AVC COLLEGE OF ENGINEERING
MANNAMPANDAL, MAYILADUTHURAI – 609 305

COURSE MATERIAL FOR THE SUBJECT

NUCLEAR ENGINEERING

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CERTIFICATE

This is to certify that this Course material for the Subject **Nuclear Engineering** covers the entire syllabus prescribed by Anna University, Trichy

HEAD OF THE DEPARTMENT **FACULTY IN-CHARGE**

PRINCIPAL

UNIT I

INTRODUCTION:

In 1803 John Dalton, attempting to explain the laws of chemical combination, proposed his simple but incomplete *atomic hypothesis.* He postulated that all elements consisted of indivisible minute particles of matter, *atoms,* * that were different for different elements and preserved their in density in chemical reactions. In 181I Amadeo Avogadro introduced the molecular theory based on the *molecule,* a particle of matter composed of a finite number of atoms. It is now known that the atoms are themselves composed of subparticles, common among atoms of all elements.

An atom consists of a relatively heavy, positively charged *nucleus* and a number of much lighter negatively charged *electrons* that exist in various orbits around the nucleus. The nucleus, in turn, consists of subparticles, called *nucleons.* Nucleons are primarily of two kinds: the *neutrons,* which are electrically neutral, and the *protons,* which are positively charged. The electric charge on the proton is equal iri magnitude but opposite in sign to that on the electron. The atom as a whole is electrically neutral: though number of protons equals the number of electrons in orbit. One atom may be transformed into another by losing or acquiring some of the above subparticles. Such reactions result in a change in mass Δm and therefore release . (or absorb) large quantities of energy ΔE, according to Einstein's law

$$\Delta E = \frac{1}{g_c} \Delta mc^2 \qquad \text{-----------------------} \ (1.1)$$

where c is the speed of light in vacuum and g_c is the familiar engineering conversion The above equation applies to *all* processes, physical, chemical, or nuclear, in which energy is released or absorbed. Energy is, however, classified as *nuclear* if it is associated with changes in the atomic nucleus. Figure 1.1 shows three atoms. Hydrogen has a nucleus composed of one proton, no neutrons, and one orbital electron. It is the only atom that has no neutrons. Deuterium has one proton and one neutron in its nucleus and one orbital electron. Helium contains two protons, two neutrons, and two electrons. The electrons exist in Orbits and each is quantitized as a lumped unit charge as shown.

Most of the mass of the atom is in the nucleus. The masses of the three primary atomic subparticles are

Neutron mass $m_n = 1.008665$ amu
Proton mass $m_p = 1.007277$ amu
Electron mass $m_e = 0.0005486$ amu

The abbreviation amu, *for atomic mass unit,* is a unit (mass approximately equal to1.66 X 10^{-27}kg, or 3.66 X 10^{-27} Ibm). These three particles are the primary building blocks of all atoms. Atoms differ in their mass because they contain varying numbers. Atoms with nuclei that have the same number of protons have similar chemical and physical characteristics and differ mainly in their masses. They are called *isotopes. For* example, deuterium, frequently called *heavy hydrogen,* is an isotope of hydrogen. It exists as one part in about 6660 in naturally occurring hydrogen. When combined with oxygen, ordinary hydrogen and deuterium form *ordinary water (or* simply water) and *heavy water,* respectively. The number of protons in the nucleus is called the *atomic number* Z. The total number of nucleons in the nucleus is called the *mass number A.* As the mass of a neutron or a proton is nearly 1 amu, *A* is the integer nearest the mass of the nucleus,

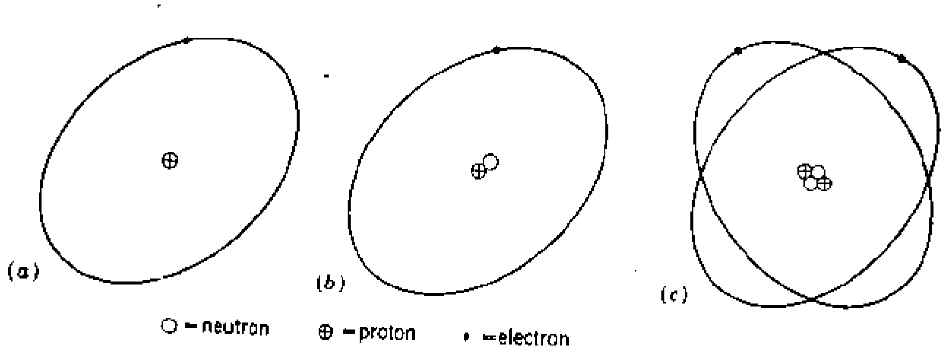


Figure 1.1 Structure of some light atoms: *(a)* hydrogen; *(b)* deuterium or heavy hydrogen, and *(c)* helium.

which in turn is approximately equal-to the atomic mass of the atom. Isotopes of the same element thus have the same atomic number but differ in mass number. Nuclear symbols are written conventionally as

$${}_{z}X^{A}$$

where X is the usual chemical symbol. Thus the hydrogen nucleus is H_1, deuterium is H_2 (and sometimes D), and ordinary helium is ${}_2He^4$. For particles containing no protons, the subscript indicates the magnitude and sign of the electric charge. Thus an electron is ${}_{\_}e^{o}$ (sometimes *e-* or (J) and a neutron is ${}_{o}n^{1}$. Symbols are also often written in the form He-4, helium-4, etc

Many elements (such as hydrogen, above) appear in nature as mixtures of isotopes of varying abundances. *For* example, naturally occurring uranium, called *natural uranium,* is composed of 99.282 mass percent U^{238}, 0.712 mass percent U^{235} and 0.006 mass percent U^{234}, where the atomic number is deleted. It is 92 in all cases. Many isotopes that do *not* appear in nature are synthesized in the laboratory or in nuclear reactors. *For* example, uranium is known to have a total of 14 isotopes that range in mass numbers from 227 to *240.*

Two other particles of importance are the positron and the neutrino. The *positron* is a positively charged electron having the symbols ${}_{+l}e^{o}$, *e+,* or β^+. The *neutrino* (little neutron) is a tiny electrically neutral particle that is difficult to observe experimentally. Initial evidence of its existence was based on theoretical considerations. In nuclear reactions where a β particle of either kind is emitted or captured, the resulting energy (corresponding to the lost mass) was *not* all accounted *for* by the energy of the emitted β particle and the recoiling nucleus. It was first suggested by Wolfgang Pauli in 1934 that the neutrino was simultaneously ejected in these reactions and that it carried titled balance of the energy, often larger than that carried by the β particle itself. The importance of neutrinos is that they carry some 5 percent of the total energy produced in fission. This energy is completely lost because neutrinos do *not* react and are *not* stopped by any practical structural material. The neutrino is given the symbol *v.* There are many other atomic subparticles. An example is the *mesons,* unstable positive, negative, or neutral particles that have masses intermediate between an electron and a proton. They are exchanged between nucleons and are thought to account *for* the forces between them. Electrons that orbit in the outermost shell of an atom are called *valence electrons.* The outermost shell is called the *valence shell.* Thus, hydrogen has one valence electron and its K shell is the valence shell, etc. Chemical properties of an element are a function of the number of valence electrons. The electrons play *little* or *not* part in nuclear interactions.

CHEMICAL AND NUCLEAR EQUATIONS

Chemical reaction involve the combination or separation of whole atoms. For example

$$C + O_2 \rightarrow CO_2 \quad \text{-------------------} \ (1.2)$$

This reaction is accompanied by the release of about 4 electron volts (eV). An *electron volt* is a unit of energy in common use in nuclear engineering. 1 eV = 1.6021 X 10^{-19} joules (1) = 1.519 X 10^{-22} Btu = 4.44 X 10^{-26} kWh. 1 *million electron volts* (1MeV) = 10^6 eV.

In chemical reactions, each atom participates as a whole and retains its identity. The molecules change. The only effect is a sharing or exchanging of valence electrons. The nuclei are unaffected. In chemical equations there are as many atoms of each participating element in the products (the right-hand side) as in the reactants (the left hand side). Another example is one in which uranium dioxide ($U0_2$) is converted into uranium tetrafluoride (UF_4), called green salt, by heating it in an atmosphere of highly corrosive anhydrous (without water) hydrogen fluoride (HF), with water vapor (H_2O) appearing in the products

$$UO_2 + 4HF \rightarrow 2H_2O + UF_4 \quad \text{-------------------} \ (1.3)$$

water vapor is driven off and UF_4 is used to prepare gaseous uranium hexafluoride (UF_6), which is used in the separation of the $U^{235}$and U^{238} isotopes of uranium by the gaseous diffusion method. (Fluorine has only one isotope, F19, and thus combinations of molecules of uranium and fluorine have molecular masses depending only on the uranium isotope.)

Both chemical and nuclear reactions are either *exothermic* or *endothermic,* that is, they either release or absorb energy. Because energy and mass are convertible, (Eq.1), chemical reactions involving energy *do* undergo a mass decrease in exothermic reactions and a mass increase in endothermic ones. However, the quantities of energy associated with a chemical reaction are very small compared with those of a nuclear reaction, and the mass that is lost or gained is minutely small. This is why we assume a preservation of mass in chemical reactions, undoubtedly an incorrect assumption but one that is sufficiently accurate for usual engineering calculations. In nuclear reactions, the reactant nuclei do not show up in the products, instead we may find either isotopes of the reactants or other nuclei. In balancing nuclear equations it is necessary to see that the same, or equivalent, nucleons show up in the products as entered the reaction. For example, if K, L, M, and N were chemical symbols, the corresponding nuclear equation might look like

$$_{Z1}K^{A1} + {}_{Z2}L^{A2} \rightarrow {}_{Z3}M^{A3} + {}_{Z4}N^{A4} \quad \text{------------------ (1.4)}$$

To balance Eq. (1-4), the following relationships must be satisfied.

$$Z_1 + Z_2 = Z_3 + Z_4$$

$$A_1 + A_2 = A_3 + A_4 \quad \text{----------------------(1.5)}$$

Sometimes the symbols γ or v are added to the products to indicate the emission of electromagnetic radiation or a neutrino, respectively. They have no effect on equation balance because both have zero Z and *A*, but they often carry large portions of the resulting energy.

Although the mass numbers are preserved in a nuclear reaction, the masses of the isotopes on both sides of the equation do not balance. Exothermic or endothermic energy is obtained when there is a reduction or an increase in mass from reactants to products, respectively.

Example 1-1 One exothermic reaction occurs when common aluminum is bombarded with high-energy a particles (helium-4 nuclei), resulting in Si ^{30} (a heavy isotope of silicon whose most abundant isotope has mass number 28). In the reaction, a small particle is emitted. Write the complete reaction and calculate the change in mass

SOLUTION: The reaction is

$$_{13}Al^{27} + {}_{2}He^{4} \rightarrow {}_{14}Si^{30} + {}_{Z,4}X^{A.4}$$

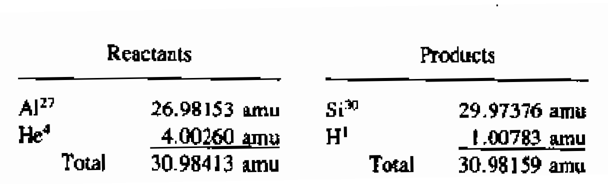
where X is a symbol of a yet unknown particle. Balancing gives

$$Z_4 = 13 + 2 - 14 = 1 \qquad \text{and} \qquad A_4 = 27 + 4 - 30 = 1$$

The only particle satisfying these is a proton. Thus the complete reaction is

$$_{13}Al^{27} + {}_{2}He^{4} \rightarrow {}_{14}Si^{30} + {}_{1}H^{1} \quad \text{------------ (1.6)}$$

The isotope masses of the nuclei showing up in this reaction are:



Reactants		Products	
Al^{27}	26.98153 amu	Si^{30}	29.97376 amu
He^4	4.00260 amu	H^1	1.00783 amu
Total	30.98413 amu	Total	30.98159 amu

Thus there is a *decrease* in mass, as Δm = 30.98159 - 30.98143 = *-0.00254* amu. The corresponding energy is negative; i.e., energy is released or is exothermic. In nuclear reactions, the results depend on a small difference between large numbers, which makes it necessary to carry the isotope masses to the fourth or fifth decimal places.

An example of an endothermic nuclear reaction is

$$ {}_7N^{14} + {}_2He^4 \rightarrow {}_8O^{17} + {}_1H^1 \quad \text{----------------------------(1.7)} $$

The sum of the masses of these reactants and products are *14.00307* + 4.00260 = *18.00567* amu and 16.99914 + 1.00783 = *18.00697* amu, respectively. Thus there is a net gain in mass of 0.00130 amu, which means that energy is absorbed and the reaction is endothermic.

In the above two reactions, the positively charged a particles must be accelerated to high kinetic energies to overcome electrical repulsion and bombard the positively . charged aluminum or nitrogen nuclei. The reactants possess initial kinetic energy equal to the kinetic energy of the a particle plus the kinetic energy of the nucleus, though the latter is usually negligible. (This process is analogous to raising a fuel-air mixture to its ignition temperature by adding activation energy before combustion can take place.) When the reactions are completed, the energy released will be equal to the initial energy of the reactants plus the energy corresponding to the lost mass (or minus the energy corresponding to the gained mass).

This energy shows up in the form of kinetic energy of the resultant particles, in the form of kinetic energy, and sometimes as *excitation energy* of the product nucleus, if any become so excited. The total kinetic energy of the products is divided among the nuclei and particles in such a manner that the lighter particles have higher kinetic energies than the heavier ones.

The isotope masses used above included the masses of the orbital electrons. The nuclear masses can be computed by subtracting the sum of the masses of 2 orbital electron. For example, the mass of the AI^{27} nucleus = 26.98153 -13 x *0.000548*= 26.97440 amu, and so on. Such corrections are unnecessary in most cases because the same number of electrons show up on both sides of the equation. . For example, in Eq. (1-6), the energy produced corresponds to the change in masses of the nuclei as given by

$$ \Delta m = [(M_{Si} - 14m_e) + (M_H - m_e)] - [(M_{Al} - 13m_e) + (M_{He} - 2m_e)] \quad \text{--------(1.8)} $$

where M is the isotope atomic mass and m, the mass of the electron. It can be seen that the number of electrons balance and that

$$ \Delta m = (M_{Si} + M_H) - (M_{Al} + M_{He}) \quad \text{----------------------------(1.9)} $$

The principle holds even if neutrons (whose mass, 1.008665 amu, does not include any electrons) are involved. In general then

$$\Delta m = \Sigma M_{\text{products}} - \Sigma M_{\text{reactants}} \quad \text{-------------------------(1.10)}$$

and the "electron masses" are neglected. This rule applies even if an electron appears on either side of the equation. An example is

$$_{16}S^{35} \rightarrow {}_{17}\mathrm{Cl}^{35} + {}_{-1}e^{0} \quad \text{----------------------------(1.11)}$$

In this case

$$\Delta m = [(M_{\mathrm{Cl}} - 17m_e) + m_e] - (M_S - 16m_e) = M_{\mathrm{Cl}} - M_S \text{ ---------(1.12)}$$

An *exception,* however, is in reactions involving positrons

In this case $$_{6}\mathrm{C}^{11} \rightarrow {}_{5}\mathrm{B}^{11} + {}_{+1}e^{0} \quad \text{----------------------------(1.13)}$$

$$\Delta m = [(M_{\mathrm{B}} - 5m_e) + m_e] - (M_{\mathrm{C}} - 6m_e) = M_{\mathrm{B}} - M_{\mathrm{C}} + 2m_e \text{ ----(1.14)}$$

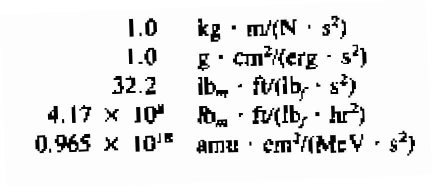
Two electron masses are added if the positron is on the right-hand side and subtracted if it is on the left-hand side of the equation.

ENERGY FROM NUCLEAR REACTIONS

The energy corresponding to the change in ma

$$\Delta E = \frac{1}{g_c} \Delta m c^2 \quad \text{----------------------------(1.15)}$$

where g_c is a conversion factor that has the following values



1.0	$\mathrm{kg \cdot m/(N \cdot s^2)}$
1.0	$\mathrm{g \cdot cm^2/(erg \cdot s^2)}$
32.2	$\mathrm{lb_m \cdot ft/(lb_f \cdot s^2)}$
4.17×10^{8}	$\mathrm{lb_m \cdot ft/(lb_f \cdot hr^2)}$
0.965×10^{18}	$\mathrm{amu \cdot cm^2/(MeV \cdot s^2)}$

Thus if *dm* is in kilograms and c in meters per second, Δ*E* will be in joules. Since c = 3 X10^8 m/s, Eq. (1-15) can be written in the form

$$\Delta E \text{ (in J)} = 9 \times 10^{16} \Delta m \text{ (in kg)}$$

But as it is convenient to express the masses of nuclei in amu = 1.66 X 10-^{27} kg and the energy in joules (1) and MeV, Eq. (1-15) becomes

$$\Delta E \text{ (in J)} = 1.49 \times 10^{-10} \Delta m \text{ (in amu)}$$

or

$$\Delta E \text{ (in MeV)} = 931 \Delta m \text{ (in amu)}$$

a useful relationship to remember. The reaction in Example 1-1 thus produces -0.00254 X 931 =- 2.365 (in MeV).

NUCLEAR FUSION AND FISSION

Nuclear reactions of importance in energy production are *fusion, fission.* and *radioactivity. Infusion,* two or more light nuclei fuse to form a heavier nucleus. In *fission.* a heavy nucleus is split into two or more lighter nuclei. In both, there is a decrease in mass resulting in exothermic energy.

Table 1.1

	Energy				
Mass	MeV	J	Btu	kWh	MW · day
amu	931.478	1.4924×10^{-10}	1.4145×10^{-13}	4.1456×10^{-17}	9.9494×10^{-13}
kg	5.6094×10^{29}	8.9873×10^{16}	8.5184×10^{13}	2.4965×10^{14}	5.9916×10^{19}
lb_m	2.5444×10^{29}	4.0766×10^{16}	3.8639×10^{13}	1.1324×10^{14}	2.7177×10^{14}

FUSION

Energy is produced in the sun and stars by continuous fusion reactions in which four nuclei of hydrogen fuse in a series of reactions involving other particles that continually appear and disappear in the course of the reactions, such as He^3, nitrogen, carbon, and other nuclei, but culminating in one nucleus of helium and two positrons

$$4\,{}_1H^1 \rightarrow {}_2He^4 + 2\,{}_{+1}e^0 \quad \text{----------------------------(1.16)}$$

resulting in a decrease in mass of about 0.0276 amu, corresponding to 25.7 MeV. The heat produced in these reactions maintains temperatures of the order of several million degrees in their cores and serves to trigger and sustain succeeding reactions. On earth, although fission preceded fusion in both weapons and power generation, the basic fusion reaction was discovered first, in the 1920s, during research on particle accelerators. Artificially produced fusion may be accomplished when two light atoms fuse into a larger one as there is a much greater probability of two particles colliding than of four. .The 4-hydrogen reaction requires, on an average, billions of years for completion, whereas the deuterium-deuterium reaction requires a fraction of a second. To cause fusion, it is necessary to accelerate the positively charged nuclei to high kinetic energies, in order to overcome electrical repulsive forces, by raising their temperature to hundreds of millions of degrees resulting in a plasma. The plasma must be prevented from contacting the walls of the container, and must be confined for a period of time (of the order of a second) at a minimum density. Fusion reactions are called *thermonuclear* because very high temperatures are required to trigger and sustain them Table 1-2 lists the possible fusion reactions and the energies produced by them. *n, p,* D, and T are the symbols for the neutron, proton, deuterium *(W)*, and tritium (H3), respectively. Many problems have to be solved before an artificially made fusion reactor becomes a reality. The most important of these are the difficulty in generating and maintaining high temperatures and the instabilities in the medium (plasma), the conversion of fusion energy to electricity, and many other problems of an operational nature.

Table 1.2

Number	Fusion reaction		Energy per reaction, MeV
	Reactants	Products	
1	D + D	T + p	4
2	D + D	He^3 + n	3.2
3	T + D	He^4 + n	17.6
4	He^3 + D	He^4 + p	18.3

FISSION

Unlike fusion, which involves nuclei of similar electric charge and therefore requires high kinetic energies, *fission* can be caused by the neutron, which, being electrically neutral, can strike and fission the positively charged nucleus at high, moderate, or low speeds without being repulsed. Fission can be caused by other particles, but neutrons are the only practical ones that result in a sustained reaction because two or three neutrons are usually released for each one absorbed in fission. These keep the reaction going. There are only a few fissionable isotopes. U^{235}, PU^{239},and $U^{233}$are fissionable by neutrons of all energies. U^{238}, Th^{232} and PU^{240} are fissionable by high-energy neutrons only. An example, shown schematically in Fig. 1-3, is

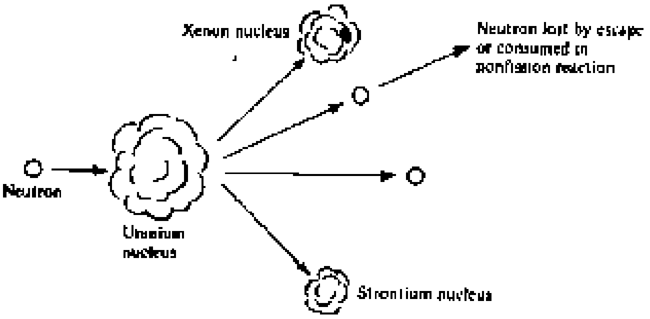


Fig. 1.3 A typical fission reaction.

$$ {}_{92}U^{235} + {}_{0}n^{1} \rightarrow {}_{54}Xe^{140} + {}_{38}Sr^{94} + 2{}_{0}n^{1} \text{ --------------------(1.17)} $$

The immediate (prompt) products of a fission reaction, such as Xe^{l40} and Sr^{94} above, are called *fission fragments.* They, and their decay products are called *fission products.* Figure 1-4 shows fission product data for U^{235} by thermal and fast neutrons and for $U^{233}$and $Pu^{239}$by thermal neutrons. The products are represented by their mass numbers.

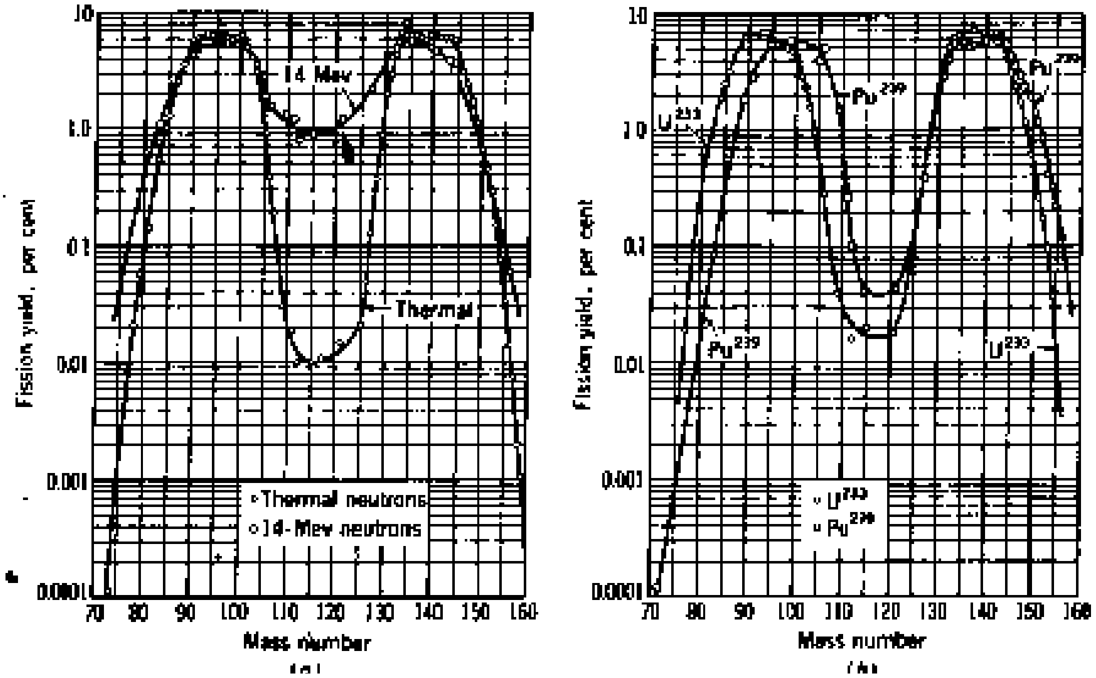


Fig 1.4 Fission product yield data for *(a)* U^{235} by thermal and 14 MeV neutrons and *(b)* $U^{233}$and PU^{239}. by thennal neutrons

ENERGY FROM FISSION AND FUEL BURNUP

There are many fission reactions that release different energy values. The one in Eq. (9-15), for example, yields 196 MeV. Another

$$ {}_{92}U^{235} + {}_{0}n^{1} \rightarrow {}_{56}Ba^{137} + {}_{36}Kr^{97} + 2{}_{0}n^{1} \text{ ---------------------------(1.18)} $$

has the mass balance

$$ 235.0439 + 1.00867 \longrightarrow 136.9061 + 96.9212 + 2 \times 1.00867 $$

$$ 236.0526 \longrightarrow 235.8446 $$

$$ \Delta m = 235.8446 - 236.0526 = -0.2080 \text{ amu} $$

Thus

$$ \Delta E = 931 \times -0.2080 = -193.6 \text{ MeV} = -3.1 \times 10\text{-}11\text{J} $$

$$ = -2.937 \text{ X } 10\text{-}11\text{Btu} $$

On the *average* the fission of a U^{235} nucleus yields about 193 MeV. The same figure roughly applies to $U^{233}$and PU^{239}.This amount of energy is *prompt,* i.e., released at the time of fission. More energy, however, is produced because of (1) the slow decay of the fission fragments into fission products and (2) the non fission capture of excess neutrons in reactions that produce energy, though much less than that of fission though much less than that of fission.

The *total energy.* produced *per* fission reaction, therefore, is greater than the prompt energy and is about 200 MeV, a useful number to remember.

The complete fission *of* 1 g *of* U235nuclei thus produces

$$\frac{\text{Avogadro's number}}{U^{235}\text{ isotope mass}} \times 200\text{ MeV} = \frac{0.60225 \times 10^{24}}{235.0439} \times 200$$

$$= 0.513 \text{ X } 10^{24}\text{ MeV}$$
$$= 2.276 \text{ x } 10^{24}\text{ kWh}$$
$$= 8.190 \text{ X } 10^{10}\ J$$
$$= 0.948\text{ MW-day}$$

Another convenient figure to remember is that a reactor burning 1g *of* fissionable material generates nearly 1 MW-day *of* energy. This relates to fuel *burnup.* Maximum theoretical burnup would therefore be about a million MW-day/ton (metric) *of* fuel. This figure applies if the fuel were entirely composed *of* fissionable nuclei and all *of* them fission. Reactor fuel, however, contains other non fissionable isotopes *of* uranium, plutonium, *or* thorium. *Fuel* is defined as all uranium, plutonium, and thorium isotopes. It does not include alloying *or* other chemical compounds or mixtures. The term *fuel material* is used to refer to fuel plus such other materials. Even the fissionable isotopes cannot be all fissioned because *of* the accumulation *of* fission products that absorb neutrons and eventually stop the chain reaction. Because *of* this and owing to metallurgical reasons such as the inability *of* the fuel material to operate at high temperatures *or* to retain gaseous fission products such as Xe and Kr, Eqs. (1-17) and (1 18) in its structure except *for* limited periods *of* time-burn up values are much lower than this figure. They are, however, increased somewhat by the fissioning *of* some fissionable nuclei, such as Pu^{239},which are newly converted *from* fertile nuclei, such as U^{238} (Depending upon fuel type and *enrichment (mass* percent *of* fissionable fuel in all fuel), burnups may vary *from* about 1000 to 100,000 MW . day/ton and higher.

RADIOACTIVITY

Radioactivity is an important source *of* energy *for* small power devices and a source *of* radiation for use in research, industry, medicine, and a wide variety *of* applications, as well as an environmental concern. Most *of* the naturally occurring isotopes are *stable.* Those that are not stable, i.e., *radioactive,* are some isotopes *of* the heavy elements thallium (Z = 81),lead(Z = 82), and bismuth (Z = 83) and all the isotopes *of* the heavier elements beginning with polonium (Z = 84). A few lower-mass naturally occurring isotopes are radioactive, such as K^{40},Rb^{87} and In^{115}. In addition, several thousand artificially produced isotopes *of* all masses are radioactive. Natural and artificial radioactive isotopes, also called *radioisotopes,* have similar disintegration rate mechanisms. Figure 1-5 shows a *Z-N* chart *of* the known isotopes.

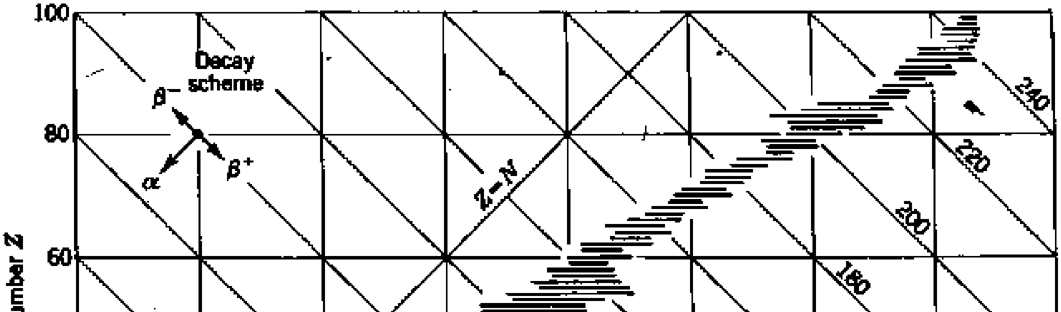


Fig 1.5. *Z-N* chart *of* the known isotopes

Radioactivity means that a radioactive isotope continuously undergoes spontaneous (i.e., without outside help) disintegration, usually with the emission of one or more smaller particles from the *parent* nucleus, changing it into another, or *daughter*, nucleus. The parent nucleus is said to *decay* into the daughter nucleus. The daughter mayor may not be stable, and several successive decays may occur until a stable isotope is formed. An example of radioactivity is

$$_{49}In^{115} \rightarrow {}_{50}Sn^{115} + {}_{-1}e^{0} \qquad \text{----------------------------(1.19)}$$

where the part, In^{115}, is a naturally occurring radioisotope and its daughter, Sn^{ll5} is stable.

Radioactivity is *always* accompanied by a *decrease* in mass and is thus always exothermic. The energy liberated shows up as kinetic energy of the emitted particles and as γ radiation. The light particle is ejected at high speed, whereas the heavy one recoils at a much slower pace in an opposite direction. Naturally occurring radioisotopes emit α, β. or γ particles or radiations. The artificial isotopes, in addition to the above, emit or undergo the following particles or reactions: positrons; orbital electron absorption, called K capture; and neutrons. In addition, neutrino emission accompanies β emission (of either sign).

Alpha decay Alpha particles are helium nuclei, each consisting of two protons and two neutrons. They are commonly emitted by the heavier radioactive nuclei. An example is the decay of PU^{239}into fissionable U^{235}

${}_{94}Pu^{239} \rightarrow {}_{92}U^{235} + {}_{2}He^{4}$ ----------------------------(1.20)

Beta decay An example of β decay, besides Eq. (1-21), is

${}_{82}Pb^{214} \rightarrow {}_{83}Bi^{214} + {}_{-1}e^{0} + \nu$ ----------------------------(1.21)

where v, the symbol for the neutrino, is often dropped from the equation. The penetrating power of β particles is small compared with that of γ- rays but is larger than that of α particles. β - and α- particle decay are usually accompanied by the emission of' γ radiation.

Gamma radiation This is electromagnetic radiation of extremely short wavelength and very high frequency and therefore high energy γ rays and X-rays are physically similar but differ in their origin and energy: γ rays from the nucleus and X-rays from the atom because of orbital electrons changing orbits or energy levels. Gamma wavelengths are, on an average, about one-tenth those of X-rays, although the energy ranges overlap somewhat. Gamma decay does not alter either the atomic or mass numbers.

Positron decay Positron decay occurs when the radioactive nucleus contains an excess of protons. It effectively converts a proton into a neutron. An example is

${}_{7}Ni^{13} \rightarrow {}_{6}C^{13} + {}_{+1}e^{0}$ ----------------------------(1.22)

Because the daughter has one less proton than the parent, one of the orbital electrons is released to maintain atom neutrality. It combines with an emitted positron according to

The two particles therefore undergo an *annihilation* process, which produces γ energy equivalent to the sum of their rest masses $2\ m_e c^2$, or - (2 X 0.0005486)931 = -1.02 MeV. The reverse of the annihilation process is called *pair production.* In this, a γ ray Photon of at least 1.02 MeV energy forms a positron-electron pair. This is an endothermic process that Converts energy to mass. It occurs in the presence of matter and Never in a vacuum.

K capture K capture also takes place when a nucleus possesses an excess of protons but not the threshold of 1.02 MeV necessary to emit a positron. Instead it captures an orbital

${}_{29}Cu^{64} + {}_{-1}e^{0} \rightarrow {}_{28}Ni^{64}$ ---------------------------(1.23)

electron from the orbit or shell nearest to the nucleus, called the K shell; hence the name K capture. The vacancy in the K shell is filled by another electron falling from a higher orbit. Thus K capture is accompanied by x-ray emission from the atom. K capture also effectively changes a proton into a neutron. The process is shown in Fig. 9-6. An example of K capture is in which $Ni^{64}$is stable. Because the parent acquires an electron, the electron symbol is on the left-hand side.

Neutron emission This occurs when a nucleus possesses an extremely high excitation energy. The *binding energy* of a neutron in a nucleus (the energy that would have to be added to a nucleus to expel a neutron) varies with mass number but averages about 8 MeV. Thus, if the excitation energy of a nucleus were at least 8 MeV, it could decay by the emission of a neutron. An example is

$${}_{54}Xe^{137} \rightarrow {}_{54}Xe^{136} + {}_{0}n^{1} \quad \text{----------------------------(1.24)}$$

The parent $Xe^{137}$is a fission product resulting from the *β* decay of the fission fragment I^{137}(called a *precursor).* In neutron decay the daughter is an isotope of the parent. It is a rare occurrence except in nuclear reactors where it is the source of *delayed fission neutrons,* which are of utmost importance in reactor control.

DECAY RATES AND HALF -LIVES

There can be no indication of the time that it takes anyone particular radioactive nucleus to decay. However, if there are a very large number of radioactive nuclei of the same kind, there is a definite statistical probability that a certain fraction will decay in a certain time. Thus if we have two separate samples, one containing 1020and the other 1030of the same radioisotopes, we will find that the same fraction in each, say one-half or 1020/2and 103012,will decay in the same time. In other words, the rate of decay is a function only of the number of radioactive nuclei present at any time, provided that the number is large (true in most cases of practical interest). Radioactive-decay rates, unlike chemical reaction rates, which increase exponentially with temperature, are practically unaffected by temperature, pressure, or the physical and chemical states of matter, i.e., whether in a gaseous, liquid, or solid phase or in chemical combinations with others.

If N is the number of radioactive nuclei of one species at any time *θ*, and if *dN* is the number decaying in an increment of time d*θ*, at *θ*, the rate of decay *-dN/dθ* is directly proportional to N

$$-\frac{dN}{d\theta} = \lambda N \quad \text{----------------------------(1.25)}$$

A is a proportionality factor called the *decay constant.* It has different values for different isotopes and the dimension time^{-1}, usually S^{-1}. Integrating between an arbitrary time, *θ* = 0, when the number of radioisotopes
was No, gives

$$-\int_{N_0}^{N}\frac{dN}{N} = \lambda\int_{0}^{\theta} d\theta \quad \text{----------------------------(1.26)}$$

thus

$$N = N_0 e^{-\lambda\theta} \quad \text{----------------------------(1.27)}$$

The rate of decay λN is also called the *activity* A and commonly has the dimension disintegration per second (dis/s) or S^{-1}. The initial activity Ao is equal to λ*No.* Thus

$$A = A_0 e^{-\lambda\theta} \qquad \text{----------------------------(1.28)}$$

A common way of representing decay rates is by the use of the *half-life,* 81/2'. This is the time during which one-half of a number of radioactive species decays or one-half of their activity ceases. Thus

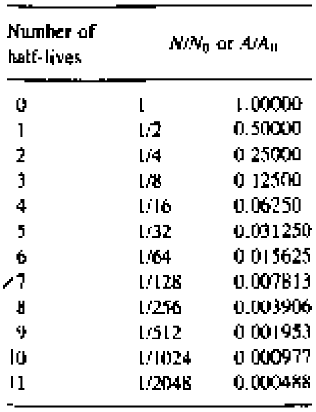
$$\frac{N}{N_0} = \frac{A}{A_0} = \frac{1}{2} = e^{-\lambda\theta_{1/2}} \qquad \text{----------------------------(1.29)}$$

and

$$\theta_{1/2} = \frac{\ln 2}{\lambda} = \frac{0.6931}{\lambda}$$

and the half-life is inversely proportional to the decay constant. Starting at = 0 when N = No, one-half of No decay after one half-life; one-half of the remaining atoms, or one-quarter of No, decay during the second half-life; one-eighth of No during the third, and so on (Table 1-3 and Fig. 1-7). The fraction of the initial number of parent nuclei or activity remaining after n half-lives is equal to $(1/2)^n$. Theoretically, it takes an infinite time for the activity to cease. However, about 10 half-lives reduce the activity to less than one-tenth of 1 percent of the original negligible in many cases. Half-lives of radioisotopes vary from fractions of a microsecond to billions of years, and no two have the same half-lives. They are "fingerprints" by which a particular radioactive species may be identified. This is done by measuring

Table 1.3 Activity and half life



Number of half-lives	N/N_0 or A/A_0	
0	1	1.00000
1	1/2	0.50000
2	1/4	0.25000
3	1/8	0.12500
4	1/16	0.06250
5	1/32	0.031250
6	1/64	0.015625
7	1/128	0.007813
8	1/256	0.003906
9	1/512	0.001953
10	1/1024	0.000977
11	1/2048	0.000488

the change in activity with time and computing A from the slope of the activity history on a semilog plot (Fig. 1-8), from which 8112and the unknown specie are identified. There are cases that involve two transitions from the parent isotope with two decay rates and .two half-lives. In some cases two different half-lives represent one transition. Table 9-4 gives the half-lives and type of activity of some important radioisotopes. Note that the readily fissionable isotopes U^{233}, U^{235} and PU^{239} have extremely long half-lives, so they can be stored practically indefinitely. $U^{233}$and Pu^{239} are artificially produced (from $Th^{232}$and U^{238} respectively, themselves very long lived), whereas U^{235} is found in nature.

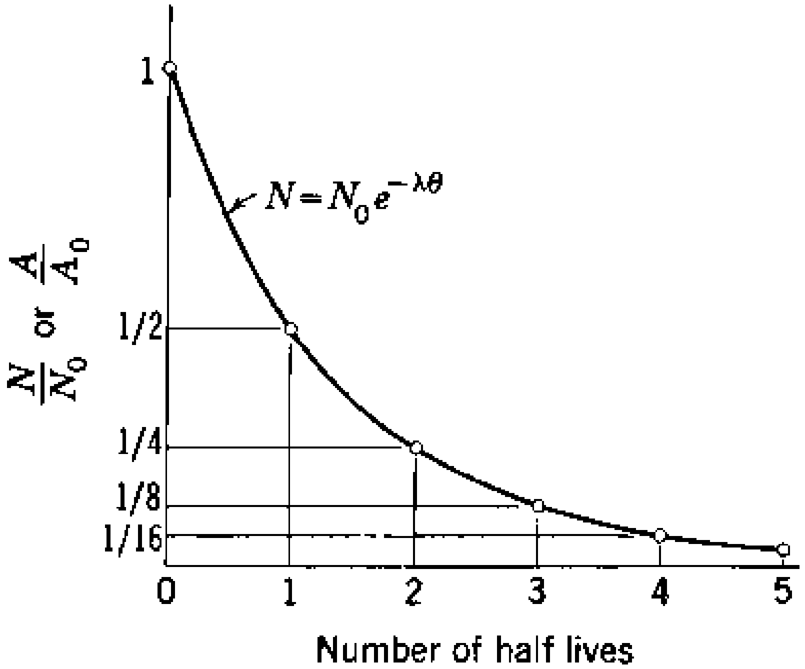


Fig 1.6. Radioactive-decay rates as a function of half-life.

The energy generated by the decaying fission products results in continuing, though much reduced, energy generation in a reactor after shutdown and must be removed by an adequate coolant system.

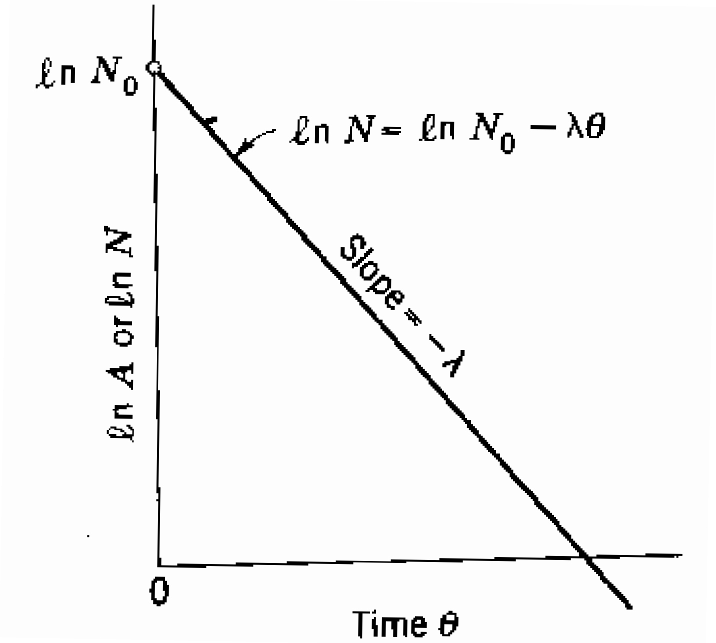


Fig 1.8. Radioactive-decay curve on a semi log plot.

Example 9-2 Radium 226 decays into radon gas. Compute (I) the decay constant and (2) the initial activity of I g of radium 226. The atomic mass is 226.0245 amu.

SOLUTION

(I) Half-life of Ra226 = 1600 yr = 5.049 x 10^{10} s (from Table 1-5)

$$\lambda = \frac{0.6931}{5.049 \times 10^{10}} = 1.3727 \times 10^{-11} \text{ s}^{-1}$$

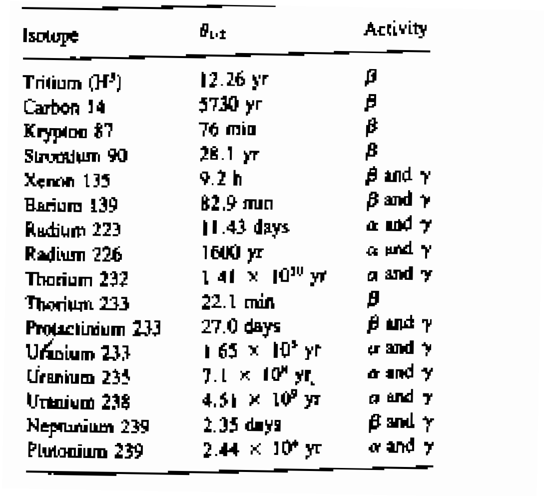
$$\text{(2) Number of atoms per gram} = \frac{\text{Avogadro's number}}{\text{atomic mass}}$$

$$= \frac{0.60225 \times 10^{24}}{226.0245} = 2.6645 \times 10^{21}$$

$$\text{Initial activity } A_0 = \lambda N_0 = 1.3727 \times 10^{-11} \times 2.6645 \times 10^{21}$$

$$= 3.6576 \times 10^{10} \text{ dis/s}$$

Table 1.4 Half lives of some isotopes



Isotope	$\theta_{1/2}$	Activity
Tritium (H^3)	12.26 yr	β
Carbon 14	5730 yr	β
Krypton 87	76 min	β
Strontium 90	28.1 yr	β
Xenon 135	9.2 h	β and γ
Barium 139	82.9 min	β and γ
Radium 223	11.43 days	α and γ
Radium 226	1600 yr	α and γ
Thorium 232	1.41×10^{10} yr	α and γ
Thorium 233	22.1 min	β
Protactinium 233	27.0 days	β and γ
Uranium 233	1.65×10^{5} yr	α and γ
Uranium 235	7.1×10^{8} yr	α and γ
Uranium 238	4.51×10^{9} yr	α and γ
Neptunium 239	2.35 days	β and γ
Plutonium 239	2.44×10^{4} yr	α and γ

Thus the activity of I g of Ra^{226} is very small compared with the number of atoms in it and may be considered practically constant, true for any species with a sufficiently long half-life. Early measurements showed the activity of 1 g of radium to be 3.7 x 10^{10} dis/s instead of the more correct value above. 3.7 x 10^{10} was adopted as a unit of radioactivity and is called a *curie* (ci). A millicurie (mci) is one-thousandth of a curie and is a common unit.

NEUTRON ENERGIES

Because neutrons are essential to the fission process, this and subsequent sections will deal with them and their interactions. As any other body, the kinetic energy of a neutron KE_n is given by

$$KE_n = \frac{1}{2g_c} m_n V^2 \qquad \text{---------------------------(1.30)}$$

where m_n = mass of neutron

v = speed of neutron
g_c = conversion factor

$$E_n = \frac{1}{2 \times 0.965 \times 10^{18}} \times 1.008665V^2 \quad \text{------------------(1.31)}$$

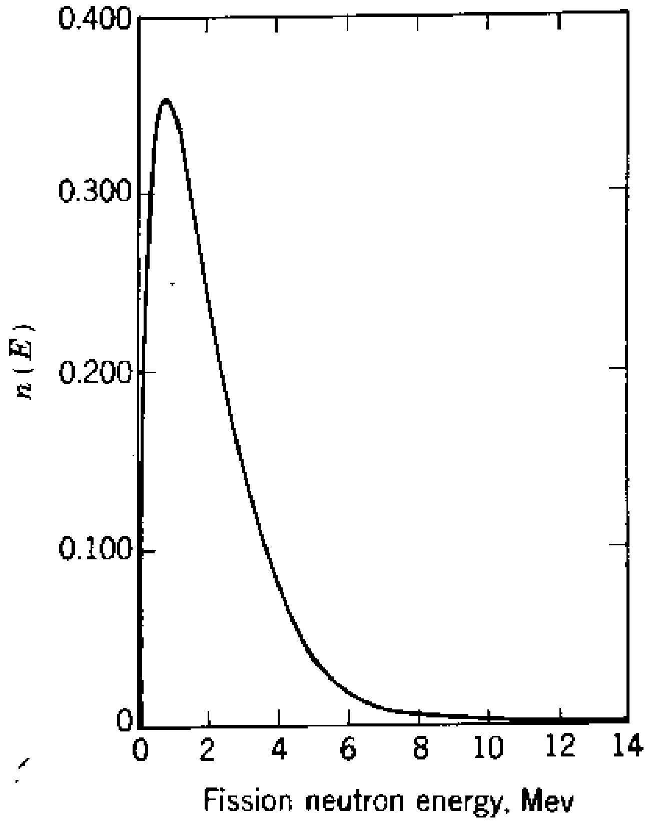
or

$$E_n = 5.227 \times 10^{-19}V^2 \text{ MeV} = 5.227 \times 10^{-13}V^2 \text{ eV} \quad \text{-----------(1.32)}$$

where V is the centimeters per second.

The newly born fission neutrons have energies ranging between less than 0.075 to about 17 MeV. When they travel through matter, they collide with nuclei and are decelerated, mainly by the lighter nuclei, thus giving up some of their energy with each successive collision. This process is called *scattering.*

Neutrons are classified into three categories according to energy *fast* (greater than 10^5eV), *intermediate,* and *slow* (less than 1 eV). One of main reactor classifications is the energy range of the neutrons causing fission. A *fast reactor* is one dependent primarily on fast neutrons for fission. A *thermal reactor* is one utilizing mostly *thermal neutrons.*



1.9. Energy spectrum of fission neutrons.

Newly born fission neutrons carry, on an average, about 2 percent of a reactor fission energy. They are either *prompt* or *delayed.* Prompt neutrons are released at the time of fission, within about 10^{-14}s (from fission fragments with a neutron-proton ratio the same as the original nucleus but greater than that corresponding to their mass number). Delayed neutrons, produced in radioactive decay of some fission products constitute only 0.645 percent of the total fission neutrons in U^{235} fission (less for $PU^{239}$and U^{233}).Their energies are small compared with those of prompt neutrons but they play a major role in reactor control.

Prompt neutrons have an energy distribution shown in Fig. 1-9 and given (for U^{235} and PU^{239} fission) by

$$n(E)\, dE_n = \sqrt{\frac{2}{\pi e}} \sinh \sqrt{2E_n}\, e^{-E_n}\, dE_n \qquad \text{--------------------(1.33)}$$

where *neE)* is the number of neutrons having energy E_n per unit energy interval dE_n.
Most of the prompt fission neutrons have energies less than1 MeV but average around
2 MeV.

THERMAL NEUTRONS

Fission neutrons are scattered or slowed down by the materials in the core. An effective scattering medium, called a *moderator,* is one which has small nuclei with high neutron scattering and low neutron-absorption probabilities, such as H and D (in H2O and D2O), C (graphite), and Be or BeO. The lowest energies they reach are those that put them in thermal equilibrium with the molecules of the medium they are in. They become thermalized and are called *thermal neutrons,* a special category of slow neutrons. Neutrons, like molecules at a given temperature, possess a wide range of energies and corresponding speeds (Fig. *9-lOa).* The velocity distribution, shown for two temperatures in Fig. *9-lOb,* is expressed by the *Maxwell distribution law*

$$n(V)\, dV = 4\pi n \left(\frac{m}{g_c 2\pi kT}\right)^{1.5} V^2 e^{-g_c(mV^2/2kT)}\, dV \qquad \text{-------------------------(1.34)}$$

where *n(V)* = number density of particles, present in given volum of medium, per unit velocity interval *dV* between *V* and *V + dV*
n = total number of particles in same volume of medium
m = mass of particle
k = Boltzmann's constant (universal gas constant divide by Avogadro's number)
= 1.3805 X 10-23J/K, or 8.61x 10-11MeV/K)
T = absolute temperature

The *most probable speed Vm* is the one that corresponds to the maximum number density evaluated by differentiating the right-hand side of Eq. (9-31) with respect to *V* and equating the derivative to zero, resulting in

$$V_m = \left(\frac{g_c 2kT}{m}\right)^{0.5} \qquad \text{---------------------------(1.35)}$$

The *energy corresponding to the most probable speed* (which is not the same as the most probable energy) is

$$E_m = \frac{1}{2g_c} m V_m^2 = kT \qquad \text{---------------------------(1.36)}$$

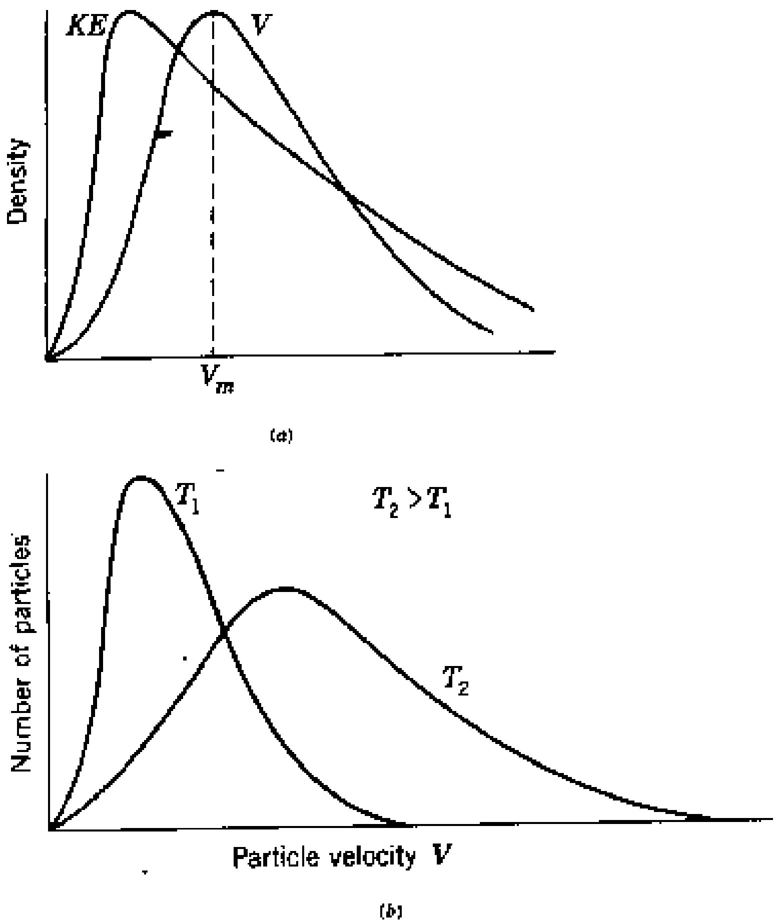


Fig 1.10 *(a)* Kinetic energy and velocity distributions of Thermal neutrons at a given temperature.*(b)*Velocity distribution at two temperatures.

Table 1.5 Thermal-neutron speeds and energies

Temperature		Most probable speed, m/s	Corresponding energy, eV
°C	°F		
20	68	2,200	0.0252
260	500	2,964	0.0459
537.8	1000	3,656	0.0699
1000	1832	4.580	0.1097

Thus the energy of the thermalized particle is independent of mass and a function only of the absolute temperature of the medium. The independence is also true for the shape of the energy-distribution curve so that when neutrons become thermalized in a medium, they possess the same energy distribution of the molecules of the medium. The speeds, however, are dependent on mass, and the speed distributions of neutrons and molecules are different. Using the neutron mass in grams and Boltzmann's constant in eV/K gives

$$Vm\text{(in m/s)} = 128.39\ T^{0.5}\ \text{(for a neutron only)-----------------(1.37)}$$

and

$$Em\ \text{(in eV)} = 8.617 \times 10^{-5}\ T\ \text{(for any particle) ---------------- (1-38)}$$

where T is in Kelvin.

Table 9-5 contains some thermal-neutron most probable energies and speeds as a function of temperature. The speed of 2200 m/s and energy of 0.0252 eV at 20°C are sometimes said to be "standard." Cross sections (Sec. 9-11) for thermal neutrons are customarily tabulated for 2200-m/s neutrons. Neutrons having energies greater than thermal, such as those in the process of slowing down in a thermal reactor, are called *epithermal* neutrons.

NUCLEAR CROSS SECTIONS

Assume a beam of mono energetic neutrons of intensity *10*neutrons/s impinging on a body having a target area A cm^2 and a nuclear density N nuclei/cm^3 (Fig. 1-11). The nuclei have radii roughly 1/1000 those of atoms and, therefore, have a cross-sectional area, facing the neutron beam, that is very small compared with the total target area. Taking one nucleus into consideration, we may use the analogy of a large number of peas (neutrons) being shot at a basketball (nucleus) in the center of a window (target area). The number of peas that will collide with the basketball is proportional to its cross-sectional area. However, the fraction colliding with the basketball, or the *probability* of collision, is equal to the cross-sectional area of the basketball divided by the area of the window. The actual cross-sectional area of a nucleus is obtained from its radius r_c

$$r_c = r_0 A^{1/3}$$

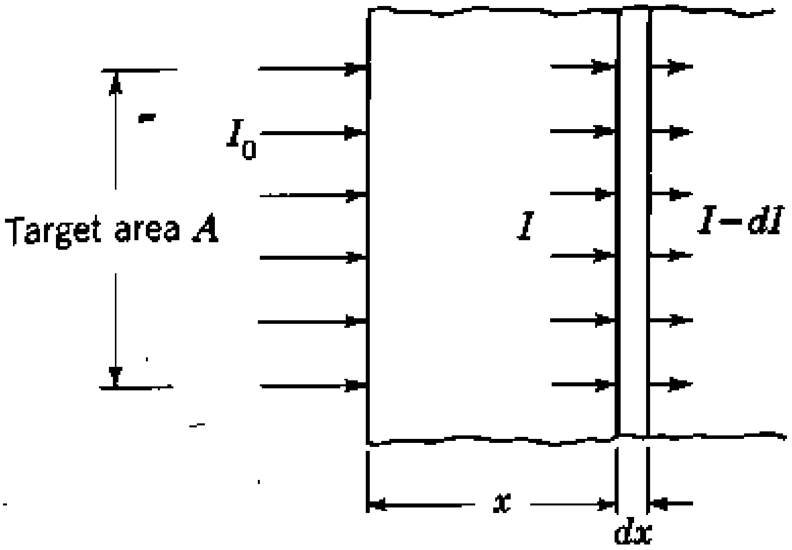


Fig 1.11. Neutron beam striking target area A.

where r_o is a constant varying for different nuclei with an average of 1.4 x 10-1^3 cm and *A* is the mass number. The average cross-sectional areas of nuclei therefore is about 10^{-24} cm^2.

The probability of neutrons colliding or interacting with nuclei is proportional to an *effective* rather than actual, cross-sectional area of the nuclei in question. This is called the *microscopic cross section,* or simply the *cross section* of the reaction and is given the symbol σ. It varies with the nucleus, type of reaction, and neutron energy.

In Fig. 1-11 the number of nuclei in volume *A dx* is *(NA dx)* As the neutron .beam passes through *dx* some of the neutrons are removed (by absorption or scatter) from the beam. The fraction removed is equal to the ratio of *effective* cross-sectional areas of the nuclei, *a(NA dX),* to the total area *A.* Thus, if at *x* and *x* + *dx* the beam intensities become *I* and *I* - *dI* respectively, it follows that, in the limit

)

$$\frac{-dI}{I} = \frac{\sigma(NA\,dx)}{A} = \sigma N\,dx$$

Integrating

$$-\int_{I_0}^{I} \frac{dI}{I} = \sigma N \int_0^x dx \qquad \text{---------------- (1-39a)}$$

From which

$$I = I_0 e^{-\sigma N x}$$

σ has the units of area. Because nuclei are small, cm^2 is too large a unit. Instead, the actual cross-sectional area of an average nucleus, 10^{-24} cm^2, was taken as the unit of the microscopic cross section and given the name *barn.* Cross-sectional values vary from small fractions of a millibarn to several thousand barns. Neutrons have as many cross sections as there are reactions. The most important are the scattering and absorption cross sections: σ_s, σ_a, σ_c, and σ_f.

where σ_s = microscopic cross section for scattering*
σ_a = microscopic cross section for absorption = $\sigma_c + \sigma_f$
σ_c = microscopic cross section for radiative (nonfission) capture
σ_f =microscopic cross section for fission

Sometimes only a *total* cross section σ_t is given, where $\sigma_t = \sigma_s + \sigma_a$ + any other. Cross-sectional energy plots for so nuclei of interest are shown in Figs. 1-12 through 1-14.

The product σ*N* is equal to the total cross sections of all the nuclei present in a unit volume. It is called the *macroscopic cross section* and is given the symbol ∑. It has the unit of length^{-1}, commonly cm^{-1} . Thus

$$\Sigma = N\sigma \qquad \text{---------------- (1-40)}$$

and Eq. *(1-39a)*can be written in the form

$$I = I_0 e^{-\Sigma x} \qquad \text{---------------- (1-39b)}$$

Macroscopic cross sections are also designated according to the reaction they represent. Thus $\sum_f = N\sigma_f$, $\sum_s = N\sigma_s$ etc. The reciprocal of macroscopic cross section for any reaction is the *mean free path* for that reaction. It has the symbol A, not to be confused with the decay constant in radioactivity. For an element of atomic mass *At* and density ρ (g/cm3), *N* (nuclei/cm3) can be calculated from

$$N = \rho \frac{\text{Avogadro's number}}{A_t}$$

Scattering is of two kinds, *inelastic* and *elastic. Inelastic scattering* occurs with high energy neutrons and the reduction in neutron kinetic energy shows up partly as kinetic energy and partly as excitation energy of the struck nucleus. *Elastic scattering* occurs with low-energy neutrons when the neutrons have slowed down and no longer possess sufficient energy to excite the nucleus. The struck neutron is not excited, and the kinetic-energy loss of the neutron is equal to the kinetic-energy gain of the nucleus.

NEUTRON FLUX AND REACTION RATES

The number of neutrons crossing a unit area per unit time in one direction is called the *neutron current* and is proportional to the gradient of *neutron density.* In a reactor core, however, the neutrons travel in all directions. If *n* is the neutron density (neutrons/cm^3) and *V* the neutron velocity cm/s, the product *nV* is the numberof neutrons ' crossing a unit area from all directions per unit time and is called the *neutron flux* φ
Thus

$$\varphi = nV \text{----------------------------------} (1\text{-}40)$$

φ has the unit neutrons/(s. cm2), which is often dropped. Because flux involves all neutrons at a given point, the reaction rate between neutrons and nuclei is proportional to it.

Fluxes are dependent upon velocity *V*, that is, upon energy, but are often quoted for broad energy ranges, such as thermal and fast. In a reactor core they vary from maximum, usually at the core geometric center, to minimum, near the edges. In thermal heterogeneous reactors, where the fuel and moderator are separate, fission neutrons are born in the fuel and thermalized in the moderator. Fast fluxes thus peak above average in the fuel, whereas thermal fluxes peak in the moderator (Fig. 1-15). Maximum full power thermal fluxes vary from $10^6$for small training reactors to as high as 10^{15} for power and research reactors.

Now, if a medium containing nuclei of density *N* is subjected to a neutron flux φ the *reaction rate,* between the nuclei and the neutrons, is given by

Reaction rate = $nVN\sigma = \sum\varphi$ reactions/(s. cm3) ------------------- (1-41)

where σ and $\sum$ are the cross sections of the particular reaction in question (i.e., absorption, scatter, etc.). Equation (1-41) simply states that the number of neutrons entering a particular reaction (the same as the number of reactions) per unit time and volume is proportional to the total distance traveled by all the neutrons in a unit volume during a unit time *(nV)* and to the total number of nuclei per unit volume *(N):* σ , the probability of the reaction, is the proportionality factor. Since in general *N* is fixed in a medium, the rates of a particular reaction (fixed ')are directly proportional to the neutron flux. It will suffice here to state that heat generation by fission at a given point in a reactor core is proportional to the neutron flux at that point. A knowledge of the neutron-flux distribution in a reactor core is therefore necessary for the study of heat generation and removal in that core.

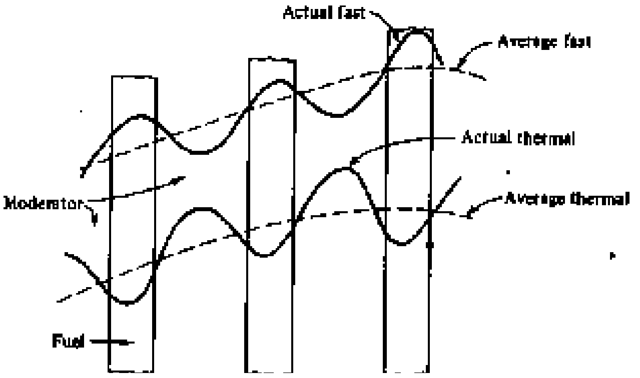


Fig.1.12. Neutron-flux distributions in fuel and moderator in a heterogeneous thermal reactor.

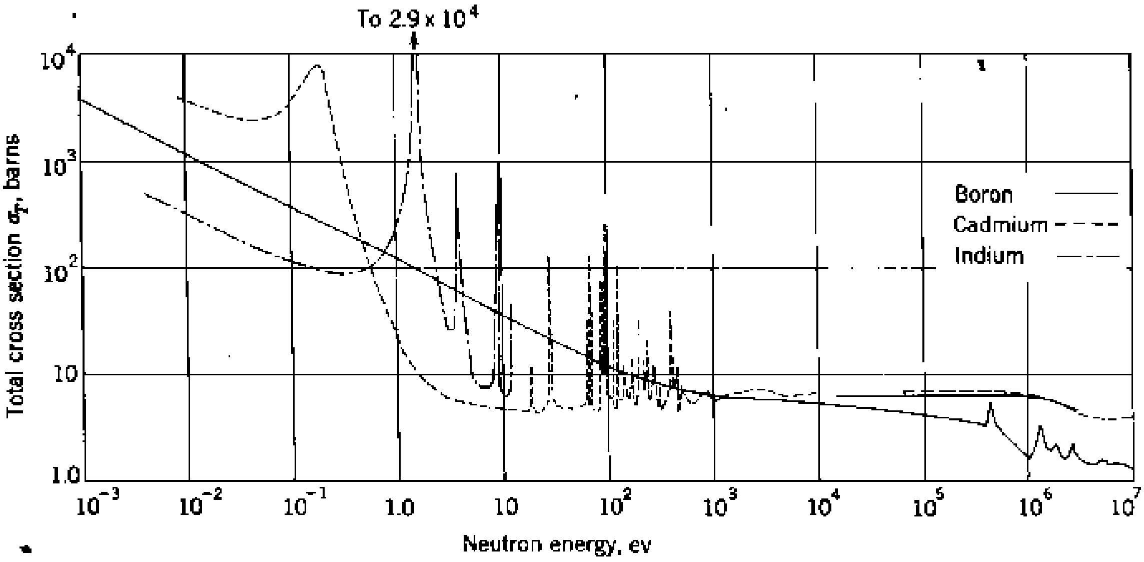


Fig 1.13. Neutron cross section for Cadmium, Indium and Boron

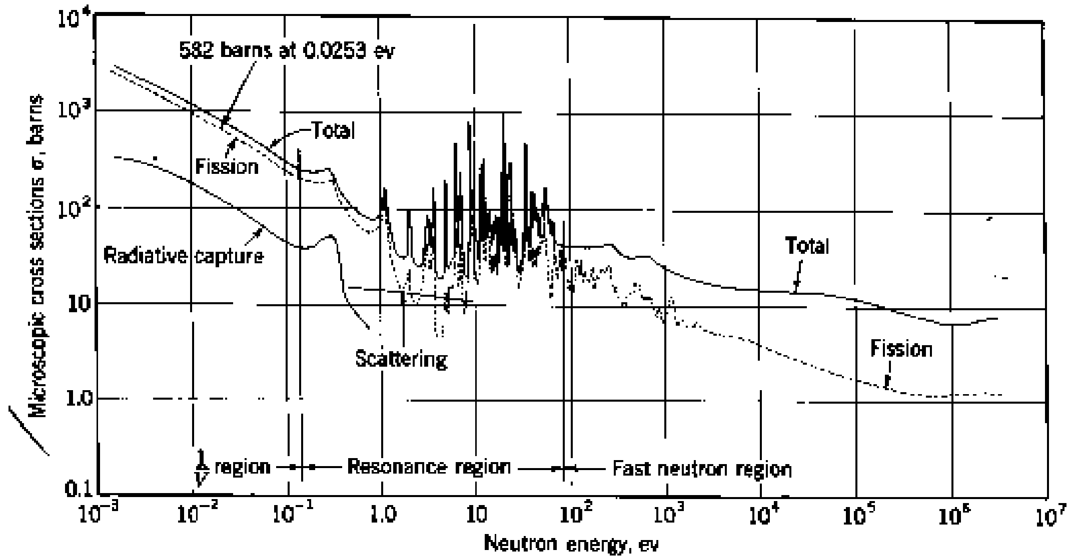


Fig 1.13. Neutron cross section for U ^{235}

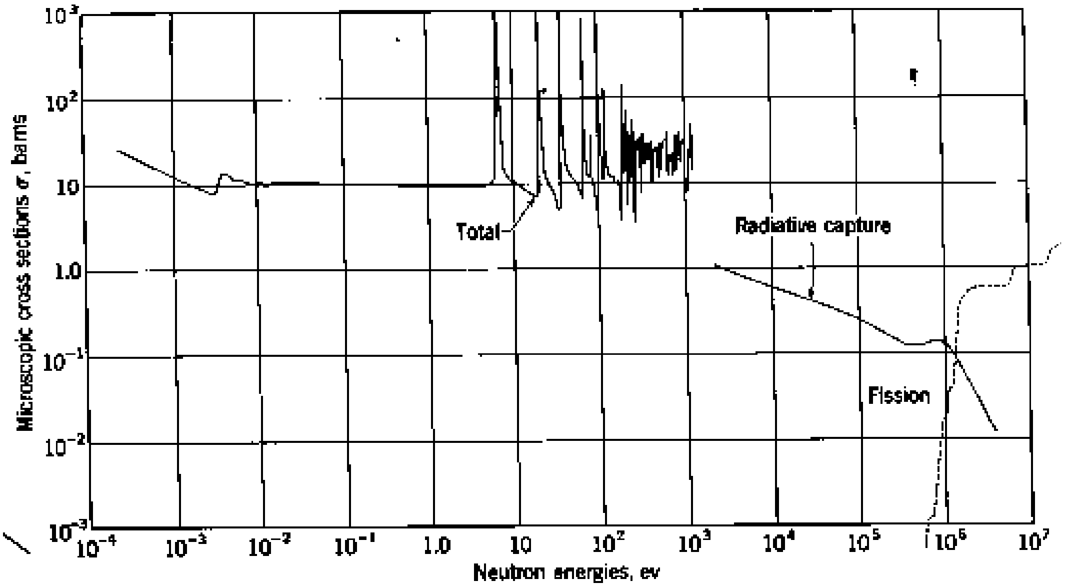


Fig 1.13. Neutron cross section for U ^{238}

THE VARIATION OF NEUTRON CROSS SECTIONS WITH NEUTRON ENERGY

σ - En plots are usually made on log-log coordinates. In many but not all cases, scattering cross sections are so small compared with absorption cross sections that the total cross sections shown are very nearly equal to the absorption cross sections. Also, for many nuclei, scattering cross sections vary little with neutron energy. Variations of absorption cross sections with neutron energy, such as those in Figs. 9-12 through 9-14 are represented by three regions which, beginning with low neutron energies, are (1) the *l/V region,* (2) the *resonance region,* and (3) the *fast-neutron region.*

I/V region In the low-energy range, the absorption cross sections for many, but not all, nuclei are inversely proportional to the square root of the neutron energy *E*

$$\sigma_a = C_1\left(\frac{1}{E_n}\right)^{0.5} \qquad \text{------------------ (1-42 a)}$$

$$\sigma_a = C_1\left[\frac{1}{m_n(V^2/2g_c)}\right]^{0.5} = C_2\frac{1}{V} \qquad \text{------------------- (1-42 b)}$$

where C_1 and C_2 are constants, m_n is the neutron mass, and *V* is the neutron velocity.

This relationship, known as the 1*IV law,* indicates that the neutron has a higher probability of absorption by a nucleus if it is moving at a lower velocity and is thus spending a longer time in the vicinity of that nucleus. The l/V law may also be written in the form

$$\frac{\sigma_{a,1}}{\sigma_{a,2}} = \frac{V_2}{V_1} = \left(\frac{E_{n,2}}{E_{n,1}}\right)^{0.5} \qquad \text{------------------- (1-42 c)}$$

where the subscripts 1 and 2 refer to two different neutron energies within the l/V range. Absorption cross sections for mono energetic neutrons, within the I/V region may thus be calculated at any energy from tabulated values at 2200 *mls.* On the log-log plots of Figs. 1-12 to 1-14, the l/V region is a straight line with a slope of - 0.5. The upper limit of the l/V region is different for different nuclei, being around 0.3 eV for indium, 0.05 eV for cadmium, 0.2 eV for U^{235}, 150 eV for boron, etc.

Resonance region Following the l/V region, most neutron absorbers exhibit one or more peaks occurring at definite neutron energies, called *resonance peaks.* They affect neutrons in the process of slowing down. Note that indium has but one peak, whereas U^{235} and U^{238} have many. Uranium-238 has very high resonance absorption cross sections, with the highest peak, about 4000 barns, occurring at about 7 eV. This fact affects the design of thermal reactors because U^{238} absorbs many of the neutrons passing through the region and affects the reactor neutron balance. Many elements, especially those of low mass numbers, do not exhibit resonance absorption and thus can be used as reactor construction materials, especially if their absorption cross sections are low.

Fast-neutron region Following the resonance region, cross sections usually undergo a gradual decrease as neutron energies increase. At very high energies, the sum of absorption and scattering cross sections approaches twice the actual cross-sectional area of the target nucleus, that is, $2\Pi\, r_c^{2.}$ Combining with Eq.(1-36)

$$\sigma_t = 2\pi r_0^2 A^{2/3}$$

Using 1.40×10^{-13} cm and 1 barn $= 10^{-24}$ cm^2

$$\sigma_t = 0.125 A^{2/3} \text{ barns} \quad \text{------------------ (1-43)}$$

In the very high neutron energy range, therefore, total cross sections are rather low, usually less than 5 barns each for the largest nuclei. Some nuclei, such as boron, carbon, and beryllium, exhibit some resonance in the high energy range (Fig. 1-12), but the peaks are rather low, and the phenomenon is of little importance.

Though, in a reactor core, neutrons are born at nil times and in all places containing fissionable material and diffuse in all directions, it will facilitate an understanding *of* the process if we examine the life cycle *of* a group *of* neutrons, assumed all horn at the Some time, which undergo scatter, leakage, absorption, and other reactions and finally cause fission and attain the same energy levels simultaneously. This group *of* neutrons is called a *generation;* The series of events *or* processes that such a group *of* neutrons undergoes *from* birth until a new generation is born by fission is called a *life cycle or* simply a cycle.

NEUTRON LIFE CYCLE

The life cycle *of* a generation of N fission neutrons will now he analyzed with the aid *of* Fig, 1-14 which shows the events that this generation undergoes in a thermal reactor. In the reactor the events do not necessarily take place in the exact order as represented in Fig. 1-14. Rather, this order is simply an effort *to* separate the events in order to be able to e\valuate them independently, The first event in which these neutrons engage is fission while still in the fast energy range. The cross sections for which a reaction with natural *or* slightly enriched uranium fuels is small. Thus only a small number *of neu*trons engage in fast fission with both U^{235} and U^{238} nuclei . Each of these produces about *2.47* new fast neutrons. The number of neu1rons in the generation thus momentarily increases by a factor called the *fast-fission factor and represented by* the symbol ε The number *of* fast neutrons in the generation will now be

$$N\,\varepsilon \quad \text{-------------------------------------- (1-44)}$$

In thermal homogeneous reactor little or no fast fission will takes place: that is, $\varepsilon = 1.0$. The reason is that the moderator is evenly dispersed through Out the Core causing the neutron rapidly slowdown to intermediates ranges with little *or* ' no fast fission taking place, ε is greatest for heterogeneous reactors having fuel elements of large cross sectional dimension since in such elements neutrons newly born inside n *fuel* element have more chances of fissioning *fuel* before coming in to contact with the moderator. An average value of t in natural uranium fueled reactors is 1.03.

The next event *to* be considered (Fig. 1-14) is leakage *of* some of the N ε *Fast* neutrons out *of* the boundaries *of* the reactor core. The fraction of the fast neutrons that remain within the core is called *the fast neutron non leakage probability* and is given the symbol .P_f. $1 - P_f$ is the fraction that leaks and may be called *the fast neutron leak probability.* Now the number of neutron that are slowed down to intermediate energies is

$N \varepsilon P_f$ -------------------------------------- (1-45)

Some of these undergo resonance absorption by U^{238}, some undergo non fission Absorption by U 235. The fraction of neutrons that escape these absorption and slow down to thermal energies is called the resonance escape probability and is given the symbol p. l-p is the *resonance absorption probability.* p, of course, is less than unity. being least for a homogeneous reactor with low enriched fuels where there is an abundance of U^{238}. It increases where the fuel to moderator decreases and approaches unity if highly enriched. fuels are used. Very low values of *p,* between 0.5 and 0.7, occur for homogeneous natural uranium fuels. This is the main reason why a sustained reaction with 'this type is impossible. The accurate determination of *p* for a particular core, design is extremely difficult, depending upon many complex factors that arc difficult to evaluate correctly. Semi empirical methods involving simplified theoretical relationships in conjunction with experimental data are usually used. The numbers of neutrons entering the thermal energy range are now

$N \varepsilon P_f\ p$--------------------------------------- (1-46)

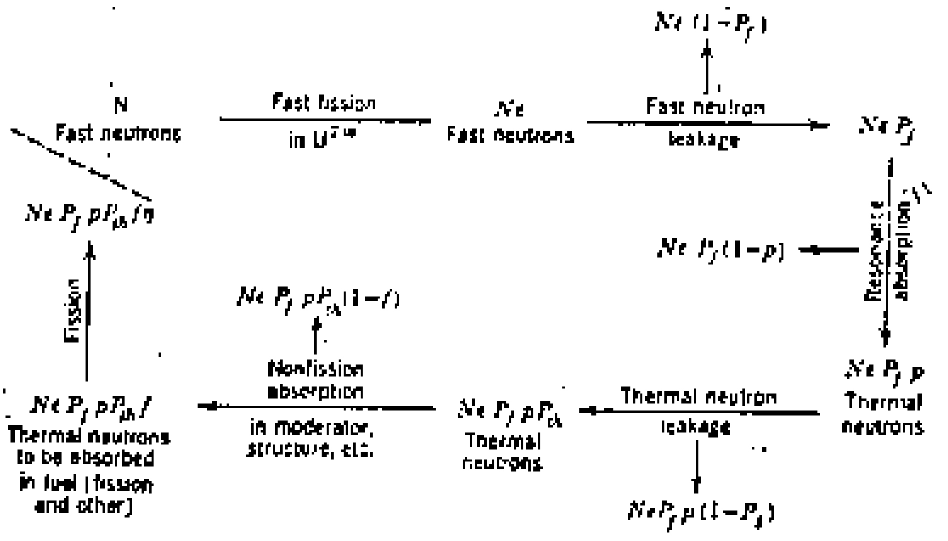


Fig 1.14. Neutron life cycle in a thermal reactor.

Again, some of these leak out of the core boundaries, leaving a fraction, P_{th} called the *thermal Neutron nonleakage probability.* 1 - P_{th} is called the *thermal neutron leakage probability.* The number of neutrons now available for absorption with in the core is given by

$$N \varepsilon P_f \; p \, P_{th} \text{ -------------------------------------- } (1\text{-}47)$$

The product of the two non leakage probabilities $P_f P_{th}$ is called simply simply the *non leakage probability a*nd is given the symbol P The above number may then written as N ε P p. Leakage is a function of the geometrical dimensions of the core as well as of the fast and thermal neutron diffusion lengths L_s and L. .A large core has a small surface to volume ratio and therefore a large *P A* large L_s. means that a fast neutron travels, on an average, a long distance between birth and thermalization and consequently has a greater probability of leakage. Like wise a large *L* means a greater probability of thermal neutron 1eakage. Except for H_2O, L_s *is* larger than *L.* This indicates that, for most moderators, there is more thermal than fast neutron leakage or that *P* is primarily influenced by P_{th}. For H_2O, the reverse is true..

Now the $N \varepsilon P_f \; p \, P_{th}$ neutrons entering the thermal-energy range a certain number will be absorbed by all but, the fuel, i.e., by moderator, cladding, structural materials etc., The fraction that remains is called the t*hermal utilization factor* and given the symbol *f.* It is the fraction that will be absorbed in the fuel. *F is* naturally a function of the volumetric composition of the different absorber in the reactor core as well as their absorption cross sections and the neutron flux in each, i.e., the absorption ratio in each. By definition, then, j in heterogeneous reactor region is given by

$$f = \frac{(N\sigma_a\phi)_U}{(N\sigma_a\phi)_U + (N\sigma_a\phi)_m + (N\sigma_a\phi)_{cl} + \cdots}$$

$$f = \frac{(\Sigma_a\phi)_U}{(\Sigma_a\phi)_U + (\Sigma_a\phi)_m + (\Sigma_a\phi)_{cl} + \cdots} \qquad \text{------------------ } (1\text{-}48)$$

where the subscripts U, m, cl indicates the fuel (U^{233}, U^{235}), moderator, Cladding, etc., constituents; $\sum_{a,}$ and σ_a are their microscopic and macroscopic absorption cross sections for thermal neutrons: and φ is the thermal neutron flux in each constituents in the core region in question. In case of homogenous reactors, the equation 1.48 reduces to

$$f = \frac{(\Sigma_a)_U}{(\Sigma_a)_U + (\Sigma_a)_m + \cdots}$$

The number of neutrons now ready for absorption is

$$N \varepsilon P_f \; p \, P_{th} f \text{ ----------------------------------- } (1\text{-}49)$$

Neutrons absorbed in fuel do not all cause fission, some engage in (n, γ) reactions with U238

In this case, '7 i8 defined as the number of fission neutrons per neutron absorbed in the fissionable fuel, and both Do not depend on the fuel enrichment. The number of fission neutrons born at the end of the life cycle, which will start a new generation and cyele, is now

$$N \varepsilon P_f \; p \, P_{th} f$$

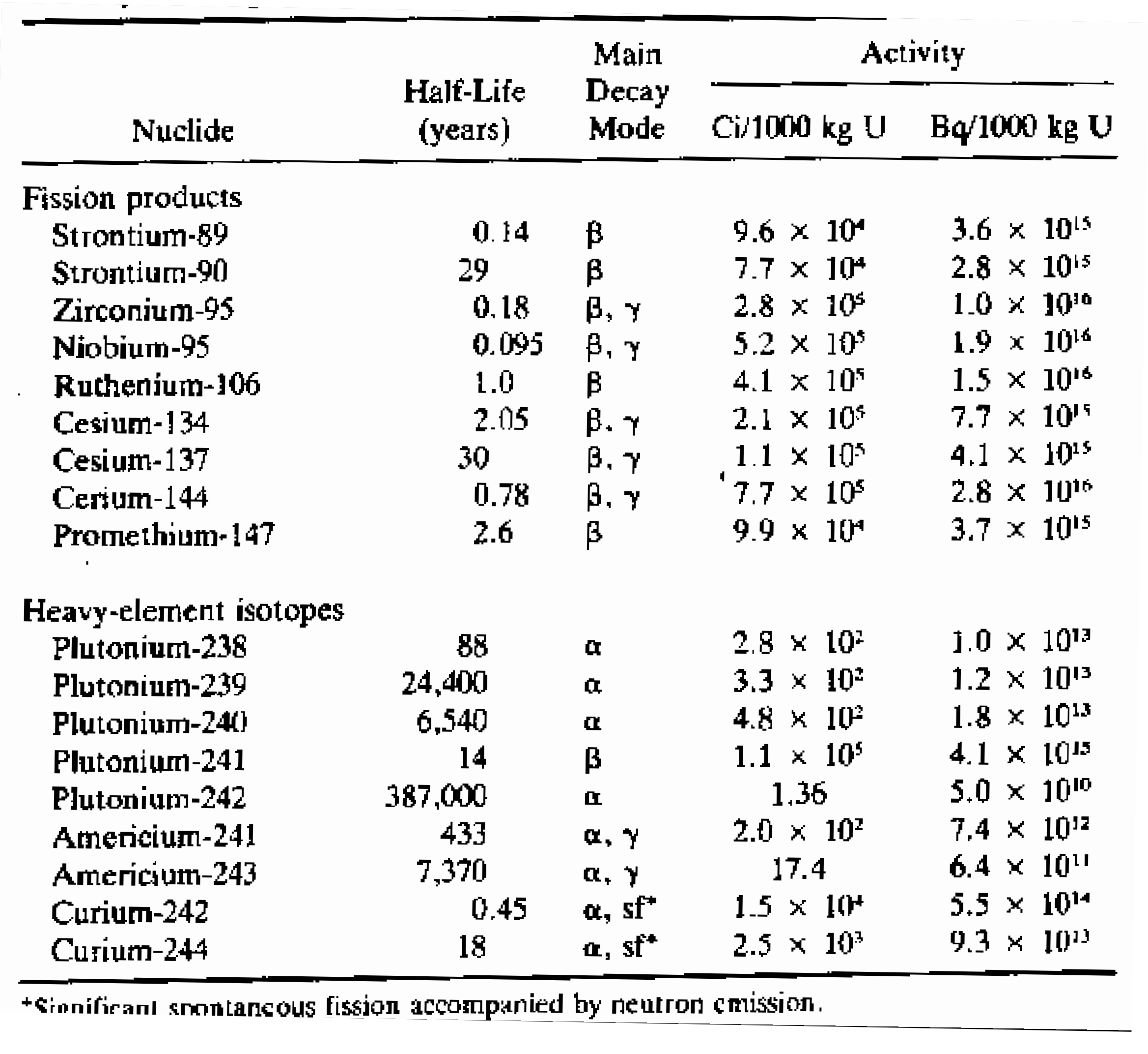
Note that both are constants for the particular fuel used and are independent of the internal arrangement or the dimensions of the core. Therefore, they are referred to as the *fuel constants*. On the other hand, f, *p,* and *f* are functions of the fuel and internal configuration of the core. In a heterogeneous core.

fuel elements are usually arranged in au order1y~eometrical fashion, and the ~*.t<.-4t.:;.:* _co~ is said to have a certain *lattice* structure.

After removal from the reactor and insertion in the storage pool, all of the radio nuclides contained in the fuel continue the decay process. Short and moderately short half-life fission products become insignificant after a few months. Therefore, a cooling period of *150* days is a useful point of reference, which just happens to be the intended cooling period when reprocessing was planned.

The major contributions to the radioactivity of spent reactor fuel after *150* days of cooling are given in Table 3.1 The values, in curies and becquerels per metric ton (1000 kg) of uranium (initially free from plutonium) charged to the reactor, *were* calculated *for* a hypothetical LWR having a thermal power of *3300* MW and a fuel burnup of 2.85 x 1012J (thermal)(2.85 TJ) per kilogram of uranium in the original fuel. Somewhat different activities would be applicable to other operating conditions, but those in the table are fairly typical. An indication of the respective activities at times after *150* days can be obtained from the half-lives It is seen that *for* a cooling period of *150* days *(or more)* a relatively few fission products, namely, strontium, zirconium, niobium, ruthenium, cesium, and some rare-earth elements, are responsible *for* nearly all of the radioactivity. These are the most important elements from which the uranium and plutonium would be separated in spent fuel reprocessing.

Table 3.1 Major Contributions to Radioactivity of Spent LWR Fuel After150 Days Cooling



Nuclide	Half-Life (years)	Main Decay Mode	Activity	
			Ci/1000 kg U	Bq/1000 kg U
Fission products				
Strontium-89	0.14	β	9.6×10^4	3.6×10^{15}
Strontium-90	29	β	7.7×10^4	2.8×10^{15}
Zirconium-95	0.18	β, γ	2.8×10^5	1.0×10^{16}
Niobium-95	0.095	β, γ	5.2×10^5	1.9×10^{16}
Ruthenium-106	1.0	β	4.1×10^5	1.5×10^{16}
Cesium-134	2.05	β, γ	2.1×10^5	7.7×10^{15}
Cesium-137	30	β, γ	1.1×10^5	4.1×10^{15}
Cerium-144	0.78	β, γ	7.7×10^5	2.8×10^{16}
Promethium-147	2.6	β	9.9×10^4	3.7×10^{15}
Heavy-element isotopes				
Plutonium-238	88	α	2.8×10^2	1.0×10^{13}
Plutonium-239	24,400	α	3.3×10^2	1.2×10^{13}
Plutonium-240	6,540	α	4.8×10^2	1.8×10^{13}
Plutonium-241	14	β	1.1×10^5	4.1×10^{15}
Plutonium-242	387,000	α	1.36	5.0×10^{10}
Americium-241	433	α, γ	2.0×10^2	7.4×10^{12}
Americium-243	7,370	α, γ	17.4	6.4×10^{11}
Curium-242	0.45	α, sf*	1.5×10^4	5.5×10^{14}
Curium-244	18	α, sf*	2.5×10^3	9.3×10^{13}

*Significant spontaneous fission accompanied by neutron emission.

The manner in which buildup of isotopes of the heavy elements occurs during reactor operation in a fuel consisting of uranium-235 anduranium-238 is illustrated in Fig. 11.2. Horizontal arrows pointing to the right represent *(n, γ)* reactions and those pointing to the *left* are *for (n,2n)* reactions with fast neutrons. Vertical arrows indicate beta decay. Where vertical arrows are absent, the nuclides are alpha-particle emitters. The decay product is then a nuclide with an atomic number two units less and a mass number of four units less than the parent. Alpha-particle decays are of minor importance in the cooling period, but they affect the build up of heavy isotopes at a later stage.

Of immediate interest is uranium-237 (half-life 6.75 days) which is formed by two *(n, γ)*stages starting with uranium-235 or by the *(n, 2n)* reaction with uranium-238. Any uranium-237 remaining in the spent fuel will be associated with the recovered uranium. Since uranium-237 is a gamma-ray emitter with a fairly short half-life, its presence makes the product difficult to handle. After a 150-day cooling period, however, the uranium-237 will have decayed to such an extent that the gamma activity is small enough to be tolerable. Moreover, the beta decay product, neptunium- 237, will be separated from the uranium during the reprocessing operation.

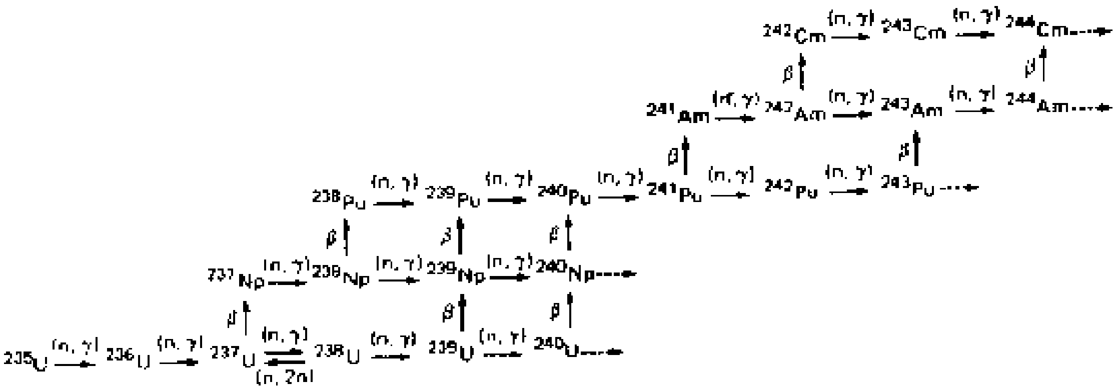


Fig 3.2 Heavy-isotope buildup in uranium.

A chart similar to Fig 3.2 showing the buildup of heavy isotopes when thorium-232 is included in the fuel is shown in Fig. 11.3. With thorium- 232 as fertile material, a long cooling time would be required to permit decay of the intermediate protactinium-233 (half life 27 days) to the fissile uranium-233. During this period, the activity of thorium-234 (half-life 24 days) would also decrease to a permissible level. However, at the same time, thorium-238 (half-life 1.9 years) would accumulate as a result of the alpha decay of uranium-232 (Fig. 3.3). Since thorium-228 has strong gamma ray emitters among its daughter products, remote handling of spent fuel would appear to be necessary regardless of the cooling time.

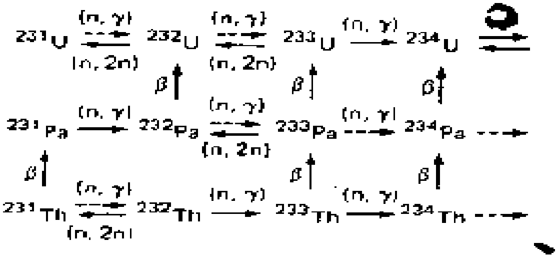


Fig 3.3 Heavy-isotope buildup in thorium

Head-End Treatment

After the cooling period, the spent-fuel assemblies would be shipped in strong metal casks to a reprocessing plant where the fission products are removed and the uranium and plutonium are recovered. The method commonly used for this purpose is based on extraction by an organic solvent, and this requires the fuel material to be dissolved in nitric acid to form a solution of nitrates. A simplified flow sheet of the reprocessing operations is given in Fig. 11.4. Each step is quite complex, but for the present purpose a brief overview will be adequate.

In the first (or "head-end") stage, the fuel rod assemblies, either with or without removal of hardware, are chopped into sections from which the spent material is leached with hot nitric acid. This process is commonly referred to as *chop-leach.* The hulls of zircaloy (or other) cladding and

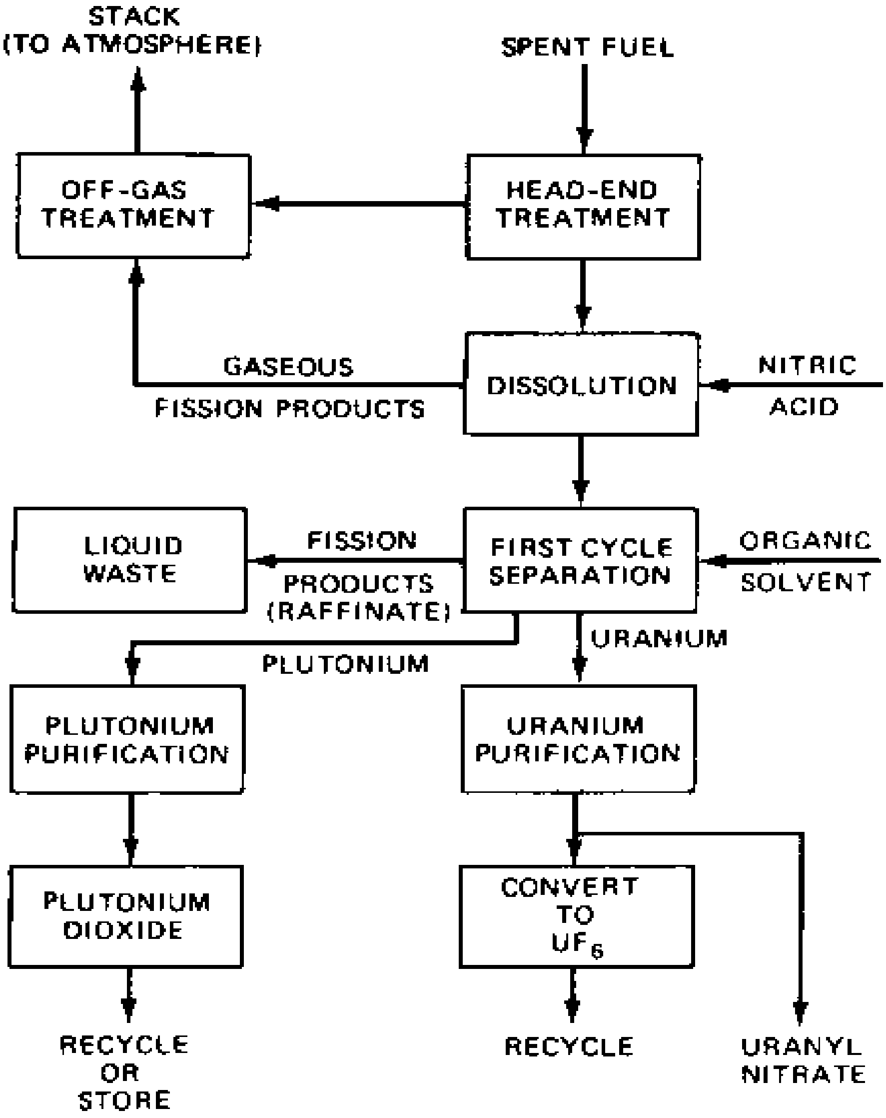


Fig 3.4 Spent-fuel reprocessing flow diagram

hardware that remain are subjected to a hot nitric acid soak to remove essentially all of the uranium and transuranic elements, i.e., elements of higher atomic number than uranium. The hardware and hulls form what are called TRU (for transuranium) wastes or alpha wastes, because they contain traces of alpha-emitting transuranium elements. These wastes have been buried temporarily in the past, but more permanent underground disposal is planned for the future.

Solvent-Extraction Separation Processes

The solvent-extraction method for separating the constituents of an aqueous solution can be used when one or more of these constituents are appreciably soluble, whereas the others are much less valuable, in an organic solvent which is essentially immiscible with water. When the organic liquid is brought into intimate contact with the aqueous solution, the substances present will distribute themselves between the organic and aqueous phases. The constituent (or constituents) with the greatest solubility in the organic medium will tend to pass into that phase whereas the others will tend to remain in the aqueous solution. Thus, a partial separation of the Constituents of the solution will have been achieved.

In choosing the organic liquid for a particular solvent-extraction procedure, an important property is its selectivity, that is its ability to extract a particular component (or components) of a solution in preference to all others that are present. The selectivity is expressed by the separation factor'; i.e., the ratio of the distribution coefficients of the wanted and unwanted species when equilibrium is attained between the two phases.
The distribution coefficient or distribution ratio D is defined as

$$D = \frac{\text{Conc. of component in organic phase}}{\text{Conc. of component in aqueous phase}}$$

the separation factor α is given by

$$\alpha = \frac{D(\text{product})}{D(\text{impurity})}.$$

A good solvent for extraction is one for which the distribution coefficient for one component is large and the separation factor is either large or small. In other words, it is desirable that D(product) shall be large, whereas D(impurity) should be 'small, or vice versa. 11.68. The extraction of an inorganic compound, such as a nitrate, from an aqueous solution by means of an organic solvent is influenced by a number of circumstances .Of particular importance are the presence of (1) salting agents, (2) complex-forming anions, and (3) oxidizing or reducing' agents.

A *salting agent* is either a salt or an acid, having the same anion as the inorganic compound to be extracted, the presence of which in the aqueous solution increases the distribution ratio. In the extraction of uranyl nitrate, for example, either nitric acid or one if its salts, such as sodium, potassium, calcium, or aluminum nitrate, can serve as a salting agent. These substances are soluble in the aqueous phase but not in the organic solvent.

The extraction of a specified element from aqueous solution by an organic medium is dependent upon the particular form in which the element is present in the solution. For example, uranyl nitrate hexahydrate can be extracted by certain organic solvents, but the corresponding sulfate is thus decrease the extractability of the uranium, since a proportion of the element will be in some form other than the nitrate. The complex –forming Anions decrease the distribution coefficient between the organic and aqueous phases, and so their effect is opposite to that of the common-io salting agents.

The solvent-extraction processes for separating uranium and plutonium from the fission products and then from each other depend on the somewhat unusual chemical behavior of the heavy elements. Starting with actinium (atomic number 89), there is a series of 15 elements, called the *actinide series,* in the sixth period of the periodic system which resemble the rare-earth (or *lanthanide)*elements in the fifth period. The lanthanide elements all have similar chemical properties, based on a positive valence of 3, resulting from the presence of three relatively loosely bound outer electrons in each atom. In the analogous actinide series, there are also marked resemblances among the elements, especially in the formation of a tripositive (III) valence state. However, because some of the actinide elements have inner electrons which are not very tightly bound, it is possible to realizetetrapositive(IV), pentapositive(V), and hexapositive(VI) states.

Although in a given valence state the various actinide elements have similar chemical properties, these properties often are very different in the different oxidation states. For example, the nitrates of the (IV) and (VI) states are appreciably soluble in certain organic liquids, but the nitrates of the (III) states are virtually insoluble in these liquids. The relative stability of the different oxidation states varies with the atomic number of the element. Hence, by the use of appropriate reagents, it is possible to shift the oxidation and reduction states so that two (or more) elements in a given solution will be in different states with differing solubility's in an organic liquid. Separation of the elements can then be accomplished by solvent extraction.

It is evident from the foregoing discussion that the nitrates of the (IV) and (VI) states of the actinide elements will have large distribution coefficients and hence will be extractable by an organic liquid, but the lower oxidation (III) state will have a smaller distribution coefficient and be less extractable. One consequence of this difference in the distribution coefficients is that after an element has been extracted into an organic medium it can be back-extracted into an aqueous solution if the (IV) or (VI) state is reduced to the (III) state. Suppose two actinide elements have been extracted into an organic solvent in the (IV) or (VI) state. If one of the elements is reduced to the (III) state, it can be separated from the other element by back-extraction into an aqueous solution. The uranium and plutonium in spent reactor fuel are separated from one another in this way.

The purex process

The "Purex" process, using n-tributyl phosphate (TBP) as the extractant, is typical of solvent-extraction procedures employed in the treatment of spent fuel. In the form of nitrates, uranium (VI) and plutonium (IV) can be readily extracted from aqueous solution by TBP, whereas the fission products are taken up to a much smaller extent. TBP is relatively stable in the presence of fairly high concentrations of nitric acid, hence, the latter is used as the salting agent.

In the first cycle of the Purex process, of which an outline flow sheet is shown in *Fig.* 11.5, the feed consists of an aqueous solution containing uranium (VI), plutonium (IV), and fission product (FP) nitrates plus an excess of nitric acid. Sodium nitrite is added to make sure that the plutonium is entirely in the (IV) state, since this form is best extracted by TBP. The feed solution enters at the middle of the first (extraction) column, while the less dense organic extractant (TBP in a kerosene-type solvent) entering from the bottom flows upward. The uranium .(VI) and plutonium (IV) nitrates are thereby extracted from the aqueous solution and pass into the organic medium. In the upper part of the column the organic phase is
scrubbed with concentrated nitric acid. Most of the fission products that may have entered the organic solvent are now back-extracted into the aqueous phase, but the nitric acid, which acts as a salting agent, largely prevents back-extraction of the uranium and plutonium. The aqueous effluent (raffinate) from the extraction column contains essentially all the fission products with little or no uranium or plutonium. The organic phase containing the uranium and plutonium next passes into the second (partitioning) column where it flows upward and meets the down flowing aqueous strip solution containing a reducing agent to reduce plutonium (IV) to plutonium (III). In a modified Purex process, the reduction is performed electrolytically. The plutonium (III) nitrate is not soluble in the organic medium and so it is back-extracted into the aqueous phase. As this flows downward it is scrubbed with fresh TBP moving upward from the bottom of the column. Any urapium (VI) that has passed into the aqueous solution is thereby returned to the organic phase. The aqueous medium, containing plutonium (III) nitrate, leaves at the bottom of the partition column.

The organic solution of uranium (VI) nitrate, from which the plutonium and fission products have been almost completely separated, is now transferred to the bottom of the third (stripping) column where it flows upward and is stripped by dilute -nitric acid flowing downward. In the absence of a salting agent, the uranium is back-extracted into the aqueous phase and then flows out of the bottom of the column. The spent solvent, leaving at the top, is sent to a recovery plant for purification and subsequent reuse in extraction.

For further purification, both the aqueous uranium (VI) and plutonium (III) nitrate solutions are submitted to a second and third cycle. The uranium purification in each cycle is essentially identical with the last two stages of the first cycle shown in Fig. 11.5. The aqueous uranium solution is first extracted into the TBP phase and scrubbed with reducil1g-;r solution; the organic phase is then stripped by dilute nitric acid in a second column. For each plutonium cycle, the plutonium (III) solution is converted into the (IV) state by means of sodium nitrite and nitric acid, extracted into the TBP medium, and scrubbed with nitric acid in the same column. The organic solution then passes to a stripping column where the plutonium is back-extracted into the aqueous phase by dilute nitric acid Ion exchange has been employed for the final purification of plutonium following the first cycle of recovery by solvent extraction. The procedure is particularly valuable for the separation of zirconium and ruthenium, and their daughter products niobium and rhodium, respectively. The common practice, however, is to use two additional stages of solvent extraction to remove residual fission products, as stated above.

Fuel Reprocessing Waste Management

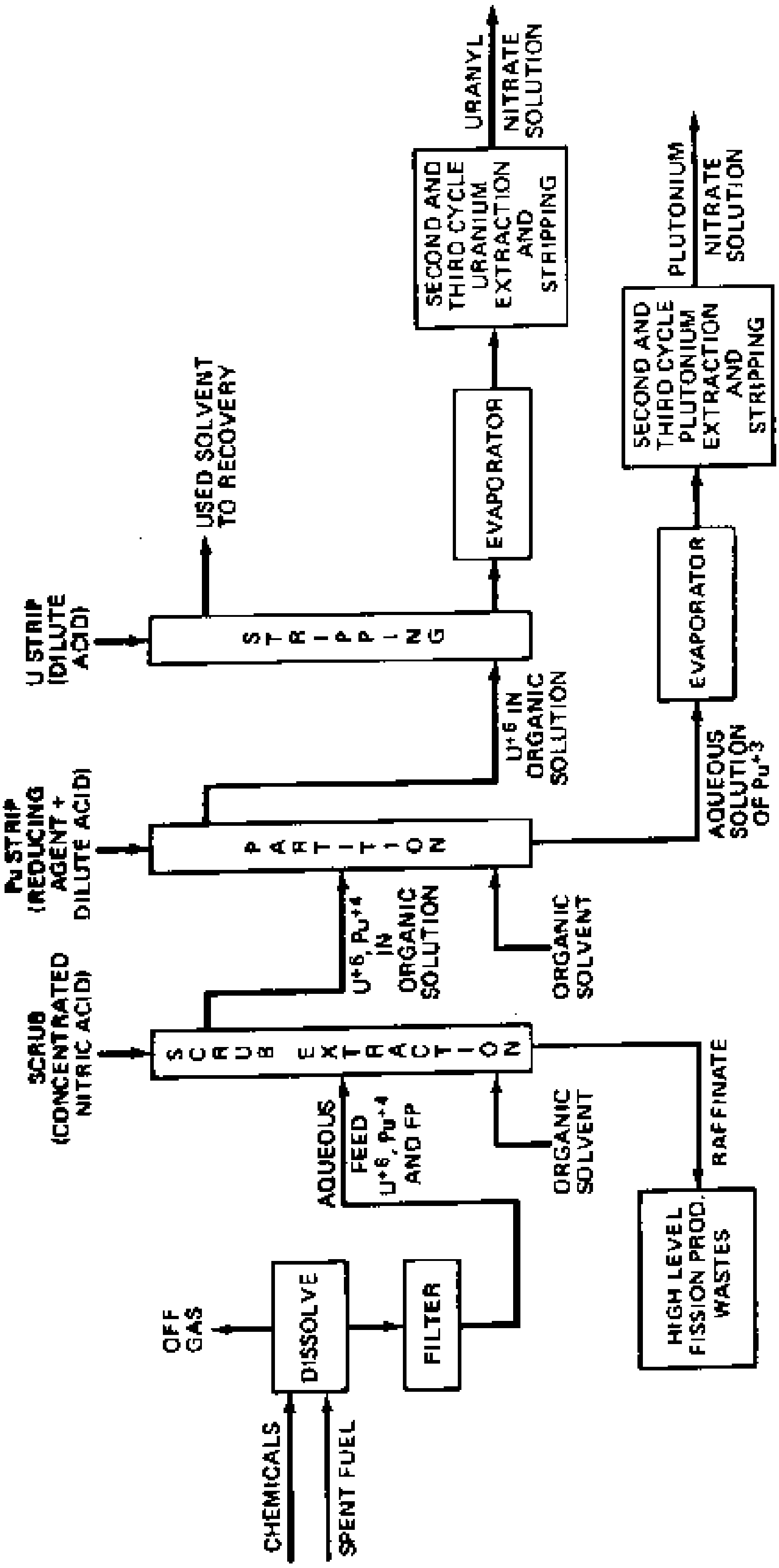
Various wastes are generated by a reprocessing plant using the Purex or a similar process. Low-radio activity wastes are handled by standard methods without problems. However, gaseous effluents that contain some radioactivity require special attention. The most highly radioactive liquid waste from the reprocessing of spent fuel by solvent e action and, in fact, from any stage of the nuclear fuel cycle-originates from the first-cycle raffinate (see Fig. 3.5). This *high-level waste* as it is called, contains about 99.9 percent of the nongaseous fission pr ducts originally present in the spent fuel; it also contains some uranium a d other actinide elements, including about 0.5 percent of the initial plutonium content. Because of its intense radioactivity, the high-level liquid waste presents a special problem.

The raffinate from the Purex process is evaporated to recover much of the nitric acid still remaining from the fuel dissolution and also to reduce the volume. The resulting solution, which contains nitrates of the metallic fission products and the actinides together with some free nitric acid, is stored, temporarily at least, steel tanks. in water-cooled, underground

Several processes have been investigated for converting high level liquid wastes into solid form. Two of the more successful are spray calcination and fluidized-bed calcination. In the former, the liquid is sprayed into the top of a cylindrical tower heater in a furnace; the solid *calcine,* consisting of a mixture of fission product and heavy element oxides, is collected at the bottom. In the other process, which has been used since 1963 for solidification of liquid wastes from the reprocessing of special highly-enriched fuels, the water is evaporated in a heated, fluidized bed of particles made from previously dried waste.

For long-term storage, the calcine (mixed oxides)may be heated with a glass-forming frit containing borax and silica. The product is a borosilicate glass which is less leachable by water and has a higher thermal conductivity than the calcine. These changes are desirable from the standpoint of the ultimate disposal of the solidified high-level wastes. However,

it appears that a sufficient increase in temperature, e.g., by radiation heating, may cause the glass to devitrify; the resulting microcrystalline material would then be more readily leachable than the original glass. This drawback could be overcome by reducing the fission-product content of the glass or by increasing the distance between the containers in storage, there by decreasing the heating rate. Nevertheless, efforts are being made to develop alternative solid forms for high-level wastes; these include crystalline products similar to stable minerals and cermets consisting of ceramic (calcined) waste particles in a metal matrix.



Outline of Purex process for spent-fuel reprocessing.

A l000-MW (electric) LWR plant will, on the average, discharge some 30 to 35 metric tons of spent fuel annually, assuming operation at full capacity. * This would result in the production of roughly 3.2 m^3 of high-level solid (borosilicate) waste. If this waste is packaged in steel cylinders 0.3 m in diameter and 3 m in length, about 15 such containers would be filled each year. Operation below the rated-capacity of the nuclear power plant, as will always be the case, will result in a smaller quantity of waste.

Characteristics of Solidified High-Level Waste

The radioactivity of the solid reprocessing wastes, as a function of time after removal from the reactor, can be inferred from Fig. 11.6, the data for which were calculated for LWR spent fuel, assuming 99.5 percent removal of plutonium (and uranium) [12J. Solid wastes would differ in the respect that the volatile (or gaseous) species, namely, iodine, krypton, tritium, and xenon, would not be present, since they are either removed or released at the reprocessing plant or during calcination of liquid wastes. The data in Fig. 11.6 refer specifically to 1000 MW(electric) of LWR operation with a burnup of 2.85 TJ/kg of fuel, initially uranium free from plutonium; 0.5 percent of the plutonium (and all the americium and curium) formed as well as 0.5 percent of the uranium are assumed to remain.

It is seen that at 1 year after discharge from the reactor, the total activity would be about 10^8 curies or more than 10^{18} Bq; it decreases by a factor of roughly ten after 10 years and another factor of about ten after 100 years. Subsequently, the activity decreases more rapidly until some 600 years have elapsed, when the decrease becomes very slow. The radioactivity then arises mainly from the long-lived, alpha-emitting transuranium nuclides, especially isotopes of plutonium and americium. There is also a small contribution from the beta-emitting fission product, technetium-99 (half-life 2.1 x 105years). Since iodine-129 is volatile, it is not present in the solid wastes.

Because the curves in Fig. 11.6 are for the total activity of all the isotopes of a given element, they do not show some interesting variations in heavy-element buildup. For example, the amount of plutonium- 238, produced in the wastes by the alpha decay of the short-lived curium- 242, increases to a maximum after some 10 years, and plutonium-240, from the decay of curium-244, attains a maximum after about 100 years. Plutonium-239, which is formed by alpha decay of americium-243 to neptunium- 239 followed by beta decay of the latter, reaches, a maximum activity after approximately 20,000 years. The sharp decrease in the activity of plutonium between about 10 and 100 years, as seen in Fig. 11.6, results from the decay of plutonium-238 (half-life 88 years) and plutonium-241 (half-life 14 years) which have made the major contribution stages. in the early 11.89. The rate of heat generation (or thermal power) in the wastes, due to radioactive decay, varies in the same general manner as the total activity. For the 1000 MW(el)~year operation of an LWR referred to, the thermal power of the solid high-level wastes would be about 350 kW after 1 year and roughly 35 kW after 10 years; at the latter time, almost 90 percent of the heat would arise from strontium-90 (and its short-lived decay product yttrium-90) and cesium-137. After 100 years, the thermal power is approximately 4 kW, and after 600 years (or more) it is less than 0.05 kW.

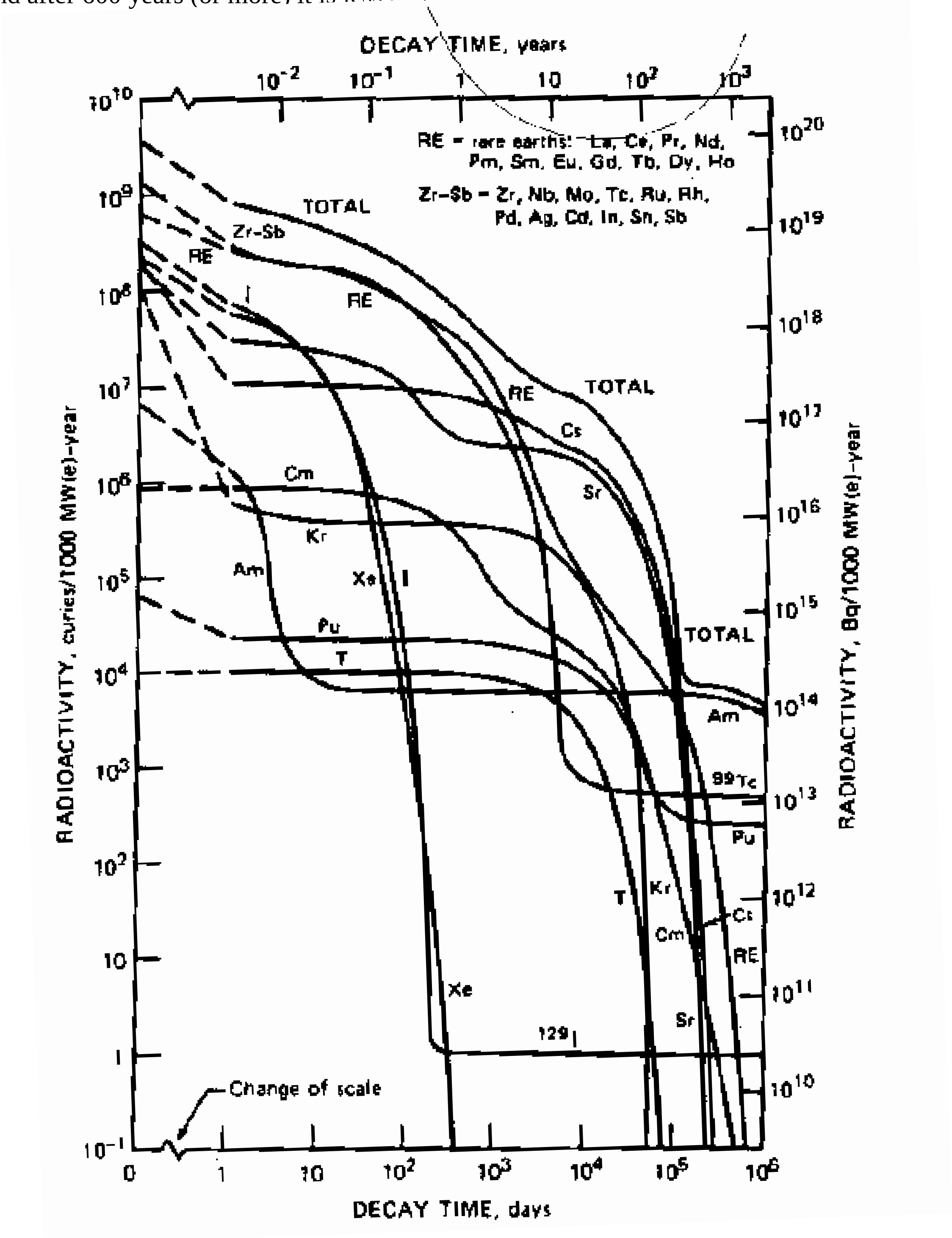


Fig 3.5 Radioactivity of waste from uranium fuel reprocessing, based on a burnup of 2.85 TJ/kg U (33,000 MW . d/1ooo kg) and 99.5 percent plutonium removal

A typical neutron-flux spectrum in a liquid-metal-cooled fast reactor compared with that in a water-cooled thermal reactor is shown in Fig. 11-1. The objective in the latter is to maintain a chain reaction with thermal neutrons having energies below I eV. In a fast-breeder reactor, the objective is to maintain a chain reaction with fast neutrons that have an average energy of about I MeV by fission in $U^{235}$and Pu^{239}.It also must provide additional fast neutrons sufficient to convert U^{238} to Pu^{239}

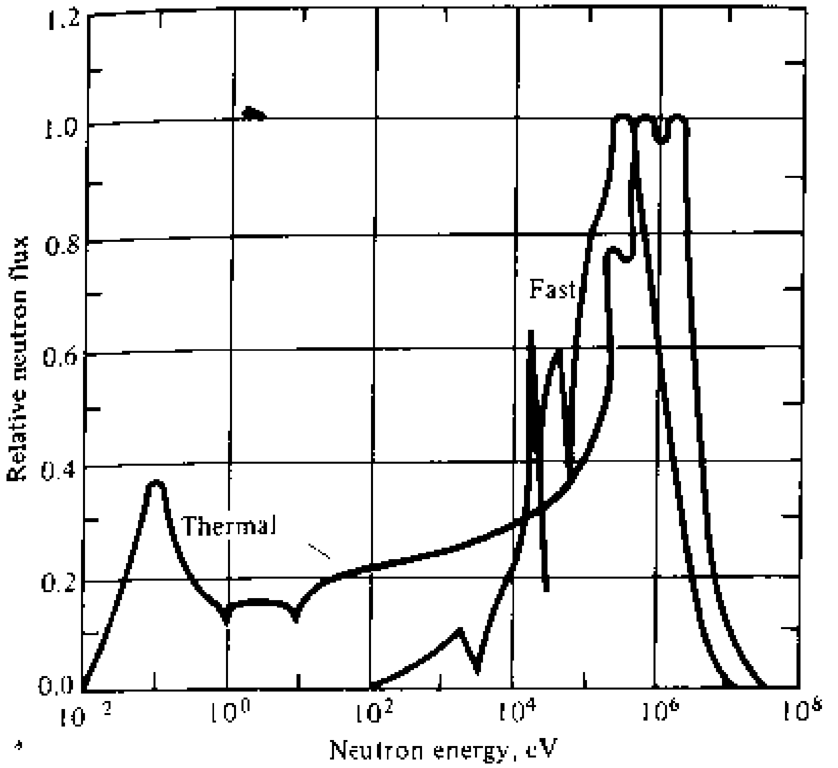


Fig 4.1 Typical neutron flux spectra for fast (LMFBR) and thermal (PWR) reactors.

Typical fission reactions in fast reactors are the same as those in thermal ones, as for example

$$_{92}U^{235} + {}_0n^1 \rightarrow {}_{56}Ba^{137} + {}_{36}Kr^{97} + 2\,{}_0n^1$$

$$_{92}U^{233} + {}_0n^1 \rightarrow {}_{56}Ba^{136} + {}_{36}Kr^{96} + 2\,{}_0n^1$$

$$_{94}Pu^{239} + {}_0n^1 \rightarrow {}_{56}Ba^{137} + {}_{38}Sr^{100} + 3\,{}_0n^1$$

---------------(4.1)

A typical non fission reaction is the one that occurs in the sodium coolant, which is composed of 100 percent Na^{23} where Na^{24} is a highly radioactive isotope that emits 2.76 MeV γ radiation and 1.3 MeV {3decays with a 14.8-h half-life to stable Mg^{24}

$$ {}_{11}Na^{23} + {}_{0}n^{1} \rightarrow {}_{11}Na^{24} $$

$$ {}_{11}Na^{24} \xrightarrow[\beta]{14.8\ h} {}_{12}Mg^{24} + {}_{-1}e^{0} $$

---------------------- (4.2)

A fast neutron reaction with U^{238} results in a series of reactions that culminate in Pu^{239},shown in Fig. 4-2 and given by

$$ {}_{92}U^{238} + {}_{0}n^{1} \rightarrow {}_{92}U^{239} + \gamma $$

$$ {}_{92}U^{239} \xrightarrow[\beta]{24\ min} {}_{93}Np^{239} + {}_{-1}e^{0} $$

$$ {}_{93}Np^{239} \xrightarrow[\beta]{2.3\ days} {}_{94}Pu^{239} + {}_{-1}e^{0} $$

------------------- (4.3)

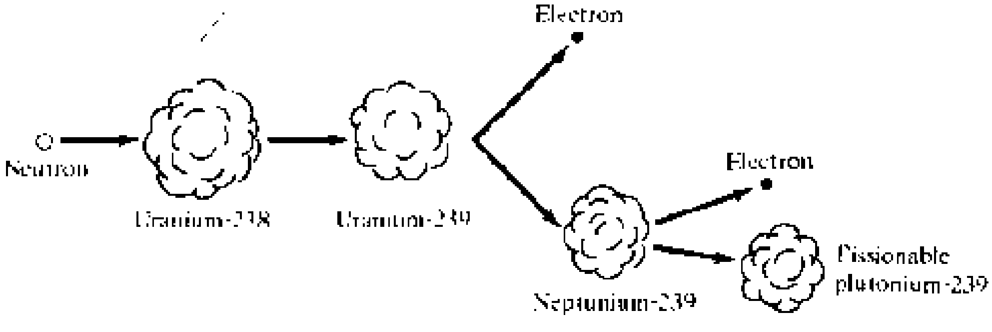


Fig 4.2 Schematic of the Pu239breeding chain.

This is a *breeding* reaction that converts *fertile* U238into *fissionable* Pu239.* We will now define two important fuel parameters. These are

υ = the average number of neutrons emitted per *fission*. *v* depends primarily on the fissile isotope, and to a lesser degree on neutron energy. It is highest for fast fission in plutonium, being about 3.0.

η = the *fission factor*. equal to the average number of neutrons emitted per neutron *absorbed* (and not necessarily causing fission).This a strong function of neutron energy (Fig. 4-3).

The Conversion and Breeding Ratios

When a neutron is *absorbed* in the fuel~ it produces 'T/neutrons. One of these must be reserved for further absorption to keep the reaction going. There will also be losses by parasitic capture in reactor coolant and materials of construction and by leakage. These losses we will designate *L,* neutrons lost per neutron absorbed.

$$C = \eta - 1 - L \qquad \text{-----------------------} \; \text{--- (4.4)}$$

The rest of the neutrons per neutron absorbed will be available for the breeding reaction and are called the *conversion* or *breeding* ratio C, or The maximum possible C, Cmax, is obtained if *L* were zero, or

$$C_{max} = \eta - 1$$

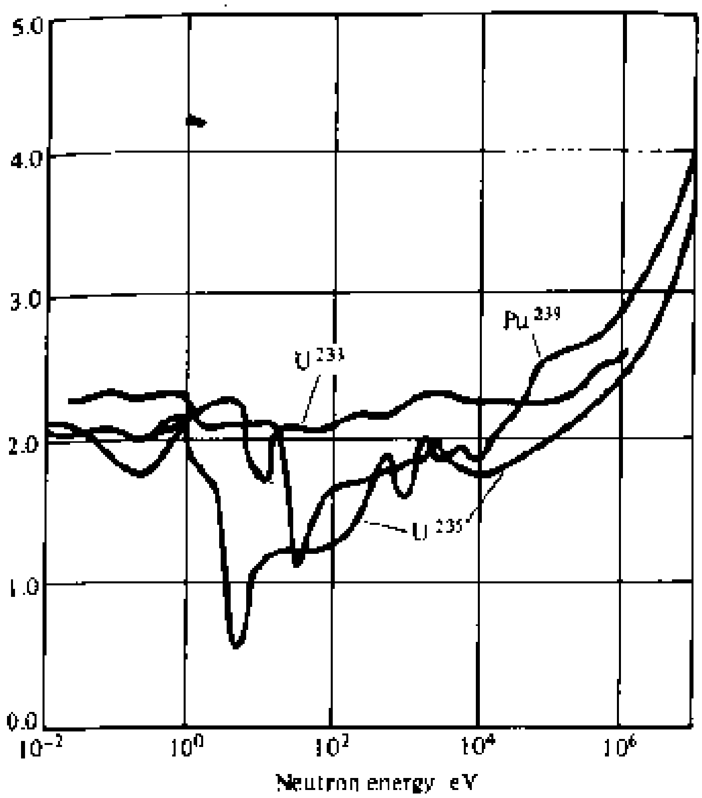


Fig 4.2 The variation of the fission factor η with neutron energy for three fissionable nuclei.

Depending upon 'T/and *L,* C can be much less than unity, and the reactor is called a *burner.* A reactor with a low C is generally called a *converter.* One with high C but less than 1.0 is called an *advanced converter.* For C less than 1.0, it can be shown that there is a maximum theoretical limit to the amount of fertile nuclei that ca be 'converted to fissionable nuclei. This maximum depends upon both C and the initial number of fissionable nuclei.

C = 1 means that the reactor is producing a number of fissionable nuclei equal to what it consumes; C > 1 means that there is no limit to conversion. In both these cases it is theoretically possible to consume all fissionable and fertile nuclei present. In practice, however, as with burner reactors, the fuel elements must be reprocessed and replaced periodically because fission products absorb neutrons and reduce reactivity or because of metallurgical considerations. A reactor with C >1 is a *breeder.*

The term *breeding ratio* has the same meaning as conversion ratio. *Breeding* (or *conversion) gain* G is the gain in fissionable nuclei per fissionable nucleus consumed. Thus

$$G = C - 1 = \eta - 2 - L \qquad \text{----------------- (4.5)}$$

and a maximum gain, Gmax, is

$$G_{\max} = C_{\max} - 1 = \eta - 2 \qquad \text{-------------------- (4.6)}$$

Average values of *υ,* η and Gmax are given in Table 4-1 for fissionable fuels for broad thermal and fast neutron energy ranges and for fertile materials for fast neutrons. Allowing for the fact that *L* is not zero, it can be seen that best breeding can occurring fast reactors fueled with Pu^{239}(and Pu^{241},which is not readily available), followed by U233and U^{235}.In thermal reactors, good conversion or breeding can be expected only from U^{233}.hence the use of Th^{232}as a fertile fuel in gas-cooled thermal reactors (Sec. 10-11). Th^{232} breeds U^{233}in reactions similar to those of U^{238} breeding Pu^{239}.

Table 4.1 Fuel production constants

	Fissionable fuels								Fertile Materials	
	Thermal				Fast				Fast	
Constants	U^{233}	U^{235}	Pu^{239}	Pu^{241}	U^{233}	U^{235}	Pu^{239}	Pu^{241}	Th^{232}	U^{238}
σ_f*	527	577	790	1000	2.2	1.4	1.78	2.54	0.025	0.112
σ_a*	580	675	1185	1400	2.35	1.61	2.05	2.83	—	0.273
ν	2.51	2.40	2.90	2.98	2.59	2.50	3.0	3.04	2.04	2.6
η	2.28	2.06	2.10	2.13	2.42	2.20	2.6	2.73	2.0	2.27
G_{max}	0.28	0.06	0.10	0.13	0.42	0.20	0.6	0.73	0	0.17

* In barn.

The Doubling Time

Of economic importance to a breeder reactor is its *doubling time.* This is the time required to produce as many new fissionable nuclei as the total number of fissionable nuclei that are both normally contained in the core and tied up in the reactor fuel cycle (i.e., in fabrication, reprocessing, etc.). In general, doubling times range between 10 and 20 years, the shorter the better.

The number density of new fissionable nuclei produced in a breeding reactor during time 8, *6Nb* nuclei/cm3, may be given by equation 4.7

Note that φ_d is shortened by operating at high power levels (high G and that it is also inversely proportional to the breeding gain $(\eta - 2 - L)$. In practice φ_d slightly increases w4h reactor life because *L* increases with fission product buildup, as a result of the finite time required after start-up to build up the fissile inventory in a breeder reactor blanket (a region surrounding the core that contains fertile material), the time taken for fuel and blanket element reprocessing, the economics of reprocessing, etc. It is economically desirable to have doubling times short enough that new Pu239 inventories are continually provided for new breeders. In other words, the plutonium is thought to be invested and the dividends are compounded. This gives rise to a *compound doubling time*, *8de*>related to θd by equation 4.8

$$\Delta N_b = \Delta N_{ff} G = \Delta N_{ff}(\eta - 2 - L)$$

$$\Delta N_{ff} = F_c (N_0)_{ff} (\sigma_a)_{ff} \bar{\phi}\, \theta \qquad \text{------------------- (4.7)}$$

where ΔN_{ff} = number of original fissionable fuel nuclei consumed (by neutron absorption) during time θ per cm^3

$(N_0)_{ff}$ = number of fissionable fuel nuclei present in the core and tied up in the fuel cycle at arbitrary time 0, nuclei/cm^3

F_c = fraction of $(N_0)_{ff}$ that is in the core, dimensionless

$(\sigma_a)_{ff}$ = microscopic absorption cross section of fissionable fuel, cm^2

$\bar{\phi}$ = average reactor neutron flux, $\text{neutrons/s} \cdot \text{cm}^2$

$$\theta_{dc} = \frac{0.6931}{\ln\,[1 + (1/\theta_d)]} \qquad \text{------------------- (4.8)}$$

θ_d and θ_{dc} may be based on core inventory only_(excluding fuel cycle inventory). Thus *Fe* = 1.0 and their values would be shortened.

LMFBR Plant Arrangement

Because sodium and other liquid metals suffer from high induced radio activities, Eq.(4-2), and are generally chemically active, intermediate coolant loops are used between the primary radioactive coolant and the steam cycle (Fig.4-2). The intermediate coolant is usually also a liquid metal, often Na or NaK. The intermediate loop guards against reactions between the radioactive primary coolant and water. Such reactions result. among other things, in the radiolytic decomposition of steam-generator water by the strong -y radiations emitted by Na^{24}. The intermediate loop also ensures against the high pressure water or hydrogen entering the reactor.

There are two primary-loop designs that are being considered. These are (1) the *loop,* or *pipe,* type and (2) the *pool, tank,* or *pot* type.,

The *loop* type, represented schematically in Fig. 11-5, is the more conventional of the two, being the design used in all U.S. operating sodium-cooled plants to date, with the exception of EBR-II.

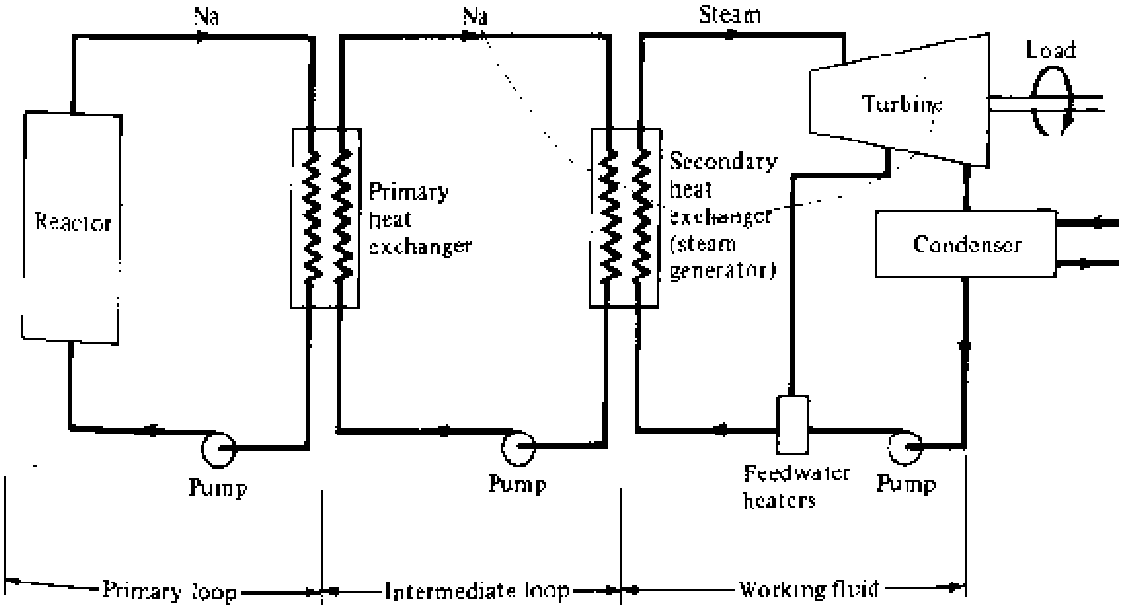


Fig 4.2 Schematic diagram of liquid metal cooled fast breeder reactor LMFBR Plant

In it, the reactor vessel, heat exchangers, liquid metal pumps, and other components of the primary system and their interconnecting piping are separated within a large building containing an inert atmosphere to preclude sodium tires in case of a sodium leak. The major advantages of this design are the accumulated experience with it and the separation or decoupling of the components of the primary system. It has the disadvantage of large and multiple shielding of pipe ways, equipment cells, and of the large and complex structure resulting from the spread of the components.

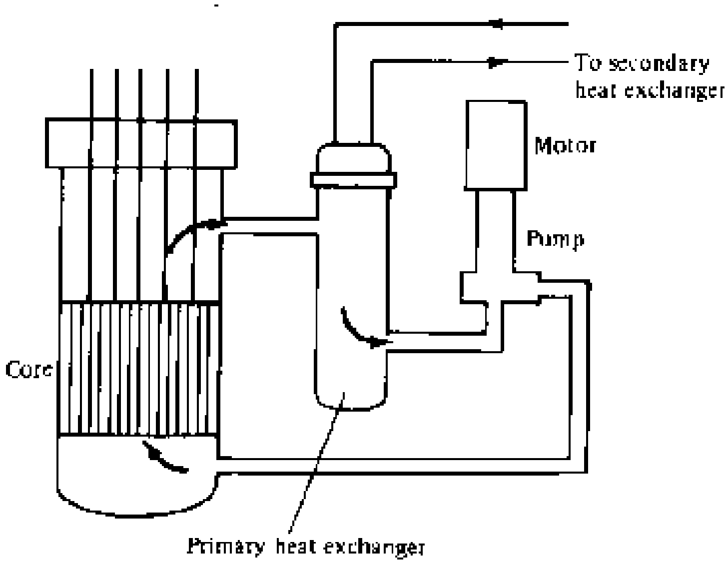


Fig 4.3 Schematic diagram *Loop* system

The design of the interconnecting piping is complex and requires expansion loops to accommodate thermal expansion. Stress concentrations at the pipe-reactor vessel joints pose critical problems. Breaks or leaks in the piping system may seriously affect reactor-core cooling. Leaks also would necessitate extended shutdowns for repairs.

In the *pool* system, represented schematically in Fig. 11-6, the entire primary system, including reactor, primary heat exchangers, and pumps, is submerged in a large tank tilled with molten sodium. That tank is part of the primary coolant loops. The heat exchangers discharge coolant directly into the tank, and the pumps receive coolant directly from it. The main advantages of the pool design are the relative insensitivity to sodium leaks in the primary system and a more compact primary-system arrangement. The disadvantages result from the close coupling of the various components, which leads to accentuated mechanical and thermal interactions, and the
rather complex structure of the pool closure that must serve the multiple functions of shield, inert gas closure, and support of equipment above and must contain all the necessary penetrations to the components.

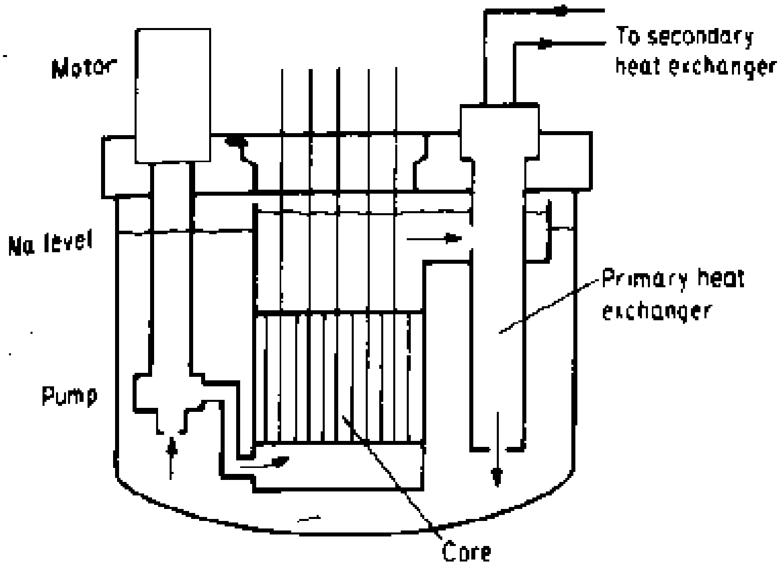


Fig 4.4 Schematic diagram *Pool* system

In general, it is now believed that the pool system has the edge in safety and economy while the loop system has the edge in that it is a straightforward mechanical design. Both types are being used in current designs. The pool is used in French, British, and recent USSR designs. The loop is used in U.S., early USSR, German, and Japanese designs (Table 11-2). The next two sections cover the Clinch River Breeder Reactor Project, an American demonstration plant of the loop design, and Super Phenix, a French commercial-size plant of the pool design.

HEAT FLOW IN A SOLID FUEL ELEMENT

Figure 8-5a shows a thin, bare (unclad) plate-type fuel element of constant cross-sectional area. The dimensions of the element are large in the y and z directions compared with that in the x direction, so that heat flow can be considered one-dimensional, i.e., in the x direction only.

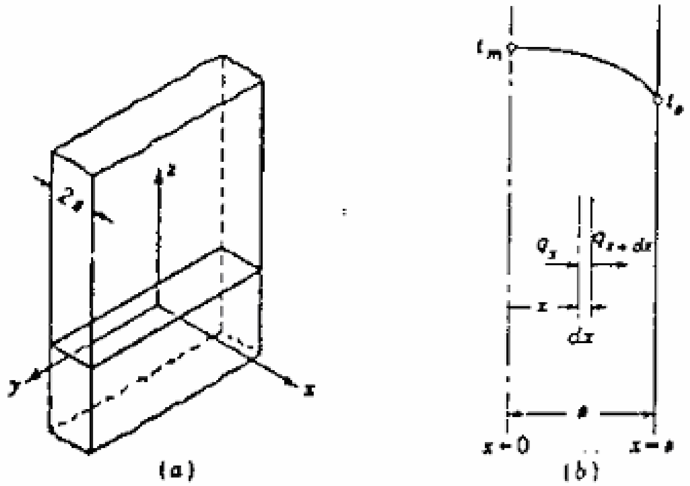


Fig. 8-5. Unclad thin plate-type fuel element.

Because ϕ and q''' are constant over the element cross section, heat is conducted equally in the $+x$ and $-x$ directions, and the mid-plane $x = 0$ (Fig. 8-5b) is the plane of highest temperatures. We shall then simply treat heat flow only in one half of the element, the $+x$ direction.

Considering a thin layer of thickness dx at distance x from the mid-plane, a heat balance may be written for the case of steady-state heat transfer as follows:

Heat generated in layer = [heat crossing plane $(x + dx)$] − [heat crossing plane (x)]

Thus

$$q''' A\, dx = q_{x+dx} - q_x \tag{8-8}$$

$$q_x = -k_f A \frac{dt}{dx}$$

and

$$q_{x+dx} = q_x + \frac{dq_x}{dx} dx$$

$$= \left(-k_f A \frac{dt}{dx}\right) + \left(-k_f A \frac{d^2t}{dx^2} dx\right)$$

where k_f is the thermal conductivity of the fuel and A is the area of the layer in the yz plane, perpendicular to the direction of heat flow. Equation (8-8) can now

$4K_f$

C_1 and C_2, the constants of the double integration, are evaluated from the boundary conditions of the system, which are

$$\frac{dt}{dr} = 0 \qquad \text{at } r = 0$$

$$t = t_m \qquad \text{at } r = 0$$

be written as

$$q'''A\,dx = -k_f A \frac{d^2t}{dx^2}\,dx$$

which reduces to

$$\frac{d^2t}{dx^2} = -\frac{q'''}{k_f} \tag{8-9}$$

Since q''' and k_f are constant and positive, Eq. (8-9) indicates that the temperature profile in the x direction is convex and has constant curvature; i.e., it is parabolic (Fig. 8-5b) except near the ends in the y direction. Equation (8-9) now integrates twice to give

$$\frac{dt}{dx} = -\frac{q'''}{k_f}x + C_1 \tag{a}$$

and

$$t = -\frac{q'''}{2k_f}x^2 + C_1x + C_2 \tag{b}$$

where C_1 and C_2 are the constants of integration. Because of symmetry around the mid-plane and because there is equal and opposite and consequently no net heat flow at that mid-plane, t_m, the temperature at the mid-point, is the maximum temperature in the section. The boundary conditions are

$$\frac{dt}{dx} = 0 \qquad \text{at } x = 0$$

and

$$t = t_m \qquad \text{at } x = 0$$

Thus $C_1 = 0$ and $C_2 = t_m$. Substituting the values for C_1 and C_2 in Eq. (b) gives an expression for the temperature t at any plane x:

$$t = t_m - \frac{q'''}{2k_f}x^2 \tag{8-10}$$

The temperature at the surface, t_s, can be obtained by putting $x = s$, where s is equal to half the element thickness in the x direction. Thus

$$t_s = t_m - \frac{q'''}{2k_f}s^2 \tag{8-11}$$

q_x, the heat conducted past any plane x, is equal to the total heat generated between $x = 0$ and x. Thus

$$q_x = q'''Ax \tag{c}$$

The same expression can be derived from $-k_fA\,dt/dx|_x$, where dt/dx is given by Eq. (a).

q_s, the heat conducted out of one surface ($x = s$), equal to the heat generated from one-half of the element, is given by

$$q_s = q'''As \tag{8-12}$$

Another expression of q_s in terms of the boundary temperatures can be obtained by combining Eqs. (8-11) and (8-12) and rearranging to give

$$q_s = 2k_fA\frac{t_m - t_s}{s} \tag{8-13}$$

showing that the heat transfer by conduction in a system with uniform internal heat generation is twice that without heat generation for the same path length s and overall temperature drop. This is because heat, produced uniformly throughout the element, has, on an average, to travel only one-half the distance between $x = 0$ and $x = s$.

The total heat generated in the two halves of the element q_{2s} (Btu/hr) is twice that given by Eq. (8-13). If A is replaced by the total surface area of the fuel element A_s (neglecting the edges), where $A_s = 2A$, it is given by

$$q_{2s} = 2k_f A_s \frac{t_m - t_s}{s} \qquad (8\text{-}14)$$

Effect of Cladding. Figure 8-6 shows a system containing a plate-type fuel element of half thickness s with cladding of thickness c and thermal conductivity k_c. The equations for the temperature variation and heat conduction within the fuel element, i.e., where $x \leqslant s$, are the same as those given above. Because of the presence of the cladding, an extra, though usually small, resistance to heat conduction results in the decrease of q_s for the same overall temperature drop (i.e., if $t_m - t_c$ were the same as $t_m - t_s$ in the previous case).

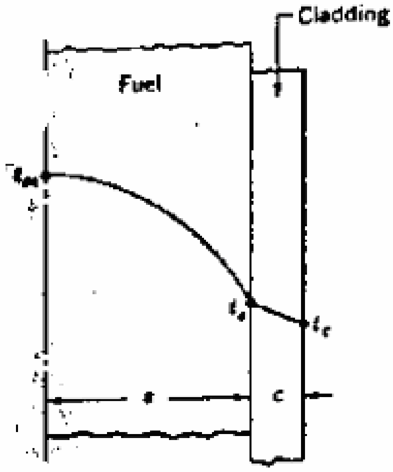


Fig. 8-6 Temperature profile in clad fuel plate ($k_f < k_c$).

In the steady state and with the assumption of no heat generated in the cladding material, the amount of heat leaving surface s must be the same as that leaving surface c. For a constant k_c, dt/dx through the cladding is constant. (The ratio of dt/dx in the fuel at $x = s$ to dt/dx in the cladding is inversely proportional to the ratio of their thermal conductivities.) Also neglecting resistance to heat flow at the fuel-cladding interface, we can write

$$q_s = q''' As = 2k_f A \frac{t_m - t_s}{s} = k_c A \left.\frac{dt}{dx}\right|_{\text{clad.}} = k_c A \frac{t_s - t_c}{c} \qquad (8\text{-}15)$$

Solving for the temperature differences,

$$t_m - t_s = \frac{q_s s}{2k_f A} = \frac{q''' s^2}{2k_f}$$

$$t_s - t_c = \frac{q_s c}{k_c A} = \frac{q''' sc}{k_c}$$

Adding,

$$t_m - t_c = \frac{q_s}{A}\left(\frac{s}{2k_f} + \frac{c}{k_c}\right) = \frac{q''' s^2}{2k_f} + \frac{q''' sc}{k_c} \qquad (8\text{-}16)$$

Rearranging gives q_s in terms of the temperatures at the boundaries of the system, that is, t_m and t_c. Thus

$$q_s = \frac{A(t_m - t_c)}{s/2k_f + c/k_c} \qquad (8\text{-}17)$$

Compare these equations with Eqs. (8-13) and (8-14), where q_s is given in terms of the temperatures at the mid-plane and surface of the fuel element.

Heat Transfer from Fuel Element to Coolant. In order to use Eqs. (8-17) and (8-18), the temperature at the outer surface of the cladding, t_c, must be known. Often the bulk temperature of the coolant fluid, t_f, is known with more certainty. Figure 8-7 shows a clad fuel plate with coolant fluid passing parallel to it. Again, in the steady state, with no heat produced in cladding or coolant, the heat passing out of the fuel-plate surface is the same as that passing out of the cladding's outer surface and into the coolant.

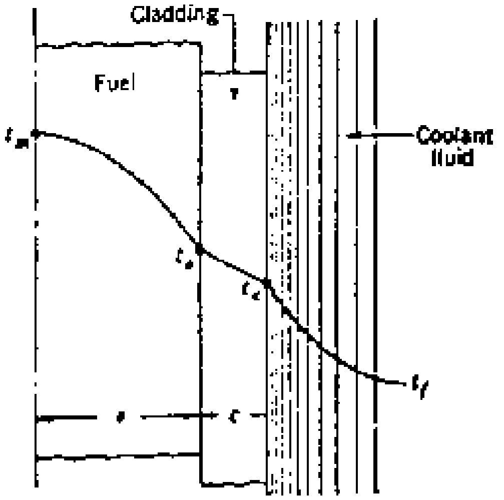


Fig. 8-7. Temperature profile in clad plate-type fuel element with coolant.

The heat passing from the cladding to the coolant is by convection. This may be natural or forced convection. In either case, the coefficient of heat transfer is given the symbol h and commonly has the units of Btu/hr ft^2 °F. Recommended methods for the evaluation of h in different systems will be presented in Chap. 10.

If q_s is again the heat leaving the fuel element (see Fig. 8-7), the following relationships hold:

$$q_s = q'''As = 2k_f A \frac{t_m - t_s}{s} = k_c A \frac{t_s - t_c}{c} = hA(t_c - t_f)$$

Solving for temperature differences,

$$t_m - t_s = \frac{q_s s}{2k_f A} = \frac{q''' s^2}{2k_f} \tag{d}$$

$$t_s - t_c = \frac{q_s c}{k_c A} = \frac{q''' sc}{k_c} \tag{e}$$

and

$$t_c - t_f = \frac{q_s}{hA} = \frac{q''' s}{h} \tag{f}$$

Adding

$$t_m - t_f = \frac{q_s}{A}\left(\frac{s}{2k_f} + \frac{c}{k_c} + \frac{1}{h}\right) = \frac{q''' s^2}{2k_f} + q''' s\left(\frac{c}{k_c} + \frac{1}{h}\right) \tag{8-19}$$

If only t_m and t_f are known or assumed, the intermediate temperatures t_s and t_c can be obtained by calculating q_s from Eq. (8-19) and then substituting into Eqs. (*d*) and (*e*) or (*f*).

Equation (8-19) may be used to explain some of the limitations on heat generation in nuclear reactors. For any constant value of q''', if t_f is to be kept as high as possible for good plant thermal efficiency, h has to be increased materially, since it affects only part of the equation, to keep the maximum fuel temperature

t_m or the maximum cladding temperature t_c from becoming excessively high. On the other hand, if the temperatures are fixed, h again has to be increased materially to allow a high value of q'''. q''' may be regulated by changing the neutron flux with the help of the control rods. In essence, there is no limit to the quantity of heat that a nuclear reactor is capable of generating, so long as adequate cooling is provided to keep the temperatures in the system from exceeding their safe limits.

It can be seen that these values of h, the minimum necessary to keep the maximum fuel temperature within 700°F, increase rapidly with neutron flux ϕ. If ϕ is increased beyond a certain point, however, it is impossible to maintain the specified maximum fuel temperature (unless the coolant temperature is lowered), no matter what value of heat-transfer coefficient is designed into the system. This theoretical maximum value of flux, ϕ_{max}, is obtained by putting $h = \infty$ in Eq. (8-19) and evaluating a corresponding value of volumetric thermal source strength, q'''_{max}, to give

$$q'''_{max} = \frac{t_m - t_f}{s^2/2k_f + sc/k_c} \qquad (8\text{-}20)$$

(In effect, this makes the temperature of the coolant equal to the temperature of the cladding's outer surface.)

ϕ_{max} can then be evaluated from q'''_{max} by the use of Eq. (8-2). Equation (8-20) shows that, to increase ϕ beyond ϕ_{max} for a given fuel and cladding configuration, it is necessary to lower t_f (with its undesirable effect on plant thermodynamic efficiency) or to increase t_m (at the expense of reduced fuel burnup or otherwise by fuel alloying or using ceramic fuels), or both.

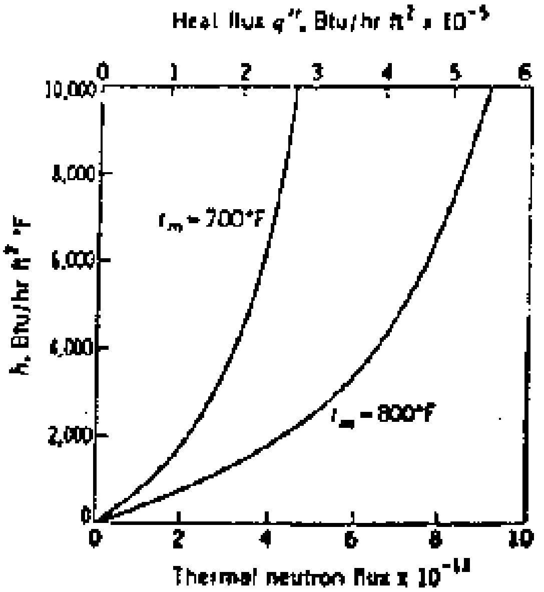


Fig. 8-8. Variation of h with ϕ and q'' for fuel element and coolant temperatures of Examples 8-3 and 8-4.

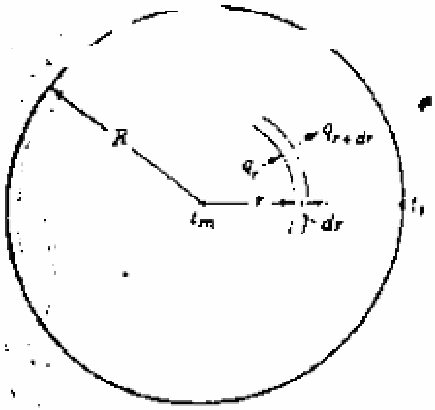


Fig. 8-9. Cross section of solid cylindrical fuel element.

8-7. Heat Flow out of Solid Cylindrical Fuel Elements

Here again a short section of a cylindrical or pin-type fuel element of a diameter small compared with that of the reactor core will be treated. The neutron flux will be assumed not to change appreciably in either the axial or radial direction. The heat flow out of the fuel element will substantially be radial and will be equal in all directions.

Figure 8-9 shows the cross section of an unclad element having a radius R and length (in the axial direction, not shown) L. Consider a thin cylindrical layer at r of thickness dr. A heat balance may be written for the case of steady-state heat transfer as follows:

$$q'''2\pi r\,dr\,L = q_{r+dr} - q_r \tag{8-21}$$

where

$$q_r = -k_f A\frac{dt}{dr} = -2\pi L k_f r\frac{dt}{dr}$$

and

$$q_{r+dr} = q_r + dq_r = q_r + \frac{dq_r}{dr}dr$$

$$= -2\pi L k_f r\frac{dt}{dr} - 2\pi L k_f\left(r\frac{d^2t}{dr^2} + \frac{dt}{dr}\right)dr$$

Thus

$$q'''2\pi r\,dr\,L = -2\pi L k_f\left(r\frac{d^2t}{dr^2} + \frac{dt}{dr}\right)dr$$

This reduces to

$$k_f\frac{d^2t}{dr^2} + \frac{k_f}{r}\frac{dt}{dr} + q''' = 0 \tag{8-22}$$

The solution of this differential equation is

$$t = -q'''\frac{r^2}{4k_f} + C_1\ln r + C_2 \tag{8-23}$$

C_1 and C_2, the constants of the double integration, are evaluated from the boundary conditions of the system, which are

$$\frac{dt}{dr} = 0 \qquad \text{at } r = 0$$

and

$$t = t_m \qquad \text{at } r = 0$$

so that the solution expressing t in terms of r is

$$t = t_m - \frac{q'''r^2}{4k_f} \tag{8-24}$$

The temperature distribution here, as in the flat plate, is parabolic. The maximum temperature difference in the system may be obtained by putting $r = R$ and $t = t_s$ in Eq. (8-24), so that

$$t_m - t_s = \frac{q'''R^2}{4k_f} \tag{8-25}$$

q_r, the heat flow at any radius r, is equal to the total heat generated by the element within that radius. Thus

$$q_r = \pi r^2 L q''' \tag{8-26}$$

q_s, the heat conducted out of the periphery of the element, is equal to the heat generated in the entire element and is given by

$$q_s = \pi R^2 L q''' \tag{8-27}$$

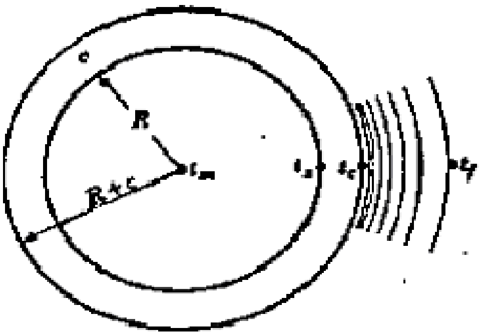
Another expression of q_s, in terms of the boundary temperatures, can be obtained by combining Eqs. (8-25) and (8-27) and rearranging to give

$$q_s = 4\pi k_f L(t_m - t_s) \tag{8-28}$$

Putting $2\pi RL = A_s$, the peripheral area of the element through which the total heat generated by the element must pass, and rearranging give

$$q_s = 2k_f A_s \frac{t_m - t_s}{R} \tag{8-29}$$

Note the similarity in form between Eqs. (8-14) and (8-29) for the total heat transfer of a plate-type and a cylindrical fuel element, respectively.



'ig. 8-10. Cross section of cylindrical uel element with cladding and coolant.

Effect of Cladding and Coolant. Figure 8-10 shows the cross section of a cylindrical fuel element of radius R and axial length L, having a center temperature t_m and surrounded by cladding of radial thickness c and coolant fluid having bulk temperature t_f.

In the steady state with no heat generated in cladding or coolant, and with negligible resistance to heat flow at the fuel-cladding interface, the heat passing out of the fuel surface is the same as that passing through the ladding and into the coolant fluid. (The expression for heat flow through the ladding is that for heat flow by conduction through a hollow cylinder.) Thus

$$q_s = \pi R^2 L q''' = 4\pi k_f L(t_m - t_s) = \frac{2\pi k_c L(t_s - t_c)}{\ln[(R + c)/R]} = 2\pi(R + c)Lh(t_c - t_f) \tag{8-30}$$

Solving for the temperature differences and adding, in the same fashion as was done in the case of the plate-type fuel element, give

$$t_m - t_f = \frac{q'''R^2}{4k_f} + \frac{q'''R^2}{2}\left[\frac{1}{k_c}\ln\frac{R+c}{R} + \frac{1}{h(R+c)}\right] \tag{8-31}$$

Compare this equation with Eq. (8-19) for a plate-type fuel element. An expression for the maximum theoretical flux attainable with fixed configuration and given fuel and coolant temperatures can be obtained by a procedure similar to that in the previous case. It should be noted here that in some instances, especially when ceramic fuels are used, unbonded fuel-cladding elements are built. The space between the meat (fuel) and the cladding may be filled with an inert gas such as helium (during the jacket welding process) or by a liquid metal. The space presents an extra resistance to heat flow. Because the thickness of this space is quite small, no convection effects, even in the case of a gas, need be taken into account, and heat transfer through it may be assumed to occur totally by conduction. In the case of the gas layer, thermal conductivity of the gas changes with fuel burnup because of the liberation of fission gases which mix with it. In the case of helium the effect is to decrease the gas thermal conductivity [60].

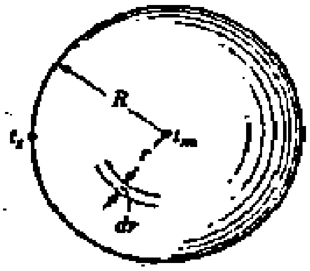


Fig. 8-11. Spherical fuel element.

8-8. Heat Flow out of Spherically Shaped Fuel

A treatment analogous to that for cylindrical fuel elements for the case of a bare (unclad) sphere of radius R, shown in Fig. 8-11, and having a uniform volumetric thermal source strength q''' yields the following differential equation:

$$k_f\frac{d^2t}{dr^2} + \frac{2k_f}{r}\frac{dt}{dr} + q''' = 0 \tag{8-32}$$

Integration of this equation for the boundary conditions $t = t_m$ at $r = 0$ and $t = t_s$ at $r = R$ gives

$$t_m - t_s = \frac{q'''R^2}{6k_f} \tag{8-33}$$

The total heat generated, q_t, is given by the expression

$$q_t = \tfrac{4}{3}\pi R^3 q''' \tag{8-34}$$

Combining Eqs. (8-33) and (8-34) gives

$$q_t = 8\pi R k_f(t_m - t_s) \tag{8-35}$$

Putting $4\pi R^2 = A_s$, the total peripheral area of the spherical element, and rearranging give

$$q_t = 2k_f A_s \frac{t_m - t_s}{R} \tag{8-36}$$

Again note the similarity in form between this expression and those given by Eqs. (8-14) and (8-29).

Expressions relating the temperatures and heat generation in the case of a spherical fuel element with cladding and coolant can be derived with a treatment similar to the preceding two cases. Also an expression for the maximum theoretical flux for given configuration and fuel and coolant temperatures can be derived in a manner similar to the above.

8-9. Axial Temperature Distribution of Coolant and Fuel Element

The temperatures expressed by the relationships of the previous three sections are those that exist in the case of uniform volumetric thermal source strength q'''. This is true in the case of short sections of a long fuel element in which ϕ does not change appreciably in the axial direction. If, however, we are to deal with a whole fuel element placed in a reactor core, as in Fig. 8-12, the axial variations in ϕ and q''' in the fuel element follow that in the core and must be taken into account.

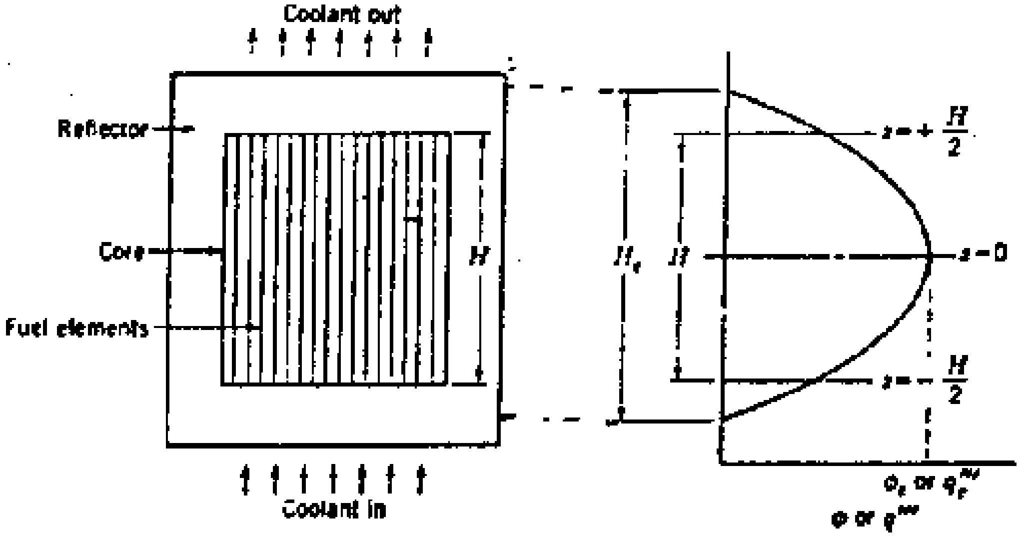


Fig. 8-12. Axial flux distribution in a reactor core. H = active core height; H_e = extrapolated height.

In the discussion that follows we shall treat the case of a single fuel element whose length equals the core height H. The element is supposed to be situated somewhere in the core where the effective neutron flux at its center is ϕ_c. From considerations of reactor physics (Chap. 5), the neutron flux drops to zero at some extrapolated height H_e, that is, $\phi = 0$ at $z = \pm H_e/2$, where $z = 0$ corresponds to the core (and fuel element) center plane. The coolant fluid is assumed to pass upward through the core and parallel to the fuel element (see Fig. 8-13). We shall also make the following simplifying assumptions:

1. The variation of ϕ in the axial direction is a pure cosine function of z. The maximum values of ϕ and q''' occur in a single fuel element at its center and will be designated ϕ_c and q'''_c. These, of course, are the maximum values for that element only. Other fuel elements closer to the center of the core normally have higher values of ϕ_c and q'''_c. It should be indicated here that when the reactivity varies with z, because of a large axial temperature rise in a water-moderated and -cooled core or because of a change in phase, such as in the boiling-water reactor, the axial flux may deviate appreciably from the cosine function. In such cases some other relationship, between q''' and z should be used. If the q''' variation is compli-

cated, the problem may be solved by separately treating segments of the fuel elements where the q''' variation can be approximated by a simple or linear function of z.)

2. (An average value of the convective coefficient of heat transfer between coolant fluid and fuel element h will be used for the entire length of the fuel element. Actually a small variation in h takes place because of the dependence of h on the physical properties of the coolant, which normally vary with temperature, and on the operating characteristics, such as velocity, which vary with both temperature and coolant-channel geometry (entrance effects, etc.). In order to maintain as uniform moderating power as possible (in a water-cooled core) the coolant temperature rise is deliberately kept to a minimum) (by increasing its mass rate of flow). To introduce h in the analysis as a variable function would unnecessarily complicate the analysis.

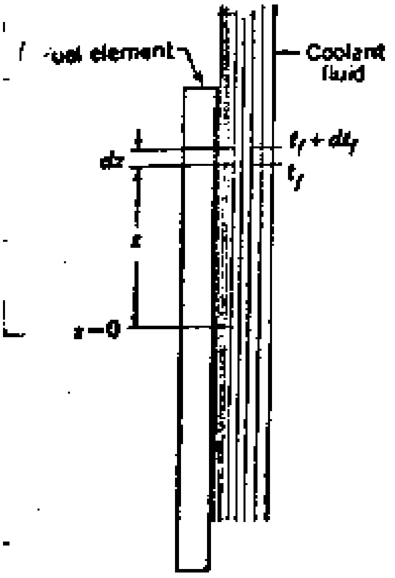


Fig. 8-13. Fuel element and associated coolant (shown on one side for simplicity).

3. (Similarly, the thermal conductivity of the fuel and the cladding as well as the physical properties of the coolant fluid will all be considered constant and independent of z.)

We were so far able to evaluate the temperatures of fuel, cladding, and coolant at any one cross section. In the following analysis we shall evaluate the axial temperature variation of the coolant and cladding surface. The position of the cross section at which maximum cladding-surface temperature takes place will also be evaluated. The cross sections at which maximum center fuel and cladding temperatures occur will be discussed later. It is these temperatures that pose metallurgical limitations on reactor operation.

The axial variation of neutron flux along the fuel element is given (Sec. 5-9) by

$$\phi = \phi_c \cos \frac{\pi z}{H_e}$$

and, since each fuel element is usually uniform in cross-sectional dimensions and in fuel type and enrichment,

$$q''' = q_c''' \cos \frac{\pi z}{H_e} \qquad (8\text{-}37)$$

where q''' and q_c''' are the volumetric thermal source strengths at any point z and the center of the fuel element, respectively.

If we consider a heat balance for a differential section of the fuel element of height dz (Fig. 8-13), at z the sensible heat gain by the coolant fluid (assuming no change in phase) is equivalent to the heat generated in the differential fuel element. Thus

$$\dot{m} c_p \, dt_f = q''' A_c \, dz \qquad (8\text{-}38)$$

where $\dot{m}$ = mass-flow rate of coolant ...

c_p = specific heat of coolant fluid, Btu/lb$_m$ °F

dt_f = coolant-fluid temperature rise between z and $z + dz$, °F

A_c = cross-sectional area of one fuel element, ft^2

Equation (8-38) combines with (8-37) and integrates between the limi $z = -H/2$ and z as follows:

$$\dot{m}c_p \int_{t_{f_1}}^{t_f} dt_f = q_c'''A_c \int_{-H/2}^{z} \cos\frac{\pi z}{H_e}\,dz$$

where t_{f_1} and t_f are the coolant-fluid temperatures at $z = -H/2$ and z.

Thus
$$t_f = t_{f_1} + \frac{q_c'''A_cH_e}{\pi c_p\dot{m}}\left(\sin\frac{\pi z}{H_e} + \sin\frac{\pi H}{2H_e}\right) \qquad (8\text{-}3$$

This equation gives the temperature of the coolant fluid t_f as a function of The coolant temperature at the center of the element can be obtained by putti $z = 0$. The exit temperature of the coolant fluid, t_{f_2}, can be obtained by putti $z = +H/2$ to give

$$t_{f_2} = t_{f_1} + \frac{2q_c'''A_cH_e}{\pi c_p\dot{m}}\sin\frac{\pi H}{2H_e} \qquad (8\text{-}40$$

In case the extrapolation lengths can be ignored, $H \cong H_e$, and Eq. (8-40 reduces to

$$t_{f_2} = t_{f_1} + \frac{2q_c'''A_cH}{\pi c_p\dot{m}} \qquad (8\text{-}40$$

The axial temperature variation of the cladding surface, t_c, will now be eva ated. The heat transferred between cladding and coolant, at any point z alo the fuel element, per unit area of cladding surface (circumferential) is give $h(t_c - t_f)$. Since h has been assumed constant along the fuel element, the te perature difference $t_c - t_f$ is directly proportional to the volumetric ther source strength q''' at that point. Since the latter is a cosine function of z, above temperature difference also has the form

$$t_c - t_f = (t_c - t_f)_c \cos\frac{\pi z}{H_e} \qquad (8\text{-}4$$

but
$$q_c'''A_c\,dz = hC\,dz\,(t_c - t_f)_c \qquad (8\text{-}4$$

where C is the circumferential length of the clad fuel element in feet, a const value, and $(t_c - t_f)_c$ is the temperature difference between cladding surface coolant at $z = 0$. Combining Eqs. (8-39), (8-41), and (8-42) and rearranging

$$t_c = t_{f_1} + q_c'''A_c\left[\frac{H_e}{\pi c_p\dot{m}}\left(\sin\frac{\pi z}{H_e} + \sin\frac{\pi H}{2H_e}\right) + \frac{1}{hC}\cos\frac{\pi z}{H_e}\right] \qquad (8\text{-}$$

The variation of q''' (same as ϕ), t_f, and t_c along the coolant path (in the direction) is shown in Fig. 8-14. Note that $t_c - t_f$ is a cosine function of z has a maximum value at the center.

The other temperatures within the fuel element, such as t_s and t_m, may the evaluated at any point z from t_f and t_c at that point by working backward the fuel element, using the appropriate relationships given in Secs. 8-6 to 8-8. It should be remembered that this technique is not entirely correct inasmuch as the axial flux change actually causes the heat flow out of the fuel element to deviate

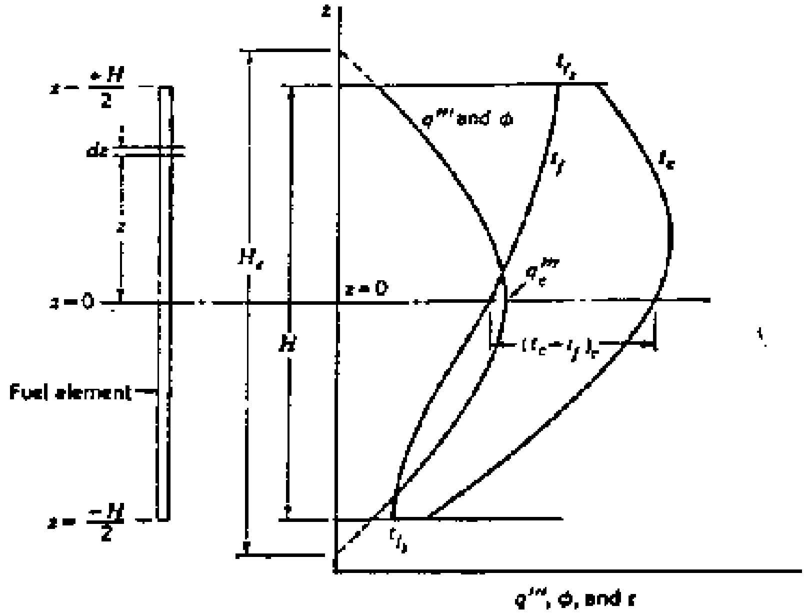


Fig. 8-14. Axial distribution of fuel-surface and coolant temperatures.

from one-dimensional (radial). However, for a long, thin fuel element, the error involved is quite small.

8-10. Maximum Temperatures in Fuel Element

The point at which the maximum cladding-surface temperature occurs, called z_c, can be obtained by differentiating Eq. (8-43) with respect to z, equating to zero, and solving for z_c as follows:

$$\frac{dt_c}{dz} = 0 + q''A_c\left[\frac{H_e}{\pi c_p \dot{m}}\frac{\pi}{H_e}\left(\cos\frac{\pi z}{H_e} + 0\right) + \frac{1}{hC}\frac{\pi}{H_e}\left(-\sin\frac{\pi z}{H_e}\right)\right] = 0$$

Dividing by $\cos \pi z/H_e$ and rearranging give

$$\tan\frac{\pi z}{H_e} = \frac{hCH_e}{\pi c_p \dot{m}}$$

Solving for z in the above equation gives z_c as follows:

$$z_c = \frac{H_e}{\pi}\tan^{-1}\frac{hCH_e}{\pi c_p \dot{m}} \tag{8-44}$$

In this equation all components of the arc tangent are positive, indicating that it can assume values only in the first quarter, i.e., between 0 and $\pi/2$. This means that z_c is a positive value or that the point of maximum cladding-surface temperature occurs past the mid-plane ($z = 0$), as shown in Fig. 8-15. This is to be expected, since the coolant, coming in from the bottom at low temperature, helps to keep the bottom half of the fuel element cooler than the top half. Equation (8-44) also indicates that increasing the heat-transfer coefficient or decreasing the coolant mass-flow rate shifts the point of maximum cladding-surface temperature farther away from the fuel-element mid-plane. It also indicates that z_c is independent of q''' or ϕ.

The maximum cladding-surface temperature t_{cm} can now be obtained by substituting the value for z_c from Eq. (8-44) into Eq. (8-43).

It is also of interest to evaluate the maximum values of the cladding inner temperature t_s and the fuel-center temperature t_m since these are the ones that pose limitations on the maximum allowable heat generation from the fuel element. The positions of these two maxima along the fuel element can be evaluated by a technique similar to that above. An expression for $t_m - t_f$ or $t_s - t_f$ for the particular fuel-element shape used, such as Eq. (8-31) for a cylindrical fuel element, should be combined with Eq. (8-39) and an expression for t_m or t_s in terms of t_{f_1}, obtained. The remaining steps are similar to those in which z_c and t_{cm} were obtained.

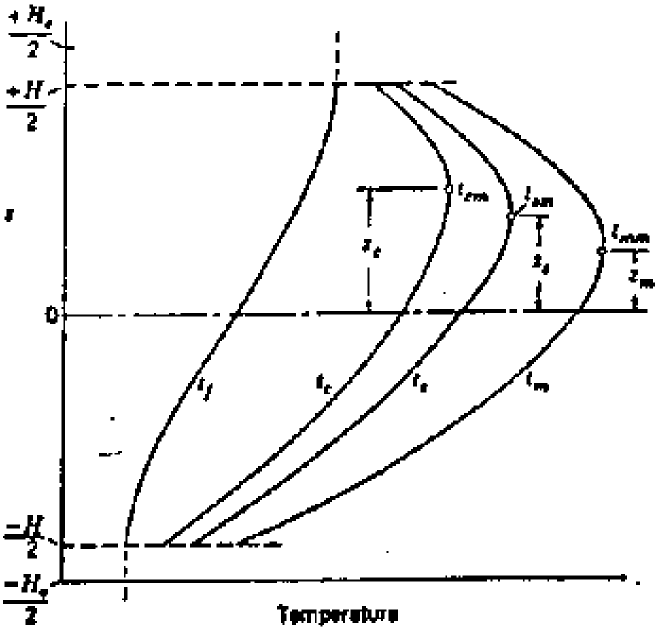


Fig. 8-15. Variation of fuel-element center, surface, and cladding temperatures with height.

The positions of the maximum values of t_m and t_s, t_{mm} and t_{sm}, occur at z_m and z_s, closer to the fuel-element mid-plane, as shown in Fig. 8-15. The reason for this is that the radial heat flow through the fuel and cladding, at any section z, is proportional to q''' at that section. This is a cosine function of z and a maximum at the fuel-element mid-plane ($z = 0$). Thus in the upper half of the fuel element, while t_f increases with z, $t_c - t_f$ is a cosine function of z, that is, decreases with z, giving a maximum for t_c at some position z_c. $t_s - t_c$ is also a cosine function of z. Thus when $t_s - t_c$ is added to t_c it causes the maximum t_{sm} to shift further below t_{cm}. Similarly, $t_m - t_s$ is a cosine function of z and when added to t_s causes another shift in the maximum t_{mm} to a point below t_{sm}. An interesting and simple graphical solution of the axial temperature variations has been worked out by French [61].

THERMAL AND BIOLOGICAL SHIELD

Essentially all the energy absorbed in the shield front the fast neutrons and gamma rays is ultimately degraded into heat. This means that, *for* a reactor of moderate or high power, a considerable amount of heat is generated within the shield. Since the absorption of both neutrons
and gamma radiation is at least approximately exponential in character, a large proportion of the total heat liberated will be released in those parts of the shield closest to the reactor.absorption of the gamma-ray energy in the first 10 percent of the shield. Similarly, 99 percent of the absorption will Occur within the first 20percent shield thickness

Pressurized-water (PWR) an boiling-water (BWR) reactors are contained in thick-walled steel pressure vessels heat generated in the walls of these vessels as a result of radiation absorption could cause unacceptable thermal stresses. In order to prevent damage to the pressure 'vessel as well as to the main shield (or *biological shield)* from excessive heating, some form of *thermal shield* is introduced core and the containing vessel between the reactor ~. e thermal shield, usually made of steel, attenuates the radiation from t e core and thus reduces the flux impinging \ on the pressure vessel and th biological shield. The energy deposited in I the thermal shield as heat by the absorbed radiations is removed by the mm, respectively, and they serve as thermal shields.

The need for thermal shielding is less in BWRs than in PWRs. Not only is the pressure vessel wall thinner in a BWR (150 to 180 mm versus 230 mm), but for the same total thermal power and power density, is substantially lower because of the larger core volume. For a given power
output, the pressure and thermal stresses in the wall of the pressure vessel are lower in a BWR than a PWR. Furthermore, the steel core shroud and the jet pumps which surround the core provide some thermal shielding.

Since the thermal shield is a heavy metal, it is effective for attenuating gamma rays and for the inelastic scattering of fast neutrons. These two types of radiation carry most of the energy leaking from the core. The heat deposited by these radiations in the thermal shield is removed by the coolant in a water-cooled reactor, and so it contributes to the energy available for power generation. In reactors of low or moderate power, such as are used for educational and research purposes, thermal shields are not Reactor Shielding Requirements

In principle, the problem of shielding the reactor itself has three aspects: (1) slowing down of fast neutrons, (2) capture of the slowed down (or initially slow) neutrons, and (3) attenuation of all forms of gamma radiation, including primary radiations from the reactor core and secondary radiations formed as a result of. various interactions between neutrons and nuclei outside the core. .

Since reactor shielding involves the attenuation of fast neutrons and gamma radiations, the problems are essentially the same for both thermal and fast reactors. A significant difference is that the number of ' neutrons escaping from a fast reactor, for a given operating power, is generally greater than from a thermal reactor. Furthermore, the breeder blanket of a fast reactor can be a major source of radiation, as a result of fission by fast neutrons and *(n, ,,/)*reactions withur~nium-238. The source distribution is therefore more complicated than for a thermal .reactor.

SHIELDING MATERIALS

Materials used in shielding may be divided into three broad categories, according to their function: (1) heavy or moderately heavy elements to attenuate the gamma radiation and to slow down very fast neutrons to about 1 MeV by inelastic collisions, (2) hydrogenous substances to moderate neutrons having energies in the range below about 1MeV by elastic, collisions and (3) materials, notably those containing boron, which capture neutrons without producing high energy gamma rays. Although these three classes will be considered separately, it will be seen that the
same material may often serve two or all three purposes.

Iron, as carbon steel or stainless steel, has been commonly used as the material for thermal shields; as already seen, such shields can absorb gamma rays and fast neutrons escaping from the reactor core. In fact, two or three layers of steel with water between them,' such as are present in PWRs, represent a –very effective shield for both neutrons and gamma rays. Apart from the use of . massive iron in shield construction, iron turnings or punchings and iron oxides have been incorporate (fine concrete for shielding purposes. further reference special concretes, which include heavy elements such as iron or barium n some form, will be made shortly.

Because of their hi density and ease of fabrication, lead and lead alloys b: ve been used t some extent in nuclear reactor shields. For gamma rays with energies .the region of 2 MeV, roughly the same mass of lead as of iron is required to remove a specified fraction of the radiation. However, at both higher and lower energies the mass-attenuation efficiency of lead is appreciably greater than that of iron. Because of its low melting point, lead can be used only where the temperatures are not too high.

Fusion reactors

A fusion reactor *would have* several of the attractive features of present-day nuclear power plants:
Abundant fuel supply
Efficient conversion of mass to energy
No air pollution or contribution to global warming But it would *not have* the critical negative concerns that have stopped the development of fission reactors:
No risk of uncontrolled energy release
Greatly reduced quantity of high level waste
Greatly reduced threat to non-proliferation of weapons material

A Fusion Reactor Would Have Abundant Fuel Supply

The first fusion reactor would probably burn two isotopes of Hydrogen, Deuterium and Tritium, which differ from Hydrogen in the number of neutrons in the nucleus.
D + T ------- He + neutron + 17.6 MeV

The ratio of naturally occurring Deuterium to Hydrogen atoms is about 1:6000. Said another way, there are more than 10^{13} *tons* of Deuterium in the surface waters of the earth. This is truly an inexhaustible fuel supply

- Tritium is radioactive (β-emitter) and decays with a half-life of about 12 years. Therefore it does not occur naturally and must be bred in the fusion reactor, for example through the nuclear reaction (Natural Lithium is ~ 92.5% ^{7}Li)

^{7}Li + neutron ------- ^{4}He + T + neutron – 2.49 MeV

^{6}Li + neutron -------- ^{4}He + T + 4.8 MeV

The Lithium from land deposits or sea water is sufficient for *thousands of years'* energy supply

While the first generation fusion reactor would burn D-T, a second generation reactor would likely burn only Deuterium. This requires some advance of present day plasma physics, but would greatly simplify the fuel cycle.

ADVANTAGES OF FUSION

In Fission, the energy release is about 1 MeV per Nucleon
In D-T Fusion, the energy release is about 3.5 MeV per Nucleon

By comparison, fossil fuels are grossly inefficient, less than 1 eV per Nucleon!

This efficiency translates to correspondingly smaller requirements for the fuel supply system and waste disposal:
To generate 1000 MW in one day requires *9000 tons of coal* and generates *30,000 tons of* CO_2 (plus other noxious gases, e.g., SO_2 and NO_2). To generate the same amount of energy from fusion requires about *2.5 pounds of D + T and makes 2 pounds of He!*

The amount of fuel in the reactor at any time is sufficient to burn for only ~ 10 s. Fuel is continuously supplied and can be shut off by a simple valve, much like the way fuel is supplied to a gas burner.

In addition, fusion reactions are easily extinguished by injection of heavier impurity gases or a "poison pellet".
A "melt-down" cannot occur! Fusion reactors are passively safe.

No Actinides are produced in the fusion fuel cycle

The neutron from D-T fusion will cause some activation as it interacts with material walls. But structural materials can be used which minimize the production of radioactive isotopes and their environmental consequences.
Advanced designs for fusion reactors permit recycling of materials or shallow burial about 100 years after shutdown.

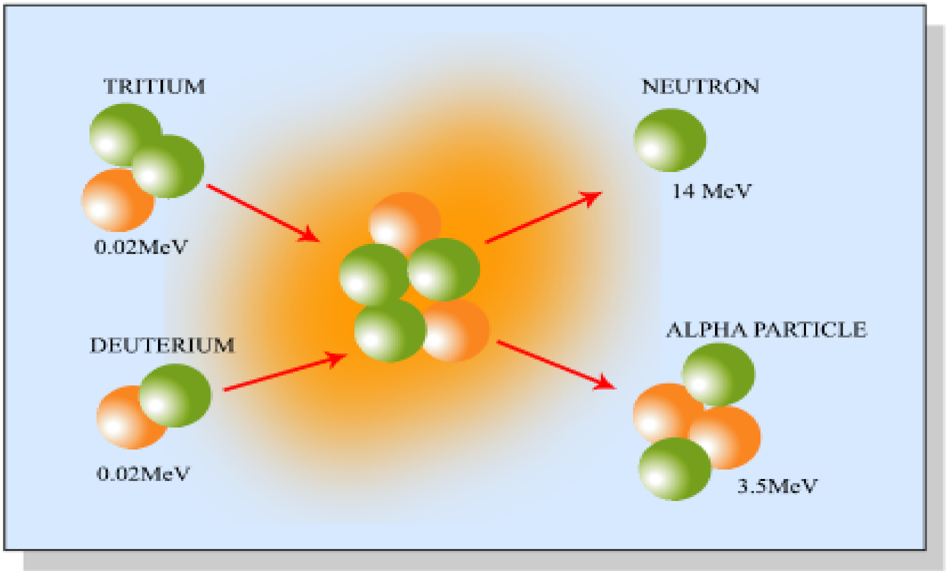


Fig 4.7 Fusion Process

THE FUSION PROCESS

D and T are both positively charged and repel each other through the Coulomb force

If D and T ions manage to touch then the nuclear force takes over and turns

D + T □He(3.5MeV) + n(14.1MeV)

Repulsion is overcome by increasing the speed or energy of D and T…in a hot fusion plasma, the D and T nuclei are hot enough to fuse…

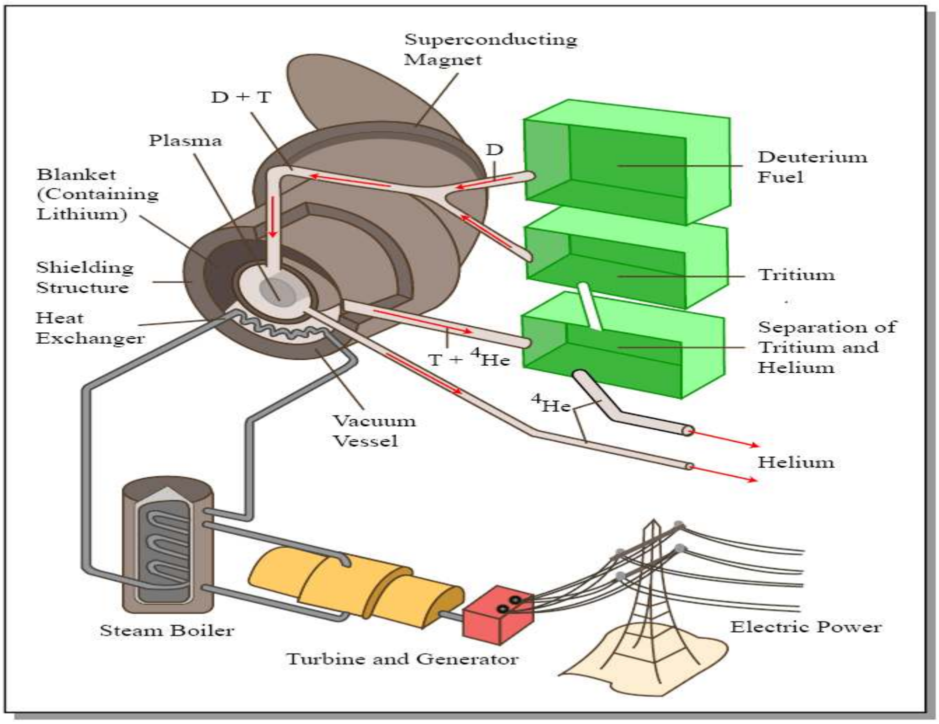


Fig 4.8 Fusion reactor

FUSION POWER GENERATION

In steady-state, the energy from the fusion reactions are radiated to the first wall and blanket.

Tritium is not available naturally but can be readily bred with lithium.

The 14.1MeV neutron is absorbed into a blanket containing lithium in order to breed tritium.

The hot blanket and first wall is cooled; the thermal energy Is extracted from the coolant for power generation.

FUSION ENERGY REQUIREMENTS

Need enough particles to fuse: n

- Need to keep the plasma together long enough for fusion reactions to occur: τ
- Need enough energy for the particles to fuse : T

So the product n τ, T must be large enough to sustain fusion events at a sufficient level to produce net power

FORCES OF FIELDS ON PLASMA

The plasma now feels a force from the magnetic field

Ions and electrons follow the field lines

- Plasma is confined
- More organization

A linear device, however, has end losses

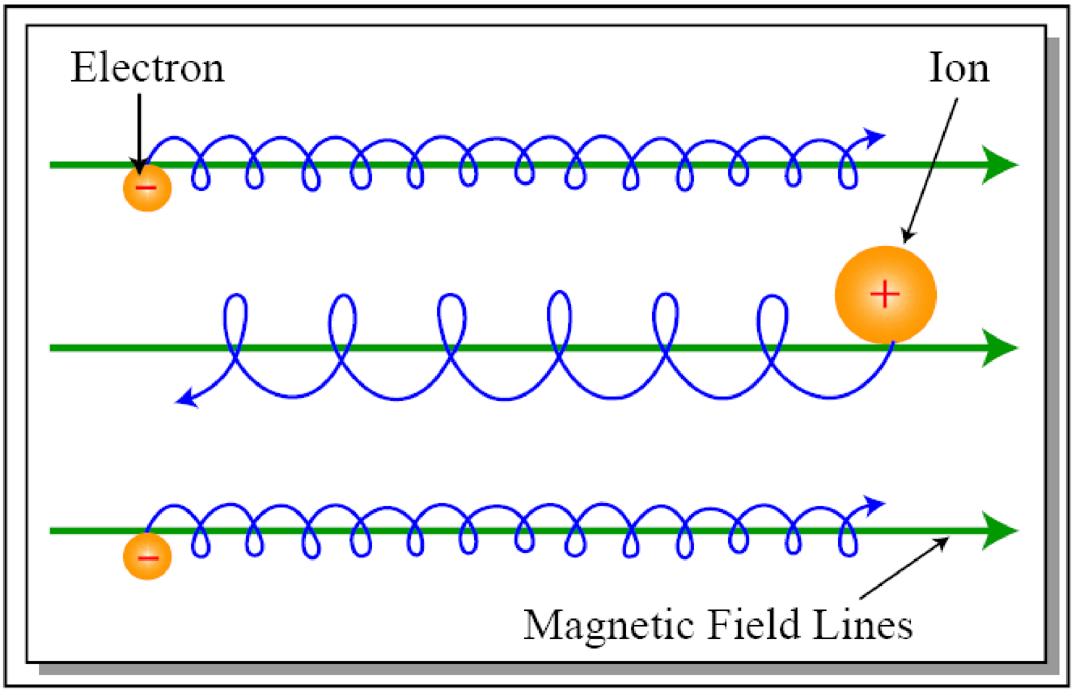


Fig 4.9. Plasma field

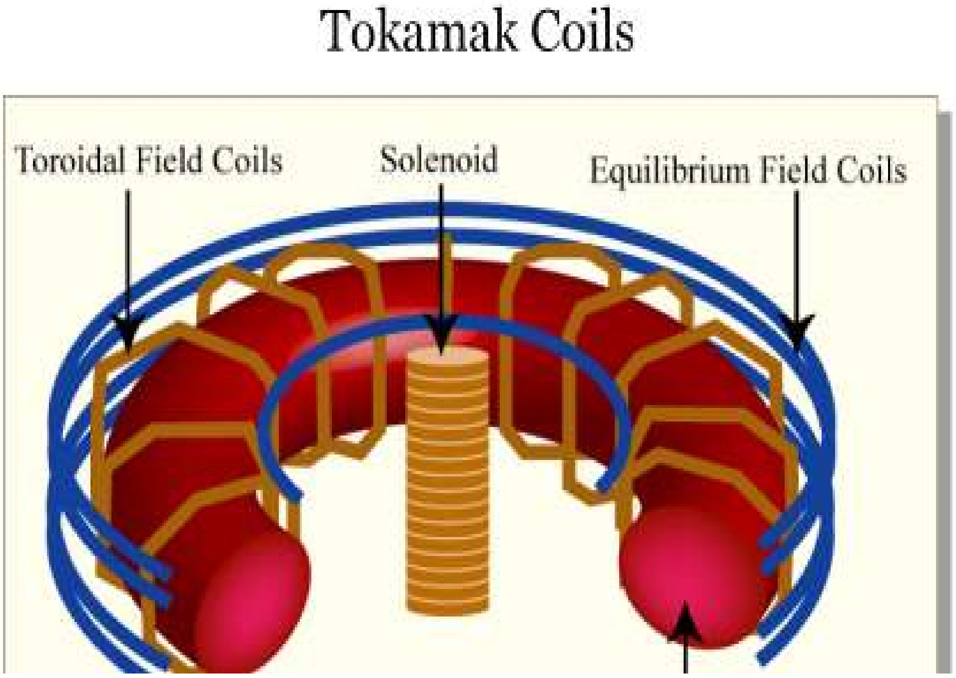


Fig 4.9 View of Toroidal coil

UNIT V

ACCIDENT PREVENTION

Introduction

The first goal in reactor safety is to prevent accidents from occurring. This goal has two aspects. First, the reactor system needs to be designed, constructed, and operated so that the chances of a malfunction or operational error are very small. Since some equipment failures and operational mistakes are inevitable during the service lifetime of such a complex system as a nuclear power plant, the second aspect of the prevention goal is to provide "self-healing" features that will cope with such incidents.

The attainment of a reliable system by conservative design requires anticipating possible modes of failure and making provision for them, meeting demanding quality standards, and adhering to applicable regulatory requirements. Details of system design are beyond the scope of this treatment. However, we will examine some accident scenarios, the prevention of which must be demonstrated as part one licensing procedure. The need for quality assessment is also enforced as a regulatory requirement.

QUALITY ASSURANCE: CODES AND STANDARDS.

Engineering codes and standards are an important aspect of quality assurance, since they represent the' recognized practice for assuring acceptable levels of quality and performance in materials and components. A number of preexisting codes, such as the ASME Boiler and Pressure Vessel Code have been improved, supplemented, or completely revised to satisfy requirements of the nuclear industry.

Hundreds of standards are relevant to the practice of nuclear' engineering. These cover all aspects of nuclear power plant design, construction, equipment performance, and instrumentation, as well as the manufacturing of nuclear fuels. Many standards also deal with computer codes and information transfer. Since standards document accepted practice, it is important for a nuclear reactor engineer to become familiar with those standards that are relevant to a given professional assignment. These are generally available in technical libraries or through ANSI.

Special code requirements are established for *nuclear safe y grade* components, which are considered vital 10 plant safety. When such components such as valves are manufactured to meet these stringent requirements, they may be labeled with an "N" stamp. Categorization into classes, depending upon safety significance, with differing requirements for quality assurance and in service inspections is as follows:

Safety Class 1. This, the most vital category applies to components of the primary coolant system, whose failure would cause a major coolant loss.

Safety Class 2. In this category, we have structures and components that are required to fulfill a safety function such as shutting down the reactor, cooling other safety systems, and controlling the release of radioactivity.

Safety Class 3. This applies to systems whose failure would allow release to the environment of gaseous radioactivity that would normally be held for decay with in the plant.

Non safety grade components must meet "high-quality industrial standards." Since a significant additional expense is associated with safety grade components, there has been in the past a design incentive to specify such components only for essential safety functions.

Many standards have been incorporated into Federal Regulations and Regulatory Guides issued by the U.S. Nuclear Regulatory Commission (NRC). General quality assurance criteria are specified in Title 10 of the Code of Federal Regulations, Part 50, Appendix B (10 CFR 50). Specific quality assurance requirements appear in the Regulatory Guide dealing with the particular topic considered.

There are five major aspects of a quality assurance program:

1. The actual formulation of the program itself includes specific detailed policies and procedures.
2. Should design be required, quality control of the necessary practices is needed.
3. Component and material procurement activities require the enforcement of quality assurance requirements. .
4. Inspection and test procedures are prescribed to assure that all specifications are met.
5. Auditing and record-keeping procedures are provided to assure adherence to requirements.

INHERENT REACTOR STABILITY

In selecting a safe reactor design, an essential requirement is that he concept should provide inherent stability against an increase in reactivity. This can be realized if the reactor has a quick-acting negative temperature (or power) coefficient of reactivity. As seen in Chapter 5, there is then a self-limiting effect on disturbances that lead to an increase in temperature (or power level). This self-limiting feature is generally the result of the fuel Doppler coefficient, although in water-cooled react<?rs the expansion of the coolant-moderator contributes to the overall negative coefficient. However, the moderator contribution is delayed because of the time required for heat transfer from the fuel to the moderator (§5.92). 12.21. It should be understood that whereas a negative temperature (or power) coefficient is a requirement for reactor safety, it does not guarantee safety. If there were a sudden increase in reactivity, the power excursion would be terminated automatically after a finite time because of the negative temperature coefficient. But during this time the thermal power, and hence the fuel temperature, may have risen o such a high level that the fuel rods would suffer damage. This aspect of safety is taken into consideration in reactor system design. Also, should the system have a positive coolant void coefficient, as was the case in the hernobyl4 reactor, the danger would be exacerbated.

REACTOR PROTECTION SYSTEM

Nuclear power reactors are designed to produce heat to satisfy he demand for steam by 'a turbine-generator, up to a specified limit. The reactor control system, with its automatic and manual controls, serves to maintain safe operating conditions as the demand is varied because excess cooling capability is provided in the design of the reactor system, an overpower equal to about 118 percent (in a PWR) or 120percent (in a BWR) of the rated (or design)power can be tolerated without causing damage to the fuel rods. If the thermal power should exceed the limiting value or if other abnormal conditions which might endanger the system should arise, the reactor protection system would cause reactor trip (orscram)

In reactor operations, the term *transient* describes in general, any significant deviation from the normal value of one or more of the important operating parameters, e.g., system temperatures and pressures, thermal power level, coolant flow rate, turbine trip, equipment failure, etc. If the transient is a minor one, within the permissible operating limits
of the system, the controls will be adjusted automatically to compensate for the deviation. A severe transient, however, will activate the reactor protection system.

The purpose of the protection system is to shut the reactor down and maintain it in a safe condition in the event of a system transient or malfunction that might cause damage to the core, most likely from overheating. The protection system includes a wide variety of instruments for measuring operating variables and other characteristics of the overall nuclear
plant system. If the instruments indicate a transient that cannot be corrected immediately by the control system, the reactor is shut down automatically by the protection system. In addition, the reactor operator can cause an independent (manual) trip if there are indications that an unsafe condition may be developing.

When a reactor trip signal is received in a PWR, the electromagnetic clutches holding up the control rods are deenergized by an automatic cutoff of electric power. The rods then drop into the reactor core. Borated water (boric acid solution) can be injected from the chemical and volume control system or CVCS by manual action to provide a
backup to the control rods if required. In a BWR, a rapid shutdown is achieved by forcing the control rods up into the core by hydrostatic pressure; at the same time, power to the recirculation pumps is cut off. The reactivity in a BWR can also be decreased by injection of an aqueous solution of sodium penta borate.

. An essential requirement of the reactor protection system is that it must not fail when needed; on the other hand, an error in the instrumentation or other malfunction with the system should not cause an unnecessary ("false") reactor trip. In order to avoid such false trips, three or more redundant channels, consisting of detector and actuator, are used to monitor operating variables. A reactor trip will occur only when two or more channels call for action simultaneously. The availability of several independent channels permits regular settings of the channels, one at a time, without impairing the effectiveness of the protection system.

REACTOR TRIP SIGNALS

Some of the signals that would cause actuation of the protection system.. A more complete listing is given here for water-cooled reactors; unless otherwise indicated, the trip signals apply to both PWRs and BWRs.

1. Rapid increase in the neutron flux during startup, resulting in a too rapid rise in the thermal power .
2. High neutron flux during power operation, indicating an overpower above the permissible level
3. Abnormal reactor system temperature or pressure
4. Loss (or decrease) of coolant flow, e.g., from a pump failure
5. High steam flow, e.g., from a break in a steam line
6. Closure of a main steam isolation valve. especially in a BWR
7. Turbine-generator trip, e.g., from a loss of load
8. Loss of power supply for instruments (dc) or for pumps, valves, etc. (ac)
9. High water level in the pressurizer (in a PWR)
10. Low water level in the reactor vessel (in a BWR)
11. Low feed water flow or low water level in a PWR steam generator
12. High radioactivity in the steam from a BWR

SHUTDOWN COOLING

Although the reactor shutdown cooling system is not generally regarded as a component of the protection system, shutdown cooling is nevertheless an essential aspect of reactor protection. When a reactor is shut down, either deliberately or in response to a severe transient, the self sustaining fission chain reaction is terminated but a considerable amount of sensible (or stored) heat is still present in the fuel rods. Furthermore, heat continues to be generated by decay of the fission products and (for a short time) by fissions caused by delayed neutrons. Hence, cooling of the reactor core mU3tbe maintained for many days after shutdown. The sensible heat and the delayed fission heat are removed within about half a minute and then only type decay heat determines the cooling requirements.

The thermal power from this source is initially about 7 percent of the operating power of the reactor at shutdown, assuming the reactor has been operating for a substantial time. The decay power decreases to about 1.3 percent after an hour, 0.4 percent after a day, and 0.2 percent at the end of a week (see Fig. 2.33).

If the normal heat removal system is still operative when the reactor is tripped, cooling will, of course, be adequate. Steam, produced in the steam generator of a PWR or in the reactor vessel of a BWR, bypasses the turbine and goes directly to the condenser. The condensate then returns .to the steam generator (in a PWR) or to the reactor vessel (in .. a BWR) in the usual way. When the system temperature and pressure have decreased to a sufficient extent, the cooling function is transferred to the residual heat removal (or shutdown cooling) system which circulates primary system coolant through the reactor vessel and independent heat exchangers cooled by a separate service water supply system. This service water is obtained from the so-called ultimate heat sink which usually also provides the condenser cooling water. Since the heat sink is required for safe emergency shutdown of the reactor and subsequent dissipation of the residual heat, the system must be capable of operating for at least 30 days even if the most severe natural phenomenon expected at the plant site should occur. *

In some situations, the reactor would be tripped and isolation valves in the steam supply would close automatically to prevent the escape of possibly radioactive steam to the environment. The normal heat removal and condenser system would then not be available for cooling the fuel. In a PWR, a large condensate tank can provide an auxiliary supply of feed water to the steam generators to permit reactor cool down for about 8 hours. By this time, the conditions would be suitable for the residual-heat removal system to function. In a BWR, a pressure-relief valve permits controlled release of steam to the pressure-suppression pool where it would be condensed (§ 12.51). The level of the water in the reactor vessel is then maintained by the reactor core isolation system which is supplied by water from a condensate storage tank. Finally, the residual-heat removal system can be actuated.

If electric power is lost when the reactor is tripped, so that the pumps cannot operate, overheating of the fuel can be prevented by releasing steam from the safety valves. This can be continued as long as feed water is available (from the auxiliary system) to the steam generators in a PWR or to the primary system in a BWR. In the meantime, startup of the emergency (diesel) generators will permit operation of the essential pumps, pending the restoration of offsite power.

ENGINEERED SAFETY FEATURES

INTRODUCTION

The purpose of the engineered safety features is to prevent or limit the escape of radioactivity to the environment in cases of a highly unlikely transient or accident that is too severe to be accommodated by the reactor protection system alone. The major engineered safety features are: (1) the emergency core-cooling system to supply water to the reactor core in the event of a loss-of-coolant accident, (2) the containment vessel (or structure) to provide a barrier to the escape to the environment of radioactivity that might be released from the reactor core, (3) the cleanup system for removing part of the radioactivity and heat that may be present in the containment atmosphere, and (4) hydrogen control to prevent formation of an explosive hydrogen-oxygen (air) mixture in the containment.

The Emergency Core-Cooling System

If a substantial break should occur in the primary coolant circuits of a water-cooled reactor, the system pressure would drop and the emergency core-cooling system (ECCS) would become operative. This system consists of a number of independent subsystems which would be actuated in sequence as depressurization proceeds. The subsystems introduce water into the core to provide cooling when the flow of primary coolant is significantly decreased or lost. The water is supplied from different sources,

with larger volumes becoming available to flood the core during the later stages of depressurization. Redundancy of equipment and flow paths is provided to assure reliability of operation. Since the primary cooling systems are different in PWRs an BWRs, so also are the ECCS subsystems. There are also variations among PWRs fabricated by different vendors and also among BWRs of different designs. The following descriptions may be
regarded, however, as being fairly typical of current practice

P*ressurized-Water Reactors*

The *high-pressure injection system* (HPIS) comes into operation when the reactor system pressure suffers a moderate drop, from th normal operating pressure of 15.5 MPa to about 11 MPa. Such a decrease could occur as the result of a small break in the primary coolant circuit of a PWR or if, following an over pressure condition, the pressurizer relief valve failed to close after the normal pressure had been re . 6). The HPIS would then rapidly compensate for the loss of coolant (water or steam). This ECCS subsystem, which utilizes the same pumps as the CVCS, injects borated water into the reactor coolant inlet lines (cold legs). A slow-acting backup system may be included to inject borated water into the reactor
vessel outlets (hot legs).

In the event of a large break, the system pressure would drop rapidly and then another ECCS subsystem, the *accumulator injection system,* would be actuated. The accumulators are two or more independent tanks containing cool borated water stored under nitrogen gas at a pressure of about 1.4 to 4.1 MPa, according to the system design. The tanks are connected through check valves to the reactor cold legs or sometimes ~ directly into the upper part of the down comer system of the reactor vessel (see Fig. 1.4). As the-primary system pressure drops below the pressure in the accumulator, roughly 20 to 25 s after a large break, the check valves would open automatically and borated water would be rapidly-injected into the reactor vessel. These accumulator tanks, one for each coolant
loop, constitute a "passive" system since it can function without pumps which require electric power.

As the pressure is reduced further, a *low-pressure injection system* (LPIS), which utilizes the pumps and heat exchangers of the auxiliary cooling (residual-heat removal) system, is actuated. This (like the HPIS) is an "active" system, which requires electrically driven pumps. If the offsite power should fail, there would be ample time (30 s from the time of the major break in the coolant line) for the emergency diesel generators to start. Details of the systems vary among reactor vendors, but generally the design provides for the water to be drawn initially from the refueling storage water tank* followed, if necessary, by water from a sump in the containment structure.

A typical flow scheme for the ECCS components of a PWR is shown in Fig. 12.1. In this particular case, the HPIS circuit includes a tank of concentrated boric acid solution. The contents of this tank are injected into the reactor vessel as the solution is displaced by water forced in by the CVCS pumps. The safety injection pump is a component of the HPIS
backup system.

The most severe design basis accident (§ 12.73) for a PWR is taken to be a major break in one of the cold legs of the primary coolant system. In order to establish a conservative design for the ECCS, it is usually assumed that all the water injected into the broken loop will be lost out of the break. Only water injected into the other loops is regarded as being available for cooling the reactor core.

Boiling-water reactors

The design basis loss-of-coolant accident postulated for a BWR has large break in the intake line to one of the recirculation pumps, Le., 10the line between the reactor vessel and the pump (see Fig. 1.5). Water and steam would escape from the break, and the system pressure would drop. However, the depressurization rate is less than in a PWR (cf. §12.87).

12.40. In more recent BWRs, the ECCS consists of three subsystems designed to be effective over the full range of potential pipe rupture sizes.

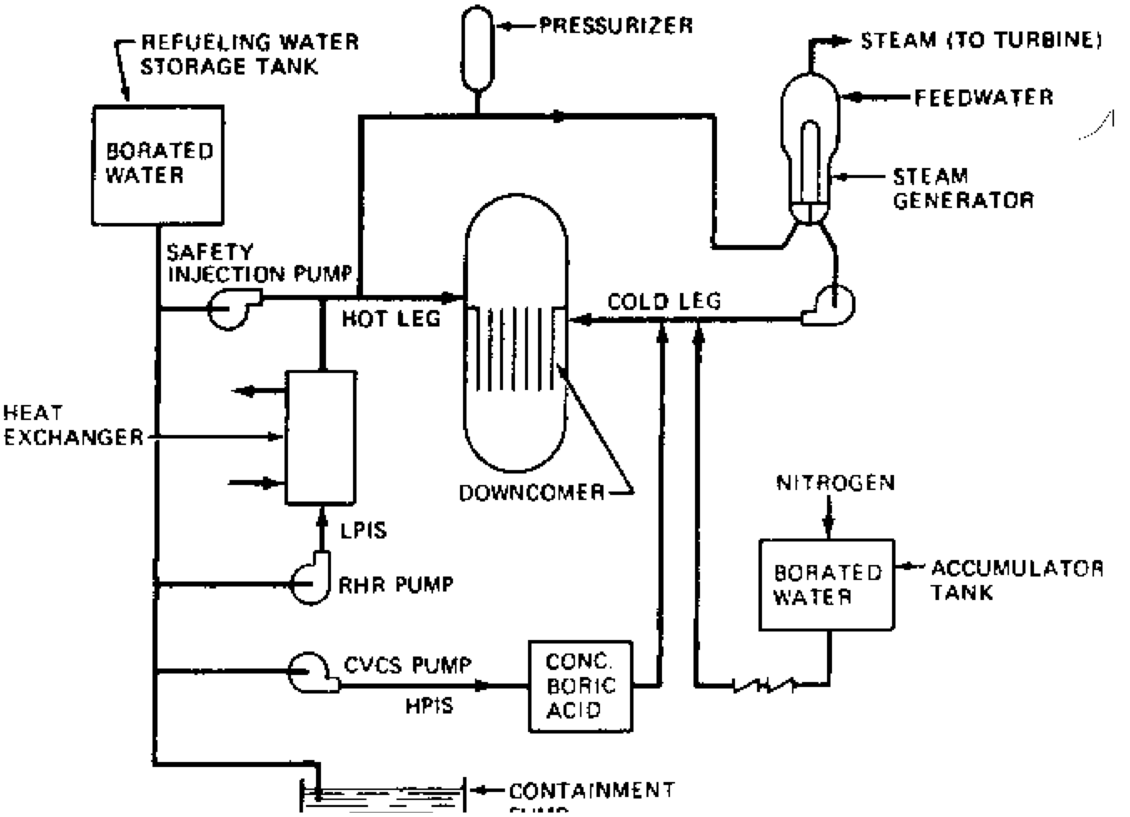


Fig 5.1 Emergency core cooling system of PWR

Redundancy of components assures a high level of operational reliability. A high-pressure core-spray system (HPCS), using a single pump, forces water through a sprayer above and around the periphery of the reactor core through nozzles. This system can function from the full operating pressure of the reactor (7.24 MPa) downward. A low-pressure core-spray system (LPCS), also with a single pump, provides protection against large breaks that would cause more rapid depressurization of the reactor vessel. Finally, a low-pressure injection system (LPIS), consisting of three separate pumping subsystems, can deliver a large volume of water to refill the vessel once it is depressurized.

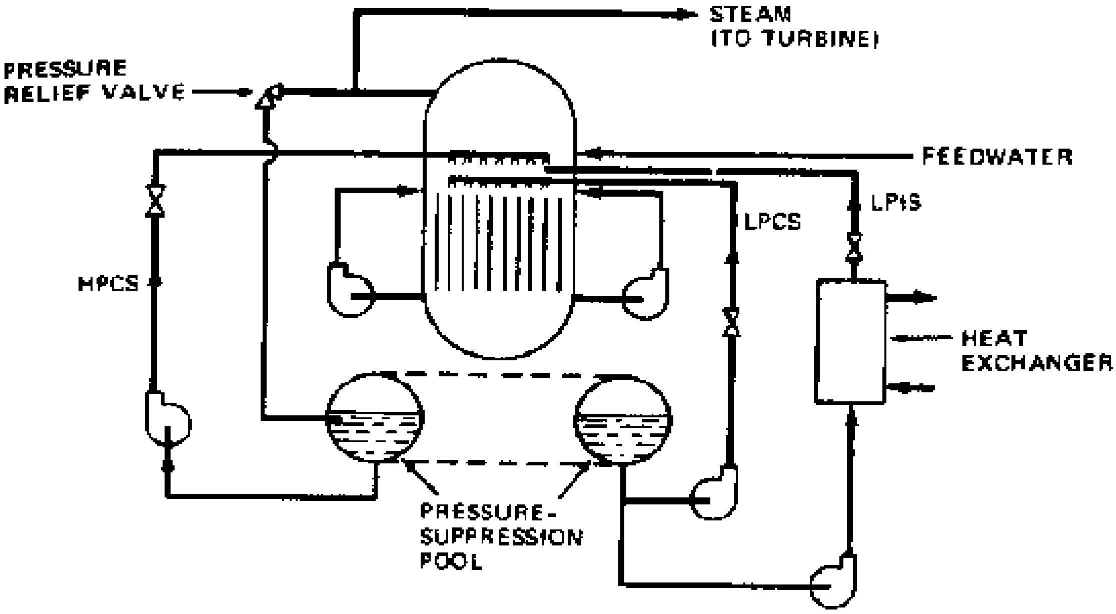


Fig 5.1 Emergency core cooling system of PWR

Various ECCS subsystems would e actuated upon receipt of system pressure signals. An essentially continuous supply of water is' available for a U the subsystems from the pressure- supression pool (§12.52). This pool would also collect water and steam lost fr the reactor vessel through a break. An additional safeguard is an automatic relief valve to prevent steam to the suppression pool. This would reduce the system pressure in case of small breaks so that the LPCS and LPIS could become effective. A schematic flow scheme of the ECCS subsystems for a BWR is given in *Passive ECCS* So-called "next-generation" reactor plants, to be described in feature emergency cooling provided by passive means. For example, in a gravity-fed system used to flood the core, cooling will be available even in the event of loss of station power, thus providing some additional safety backup.

CONTAINMENT SYSTEM

A containment structure enclosing the reactor primary system acts as the final backup barrier to fission product release to the environment in the defence-in-depth design of the reactor safety systems. Although containment design in LWRs is traditionally based On holding the pressure resulting from the release of the primary

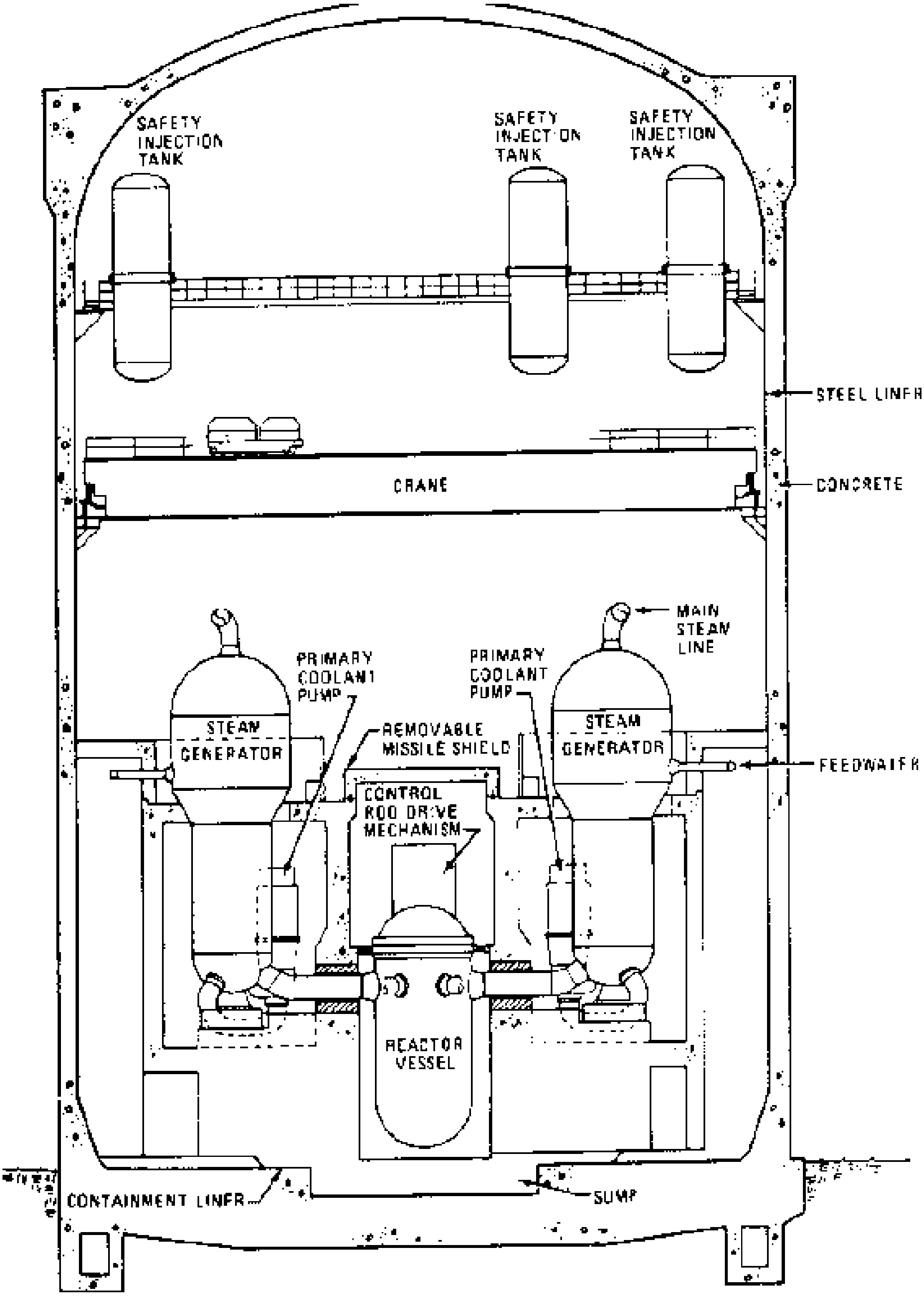
coolant in a loss-of coolant accident (LOCA) and to withstand the impact of internally generated missiles, margins are such that substantially higher pressures can be accommodated before failure. As we shall see later (§12.95), additional challenges are provided by so-called severe accidents in which the core is degraded. Fission product transport into the containment can be modeled
analytically with various uncertainties involved. Therefore, it is useful to view the containment as the essential physical envelope which will indeed prevent the fission products from entering the environment despite the modeling uncertainties that may be present. A

Containments for PWRs are large cylindrical or spherical pressure vessels designed for pressures on the order of 345 kPa(g) (50 psig). Since the PWR containment must be large enough anyway to enclose all of the components of the entire primary system, which may include four steam generators and a pressurizer in addition to the reactor pressure vessel
itself, it is practical to achieve the necessary pressure reduction merely by allowing steam formed by coolant flashing to vent into the large volume. On the other hand, the BWR primary systemis contained essentially within the pressure vessel with much less building volume needed for housing it. Therefore, BWRs use means to condense the released steam in pressure suppression pools so that the containment building need not be much larger
than that required to hold the system.

Containment structures for PWRs vary to some extent from plant to plant, but they are commonly cylindrical(roughly 37m diameter) with a domed top (overall height some 61 m). They are usually made of reinforced concrete, about 1.07m thick, with an internal steel liner, roughly 38 mm thick. As shown in Fig. 5.1, the entire primary coolant system is enclosed as well as elevated injection tanks.

A spherical design is shown in Fig. 5.2. Compare with a cylindrical design of equivalent free volume, the spherical configuration provides additional operating floor area and efficient place me of auxiliary and maintenance activities. An in-continment refueling was storage tank (IRWST) shown in the figure provides water for both safety injection and
severe accident core debris cooling. Sphere me vary from about 40 m for a 2600-MW(t)plant to about 60 m for a plant rated at 3800MW(t). Corresponding free volumes are about 57,000and 96,000 m3,respectively.

If there were to be a complete loss of coolant, nearly all the heat content of the coolant and fuel prior to the accident would be released to the containment atmosphere. The volume and strength of the structure are such that it can withstand the maximum containment temperature and pressure that would be expected from the steam produced by the flashing
of all the water in the primary circuit and from the effects of the ECCS. Typically, the calculated maximum pressure would be about 280 kPa (g); the containment structure is thus designed to withstand 310 kPa(g) and is tested at 350 kPa(g). At the design pressure, the leakage rate should not



5.3 Typical PWR Containment Structure

In order to cool the containment atomosphere and reduce the pressure by condensing part of the steam after a loss-of-coolant accident, water would be sprayed through nozzles near the top of the structure. The water, which collects in the containment sump, can be recirculated through the heat exchangers of the residual-heat removal system(§12.29)to provide continuous cooling of the containment atmosphere. 12.49. The containment sprays also serve to remove some of the radio- ---A activity from the atmosphere. Sodium hydroxide or alkaline sodium thiosulfate in the water facilitates the removal of radio iodines which are generally the determining factor in the environmental hazard that would result from a large radioactive release (§12.160 et seq.). In some PWR installations, the radioactivity level in the containment atmosphere would be reduced by using blowers to circulate the air through iodine absorbers and particulate filters.

A special type of PWR containment system makes use of an ice condenser to reduce the interior pressure. Instead of the usual steel-lined concrete structure, the steel liner (or shell) is separated from the outer concrete (or shield) building. The annular space between the liner and the concrete, above the level of the reactor vessel, contains cells filled with refrigerated borated ice. In the event of a loss-of-coolant accident, condensation of the released steam by the ice would limit the pressure in the containment. Consequently, the structure may be designed for a pressure of only 69 kPa(g) and it can have a smaller volume than a conventional PWR containment building. Some fission products would also be removed in the condenser. Furthermore, the borated water formed by the melting ice would collect in the containment sum~ would be available for core cooling.

Boiling-water reactors

Containment structure designs for BWRs have been modified somewhat over the years. However, all designs use p*ressure suppression,* in which the steam released by the flashing of the coolant in a LOCA is forced into a pool of water and condensed. General, designs call for the condensation of 80 to 90 percent of the steam to keep the pressure from exceeding building specifications.

As shown in Fig. 12.5, an inner concrete cylindrical *drywell,* which encloses the reactor vessel and the recirculation system, acts as a pressure envelope and supports an upper refueling pool. In the event of a LOCA, steam passes through horizontal vents near the bottom of the drywell to an annular *suppression pool,* which also acts as a water source for emergency cooling. The building arrangement is shown\in Fig. 12.6. For refueling, the center portion of the pool is drained so that the reactor vessel head will be accessible.

As a result of the action of the suppression pool, the system can be designed for modest pressures, about 175kPa(g) for the drywell, and 100kPa(g) for the suppression chamber. Leakage must be limited to 0.1 percent per day. Typical volumes for the drywell and pool are 8000 m3 and 4300m^3 respectively. The suppression chamber volume is about 33,000 m3.

A containment building for an advanced large BWR (§13.48) being marketed in the 1990sis shown in Fig. 12.7.The compartmentalized pool system shown in Fig. 12.5 is retained. In all BWRs, the suppression pool removes part of the radioisotopes that may be present in the blow down, particularly radioiodine. Also, the suppression chamber is provided with various absorbers and particulate filters to remove isotopes that may be released from the pool.