


Fig. 28.15. Steam cycle used at Rana Pratap Sagar Atomic Power Plants.

2. **Calandria.** The calandria which is the heart of the reactor contains the fuel and the moderator. Calandria is made of austenatic stainless steel and is 507 mm long and 605 mm in diameter. The tube plates at the end have 306 holes of 228.6 mm square pitch to accommodate the nickel-zircaloy tubes. These tubes have a wall thickness of only 1.24 mm to minimize neutron absorption while retaining sufficient strength.

The pressure tubes made of zircaloy-2 have a wall thickness of 3.9 mm this being based on a maximum allowable design stress of 11 bar at 299°C. The pressure tubes are centred in the calandria tubes by spacers and the annular clearance between the two is filled with dry air to act as layer of insulation and minimise the loss of heat to the moderator.

3. **Moderator.** The moderator has about 37 MW of heat deposited into it and this heat is removed by continuous circulation through the heat exchangers. There is about 140 tonnes heavy water in the moderator system, of which calandria normally contains 130 tons. The flow through the heat exchangers (6000 gal/min) is made up 5500 gal/min directly from the calandria and 500 gal/min from the dump tank. Some 3900 gal/min are fed to the circuits cooling the control rods and 100 gal/min is fed to ion exchangers to remove impurities (boron poison) for starting up reactivity control.

4. **Primary Cooling System.** The arrangement is shown in Fig. 28.14. The flow through the reactor pressure tubes is in opposite directions through adjacent tubes as shown in figure. Actually, the coolant pipes to each pressure tube at either end are grouped into inlet and outlet headers, across which the boilers and circulating pumps are connected. There are 8 boilers, arranged 4 in each group, and circulating pumps, one for each boiler. Each boiler consists of 10 heat exchanger units of the inverted U-shell and tube type.

5. **Secondary Cooling System.** The secondary coolant which is light water is passed to the shell side of each inverted U-boiler at the bottom and preheated to saturation temperature. It reaches the top of the inverted U, enters the boiler drum as a steam and water mixture. The steam is separated and fed to the turbine at 11 bar and 250°C and separated water flows from the bottom of the drum to other leg of the U.

The generator is directly coupled to the turbine. The stator windings are water-cooled and stator iron and rotor are hydrogen cooled.

6. **Waste Disposal.** Irradiated fuel bundles which are dangerously radioactive are transferred from fuelling machine to a fuel transfer room via underground passage to the spent fuel storage bay located in service building which has sufficient capacity to cool and shield the irradiated fuel for 25 years. The depth of demineralised water in the bay is 7.16 metres which gives a minimum cover of 4.11 metres on the stored fuel, where 0.45 metre is considered more than satisfactory for irradiated fuel from shielding and cooling considerations.

All effluent waste gases are mixed with inactive air discharged from the ventilation systems and after passing through the filters are released through the main stack at a height of 100 metres above ground. The stack flow is about 1750 m^3/min.

A small inspection bay is provided adjacent to the storage bay. Here, underwater inspection, leak testing and canning of failed irradiated fuel can be carried out.

7. **Condenser Cooling Water System.** The requirement of cooling water for the condenser is as high as 150 thousand tonnes per hour for both units. The terrain at the power plant site is traversed by natural nallas carrying monsoon run-off from upper catchment area to the river. A natural advantage of one such nalla at intake end and another some thousand feet downstream at the outfall end for the condenser circulating water is also taken in locating the site of the power plant.

The cost of setting up I-Unit is estimated to be about 73.5 crores and for second unit about 66.5 crores. Out of total 140 crores, nearly 60 crores are in the form of foreign exchange.

The outline of the plant is shown in Fig. 28.15.

Unit I of this plant has been shut down from March 82 due to the water leak detected in one of the end shields. Unit II had a continuous run from October 82 to Jan. 83 and it has generated 433 million units in 1982.

(C) Kalpakkam Atomic Power Station. The third atomic power project which is totally designed by Indian engineers using indigenous materials is Kalpakkam Atomic Power Project of 470 MW capacity at Kalpakkam near Madras in Tamil Nadu state.

The reactor construction is of CANDU type is exactly similar to the reactor at Rana Pratap Sagar. Therefore, the details of the reactor are not described to avoid the repetition of the matter.

The outline of the components at Kalpakkam power station is shown in Fig. 28.16. Two identical reactors each of 240 MW, capacity are designed to generate 440 electrical MW. The thermal generating capacity of both reactors is approximately 700 MW. Each reactor is 42 metres high and 5.2 metres in diameter. Two slabs of 2.5 metres thick made of prestressed concrete with 150 mm gap in between are provided to eliminate completely the possibility of nuclear radiation to the atmosphere.

A generator turbine house is located near the condenser pumping house as shown in figure because the condensers are located just below the generator-turbine house. Two small service buildings are provided to either side for the storage of material and minor repair of equipments. A big service building is also provided in between two reactors where all major repair works are carried out. The waste material from the reactor is also processed in this building to recover the useful isotopes and Pu^{239}. The fuel required for charging is also supplied through this building to the reactors as the access to the reactors is provided through this building.

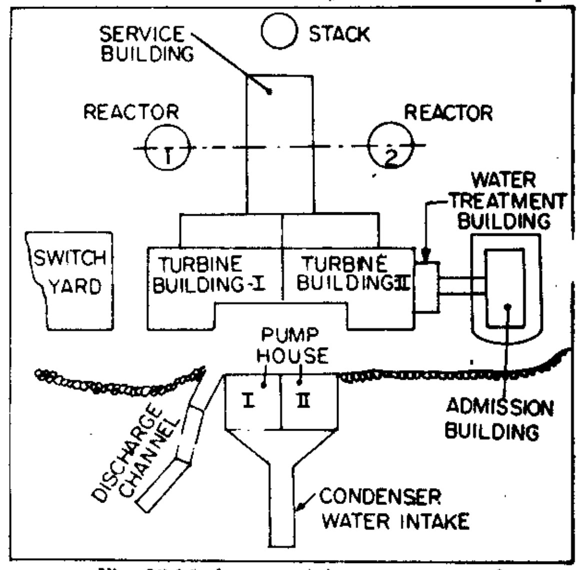


Fig. 28.16. Layout of the components of Rana Pratap Sagar Power Station.

One storage tank for storing D_2O near each reactor is provided as shown in figure. The upper portion of each tank is filled with Helium at high pressure to maintain 100% purity of D_2O and to prevent the exposure to the atmospheric air. A D_2O upgrading plant is also provided to supply high purity D_2O, as it gets contaminated due to radioactivity in the reactors. A separate water demineralising plant (secondary

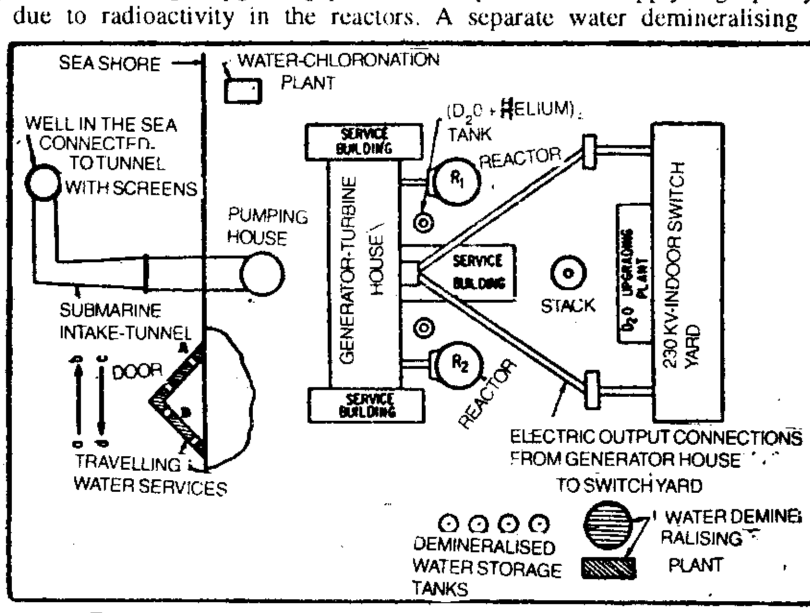


Fig. 28.17. Layout of the components of Kalpakkam Power Station.

circuit) is also provided to maintain the required purity of water passing through the closed circuit. The demineralised water storage tanks are located near the demineralising plant.

With the past experience of Tarapur power plant with outdoor switchyard getting corroded due to salty atmosphere, an indoor switchyard is provided at this power station.

The unusual feature at Kalpakkam is the difficulty experienced in the design and construction of condenser cooling water system. The shore-line on the eastern coast of India is subjected to heavy littoral drift and also heavy cyclones. This has necessitated the adoption of submarine tunnel 22 metres below the rock bed and 150 m in length from the sea shore for drawing cooling water for the condensers ; and construction of approach jetty extending nearly 500 metres into the open sea to facilitate the construction of the intake structure. The condenser water pump house is located near the turbine-generator building as shown in figure. The water through the tunnel flows under gravity upto pump house as the tunnel has been given inclination towards the pump house and after that the water is circulated through the condenser with the help of pumps. The water coming out from the condenser is discharged to the sea through the travelling water screens (either A or B) according to the flow direction of water along *ab* or *cd* as shown in figure during different tidal periods. This arrangement is necessary to avoid the mixing of hot water with the intake cold water which may affect the performance of condenser very badly. A quantity of water circulated would be of the order of 1.5 lac gallons per minute. A chlorination plant for treating the condenser water is also provided to avoid the formation of algae in the condenser tubes otherwise it may affect the performance of the condenser.

The foundation of unit I and other units as reactor vault, containment building, turbine building, turbo-generator are completed. The construction of the sea water intake structure, the submarine tunnel, pump house are completed. Major equipments like turbo-generator, reactor vessel, end shields, condenser water pumps, moderator heat exchangers, compressed air plant and water treatment plant are already purchased from the Indian manufacturers. All major electrical equipments like transformers, switchgears, isolators and electric control systems are also completed in India. The first unit of this power plant was commissioned on 23rd July 1983. With the commissioning of this plant, India has become 7th country in the world that has the capability to design, construct and commission nuclear plants by indigenous efforts. But unfortunately, this may prove to be the most expensive power plant in the country. The high cost is due to several factors as plant machinery costs 118 crores, heavy water costs 6 million per ton for its requirement of 230 tons and actual output of 180 MW against designed 235 MW. Added to these is the colossal delays resulting in the loss of Rs. 50 crores and needing 16 years to build the plant. The capital cost of the plant goes to Rs. 13,000/kW, a very high figure. To meet its cost, power has to be sold out at 80 to 90 paise per kWh which is double the rate of power available from thermal plants (1985). The work of the second unit is also completed. The overall progress of the plant construction was considerably behind schedule due to technical and economical difficulties. The second unit went critical by 1985.

The other two plants which are also CANDU-type are proposed at Narora in U.P. and Kakrapara in Gujarat near sea shore, each of 235 × 2 MW capacity. The Narora's work is progressing well. Two units of this project were commissioned during 1986-87 and 1987-88. Financial sanction for two units at Kakrapar was accorded in July 1981. Kakrapar atomic power station unit-II has started generating power. It is presently operating at 100 MW and supplying electricity to the Western region. This is the 10th reactor commissioned in India. Beyond Kakrapar, four more units of 235 MWe capacity had been also included in the VIII-plan period. A fast breeder test reactor was commissioned by the middle of 1990.

The enormous problems involved with thermal plants and natural limitations on hydrogenerator, a nuclear power programme of 10,000 MWe capacity by the end of the century constituting 10% of the electrical energy need of the country has been drawn by the Government.

(D) Narora Atomic Power Plant

This power plant is situated in U.P. and works on the same cycle as atomic plant at Kota and Kalpakam. The working technology is based on CANDU-type reactor. The pictorial view of the plant is shown in Fig. (28.18).

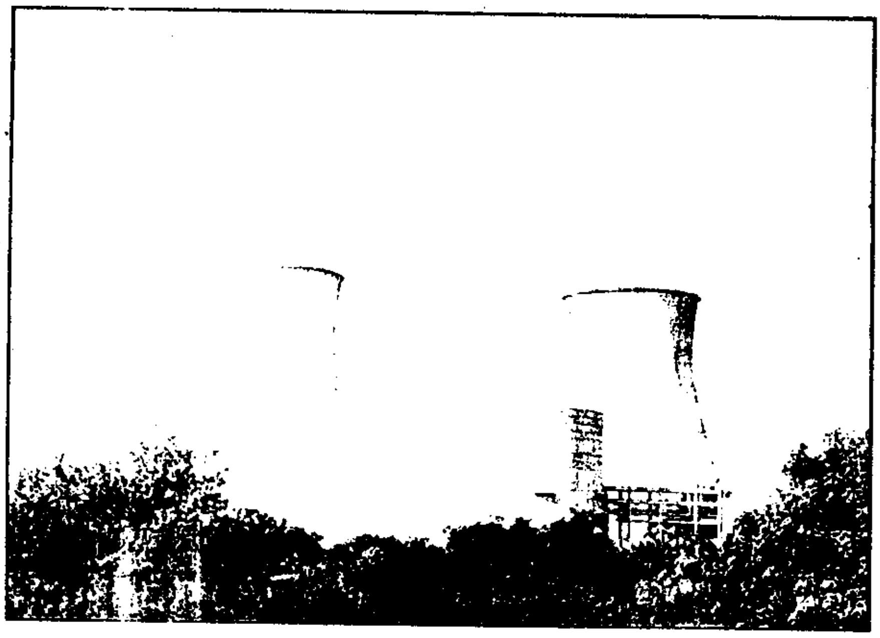


Fig. 28.18. Narora Atomic Power Station-1 & 2.

(E) Kakrapar Atomic Power Plant

This atomic power plant is located in Gujrat and it also works on the principle of CANDU-type reactor. It pictorial view is shown in Fig. (28.19).

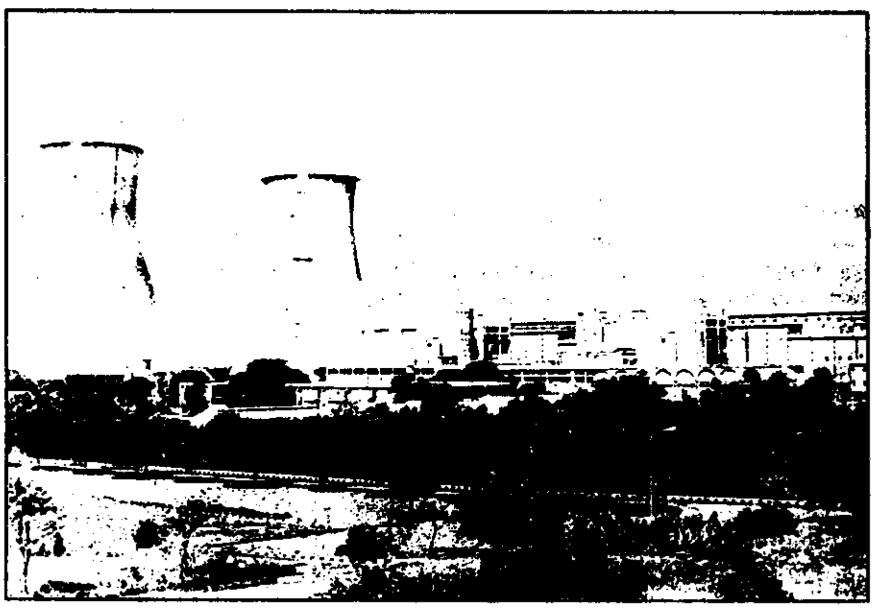


Fig. 28.19. Kakrapar Atomic Power Station-1 & 2.

(F) Kaiga-Atomic Power Plant

In this power plant, the same technology is used as used in earlier PHWRs at Narora and Kakrapar as well as Kalppakam and Rana Pratap Sagar plants.

Kaiga is situated at 56 km east of Karvar and 13 km upstream of Kadra dam on the left bank of Kali river in Karnataka state. It is a beautiful location surrounded from all sides by evergreen hills with thick forest cover. The site has been selected for having six reactor units of PHWRs, each of 220 MWe.

The first unit at Kaiga attained criticality on 24th Sept. 99 and was synchronised to the grid on 2nd Dec. 1999. It is dedicated to the nation on 6th March 2000 by Prime Minister Atal Bihari Vajpayee. This reactor is expected to generate 2420 Million units annually. Kaiga-2, India's 11th Nuclear reactor went critical on 14th Sept. 2000.

Based on the operating experience of the earlier pressurised heavy water reactors in India, a number of modifications have been incorporated in Kaiga project as listed below :

(1) In Narora and Kakrapar stations, there is only one dome for outer containment and the inner containment top consists of concrete slab. A double dome design was evolved for Kaiga for both inner and outer containment.

(2) Structural wall is provided in Reactor Building for supporting all the floors to avoid equipment loading on containment wall.

(3) A separate diesel generating building, contral building, spent storage building, and emergency core cooling system are provided.

(4) Two separate turbo-generator buildings are provided from missile considerations.

(5) The end shields are provided with venting arrangement at the topmost point which ensures proper venting and avoids the problem of high radiation fields in the fuelling machine vaults.

The general view of Kaiga plant is shown in Fig. (28.20).

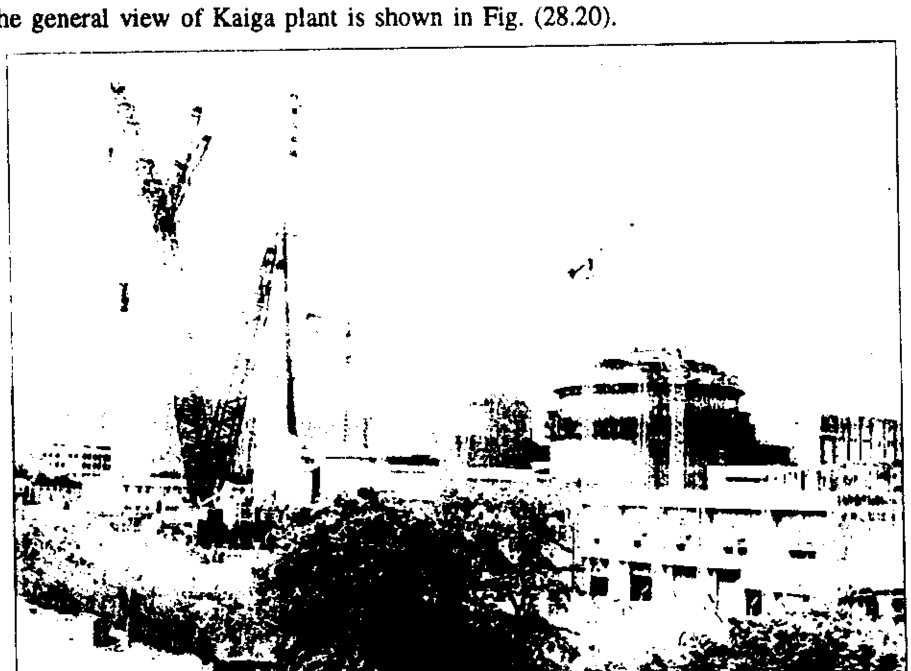


Fig. 28.20. General view of Kaiga Project. This picture shows Reactor Buildings-1 & 2, chimney stack, 640 tonne crane and upgrading plant.

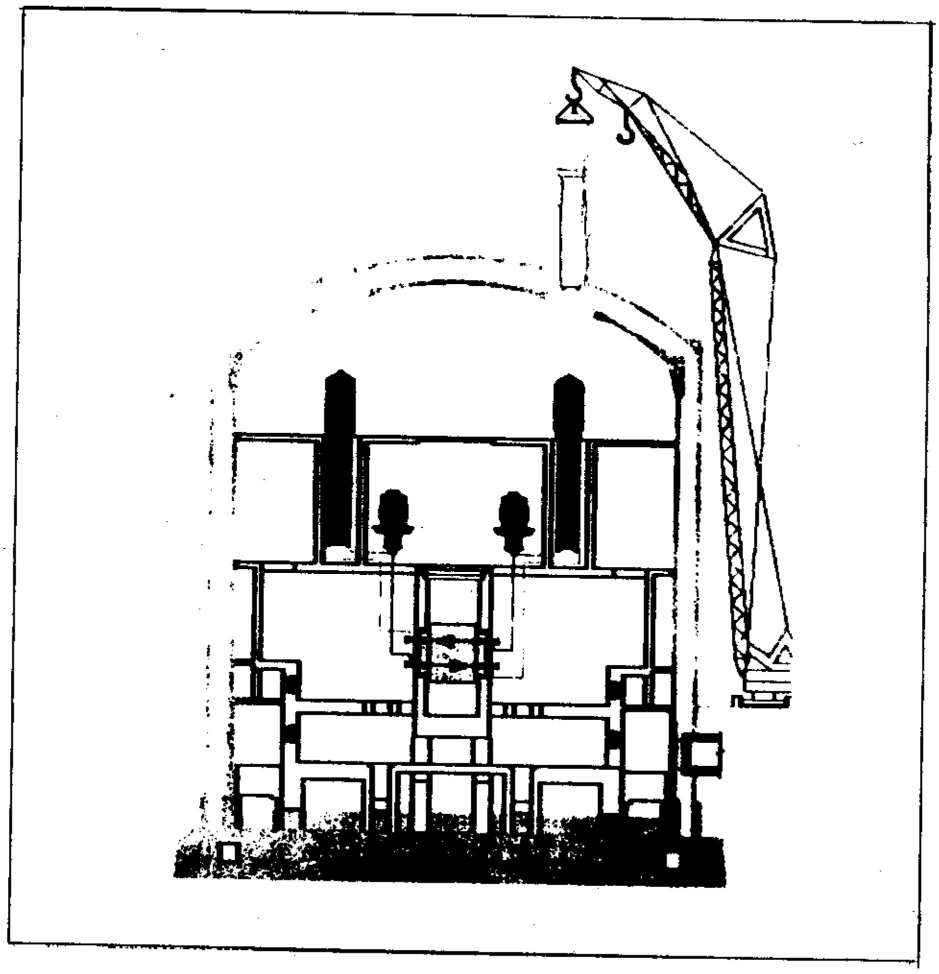


Fig. 28.21. PHWR Double Containment.

(6) The primary heat transport (PHT) system is valveless system, eliminating the steam generator isolating valves and PHT pump discharge valves. This reduces the maintenance work load and radiation exposure. The radiation escape from coolent system will also be lesser.

(7) The Kaiga station confirms to international standards and for the first time a new high performance concrete has been used.

(8) On-load test facility for the primary shut-off rods is provided.

(9) The safety features include an advanced highly reliable and automatic reactor shut-down system and a microprocessor-based system for reactor protection.

(10) A number of computerised systems have been provided for increasing the reliability of plant operation and online information concerning plant operation as listed below :

(*a*) Duel process control system is provided to control 5 important parameters in coolent system.

(*b*) Reactor control system to control reactor power.

(*c*) Programmable digital comparator system to compare various parameters with set points. It gives alarm when set points are exceeded and protective actions are taken through primary shut-down system.

(*d*) A computerised operator information system for getting data on 3000 parameters. All parameters are kept in memory for all 24-hours to analyse an incident.

(*e*) Programmable logic controllers to generate process logics and provide easy diagnosis for the problems.

The National Environment Engineering Research Institute (NEERI), Nagpur has made a study on the environmental impact on the areas surrounding Kaiga project and has given a green signal.

It is anticipated that Kaiga will operate with minimum radiation and maximum capacitor factor due to the various design changes and computerised operation system are implemented.

This reactor (Kaiga-2) of 220 MW capacity costing 2896 crore will supply electricity to the Southern grid, comprising Karnataka, A.P., Tamil Nadu, Kerala & Goa.

Evolution of Pressurised Heavy Water Reactors in India

A 3-stage programme is being pursued to develop nuclear power in India. Consistent with our unique resource position of limited uranium and large thorium reserves.

The first stage is based on pressurised heavy water reactors (PHWRs) for optimum use of the available uranium resources. These PHWRs not only use natural uranium efficiently but also provide plutonium as a by-product which will help to use our large thorium reserves for power production.

The PHWR technology developed in the country for the 220 MWe units is a commercial success. Eleven such units are in operation presently and one more is in final stage of construction and commission. Average capacity factor as high as 80.2% has been achieved by the year 1999-2000. Based on this experience of designing and operating 220 MWe units, NPC (National Power Corporation) has launched construction of two 500 MWe PHWR units at Tarapur in Oct. 1998.

The story of evolution of PHWRs in India begins with construction and operation of CIRUS, a research reactor at BARC. This reactor was fuelled with natural uranium and heavy water as moderator which extends the use of natural uranium oxide and heavy water to power reactors.

The next step was to construct the first PHWR reactor near Kota (Rajasthan), a twin unit (2 × 200 MWe) designed by Atomic Energy of Canada Limited (AECL). The design of RAPS-1 and 2 was also provided by AECL.

The design and Engineering of Madras Atomic Power Station (MAPS, 2 × 220 MWe) was taken up by PPED. Though the basic modules of the MAPS design were identical to those in RAPS, certain new design features were introduced. Prominent among those are the vapour suppression and double containment concepts. Another design feature included was a closed loop process water system, to prevent any accidental release of activity to the ultimate heat sink in case of tube failure in any heat exchanger. These features have been advanced further in subsequent power plants and have become an unique hall mark of Indian PHWRs.

For the design of Narora Atomic Power Station (NAPS, 2 × 220 MWe), the redesign was carried out based on the experience gained in construction, transportation of heavy equipments and operation of reactors in Indian power grids. The design parameters of NAPS systems also included anticipated load that could be imposed due to postulated severe earthquake. The Calandria vessel design was modified to accommodate two fast acting diverse shut down systems. The end shield was modified considerably to simplify its fabrication and transportation. Primary heat transport system incorporated state-of-the-art steam generator design. Moderator system design was simplified. Fuel handling system was redesigned to provide additional features not only to meet emerging safety philosophy but also to overcome some manufacturing and operational problems encountered in RAPS and MAPS. Among the safety systems, the NAPS design incorporated two independent diverse fast acting shut down systems, a high pressure emergency core cooling system and full double containment reactor building with innovative safety features.

The Kakrapar Atomic Power Station (KAPS, 2 × 200 MWe), the Kaiga project (1 and 2) and Rajasthan Atomic Power Project (RAPP, 3 and 4) are essentially repeat of the standardized Indian 220 MWe PHWR design. The design changes subsequently made KAPS, Kaiga 1 and 2 and RAPP 3 and 4 over NAPS have been mostly as dictated by site specific requirements only. Certain improvements in the relative arrangement and layout of buildings have been introduced in Kaiga 1 and 2 and RAPP 2 and 3 based on feed back and operational experience based from NAPS.

The steam generators have been brought fully into the primary containment covered by dome in Kaiga and RAPP. Four openings, one above each steam generator, have been provided in segmented spherical dome, to facilitate replacement of steam generator.

Details of Nuclear Power Plants in India (a)

Name of Station and Location	*Unit*	*Installed Capacity*	*Capacity Utilisation (Capacity Factor in MWe %)*				
			1997-98	*1998-99*	*1999-2000 (upto Nov. '99*	*States to which electricity is being supplied*	*Commencement of operation*
Tarapur Atomic Power Station, Tarapur, Maharashtra.	TAPS-1 TAPS-2	160 160	84 68	93 71	58 80	Maharashtra & Gujarat,	Dec. 16, 1973
Rajasthan Atomic Power Station, Rawatbhata, Rajasthan.	RAPS-1 RAPS-2	150 200	40 46	63 69	69 79	Rajasthan	April 1, 1981
Madras Atomic Power Station, Kalpakkam, Tamilnadu.	MAPS-1 MAPS-2	170 170	49 78	75 72	91 93	Tamil Nadu, Kerala, Karnataka, Andhra Pradesh, Pondicherry.	Jan. 27, 1984 Mar. 21, 1986
Narora Atomic Power Station, Narora, Uttar Pradesh.	NAPS-1 NAPS-2	220 220	90 89	68 77	80 76	Uttar Pradesh, Delhi, Punjab, Haryana, J & K, Chandigarh, Himachal Pradesh.	Jan. 1, 1991 July 1, 1992
Kakrapar Atomic Power Station, Kakrapar, Gujarat.	KAPS-1 KAPS-2	220 220	48 63	72 78	79 86	Gujarat, Madhya Pradesh, Maharashtra, Goa, Daman & Diu,	May 6, 1993 Sept. 1, 1995
Kaiga-1 & 2	KA-1 KA-2	220 220	—	—	—	—	Mar. 16, 2000 and Sept. 5, 2000

(*a*) RAPS-3 (220 MWe) went critical on June 1, 2000 and RAPS-4 is expected to go critical in Nov. 2000.

(*b*) The contribution of nuclear power is about 2% of the total power produced in the country from all sources.

(*c*) Proposals for nuclear power development in the Nineth Five Year Plan include 2 × 500 MWe plant at Tarapur (TAPP 3 & 4), additional 2 × 220 MWe Unit at Kaiga (Kaiga 3 & 4) and commissioning of the Detailed Project Report (DPR) for the 2 × 1000 MWe Nuclear Power Station at Kudankulam in Tamil Nadu with Russian assistance, apart from completing and commissioning the ongoing projects of a total capacity of 880 MWe, comprising the Kaiga Atomic Power Project Units-1 & 2 (2 × 220 MWe), and Rajasthan Atomic Power Project Units-3 and 4 (2 × 220 MWe).

World's Leading PHWRs in 1998

In 1998 there were 36 units of pressurised heavy water reactors (PHWRs) operating in the world with a total installed capacity of 21255 MWe.

Sr. No.	*Country*	*Reactor*	*Capacity (MWe)*	*C.F. (%) 1998*
1.	Canada	Darlington-3	935	94.64
2.	Argentina	Atucha-1	357	86.07
3.	India	Narora-1	220	86.00
4.	Romania	Cernavoda-1	700	85.81
5.	Canada	Darlington-4	935	85.09
6.	Canada	Darlington-1	935	84.00
7.	India	Madras-2	170	84.00
8.	Republic of Korea	Wolsong-2	679	81.81
9.	Canada	Darlington-2	935	81.41
10.	Argentina	Embalse	648	80.95
11.	Republic of Korea	Wolsong-1	679	78.34
12.	Canada	Pickering-5	540	78.17
13.	Canada	Pickering-8	540	77.85
14.	India	Narora-2	220	77.00
15	India	Kakrapar-2	220	77.00

28.9. INDIA'S 3-STAGE PROGRAMME FOR NUCLEAR POWER DEVELOPMENT

There is lot to think for developing countries to go for nuclear power generation as the capital and foreign exchange required is extremely large. Once decision has been taken to develop the nuclear power, the next problem is the programme of development and selection of reactor, fuel, material, industrial capability and skilled technique.

The first and major problem is the choice of reactor and fuel. The choice between the natural uranium against enriched uranium depends upon the availability of natural uranium resources in the country. A country going to install 5000 MW of nuclear power has to spend about Rs. 260 million per year in foreign exchange for the purchase of enriched uranium and total of 1.9 billion over the entire life of the plant. A country using its natural resources of indigenous uranium if available or import from supplier countries has to spend Rs. 1.875 billion for the entire life span of the power plant of 5000 MW capacity. Therefore, it is obvious that natural uranium fuelled reactors would be preferred in developing countries.

India realised that with her limited fuel resources, particularly oil, and many areas, remote from all sources of conventional thermal or hydel, economical nuclear power could play a vital role in the growth and development of the country's electric power system.

Late Dr. Bhabba introduced a 3-stage nuclear programme in 1959 for the development of nuclear power in the country.

1. In the first stage, natural uranium fuelled reactor would be established in the country which would produce a fissionable material plutonium (Pu^{239}).

2. In the second stage, the plutonium available from the first stage would be used as fuel in fast breeder reactors either to produce more plutonium than consumed from U^{238} or to produce U^{233} from thorium.

3. In the third stage, U^{233} would be used only as fuel where thorium would be again converted into fissible U^{233}. It is already mentioned that U^{233} is best among all fissile fuels for breeder reactors. Therefore, all the reactors in the third stage would be fuelled with U^{233} and only thorium would have to be fed.

Three stage programme of India is represented in Fig. 28.22.

The development of each stage depends upon the development of previous stage. To go from one stage to another needs an adequate installed capacity as well as several years (20 or more) of operation.

The cause of requiring number of years to develop such a programme is low production of Pu^{239} even from large capacity nuclear plants. One tonne of uranium contains only 245 kg of Pu^{239} after an irradiation of 7000 MWD/tonne. A 1000 MW capacity nuclear power station would yield only 380 kg of Pu^{239} per year with an annual feed of 150 tonnes of natural uranium. In the second stage, with the feed of 380 kg of Pu^{239}, about 170 kg of U^{233} would be produced per year.

Considering the limited natural uranium resources of India, natural uranium fuelled heavy water moderated reactor would be the best choice as heavy water reactors produce the maximum amount of plutonium per tonne of uranium mined. This would also help to start the breeder reactors earliest possible.

The estimates of heavy water demand and supply prepared by the Department of Atomic Energy (DAE) show that India would have to depend for a long time on imports of this material in its nuclear power programme. The Soviet Union would be the possible supplier of heavy water. The estimated heavy water requirement in 1984 was 256 tons while the capacity of the working heavy water plants is 200 tons per year which may be able to produce maximum 100 tons/year. Of the five heavy water projects, Nangal, Baroda and Tuticorin are under operation and Talcher and Kota are under commissioning. The Nangal plant was shut down at the end of Sept. 1982 due to total power cut. The Baroda plant has been in continuous operation since Dec. 1982. The Tuticorin plant was shut down due to massive power cut imposed by Tamil Nadu Board since Nov. 82. The plant was restarted in Feb. 83. In Talcher plant, modifications are incorporated and its commissioning depends on availability of NH_3 and synthesis gas from fertilizer plant. In Kota plant, enrichment was demonstrated in low pressure operations. The DAE proposes to set up 7 new plants for making heavy water. Only two have been sanctioned, namely, Thal Vaishet and Manuguru with a capacity of 140 and 200 tons respectively per year.

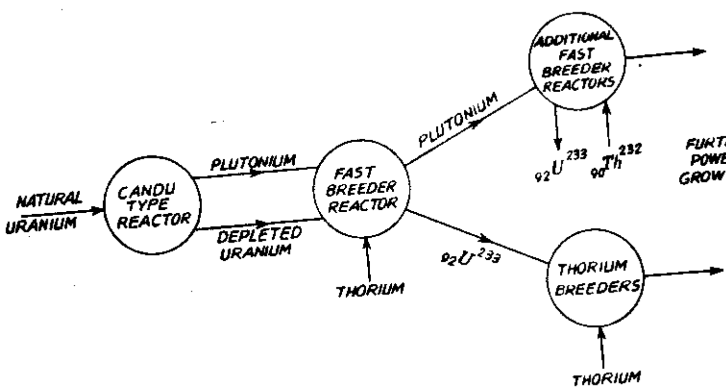


Fig. 28.22. Three-stage programme of nuclear power development of India.

Atomic Energy Commission of India proposes to construct 22 atomic reactors with total capacity of 9350 MW by the end of this century at the cost of 14000 crores to overcome the country's power problem. The current uranium economically exploitable resources are 15×10^3 tons out of 34×10^3 tons, which will be sufficient for the present nuclear programme. Estimated additional resources are 27×10^3 tonnes. The major resources are located in Singhbhum region of Bihar, new target areas identified are M.P., Rajasthan and Central part of India. Uranium occurrences in Karnataka had been established in Dakshina Kannada and Uttara Kannada district stretching for 165 km almost upto the Goa border in the north and Mangalore in the South.

India's nuclear programme is modest compared to other industrialised countries. The nuclear capacity was 63×10^3 MWe in U.S. in 1983, in France 26×10^3 MWe, in Japan 19×10^3 MWe and USSR 20×10^3 MWe. The world average was 12% and was expected to be 25% by the end of century. In India, nuclear power contributes only 3% which would go up to 10% by the end of the century.

The necessary development efforts are initiated in India to start the breeder reactors by the year 1985 so vast resources of thorium in Nature can be used for economic power production.

Whatever the future of nuclear power development in India, all must admire the bold and imaginative step that this vast developing country took in deciding to forge ahead with its own programme of nuclear power and not be forever dependent on foreign technology and equipment in this highly sophisticated and scientific field. The Rajasthan and Kalpakkam Atomic Power Projects stand as an eloquent and elegant monuments to this India's solemn resolve.

Estimates compiled by the International Atomic Energy Agency (IAEA) indicate that the nuclear power capacity will double every 5 years and world nuclear capacity will be 4×10^6 MW by the year 2000 against the 90×10^3 MW in 1975.

By 1985, the number of countries with nuclear power plants be doubled. More number of power plants means more demand for uranium fuel, 25×10^3 tons in 1975, 35×10^3 tons in 1980 and 160×10^3 tons in 1985. There will also be greater demand for enriched uranium. 5000 tons in 1975, 30,000 tons in 1980. There was a stockpile of 150 tons of plutonium in 1975 and this will reach to 5000 tons in 1985.

The Inevitability of Nuclear Power in Future

Current electricity generation scenarios for the next 10 to 20 years often assume that natural gas will be the *fuel of choice* for new power plants, with both coal and nuclear taking a back seat due to negative public perceptions of their side effects. Even though, the real choice will be between coal and nuclear only. The real question is when ?

Prof. Radetzki (Swedish energy analyst) predicted that the global power generation will increase by 60% between 1995-2010. He further pointed out that whenever gas is available, it is likely to be the *first choice* for power generation. But when economic supplies of gas are not available, then the further need of power generation is to be made either by coal or nuclear.

The following three factors will determine the choice to be adopted in practice :

(1) **Internal Generation Cost.** This accounts for the capital cost and fuel costs.

(2) **External Cost.** This cost is related to an environmental impacts. These costs are hard to identify and more difficult to value in monetary terms.

The nuclear generates very small external cost (1% of internal cost) even taking into account the potential impact of nuclear accidents. There is *strong tilt* in favour of nuclear due to its low external cost (Coal : Nuclear = 4 : 0.025) but internal costs are more or less same for coal and nuclear as (6 : 6.4). The total cost, internal and external as (10 : 6.425), the choice goes to nuclear.

(3) **Public and Political.** With regard to coal and nuclear, the "layman's aversion" to the potential risks posed by climate change and major nuclear accidents appears to be exceptionally strong.

The climate costs of coal fired generation would be in a range of 6 to 11 cents/kWh, making coal completely uncompetitive with nuclear. But the public and political opposition in widening group of countries to existing or potential nuclear power on account of its purported dangers can be taken as a reflection of high external costs attributed by laymen and politicians.

Nuclear power is a relatively young industry, with the bulk of its history dating back only 20-30 years. In the early 1970s, it contributed only 2% of electric power demand in the world (110 TWh out of 5250 TWh). During the 1980s, after the first oil shock, nuclear power grew steadily and today represents about 17% of total electricity generation (about 2360 TWh in 1998).

Partly because of its youth, a number of problems are still on the nuclear agenda, including long-term demonstration of operational safety, the treatment and disposal of waste and risks of proliferation.

The World Energy Council (WEC) has argued in its global energy scenarios over the years that nuclear power should play a major role in contributing to electricity provision and to combat global warming. At its current level of 6% of total energy, nuclear power is well established in key market around the world, but its steps have been hesitant and its future clouded. This is ironic in the context of international negotiations to reduce green house gas emissions. Irrespective of all these things, New nuclear power is a key part of the world's energy wardrobe now and in the years to come.

Nuclear energy does not produce greenhouse gases. A nuclear unit (400 MWe) can avoid seven million tons of CO_2 annually compared with the same size coal-fired plant. This nuclear unit would produce only 20 tons of spent fuel, most of it reusable. A 1000 MW nuclear unit would need about 2 square kilometers of land while a solar park of the same capacity would need 30 square kilometers and wind-field about 100 square kilometers and a biomass about 5000 square kilometers. In Europe alone, where 35% of electricity is generated by nuclear power, it is estimated that the CO_2-emission avoided is equivalent to the total amount released each year by all the vehicles travelling European roads.

However, it is clear that the nuclear industry must take the necessary steps to bring down capital costs and satisfy public concerns about safety. If nuclear energy is to gain wide spread acceptance, these must be sustained, patient demonstration of the safety, reliability, credibility and viabilty of nuclear plants and the waste disposal process. WEC believes nuclear power will resume its growth during the next decade when over capacity in electricity generation will have disappeared and evidence of adequate safety and waste disposal has been established.

28.10. COMPARISON OF NUCLEAR PLANTS WITH THERMAL PLANTS

The comparison of nuclear plants is generally made with the thermal plants because both are presently installed with high unit capacity of 500 to 1000 MW.

Now-a-days in India, the nuclear power is preferred where there are no hydro-potentials and the coalfields are far away from the required load centres. It is estimated that, generally, nuclear power with a single unit of 200 MW capacity is competitive with thermal power at a distance of about 800 km from pitheads. No doubt the generating cost of the nuclear plant varies greatly with the type of reactor used and the availability of the fuel resources. U.S.A. still prefers pressurised or boiling water reactors as facilities for providing enriched uranium are available. India has preferred CANDU type reactors as she has large resources of natural uranium and thorium. England particularly has developed graphite moderated gas cooled reactors. England is more keen to develop the nuclear power than any other country as other fossil fuel resources are negligible compared with power demand of the nation. There is common trend throughout the world to develop the nuclear power as it is competitive in first cost as well as in running cost with thermal plants. The only difficulty faced by the developing countries is the lack of technique required initially to develop and required foreign exchange to purchase the special equipments and materials required for nuclear power plants.

The few advantages of nuclear power over the thermal are listed below :

1. The nuclear plant is more economical compared with thermal in areas which are remote from coalfields.

2. There are no fuel transportation, handling and storage charges as well as there is no problem of ash disposal.

3. The number of people required at nuclear stations is far less than thermal plants. Therefore, the cost of operation is less.

4. The nuclear plant occupies less space in comparison to thermal plants, therefore, the civil construction cost is also less.

5. The capital cost is less in bigger unit sizes as the initial expenditure on structural materials, piping, storage mechanism is less than thermal plant. The running cost is also competitive to the thermal plants.

It is so obvious from the study of nuclear power plants, either in developed countries or in developing countries, the nuclear power has far better future than thermal and it is the only source which can meet the increasing power demand of the world.

The capital and running cost of different power plants

Item	*Nuclear*	*Thermal*	*Diesel*	*Hydro*
Capital cost in rupees per kW installed	1700	1050	800	2100
Generation cost in Paise per kW-hr	4.51	5.05	3.91	1.5

These costs were calculated in 1970 when the nuclear plants were not technically developed.

EXERCISES

28.1. Draw a neat diagram of nuclear reactor and explain the functions of different components.

28.2. What are the uncommon problems faced by the engineers in the design of nuclear reactors ? Explain each in detail.

28.3. Draw a neat diagram of PWR and BWR and explain the advantages and disadvantages. What are the conditions which prefer PWR over BWR and vice versa ?

28.4. What do you understand by dual cycle used in BWR ? What are its advantages over ordinary cycle ? Dual cycle is more preferable than single cycle for variable load. Discuss the statement.

28.5. PWR is a self-regulating reactor and can be used for variable loads whereas BWR is only useful for base load plant. Justify the statement giving reasons.

28.6. What do you understand by "positive power coefficient" and negative temperature coefficient ? How these factors affect the performance of PWR and BWR ?

28.7. Draw a neat diagram of CANDU type reactor and give its advantages and disadvantages over other types. Under what circumstances this reactor is more preferable than PWR or BWR ?

28.8. What are the outstanding features of advanced gas cooled reactors over the other types ? When these are preferred ?

28.9. What do you understand by "pinch point" in gas cooled reactor ? How this affects the performance and power output of a reactor ?

28.10. Two "pinch point" steam cycles are more preferable over single pinch point cycle in gas cooled reactor. Explain.

28.11. What are the advantages of closed cycle combined with gas cooled reactor over indirect cycle ? Why these types of plants are not developed ?

28.12. What factors control the selection of a particular type of reactor ?

28.13. A developing nation has to choose among PWR, BWR, CANDU and AGR. What factors are taken into account in selecting the reactor ?

28.14. U.S.A. prefers BWR, Canada prefers CANDU and U.K. prefers AGR (Magnoz). Why ?

28.15. India has started the nuclear programme with "CANDU" type power plants at Rajasthan and Madras. Why ?

28.16. Availability of fuel and moderator generally controls the economic selection of the reactor. Justify the statement.

28.17. Draw a neat diagram of organic-liquid cooled and moderated reactor power plant and list out its advantages and disadvantages over the other reactors.

28.18. What do you understand by breeding ? What factors control the breeding ?

28.19. Breeder reactors necessarily must be fast reactors. Justify the statement.

28.20. The breeding is better with thorium than uranium. Why ?

28.21. Draw a neat diagram of breeder reactor and list out its advantages and disadvantages. Why only sodium is used as coolant in breeder reactors ?

28.22. Why the moderator is not required in breeder reactors ?

28.23. Why the reactor containment is spherical or cylindrical with spherical dome ?

28.24. What do you understand by control of a reactor ? What different controls are necessary in a thermal power reactor ?

28.25. What factors are considered in selecting an economical site for nuclear power plant ?

28.26. What do you understand by 3-stage nuclear programme of India ?

28.27. Why long periods (20 to 30 years) are required, for a nation having large resources of fertile materials to start the breeder reactor power plants ?

28.28. List out the advantages and disadvantages of nuclear plants over conventional thermal plants.