CHAPTER

NINE

PRINCIPLES OF NUCLEAR ENERGY

9-1 INTRODUCTION

There is strategic as well as economic necessity for nuclear power in the United States and indeed most of the world. The strategic importance lies primarily in the fact that one large nuclear powerplant saves more than 50,000 barrels of oil per day. At $30 to $40 per barrel (1982), such a powerplant would pay for its capital cost in a few short years. For those countries that now rely on but do not have oil, or must reduce the importation of foreign oil, these strategic and economic advantages are obvious. For those countries that are oil exporters, nuclear power represents an insurance against the day when oil is depleted. A modest start now will assure that they would not be left behind when the time comes to have to use nuclear technology.

The unit costs per kilowatthour for nuclear energy are now comparable to or lower than the unit costs for coal in most parts of the world. Other advantages are the lack of environmental problems that are associated with coal- or oil-fired powerplants and the near absence of issues of mine safety, labor problems, and transportation bottlenecks. Natural gas is a good, relatively clean-burning fuel, but it has some availability problems in many countries and should, in any case, be conserved for small-scale industrial and domestic uses. Thus nuclear power is bound to become the social choice relative to other societal risks and overall health and safety risks.

Other sources include hydroelectric generation, which is nearly fully developed with only a few sites left around the world with significant hydroelectric potential. Solar power, although useful in outer space and domestic space and water heating in some parts of the world, is not and will not become an economic primary source of electric power.

Yet the nuclear industry is facing many difficulties, particularly in the United States, primarily as a result of the negative impact of the issues of nuclear safety, waste disposal, weapons proliferation, and economics on the public and government. The impact on the public is complicated by delays in licensing proceedings, court, and ballot box challenges. These posed severe obstacles to electric utilities planning nuclear powerplants, the result being scheduling problems, escalating and unpredictable costs, and economic risks even before a construction permit is issued. Utilities had to delay or cancel nuclear projects so that in the early 1980s there was a de facto moratorium on new nuclear plant commitments in the United States.

It is, however, the opinion of many, including this author, that despite these difficulties the future of large electric-energy generation includes nuclear energy as a primary, if not the main, source. The signs are already evident in many European and Asian countries such as France, the United Kingdom, Japan, and the U.S.S.R.

In a powerplant technology course, it is therefore necessary to study nuclear energy systems. We shall begin in this chapter by covering the energy-generation processes in nuclear reactors by starting with the structure of the atom and its nucleus and the reactions that give rise to such energy generation. These include fission, fusion, and different types of neutron-nucleus interactions and radioactivity. The following two chapters will cover the two main classes of nuclear powerplants in use or under active development in the world today, the thermal and the fast-breeder reactor powerplants.

9-2 THE ATOMIC STRUCTURE

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 indentity in chemical reactions. In 1811 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 in magnitude but opposite in sign to that on the electron. The atom as a whole is electrically neutral: the 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

* Much earlier, in the fifth century B.C., the Greek Democritus declared that the simplest thing out of which everything is made is an atom. (The Greek word *atomos* means "uncut".)

$$\Delta E = \frac{1}{g_c} \Delta m c^2 \tag{9-1}$$

where c is the speed of light in vacuum and g_c is the familiar engineering conversion factor (Sec. 9-4). Equation (9-1) 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 9-1 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

$$\text{Neutron mass } m_n = 1.008665 \text{ amu}$$

$$\text{Proton mass } m_p = 1.007277 \text{ amu}$$

$$\text{Electron mass } m_e = 0.0005486 \text{ amu}$$

The abbreviation amu, for *atomic mass unit*, is a unit of mass approximately equal to 1.66×10^{-27} kg, or 3.66×10^{-27} lb_m. These three particles are the primary building blocks of all atoms. Atoms differ in their mass because they contain varying numbers of them.

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

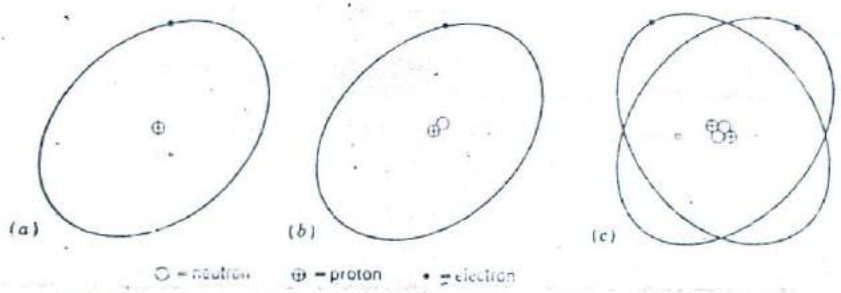


Figure 9-1 Structure of some light atoms: (*a*) hydrogen; (*b*) deuterium or heavy hydrogen, and (*c*) helium.

neutron or a proton is nearly 1 amu, A is the integer nearest the mass of the nucleus, 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 $_1H^1$, deuterium is $_1H^2$ (and sometimes D), and ordinary helium is $_2He^3$. For particles containing no protons, the subscript indicates the magnitude and sign of the electric charge. Thus an electron is $_{-1}e^0$ (sometimes e^- or β^-) and a neutron is $_0n^1$. Symbols are also often written in the form He-4, helium-4, etc. Another system of notation, written as $^{A}_{Z}X$, will not be used in this text.

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.

The known elements, their chemical symbols, and their atomic numbers are listed alphabetically in App. J. Figure 9-2 shows, schematically, the structure of H^1, He^3, and some heavier atoms and the distribution of their electrons in various orbits.

Two other particles of importance are the positron and the neutrino. The *positron* is a positively charged electron having the symbols $_{+1}e^0$, 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 the 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 ν.

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. A discussion of these and other subparticles is, however, beyond the scope of this book.

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.

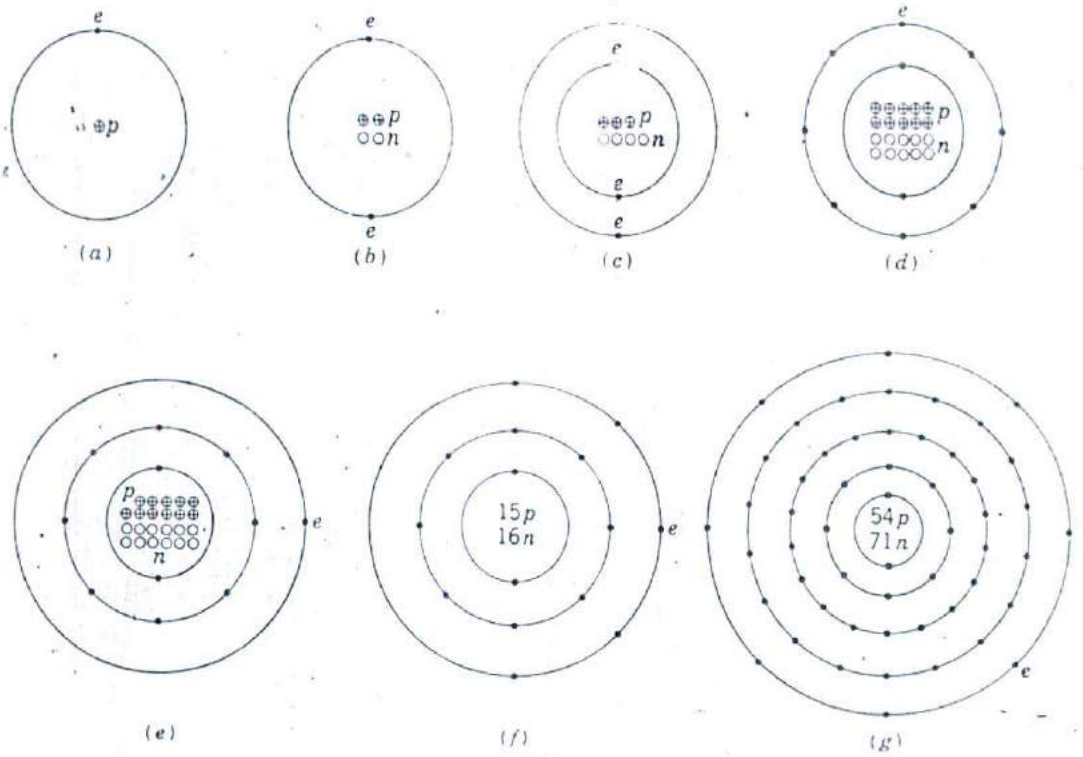


Figure 9-2 Structure of some atoms (orbit radii not to scale). (*a*) hydrogen ($Z = 1$, $A = 1$); (*b*) helium ($Z = 2$, $A = 4$); (*c*) lithium ($Z = 3$, $A = 7$); (*d*) neon ($Z = 10$, $A = 20$); (*e*) sodium ($Z = 11$, $A = 23$); (*f*) phosphorous ($Z = 15$, $A = 31$); and (*g*) xenon ($Z = 54$, $A = 125$).

9-3 CHEMICAL AND NUCLEAR EQUATIONS

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

$$C + O_2 \rightarrow CO_2 \tag{9-2}$$

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. $1\ eV = 1.6021 \times 10^{-19}$ joules (J) $= 1.519 \times 10^{-22}$ Btu $= 4.44 \times 10^{-26}$ kWh. 1 *million electron volts* (1 MeV) $= 10^6$ 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 (UO_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

$$UO_2 + 4HF \rightarrow 2H_2O + UF_4 \tag{9-3}$$

Water vapor is driven off and UF_4 is used to prepare gaseous uranium hexafloride (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, F^{19}, 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. (9-1), 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

$$_{Z_1}K^{A_1} + {}_{Z_2}L^{A_2} \rightarrow {}_{Z_3}M^{A_3} + {}_{Z_4}N^{A_4} \tag{9-4}$$

To balance Eq. (9-4), the following relationships must be satisfied.

$$Z_1 + Z_2 = Z_3 + Z_4 \tag{9-5a}$$

and

$$A_1 + A_2 = A_3 + A_4 \tag{9-5b}$$

Sometimes the symbols γ or ν 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 9-1 One exothermic reaction occurs when common aluminum is bombarded with high-energy α 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

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

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

$$_{13}Al^{27} + {}_{2}He^{4} \rightarrow {}_{14}Si^{30} + {}_{1}H^{1} \tag{9-6}$$

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 $\Delta m = 30.98159 - 30.98143 = -0.00254$ 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

$$_{7}N^{14} + {}_{2}He^{4} \rightarrow {}_{8}O^{17} + {}_{1}H^{1} \tag{9-7}$$

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 α 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 α 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 γ 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 Z orbital electrons. For example, the mass of the Al^{27} nucleus = 26.98153 − 13 × 0.0005486 = 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. (9-6), the energy produced corresponds to the change in masses of the nuclei as given by

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

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

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

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

$$\Delta m = \Sigma M_{\text{products}} - \Sigma M_{\text{reactants}} \tag{9-8}$$

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

$${}_{16}S^{35} \rightarrow {}_{17}Cl^{35} + {}_{-1}e^{0} \tag{9-9}$$

In this case

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

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

$${}_{6}C^{11} \rightarrow {}_{5}B^{11} + {}_{+1}e^{0} \tag{9-10}$$

In this case

$$\Delta m = [(M_B - 5m_e) + m_e] - (M_C - 6m_e) = M_B - M_C + 2m_e$$

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

9-4 ENERGY FROM NUCLEAR REACTIONS

The energy corresponding to the change in mass in a nuclear reaction is calculated from Einsteins law, Eq. (9-1), here repeated

$$\Delta E = \frac{1}{g_c} \Delta m c^2 \tag{9-1}$$

where g_c is a conversion factor* that has the following values

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

Thus if Δm is in kilograms and c in meters per second, ΔE will be in joules. Since $c = 3 \times 10^8$ m/s, Eq. (9-1) can be written in the form

$$\Delta E \text{ (in J)} = 9 \times 10^{16} \Delta m \text{ (in kg)} \tag{9-11}$$

But as it is convenient to express the masses of nuclei in amu $= 1.66 \times 10^{-27}$ kg and the energy in joules (J) and MeV, Eq. (9-11) becomes

$$\Delta E \text{ (in J)} = 1.49 \times 10^{-10} \Delta m \text{ (in amu)} \tag{9-12}$$

and

$$\Delta E \text{ (in MeV)} = 931 \Delta m \text{ (in amu)} \tag{9-13}$$

a useful relationship to remember. The reaction in Example 9-1 thus produces $-0.00254 \times 931 = -2.365$ MeV of energy. Mass-energy conversion factors are given in Table 9-1.

9-5 NUCLEAR FUSION AND FISSION

Nuclear reactions of importance in energy production are *fusion*, *fission*, and *radioactivity*. Radioactivity will be discussed in Sec. 9-8. In *fusion*, 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.

* The same as in force $= 1/g_c \times$ mass $\times$ acceleration.

Table 9-1 Mass-energy conversion factors

	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^{-19}
kg	5.6094×10^{29}	8.9873×10^{16}	8.5184×10^{13}	2.4965×10^{10}	5.9916×10^{8}
lb_m	2.5444×10^{29}	4.0766×10^{16}	3.8639×10^{13}	1.1324×10^{10}	2.7177×10^{8}

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

$$4\,{}_1H^1 \rightarrow {}_2He^4 + 2\,{}_{+1}e^0 \tag{9-14}$$

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 [3,83]. Table 9-2 lists the possible fusion reactions and the energies produced

Table 9-2

	Fusion reaction		
Number	Reactants	Products	Energy per reaction, MeV
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

by them. n, p, D, and T are the symbols for the neutron, proton, deuterium (H^2), and tritium (H^3), respectively.

Many problems have to be solved before an artificially made fusion reactor becomes a reality [3,83]. 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. Fusion powerplants will not be covered in this text.

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. 9-3, is

$$_{92}U^{235} + {}_0n^1 \rightarrow {}_{54}Xe^{140} + {}_{38}Sr^{94} + 2{}_0n^1 \qquad (9\text{-}15)$$

The immediate (prompt) products of a fission reaction, such as Xe^{140} and Sr^{94} above, are called *fission fragments*. They, and their decay products (Sec. 9-7), are called *fission products*. Figure 9-4 shows fission product data for U^{235} by thermal and fast neutrons (Secs. 9-10 and 9-11) and for U^{233} and Pu^{239} by thermal neutrons [84]. The products are represented by their mass numbers.

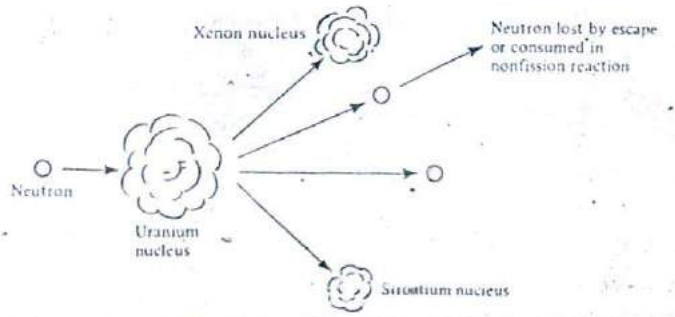


Figure 9-3 A typical fission reaction.

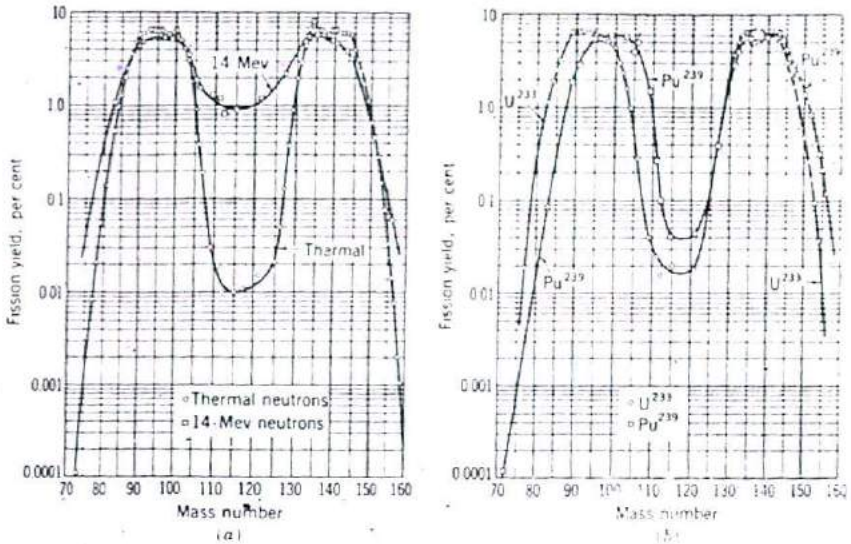


Figure 9-4 Fission product yield data for (*a*) U^{235} by thermal and 14 MeV neutrons and (*b*) U^{233} and PU^{239} by thermal neutrons [84]

9-6 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

$$_{92}U^{235} + {}_0n^1 \rightarrow {}_{56}Ba^{137} + {}_{36}Kr^{97} + 2{}_0n^1 \tag{9-16}$$

has the mass balance

$$235.0439 + 1.00867 \rightarrow 136.9061 + 96.9212 + 2 \times 1.0086^{7}$$

$$236.0526 \rightarrow 235.8446$$

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

Thus

$$\Delta E = 931 \times -0.2080 = -193.6 \text{ MeV} = -3.1 \times 10^{-11} \text{ J}$$
$$= -2.937 \times 10^{-11} \text{ Btu}$$

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 nonfission capture of excess neutrons in reactions that produce energy, 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 U^{235} nuclei thus produces

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

$$= 0.513 \times 10^{24} \text{ MeV}$$

$$= 2.276 \times 10^{24} \text{ kWh}$$

$$= 8.190 \times 10^{10} \text{ J}$$

$$= 0.948 \text{ MW-day}$$

Another convenient figure to remember is that a reactor burning 1 g 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 nonfissionable 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. (9-15) and (9-16)] in its structure except for limited periods of time—burnup 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} (Sec. 11-2). 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.

9-7 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 9-5 shows a *Z-N* chart of the known isotopes.

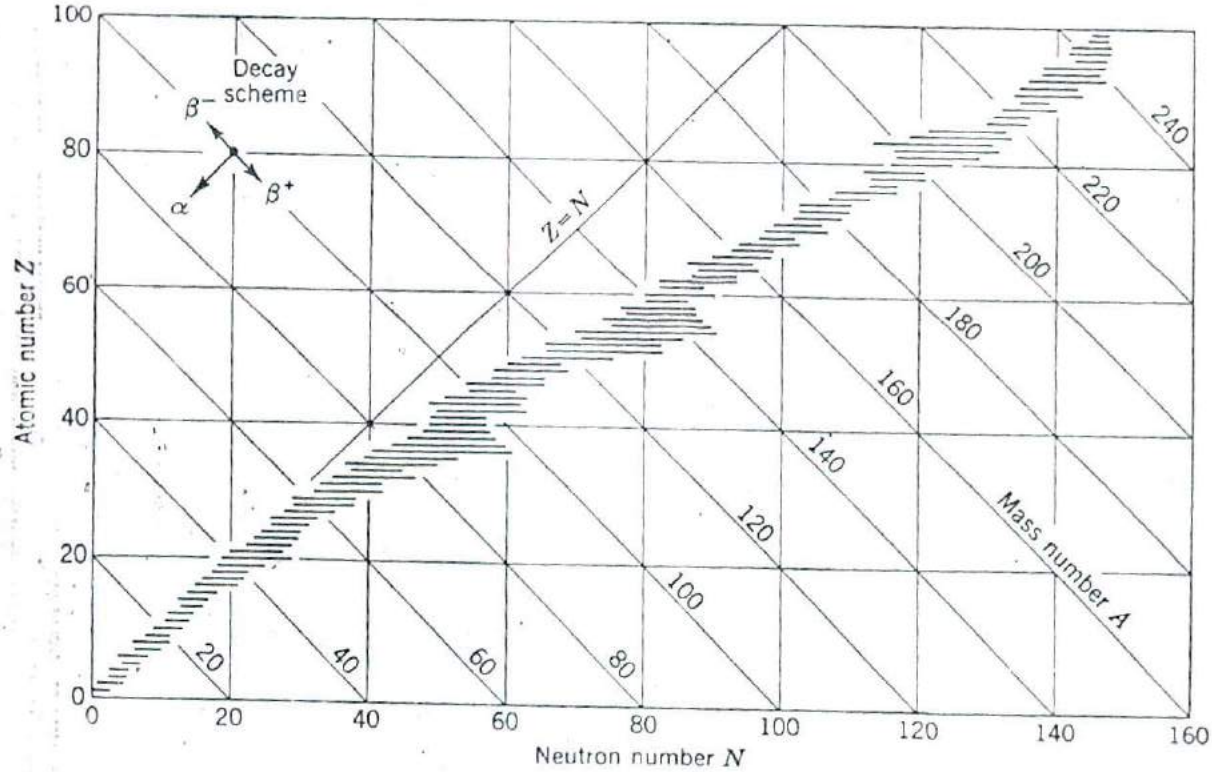


Figure 9.5 Z-N chart of the unknown isotopes.

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 may or may not be stable, and several successive decays may occur until a stable isotope is formed. An example of radioactivity is

$$_{49}In^{115} \rightarrow {}_{50}Sn^{115} + {}_{-1}e^{0} \qquad (9\text{-}17)$$

where the parent, In^{115}, is a naturally occurring radioisotope and its daughter, Sn^{115}, 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}

$$_{94}Pu^{239} \rightarrow {}_{92}U^{235} + {}_{2}He^{4} \qquad (9\text{-}18)$$

Beta decay An example of β decay, besides Eq. (9-17), is

$$_{82}Pb^{214} \rightarrow {}_{83}Bi^{214} + {}_{-1}e^{0} + \nu \qquad (9\text{-}19)$$

where ν, 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

$$_{7}Ni^{13} \rightarrow {}_{6}C^{13} + {}_{+1}e^{0} \qquad (9\text{-}20)$$

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

$$_{+1}e^{0} + {}_{-1}e^{0} \rightarrow 2\gamma \tag{9-21}$$

The two particles therefore undergo an *annihilation* process, which produces γ energy equivalent to the sum of their rest masses $2m_e c^2$, or $-(2 \times 0.0005486)931 = -1.02$ MeV.

The reverse of the annihilation process is called *pair production*. In this, a γ 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 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

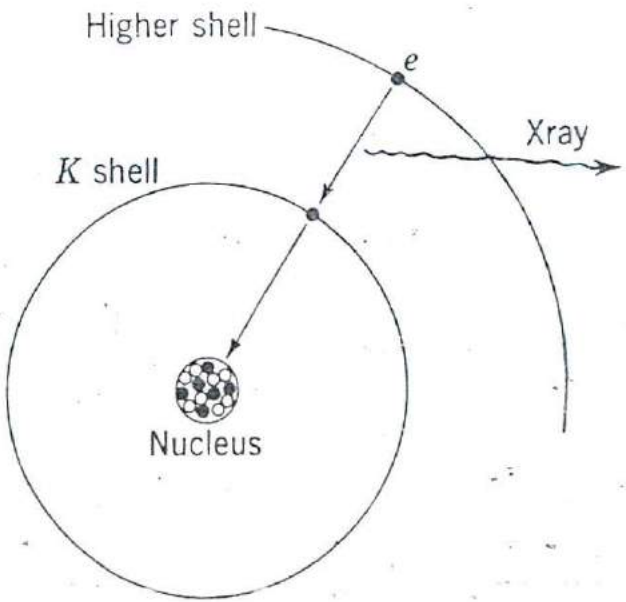


Figure 9-6 K capture.

$$_{29}Cu^{64} + {}_{-1}e^{0} \rightarrow {}_{28}Ni^{64} \tag{9-22}$$

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

$$_{54}Xe^{137} \rightarrow {}_{54}Xe^{136} + {}_{0}n^{1} \tag{9-23}$$

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.

9-8 DECAY RATES AND HALF-LIVES

There can be no indication of the time that it takes any one particular radioactive nucleus to decay. However, if there is 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 10^{20} and the other 10^{30} of the same radioisotopes, we will find that the same fraction in each, say one-half or $10^{20}/2$ and $10^{30}/2$, 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 $d\theta$, at θ, the rate of decay $-dN/d\theta$ is directly proportional to N.

$$-\frac{dN}{d\theta} = \lambda N \tag{9-24}$$

λ 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, $\theta = 0$, when the number of radioisotopes was N_0, gives

$$-\int_{N_0}^{N} \frac{dN}{N} = \lambda \int_{0}^{\theta} d\theta$$

Thus

$$N = N_0 e^{-\lambda\theta} \tag{9-25}$$

The rate of decay λN is also called the *activity* A and commonly has the dimension disintegrations per second (dis/s) or s^{-1}. The initial activity A_0 is equal to λN_0. Thus

$$A = A_0 e^{-\lambda\theta} \tag{9-26}$$

A common way of representing decay rates is by the use of the *half-life*, $\theta_{1/2}$. This is the time during which one-half of a number of radioactive species decays or one-half of their activity ceases. Thus

$$\frac{N}{N_0} = \frac{A}{A_0} = \frac{1}{2} = e^{-\lambda\theta_{1/2}}$$

and

$$\theta_{1/2} = \frac{\ln 2}{\lambda} = \frac{0.6931}{\lambda} \tag{9-27}$$

and the half-life is inversely proportional to the decay constant. Starting at $\theta = 0$ when $N = N_0$, one-half of N_0 decay after one half-life; one-half of the remaining atoms, or one-quarter of N_0, decay during the second half-life; one-eighth of N_0 during the third, and so on (Table 9-3 and Fig. 9-7). 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 9-3 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

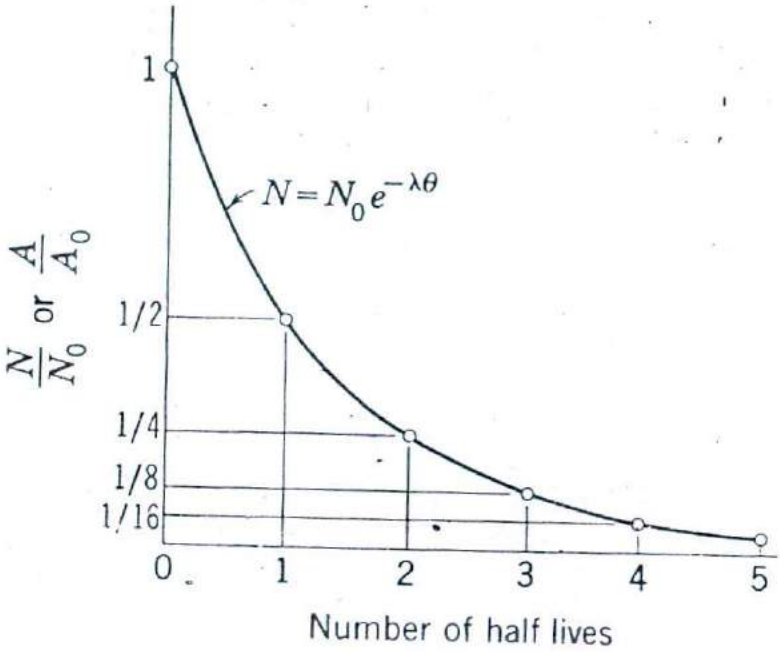


Figure 9-7 Radioactive-decay rates as a function of half-life.

the change in activity with time and computing λ from the slope of the activity history on a semilog plot (Fig. 9-8), from which $\theta_{1/2}$ and 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.

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.

Example 9-2 Radium 226 decays into radon gas. Compute (1) the decay constant and (2) the initial activity of 1 g of radium 226. The atomic mass is 226.0245 amu.

SOLUTION

(1) Half-life of Ra^{226} = 1600 yr = 5.049×10^{10} s (from Table 9-4)

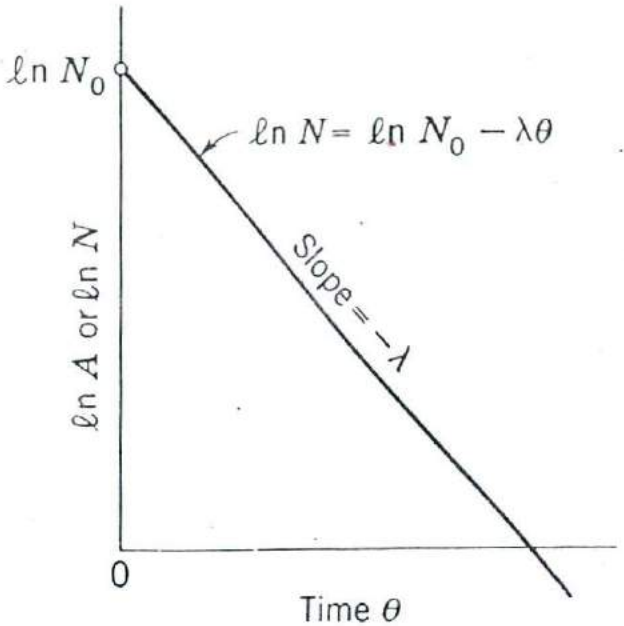


Figure 9-8 Radioactive-decay curve on a semilog plot.

Table 9-4 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

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

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

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

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

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

Thus the activity of 1 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 3.7×10^{10} dis/s instead of the more correct value above. 3.7×10^{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.

9-9 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

$$KE_n = \frac{1}{2g_c} m_n V^2 \tag{9-28}$$

where m_n = mass of neutron

V = speed of neutron

g_c = conversion factor (the same as in Sec. 9-6)

The term *kinetic* is occasionally dropped and the symbol KE_n simplified to E_n, so that neutron energy means neutron kinetic energy. E_n is commonly expressed in eV or MeV. Since $m_n = 1.008665$ amu, then

$$E_n = \frac{1}{2 \times 0.965 \times 10^{18}} \times 1.008665 V^2$$

or

$$E_n = 5.227 \times 10^{-19} V^2\ \text{MeV} = 5.227 \times 10^{-13} V^2\ \text{eV} \tag{9-29}$$

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 10^5 eV), *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* (Sec. 9-10).

Newly born fission neutrons carry, on an average, about 2 percent of a reactor

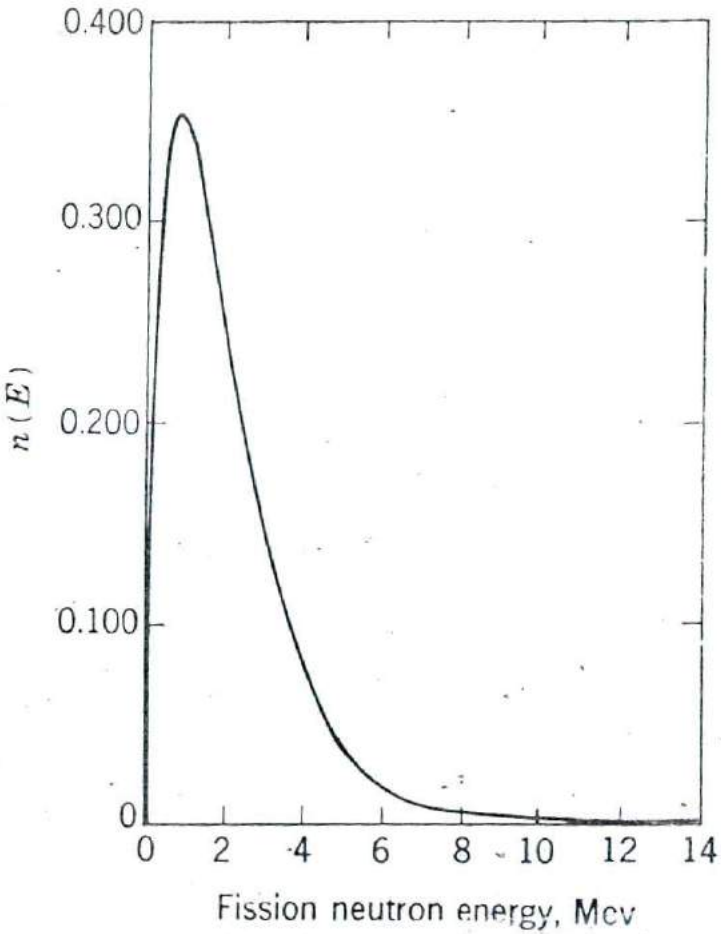


Figure 9-9 Energy spectrum of fission neutrons.

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 (Sec. 9-7), 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. 9-9 and given (for U^{235} and Pu^{239} fission) by [85]

$$n(E)\,dE_n = \sqrt{\frac{2}{\pi e}} \sinh \sqrt{2E_n}\, e^{-E_n}\, dE_n \tag{9-30}$$

where $n(E)$ is the number of neutrons having energy E_n per unit energy interval dE_n. Most of the prompt fission neutrons have energies less than 1 MeV but average around 2 MeV.

9-10 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 H_2O and D_2O), 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. 9-10*a*). The velocity distribution, shown for two temperatures in Fig. 9-10*b*, is expressed by the *Maxwell distribution law*

$$n(V)\,dV = 4\pi n \left(\frac{m}{g_c 2\pi kT}\right)^{1.5} V^2 e^{-g_c(mV^2/2kT)}\, dV \tag{9-31}$$

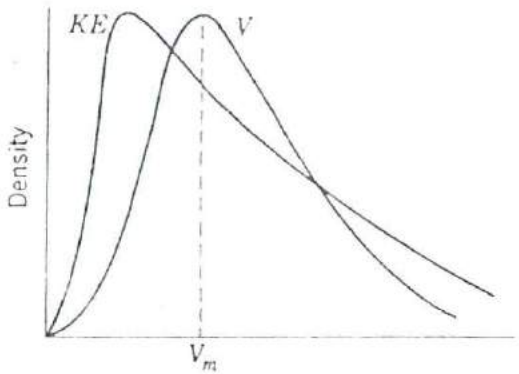
where $n(V)$ = number density of particles, present in given volume of medium, per unit velocity interval dV between V and $V + dV$

n = total number of particles in same volume of medium

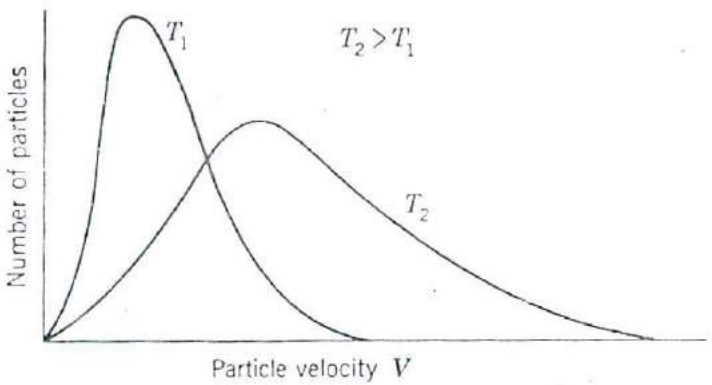
m = mass of particle

k = Boltzmann's constant (universal gas constant divided by Avogadro's number) = 1.3805×10^{-23} J/K, or 8.617×10^{-11} MeV/K)

T = absolute temperature



(a)



(b)

Figure 9-10 (*a*) Kinetic energy and velocity distributions of thermal neutrons at a given temperature. (*b*) Velocity distribution at two temperatures.

The *most probable speed* V_m 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

$$V_m = \left(\frac{g_c 2kT}{m}\right)^{0.5} \tag{9-32}$$

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

$$E_m = \frac{1}{2g_c} m V_m^2 = kT \tag{9-33}$$

Table 9-5 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

$$V_m \text{ (in m/s)} = 128.39T^{0.5} \qquad \text{(for a neutron only)} \qquad (9\text{-}34)$$

and

$$E_m \text{ (in eV)} = 8.617 \times 10^{-5}T \qquad \text{(for any particle)} \qquad (9\text{-}35)$$

where T is in kelvins. 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.

9-11 NUCLEAR CROSS SECTIONS

Assume a beam of monoenergetic neutrons of intensity I_0 neutrons/s impinging on a body having a target area A cm^2 and a nuclear density N nuclei/cm^3 (Fig. 9-11). 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, which is given by [86]

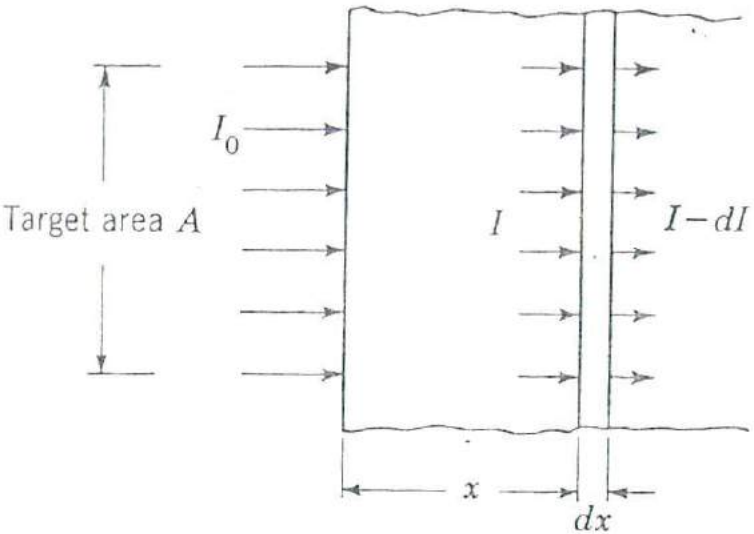


Figure 9-11 Neutron beam striking target area A.

$$r_t = r_0 A^{1/3} \tag{9-36}$$

where r_0 is a constant varying for different nuclei with an average of 1.4×10^{-13} 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. 9-11 the number of nuclei in volume $A\,\Delta x$ is $(NA\,\Delta x)$. As the neutron beam passes through Δx, 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, $\sigma(NA\,\Delta x)$, to the total area A. Thus, if at x and $x + \Delta x$ the beam intensities become I and $I - \Delta I$, respectively, it follows that, in the limit

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

Integrating

$$-\int_{I_0}^{I} \frac{dI}{I} = \sigma N \int_0^x dx$$

from which

$$I = I_0 e^{-\sigma N x} \tag{9-37a}$$

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 some nuclei of interest are shown in Figs. 9-12 through 9-14 [87].

The product σN 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 Σ. It has the unit of length^{-1}, commonly cm^{-1}. Thus

$$\Sigma = N\sigma \tag{9-38a}$$

and Eq. (9-37*a*) can be written in the form

$$I = I_0 e^{-\Sigma x} \tag{9-37b}$$

Macroscopic cross sections are also designated according to the reaction they represent. Thus $\Sigma_f = N\sigma_f$, $\Sigma_s = N\sigma_s$, etc. The reciprocal of macroscopic cross section for any reaction is the *mean free path* for that reaction. It has the symbol λ, not to be confused with the decay constant in radioactivity (Sec. 9-8).

For an element of atomic mass A_t and density ρ (g/cm^3), N (nuclei/cm^3) can be calculated from

$$N = \rho \frac{\text{Avogadro's number}}{A_t} \tag{9-39}$$

9-12 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/

* 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.

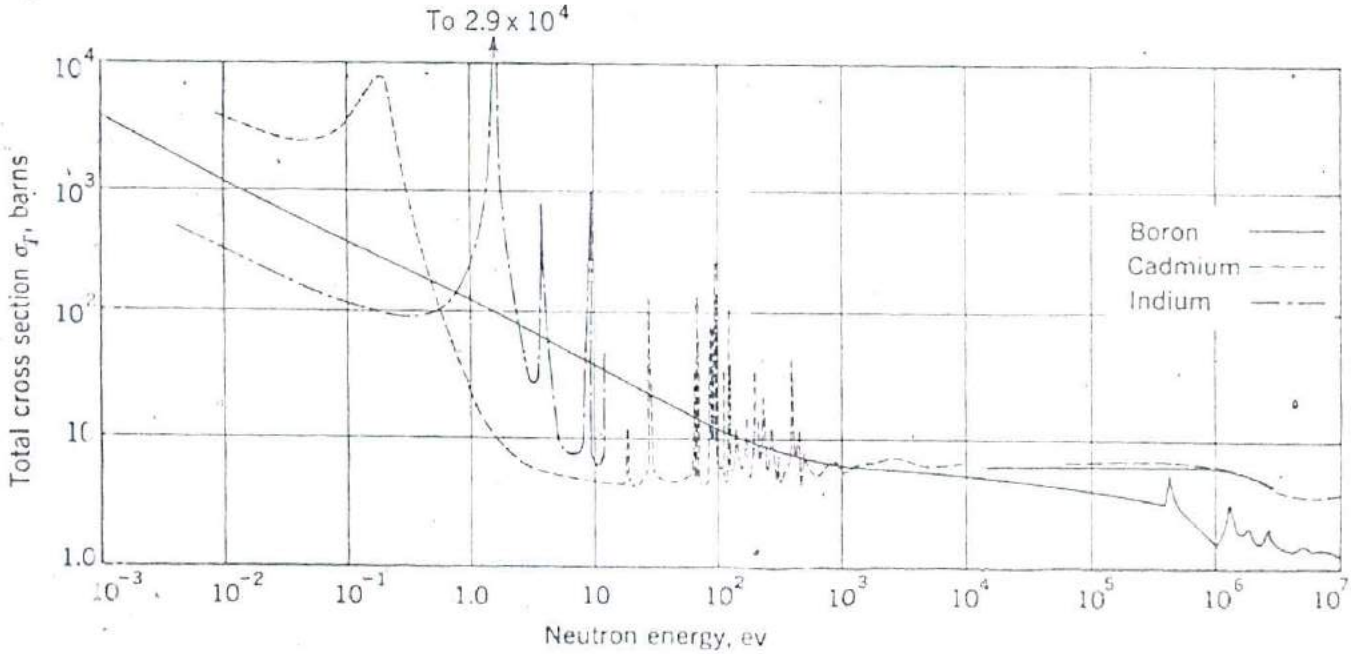


Figure 9-12 Neutron cross sections for cadmium, indium, and boron

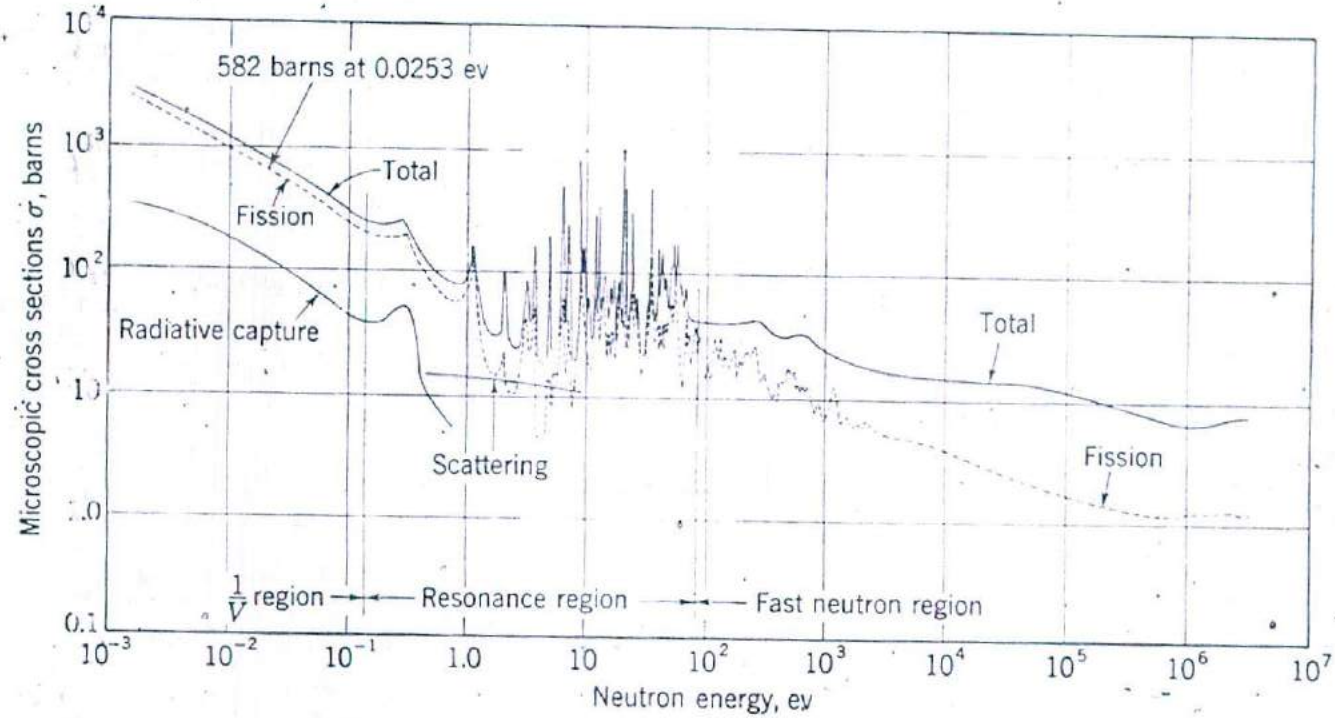


Figure 9-13 Neutron cross sections for U^{235}.

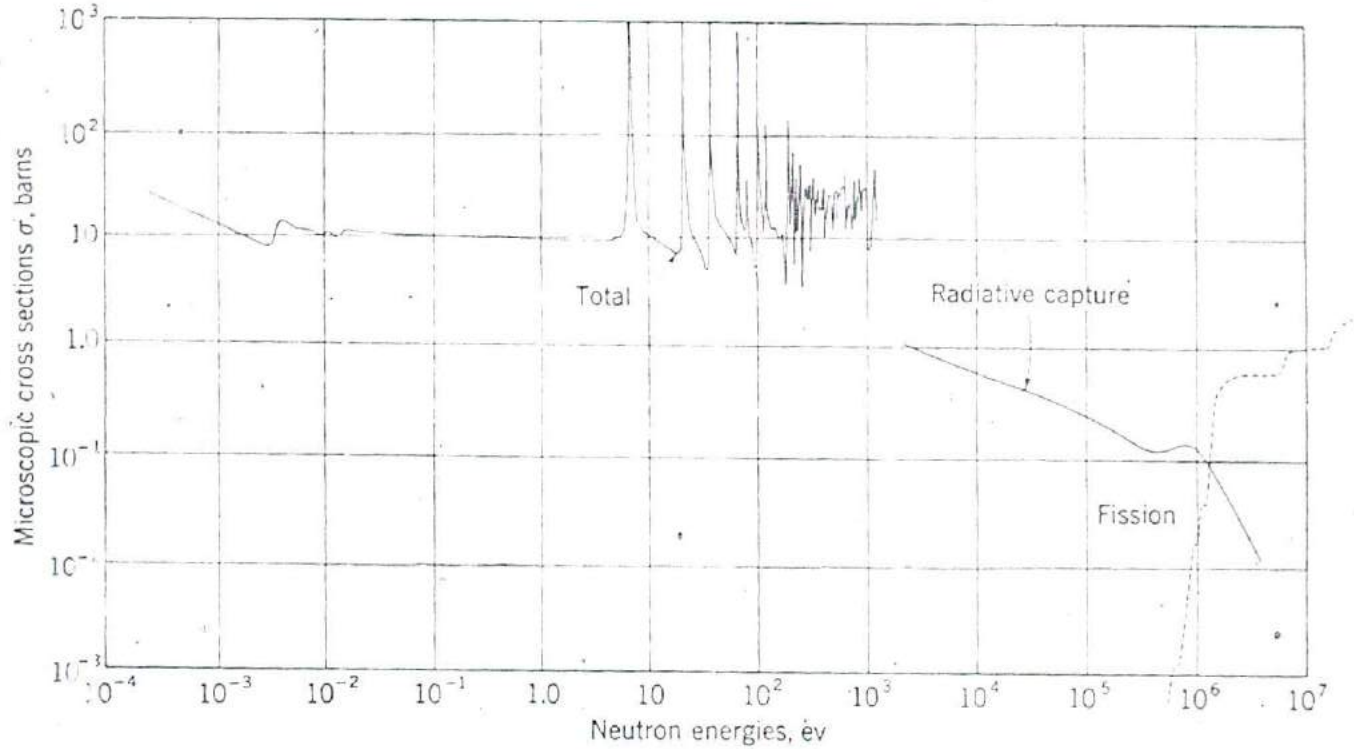


Figure 9-14 Neutron cross sections for U^{235}.

cm^3) and V the neutron velocity (cm/s), the product nV is the number of neutrons crossing a unit area from all directions per unit time and is called the *neutron flux* ϕ. Thus

$$\phi = nV \tag{9-40}$$

ϕ has the unit neutrons/(s · cm^2), 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. 9-15). 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

$$\text{Reaction rate} = nV N\sigma = \phi\Sigma \text{ reactions/(s} \cdot \text{cm}^3) \tag{9-41}$$

where σ and Σ are the cross sections of the particular reaction in question (i.e., absorption, scatter, etc.). Equation (9-41) 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 σ) 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

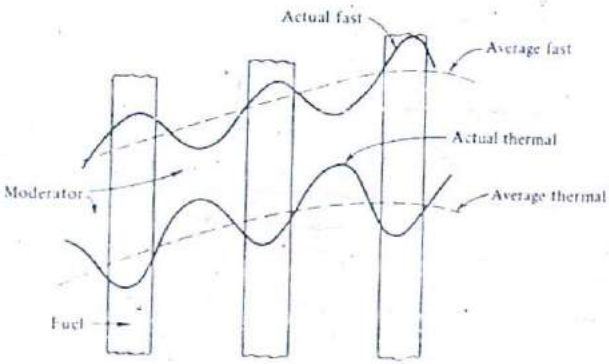


Figure 9-15 Neutron-flux distributions in fuel and moderator in a heterogeneous thermal reactor.

neutron-flux distribution in a reactor core is therefore necessary for the study of heat generation and removal in that core [2].

9-13 THE VARIATION OF NEUTRON CROSS SECTIONS WITH NEUTRON ENERGY

$\sigma - E_n$ 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 *1/V region*, (2) the *resonance region*, and (3) the *fast-neutron region*.

1/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_n

$$\sigma_a = C_1\left(\frac{1}{E_n}\right)^{0.5} \tag{9-42a}$$

Thus

$$\sigma_a = C_1\left[\frac{1}{m_n(V^2/2g_c)}\right]^{0.5} = C_2\frac{1}{V} \tag{9-42b}$$

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/V 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 1/V law may also be written in the form

$$\frac{\sigma_{a,1}}{\sigma_{a,2}} = \frac{V_2}{V_1} = \left(\frac{E_{n,2}}{E_{n,1}}\right)^{0.5} \tag{9-42c}$$

where the subscripts 1 and 2 refer to two different neutron energies within the 1/V range. Absorption cross sections for monoenergetic neutrons, within the 1/V region, may thus be calculated at any energy from tabulated values at 2200 m/s.

On the log-log plots of Figs. 9-12 to 9-14, the 1/V region is a straight line with a slope of -0.5. The upper limit of the 1/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 1/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^{233} and U^{235} 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, $2\pi r_1^2$. Combining with Eq.(9-36)

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

Using $r_0 = 1.40 \times 10^{-13}$ cm and 1 barn $= 10^{-24}$ cm^2

$$\sigma_t = 0.125A^{2/3} \text{ barns} \tag{9-43}$$

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. 9-12), but the peaks are rather low, and the phenomenon is of little importance.

9-14 FISSION REACTOR TYPES

With the information on nuclear-fuel cross sections and their variation with neutron energies in hand, we are now able to put together a qualitative picture of the effects of fuel enrichment on reactor core design and configuration.

We already know that in fission one neutron is absorbed by the fissioning nucleus but between two and three fast fission neutrons are produced. The exact number is given the symbol ν, given in Table 9-6 for the three fissionable nuclei.

In order for a reactor using U^{235} as fuel to operate at a steady rate, no more than 1.47 neutrons should be lost to the fission process. Such losses occur in two ways: (1) nonfission absorption in reactor-core materials, which include structural materials, coolant channel walls, fuel cladding, coolant, moderator, and the fuel material itself;

Table 9-6 Fuel constants

Nucleus	Number of fission neutrons per fission, ν	Number of fission neutrons per thermal neutron absorbed, η
U^{233}	2.51 ± 0.03	2.29 ± 0.02
U^{235}	2.47 ± 0.03	2.07 ± 0.03
Pu^{239}	2.90 ± 0.04	2.10 ± 0.02

and (2) leakage of the core, a function of both core materials and size. More leakage occurs if the neutron mean free path in the core materials is large and if the reactor size is small, i.e., its surface-volume ratio is large. For the purpose of discussion, we will fix the fractions of neutrons leaking and engaging in nonfission absorption in reactor materials other than the fuel itself. The only variables, then, will be nonfission and fission absorptions in the fuel.

The simplest fuel to use in a reactor is natural uranium, composed of about 0.7 percent U^{235}, about 99.3 percent U^{238}, and a trace of U^{234}. Let us now try to build a reactor core from a solid mass of this relatively cheap and plentiful fuel with possible small holes for cooling (Fig. 9-16). Let us start with 110 newly born fast neutrons within this fuel and assume that 10 will leak out of the core during their lifetime. The remaining 100 neutrons will be subjected only to scatter, radiative capture, and fission in the fuel. Because the cross-sectional curves of Figs. 9-13 and 9-14 do not contain all reactions, we will use Table 9-7, which contains the average cross sections of the above reactions for U^{235}, U^{238}, and natural uranium.

The average energy of the newly-born fast neutrons is 2 MeV. At this energy the

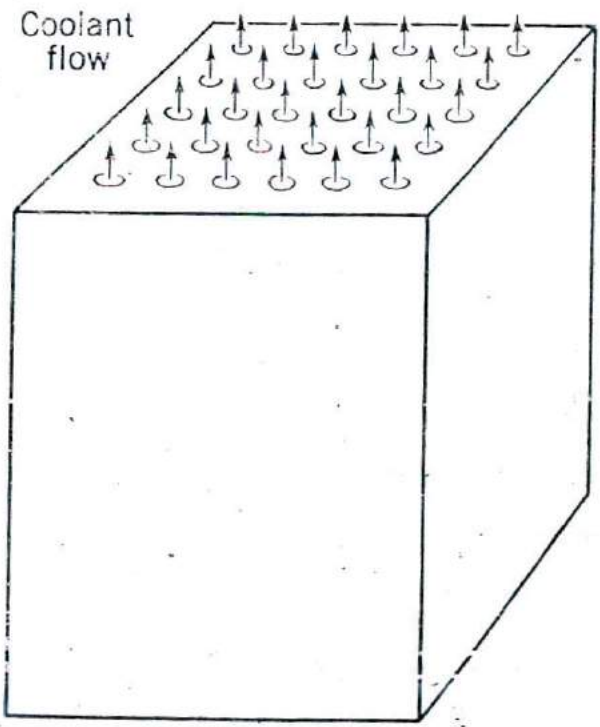


Figure 9-16 A homogeneous mass of natural uranium metal with coolant flow channels.

Table 9-7 Microscopic cross sections for uranium fuels

Neutron	Microscopic cross sections, barns	Nucleus		Natural uranium
		U^{235}	U^{238}	
Fast, 1 MeV	σ_t	5.30	6.6	6.6
	σ_c	0.093	0.14	0.14
	σ_f	1.20	0.018	0.026
Thermal, 0.0253 eV	σ_t	6.00	8.00	8.00
	σ_c	112.00	2.73	3.47
	σ_f	577.10	0	4.16

fission cross section of U^{238} is about 0.53 barn. The majority of these neutrons, however, will possess the most probable energy, just below 1.0 MeV (Fig. 9-9), where the fission cross section in natural uranium is so low compared with the scattering cross sections that only very few of the 100 original neutrons will engage in fission, producing 2.47 neutrons each. This production is so far below the original 100 that, alone, it cannot sustain the chain reaction.

The great majority of the neutrons will then be scattered down to lower energies, from the right to the left on the cross-sectional diagrams of Figs. 9-13 and 9-14. They first have to cross the resonance-absorption regions. There, the nonfission resonance cross sections of the abundant U^{238} are so great that, despite the increased fission cross sections of U^{235}, the neutrons are effectively eliminated and virtually none will reach the thermal energy region.

Now, if a moderator is homogeneously mixed with or dispersed throughout the natural-uranium lump, some of the neutrons will be slowed down past the resonance-energy region of U^{238} by the strongly scattering moderator. However, because of the presence of U^{238} throughout the core, a sufficient number of neutrons of resonance-energy range are absorbed in U^{238} that the few neutrons reaching thermal energies will be less than 1/2.47 of the original 100 and the new fission neutrons will be less than 100. This causes the reaction to die down rapidly.

Thus a sustained (critical) chain reaction is impossible in a mass of natural uranium or in a homogeneous mixture of natural uranium and moderator. Actually, it is possible to store natural uranium plates in contact with each other to any desired height without fear of a critical reaction.

In order to obtain criticality or steady power, three methods are used: (1) building a *heterogeneous reactor*, (2) *enriching the fuel*, or (3) both.

If natural uranium or, in general, low-enriched fuels are to be used, the fuel must be subdivided into separate fuel elements in the form of pins, rods, plates, hollow cylinders, pellets, spheres, etc. These are placed in the core, with space between them filled with a moderator (Fig. 9-17). This is a heterogeneous reactor core. Because the fuel elements are relatively thin, a newly born fission neutron, even near the center of the elements, has a good chance of escape before attaining resonance energies because of the low fuel-scatter cross sections and the short distance it has to travel to

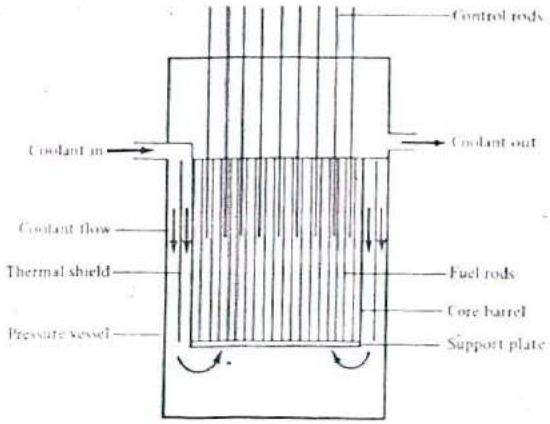


Figure 9-17 Schematic of a heterogeneous reactor.

get out of the element—a distance that is short compared with the neutron mean free path for scatter in the fuel. Thus this neutron spends the resonance-energy period outside the fuel element and escapes resonance capture. Once in the moderator, the neutron becomes thermalized in a few collisions. When it reenters the fuel, at thermal energies, the probability of fission by U^{235} far exceeds that of nonfission absorption by U^{235} and U^{238}, and a chain reaction is possible.

Actually, some resonance absorption is unavoidable, but not enough to adversely affect neutron economy. Also some fission of U^{238} by high-energy neutrons occurs in a heterogeneous reactor. Some nonfission absorption by U^{238} results in the production of Pu^{239}, a fissionable isotope, a process similar to that which occurs in fast-breeder reactors (Sec. 11-2).

Because neutron economy is difficult to come by with natural uranium, the moderator has to have a very low neutron-absorption cross section. This eliminates ordinary water and other hydrogenous materials, such as the organics, as moderators with natural uranium because they have relatively high neutron-absorption cross sections and consequently low moderating-to-absorption ratios. All water- and organic-moderated and cooled reactors must use fuels slightly enriched in U^{235}. A heterogeneous natural-uranium reactor can be built, however, with moderators of low neutron-absorption cross sections, such as graphite or heavy water, D_2O. The latter, called the *heavy-water reactor* (HWR), is the basis of the Candu reactor (Sec. 10-14).

The second method of attaining criticality is to enrich the fuel by artificially increasing the percentage of U^{235} (or other fissionable material) in it. In this case, the effect of U^{238} is less pronounced. Low-enriched fuels (of the order of a few percent), however, still have to be of the heterogeneous type for reasons similar to those given for natural uranium. Ordinary water, often incorrectly called *light water*, may be used as a moderator, resulting again in thermal heterogenous reactors.

Light-water reactors are of two main types: (1) the *pressurized-water reactor*

(PWR) (Sec. 10-6) and (2) the *boiling-water reactor* (BWR) (Sec. 10-7). Both constitute the largest number of reactors built during the first few decades of nuclear power.

A third thermal-reactor type is the *gas-cooled reactor* (GCR). The gas coolant is either CO_2, which is a poor moderator despite the presence of C because of its low density, or helium, which has no moderating capabilities. A separate solid moderator such as graphite, Be, or BeO is used, with graphite the usual choice. There are several types of such reactors, including the British Magnox and AGR types [3], the U.S. HTGR (Sec. 10-12), and the German THTR (Sec. 10-13).

Slightly higher enrichments (about 20 percent) allow the thermal homogeneous reactors to be built. The fuel is mixed with the moderator either in liquid form, called a *fluid-fueled reactor*, or solid form, such as uranium-zirconium hydride or UO_2-polyethylene mixtures. None of these have been built as power reactors.

In highly enriched fuels (beyond about 20 percent) the contribution of U^{238} resonance is no longer of prime importance. The contribution of U^{235} to the fission cross sections of the mixture outweighs U^{238} nonfission-capture effects. In this case, no moderator is necessary, and we may have a *fast reactor*, i.e., one relying primarily on high-energy (fast) neutrons for fission. Fast reactors can be homogeneous but are usually heterogeneous.

The fission energy generated in the fuel in all reactor types must be removed by a *coolant*, which leaves the reactor at a higher temperature than when it entered it. The rate of heat removal must be such that the fuel operates within safe temperature or boiling limits [2]. The energy carried out by the coolant is used in a thermodynamic cycle to generate electricity. It is fortunate that in most thermal reactors the coolant, ordinary or heavy water, can double as a moderator. Gas-cooled thermal reactors use either CO_2 or He for cooling and need a separate moderator. Fast reactors use no moderator and need a coolant that does not moderate the neutrons and that has a high heat-transfer coefficient because of the large power densities in fast-reactor cores. Liquid metals, particularly molten sodium (Na) are the most common. Such reactors are called *liquid-metal fast-breeder reactors* (LMFBR) (Secs. 11-3 to 11-5). Gases, particularly helium, have also been considered as fast-reactor coolants. Such reactors are called *gas-cooled fast reactors* (GCFR) (Sec. 11-6).

Table 9-8 contains the most common fission-power reactor types in use commercially, or under serious development, in the world today.

Table 9-8 The most-common fission-power reactor types

Neutron energy	Reactor type	Coolant	Moderator	Fuel, enrichment	Examples
Thermal	PWR	H_2O	H_2O	UO_2, low	Sec. 10-2
	BWR	H_2O	H_2O	UO_2, low	Sec. 10-7
	GCR	CO_2, He	Graphite	UC, low	Sec. 10-11
	HWR	D_2O	D_2O	UO_2, natural	Sec. 10-14
Fast	LMFBR	Na	None	$Pu^{239}O_2 + U^{238}O_2$	Sec. 11-3 to 11-5
	GCFR	He	None	$Pu^{239}O_2 + U^{238}O_2$	Sec. 11-6

9-15 REACTOR CONTROL

There are several methods to control a nuclear reactor, i.e., to start, increase, decrease, and turn off its power. The most common method is the use of *control rods*. These rods may be shaped like the fuel rods themselves and are interspersed throughout the core (Fig. 9-17). Instead of containing fuel, they contain a *neutron absorber*, also called poison, such as boron, cadmium, or indium. They have high neutron-absorption cross sections (Fig. 9-12) and do not contribute to neutron multiplication.

Such control rods are designed with sufficient absorber to change the neutron balance in the core so that less than one neutron is left for fission for each neutron engaging in fission, a situation that would lead to a power decrease and eventual shutdown. The control rods are operated by control-rod drives that can move them in and out of the core around a power equilibrium position which is usually a partially inserted position. The rods are moved out to increase power at a prescribed safe rate or are moved in to decrease power. In either case, when the required power is attained the rods are returned to the equilibrium position.

The rods can also be used to adjust power levels selectively within the core to help even out the radial power distribution. *Fuel zoning*, i.e., the use of variable fuel enrichment within the core (usually three roughly concentric zones are used) is also used to attain the same purpose.

Control rods can be suddenly and completely inserted in the core to shut it down in case of an emergency. Such an operation is called a *scram*. It can be done automatically or manually by a visible colored button on the reactor console in the control room. A number of rods may be built into the system which are usually fully withdrawn from the core during normal operation and which have the sole function of shutdown by becoming fully inserted upon demand.

The equilibrium position varies with the life of the core. As more fuel is depleted, i.e., is fissioned with time, fewer neutrons are produced and more neutrons are absorbed in nonfission reactions. The rods are then slowly moved to less-inserted positions to compensate for this loss of neutrons and to keep the core in equilibrium.

The rods are said to possess a *reactivity worth* that should be adequate to cover operational control during the life of the core, fuel depletion, safety shutdown, and such other effects as Doppler, samarium, and xenon poisoning [2].

Another method of control in pressurized-water reactors (Sec. 10-5), called *chemical shim*, is used in addition to, not in lieu of, control rods. Chemical shim is the use of a soluble absorber, usually boric acid, in the moderator coolant. The concentration of this absorber in the moderator coolant is decreased slowly during the core lifetime to overcome the effect of fuel depletion. The concentrations are sufficient to permit operating the core almost unrodded, i.e., with all control rods nearly fully withdrawn.

It should be mentioned here that boiling-water reactors do not use chemical shim and that the core of a BWR can be in equilibrium at several positions of the control rods because of the strong effect of the steam voids on reactivity (Sec. 10-8).

Another control system is the use of *reflectors*. These are mechanically operated devices, situated just outside the core, that contain material that reflects some of the neutrons escaping the core back into it. The reflectors are swung away or toward or

are axially moved with respect to the core to increase or decrease power. This method is used only for small-power reactors and in special cases, such as the SNAP 10A reactor which was operated in space [3].

Another method of control of lesser use yet is the use of some movable fuel rods. These are withdrawn from the core to decrease power and inserted to increase it, the opposite of the poison control rods.

PROBLEMS

9-1 Einstein's law (Eq. 9-11) applies to all processes: physical, chemical, and nuclear. Find the percent change in mass for the following physical and chemical processes: (*a*) when copper is heated from 100 to 1000°C, if the specific heat of copper is taken as 0.40 kJ/kgK, and (*b*) when carbon and oxygen burn to carbon dioxide, releasing 30,435 kJ/kg of carbon.

9-2 Find the percent change in mass for the following nuclear process: (*a*) when radioactive polonium-210 undergoes alpha decay for 365 days. Po^{210} has a half-life of 138.4 d and an atomic mass of 209.9829 amu. The products Pb^{206} and He^4 have atomic masses of 205.9745 and 4.00260 amu, respectively.

9-3 Find the percent change in mass for the following nuclear processes (*a*) when uranium-235 undergoes complete fission, and (*b*) when hydrogen-1 undergoes complete fusion.

9-4 A nuclear reactor powerplant operated continually for one year producing 500 MW. The powerplant efficiency is 33 percent. The reactor contained 75 metric tons of 3 percent enriched uranium dioxide fuel. Calculate (*a*) the mass of U^{235} consumed in kilograms, and (*b*) the fuel burnup, in MWd/T.

9-5 Write the complete decay reactions and calculate the energy generated, in million electron volts and joules per reaction of the following radioisotopes: (*a*) Po^{211}, an alpha emitter, (*b*) Zr^{95}, a beta emitter, (*c*) P^{30}, a position emitter, (*d*) Cu^{64}, in a K-capture reaction, and (*e*) Kr^{87}, a neutron emitter. The necessary atomic masses in amu are: Si^{30} = 29.97376, P^{30} = 29.97863, Ni^{64} = 63.9280, Cu^{64} = 63.9288, Kr^{86} = 85.9109, Kr^{87} = 86.9136, Zr^{95} = 94.9072, Nb^{95} = 94.9060, Pb^{207} = 206.9759, and Po^{211} = 210.9866.

9-6 Rutherford once postulated that when the earth was formed, it contained an equal number of atoms of U^{235} and U^{238}. From this he was able to determine the age of the earth. His answer was not far from that obtained by astronomical investigations. What is the Rutherford age of the earth?

9-7 Radioactivity exists almost everywhere, even inside our own bodies. The human body contains about 0.35 percent by mass of natural potassium, which contains 0.0118 atomic percent of radioactive potassium-40. K^{40} has an atomic mass of 39.9740 amu and a half-life of 1.28×10^9 years. Calculate the radioactivity, microcuries, in a 175-lb_m person.

9-8 When pure ordinary water is passed through a reactor core as a coolant-moderator, it becomes slightly radioactive. The most important of the radioactivities is due to the absorption of a neutron by an oxygen-16 nucleus. This absorption reaction results in a proton and a radioactive nucleus that has a 7.2-s half-life, as products. (*a*) Identify the nucleus, and (*b*) calculate the percent radioactivity left in the water 28.8 s after the above reactions.

9-9 *Carbon dating* is used to determine the age of materials of organic origin. The process involves the determination of the amount of *radiocarbon* (carbon-14) in them. When a nitrogen-14 nucleus in the atmosphere is bombarded by slow neutrons emitted in cosmic radiation, they result in carbon-14 and a proton. Carbon-14, like ordinary carbon-12, converts to $C^{14}O_2$, which constitutes 0.1 percent of all CO_2 in the atmosphere. Both are absorbed by living organisms. When absorption ceases due to the death of the organisms, the fraction of carbon-14 begins to decrease by radioactive decay. Estimate the age of an old manuscript if the amount of carbon-14 was determined by analysis to be 0.030 percent of all carbon in it.

9-10 Tritium decays by emitting low-energy β particles. This radiation acts on a phosphor producing illumination. Illuminators can thus be made by adding tritium to a phosphor in the form of paint which are sealed in a plastic container that is transparent to illumination but that blocks the β particles so that no hazard is encountered. The illuminators are used for such devices as locks, timepieces, aircraft safety markers, exit signs, etc. Regulations limit the amount of original radioactivity in such devices, depending upon service. Assuming that 4 mci are permitted for an aircraft safety device, calculate (a) the maximum

mass of tritium that can be used, in grams, and (b) the percent decrease in luminosity (which is proportional to radioactivity) after 10 years of service.

9-11 Promethium-147 is another β emitter that is used in the manufacture of illuminators when combined with a phosphor. (It has a half-life of 2.5 years. It produces more luminosity than tritium at a lower cost.) Calculate (*a*) the activity, in millicuries, of 1×10^{-4} g of Pm^{147}, and (*b*) the percent decrease in illumination (proportional to the radioactivity) after 1 year of operation.

9-12 In fast-breeder reactors, plutonium-239 is the primary fuel. A relatively stationary Pu^{239} nucleus (atomic mass 239.0522 amu) is fissioned by a 1.0-million electron volts neutron resulting in two fission fragments: krypton-93 and cerium-144. Kr^{93} undergoes five stages of β decay and Ce^{144} two stages of β decay, both to stable products which have atomic masses of 92.9060 and 143.9099 amu. (*a*) Identify all the fission products, and (*b*) calculate the total energy produced in million electron volts per Pu^{239} nucleus, and kilowatt-hours per gram of Pu^{239}.

9-13 When a spent fuel rod is removed from a reactor core, it is stored in an on-site storage pool of water so that the most intense, short-lived radioactive fission products decay, and the rod is safe for further handling and possible shipment to a reprocessing plant. Consider for simplicity only the radioisotope xenon-133 which β decays into a stable isotope with a half-life of 5.27 days. If a 30-kg fuel rod contains 0.1 percent by mass of Xe^{133} when removed from the core, what is the minimum time it should be stored in the pool so that the activity may not exceed 300 mci?

9-14 Boron-10 is used in reactor cores as a control-rod material. Natural boron has an atomic mass of 10.8110 a density of 2.3 g/cm^3 and contains 19.78 atomic percent of B^{10} which has an atomic mass of 10.01294 amu and a microscopic absorption cross section for 2200 m/s thermal neutrons of 3837 barn. Calculate the number of such neutrons absorbed per second by a 1-kg piece of natural boron.

9-15 Consider that 1 kg of pure ordinary water is subjected to an instantaneous thermal neutron pulse of 10^{16} per centimeters squared. Calculate (*a*) the radioactivity, in millicuries, generated immediately after the pulse in which oxygen-16 is converted to radioactive nitrogen-16 which has a half-life of 7.2 s, and (*b*) the radioactivity, in millicuries, one hour after the event. Ordinary water has a molecular mass of 18.01534 amu. O^{16} constitutes 99.759 atomic percent of all oxygen in ordinary water and has a microscopic cross section for thermal neutrons of 0.000178 barn.

9-16 The earth rotates around the sun with a mean radius of 149.5×10^6 km. The solar energy flux as measured just outside the earth's atmosphere (called the solar constant S, Sec. 13-3) is 1.353 kW/m^2. The reactions in the sun are of the hydrogen-1 fusion type, Eq. 9-14. Estimate (*a*) the total power generated by the sun, in megawatts, (*b*) the energy per fusion reaction, in million electron volts and joules, and (*c*) the mass of hydrogen-1 burned, in the sun in metric tons per day.

9-17 SNAPS (systems for nuclear auxiliary power) are devices that generate electric power directly from the heat generated by radioisotopic "fuels," in which case they are given odd numbers; or by fission nuclear reactors, in which case they are given even numbers. Direct generation is usually accomplished by thermoelectric energy conversion. An example is the Apollo lunar surface experiment package (ALSEP), called SNAP-27, which was placed on the lunar surface by the Apollo astronauts during their lunar landings in the late 1960s and early 1970s. SNAP-27 used plutonium-238 as "fuel" in the form of plutonium carbide PuC. Pu^{238} is an alpha emitter with an 86-year half-life. Assuming that the fuel deployed has a mass of 1 kg and that the efficiency of the thermoelectric conversion device is 8 percent, calculate (*a*) the power generated upon deployment, in watts, and (b) the power generated 5 years later. Atomic masses in amu: Pu^{238} = 238.0495, U^{234} = 234.0409, C = 12.0112. Density of PuC = 12.5 g/cm^3.

9-18 Calculate the power generated per unit volume in million electron volts per cubic centimeter, Btus per hour per cubic foot, and kilowatts per meter squared of a 3.5 percent enriched uranium dioxide fuel element in a thermal reactor if the effective fission cross section is 350 barns and the neutron flux is 10^{14}. The density of UO_2 is 10.5 g/cm^3.

9-19 A 12-ft-high fuel element has a 3.5 percent enriched uranium dioxide fuel with a U^{235} density of 8.3×10^{20} nuclei/cm^3. The fuel element diameter is 0.9 cm. The neutron flux in the fuel has a maximum value at the center plane of the element $\phi_c = 10^{14}$. It varies in the axial direction sinusoidally according to $\phi(z) = \phi_c \pi z/H$, where z is the axial distance from the center plane and H is the height of the element. The effective fission cross section is 350 barn. Calculate the power generated by the fuel element, in kilowatts, and the average linear power, in kilowatts per foot.

9-20 A reactor core contains 43,120 vertical fuel elements of the type described in Prob. 9-19. The core may, for simplicity, be considered cylindrical with a diameter of 14 ft. The element in Prob. 9-19 is situated in the center of the core. The neutron flux varies radially according to $\phi(r) = J_0(2.4048\ r/R)$ where J_0 is the Bessel function of the first kind, zero order, r is the radius measured from the core center line, and R is the radius of the core. Consider an approximation where each fuel element generates power evenly over an area $\pi R^2/n$, where n is the number of fuel elements. Calculate the total power generated by the core in megawatts. Note: $\int_0^R r\,J_0(2.4048\ r/R) = (R/2.4048)\,[r\,J_1(2.4048\ r/R)]_0^R$, where J_1 is the Bessel function of the first kind, first order, and $J_1(0) = 0$ and $J_1(2.4048) = 0.519$.

9-21 The neutron fission cross for uranium-235 for 2200 m/s neutrons is 577.1 barn. A uranium dioxide fuel pellet 0.9 cm diameter and 1.5 cm high is subjected to a monenergetic neutron flux of 10^{14}, which was thermalized to a temperature of 260°C. The U^{235} density in the pellet is 8.3×10^{20} nuclei/cm^3. The cross sections are within the $1/V$ range. (The energy of neutrons, like that of a perfect gas is given by kT, where k is the Boltzmann constant and T the absolute temperature.) Calculate the power generated by the pellet, in kilowatts.

9-22 Lithium is considered for use as a blanket material surrounding fusion reactors of the D-T type. It would receive high-energy neutrons from the D-T reaction, moderates (slows them down), absorbs them, and converts their kinetic energy to heat. That heat is then to be used in a thermodynamic cycle for power generation. Lithium also acts as a breeder of tritium for use with new deuterium fuel to keep the reactor going. Naturally occurring lithium is composed of 7.42 percent Li^6 and 92.58 Li^7. Upon neutron absorption each produces one helium nucleus and one tritium nucleus. (*a*) Write the complete nuclear equation for each. (*b*) Calculate the energy of each reaction, in million electron volts, stating whether exothermic or endothermic. (*c*) What needs to be done to naturally occurring lithium to make it a net energy generator assuming, for simplicity, that both reactions are equally probable. Atomic masses in amu: $Li^6 = 6.01512$, $Li^7 = 7.01600$, $H^3 = 3.01605$.

CHAPTER

TEN

THERMAL-FISSION REACTORS AND POWERPLANTS

10-1 INTRODUCTION

Nuclear fission was first discovered in Germany by Otto Hahn and Fritz Strassmann in 1938. Since then, great strides have led to its utilization in first, unfortunately, destructive uses, then in peaceful uses for meeting the increasing demand for abundant and reliable electric power. The time span between discovery and utilization was dizzyingly short when compared with other technologies. Because of the power development and the still lingering destructive specter, public acceptance problems have arisen.

As with all complex technologies, the first generation of nuclear-fission power plants needed improvements in design, construction, and operation. The present efforts now are concentrating on improving the design, safety, and operation of proven systems, such as, the pressurized-water, boiling-water, and gas-cooled thermal reactors, and on building demonstration plants of the fast-breeder reactor.

The Hahn and Strassmann experiments that led to the discovery of fission came about, as with many great scientific discoveries, almost by accident. It showed that uranium was split, or fissioned, into smaller elements. This was the opposite of what was expected, namely the formation of larger and heavier elements than uranium, the now called *transuranium elements*. Within a few short weeks, worldwide interpretations led to significant and far-reaching effects on the technological, economic, and political future of the world.

Ten days after publication of the Hahn and Strassmann experiments, on 16 January 1939, Lise Meitner and Otto Robert Frisch published notes in *Nature* in which they made theoretical interpretations of these experiments. On 7 April 1939, Frederic Joliet, Hans von Halban, and Lew Kowarski published the paper "Liberation of Neutrons in the Nuclear Explosion of Uranium," which dealt with the possibility of a nuclear chain reaction.

Earlier, in 1905, Albert Einstein, then a young physicist and an assistant at the patent office in Bern, published in the German journal *Annals of Physics* a 2½-page supplement to his theory of relativity entitled "Is the Inertia of a Body Dependent on Its Energy Content?" In it, Einstein arrived at his famous theory of the convertability of mass m and energy E, expressed as

$$E = mc^2$$

where c is the velocity of light (300,000 km/s). Einstein himself calculated that if mass is reduced by 1 g, an amount of energy equal to 9×10^{20} ergs is produced. He wrote at the time:

> The mass of a body is a measure of its energy content. If the energy is changed, then the mass will change in the same way.

At a later date, he wrote:

> Is it not impossible that substances whose energy content can be varied to a high degree (for example, the radium salts) will make it possible to test the theory? If the theory is in accordance with the facts, then radiation transmits inertia between the emitting and the absorbing body.

On 2 August 1939, Einstein, then living in the United States, wrote a historic letter to President Franklin D. Roosevelt that drew attention to the possibility of an atomic bomb and, considering the possibility of a German lead, urgently advised the president to make preparations for the production of nuclear weapons in the United States. This dramatic event was made possible by Hahn and Strassmann's discovery, the final necessary link in a chain of scientific discoveries that made the whole thing feasible.

Unfortunately, then, the first use of fission was for destructive purposes, a birth from which the nuclear industry continues to suffer today. However, for the sake of completeness, a word on weaponry is appropriate. The first fission bomb exploded at Hiroshima, Japan, had a uranium content of approximately 50 kg, and had the equivalent destructive power of 20,000 tons of trinitrotoluene (TNT). Of the 50-kg content, only 1 kg actually fissioned, and of that only 1 g of mass was converted to energy and disappeared. The second bomb, which exploded at Nagasaki, Japan, used plutonium as fuel. The largest known nuclear explosion was detonated by the USSR in 1961. It was a hydrogen bomb (fission plus fusion) and had the equivalent destructive power of 60 million tons of TNT. In all such explosions considerable amounts of fission products are formed, thus producing large amounts of lethal radioactive radiations as a by-product. Although this may have been "desirable" from a military standpoint, it is definitely unacceptable for peaceful, commercial uses of nuclear energy. Such radiations must be minimized and contained.

The first known thoughts regarding harnessing the tremendous explosive powers of fission for the production of energy were voiced by a 38-year-old German nuclear physicist named Werner Heisenberg (who had previously received a Nobel Prize at age 31) in a 1939 paper entitled "The Possibility of Large-Scale Energy Production Using Uranium Fission." In it, he wrote:

The data available at present indicate that the uranium fission processes discovered by Hahn and Strassmann can also be used for large-scale energy production. The most reliable method for developing a suitable machine is the enrichment of the uranium-235 isotope. The greater the degree of enrichment, the smaller the size of the machine needed. The enrichment of uranium-235 is the only method that allows the volume of the machine to stay small, that is about 1 cubic metre. Moreover, it is the only method of producing explosive substances that exceed by several decimal powers the explosive force of the strongest explosive known to date. It is, however, also possible to use normal uranium without uranium-235 enrichment, if the uranium is combined with another substance that slows down the neutrons of the uranium without absorbing them. Water is not suitable for this purpose, but present data indicate that heavy water and very pure carbon fulfill this purpose.

In a February 1940 paper, Heisenberg described the construction and operation of a nuclear "reactor." The theoretical concepts presented in that paper do not differ greatly from those currently used in present-day reactors.

Practical work on peaceful energy production required the use of enriched fuel. The enrichment process posed almost insurmountable difficulties. Several methods were considered, including the ultracentrifuge, the diffusion process, and others. In Germany the ultracentrifuge process was pursued but did not meet with much success, and the use of natural uranium and heavy water to produce plutonium was pursued. Only small amounts of heavy water were available. The large amounts needed were sought from Norsk Hydro, the Norwegian hydroelectric utility located in Vermork near Rjukan in southern Norway. This company was engaged in the production of ordinary hydrogen for ammonia synthesis, and heavy water was produced as a waste product. The plant was destroyed in a daring raid by allied forces.

In the United States, huge diffusion installations erected at considerable cost succeeded in separating the chemically identical, but nuclearly different, uranium isotopes. Also a team lead by Enrico Fermi worked with great intensity on a natural uranium reactor moderated by graphite instead of heavy water, a process tried by the Germans but not pursued further by them because their graphite was not pure enough and absorbed too many neutrons. The Germans slowed down in any case because of the indifference of the Third Reich. The United States, on the other hand, gave high priority to nuclear research. It was interested in both uranium-235, as fuel for both peaceful and military purposes, and plutonium, a transuranium element discovered in 1940 by Edwin McMillan and Glenn Seaborg (a discovery that earned them both a Nobel Prize in 1951).

On 2 December 1942, a coded message was sent to Washington. It read "The Italian sailor has arrived in the new world." This signalled that the world's first nuclear reactor went critical. It was situated under the stands of the University of Chicago football stadium. The "Italian sailor" referred to was Enrico Fermi, who had come to the new world only on 2 January 1937. That reactor, called the Chicago Pile-1 (CP1), was 9 m wide, 9.5 m long, and 6 m high. It contained about 52 tons of natural uranium and about 1350 tons of graphite. Cadmium rods were used as control devices. The experiment produced an output of 0.5 W and lasted only a few minutes. However, it was definite proof that a continuous chain reaction was possible, a feat that had eluded scientists previously. The Fermi chain reaction was the event that signalled the dawn of the nuclear age.

Parallel efforts of isotope separation at Oak Ridge, large plutonium production reactors at Hanford, and research at Los Alamos (a town constructed for atomic research in 1943 with people like Niels Bohr, James Chadwick, Enrico Fermi, Hans Bethe, and J. Robert Oppenheimer as the leader) paved the way to making the United States the leading atomic power at the time.

Many events took place after this, including the construction of hydrogen weapons by both the United States and the USSR. But the event of most concern to us here is the famous and dramatic Atoms for Peace address by President Dwight Eisenhower to the United Nations General Assembly on 8 December 1953. In part, he said:

> The United States knows that peaceful power from atomic energy is no dream of the future. That capability, already proved, is here now—today. Who can doubt, if the entire body of the world's scientists and engineers had adequate amounts of fissionable material with which to test and develop their ideas, that this capability would rapidly be transformed into universal, efficient and economic usage?
>
> To hasten the day when fear of the atom will begin to disappear from the minds of people and the governments of the East and West, there are certain steps that can be taken now . . . without irritations and mutual suspicions incident to any attempt to set up a completely acceptable system of a worldwide inspection and control. The atomic energy agency could be made responsible for the impounding, storage and protection of the contributed fissionable and other materials.
>
> The more important responsibility of this atomic energy agency would be to devise methods whereby this fissionable material would be allocated to serve the peaceful pursuits of mankind. Experts would be mobilized to apply atomic energy to the needs of agriculture, medicine and other peaceful activities. A special purpose would be to provide abundant electrical energy in the power-starved areas of the world. Thus, the contributing powers would be dedicating some of their strength to serve the needs rather than the fears of mankind.

The first reactor to produce electricity was a small 5-MW unit built near Moscow, USSR. But perhaps the most significant contribution to commercial nuclear power was the development of the nuclear submarine. The *U.S.S. Nautilus*, launched in 1954, signaled the age of controlled nuclear power. Another milestone was the commissioning of the world's first full-scale powerplant at Calder Hall, England, a 180-MW(e) gas-cooled, graphite-moderated, natural-metallic-uranium-fueled reactor that can be considered the true descendent of the Fermi pile.

In the United States, and then most of the world, development proceeded with water-cooled-and-moderated reactors that use slightly enriched fuel. The first plant had a 60-MW(e) pressurized-water reactor (PWR) that began operation in 1956 in Shippingport, Penn. It was followed by a 184-MW(e) boiling-water-reactor (BWR) plant that began operation in 1960 in Dresden, Ill. Capacities reached 500 MW(e) with the San Onofre, Calif., PWR in 1968 and the Oyster Creek BWR in 1969 and inched upward to the present 1000 to 1250 MW(e) in several plants around the world.

In the previous chapter we learned that there are two kinds of fission reactors, thermal and fast. Thermal-reactor powerplants, i.e., those using thermal reactors as a heat source, will be discussed in this chapter. Fast-breeder-reactor powerplants will be discussed in the next. Thermal reactors are those in which fission is primarily caused by thermal neutrons. They, therefore, need a moderator to thermalize the neutrons as well as a coolant to remove the heat generated by the fission process. The moderator

and coolant can be one and the same, such as light water or heavy water, or different, such as a graphite moderator and a gas coolant such as helium or carbon dioxide.

Although there have been many concepts for thermal reactors, we will limit our discussion in this chapter to four powerplant types that have been built commercially. These are

1. Pressurized-water-reactor (PWR) powerplants
2. Boiling-water-reactor (BWR) powerplants
3. Gas-cooled-reactor (GCR) powerplants
4. Heavy-water-reactor (PHWR) powerplants

There are 511 thermal reactor powerplants that are operable, under construction, or on order in the world as of early 1983. Of these there are 284 PWRs, 132 BWRs, 53 GCRs, and 39 PHWRs; the rest not yet decided. They represent about 391,600 MW of power. In addition, there are 7 fast-breeder reactor powerplants representing about 3280 MW. In the United States the number is 144 thermal and one fast breeder, representing about 135,000 MW [88]. In 1982 the United States had 76 operating plants that generated 280 billion kWh, about one eighth of the country's consumption.

10-2 THE PRESSURIZED-WATER REACTOR (PWR)

In a PWR, the coolant pressure is higher than the saturation pressure corresponding to the maximum coolant temperature in the reactor, so that no coolant boiling takes place. A PWR powerplant is composed of two loops in series, the coolant loop, called the *primary* loop, and the water-steam or *working-fluid* loop (Fig. 10-1), the coolant picks up reactor heat and transfers it to the working fluid in the steam generator. The steam is then used in a Rankine-type cycle to generate electricity.

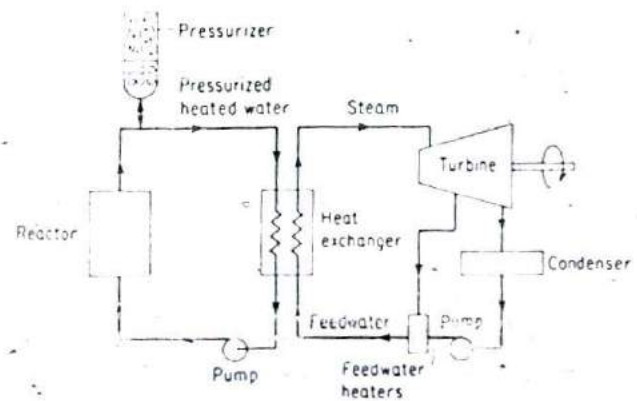


Figure 10-1 Schematic arrangement of a PWR powerplant.

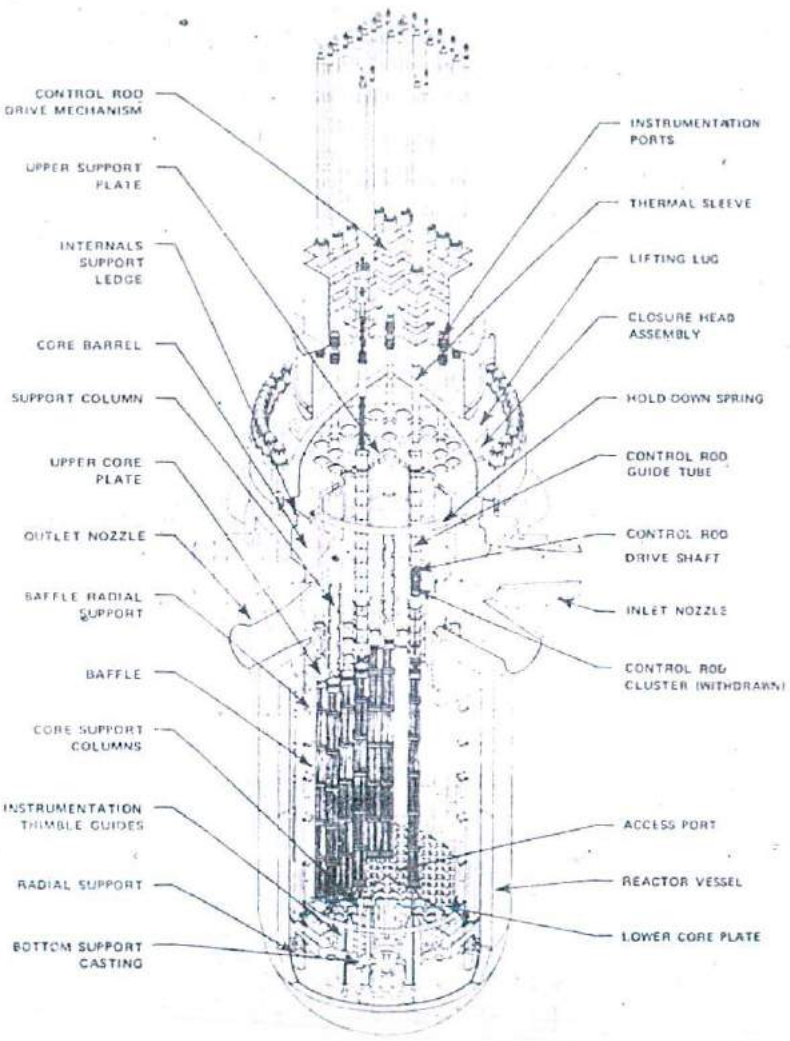


Figure 10-2 PWR reactor vessel and internals. *(Courtesy Westinghouse Electric Corporation.)*

A typical PWR reactor is shown in Fig. 10-2. The reactor core contains a total of 121 fuel assemblies of which 33 contain control-rod clusters (Fig. 10-3). The core, unlike in a BWR, is of the open type, i.e., the fuel assemblies are not enclosed in individual channels. The fuel elements are Zircaloy-clad rods, 0.422 in OD, containing UO_2 pellets, each 0.3669 in in diameter and 0.600 in long. The cladding tube is sealed at both ends by a plug welded to it. Sufficient void is left at the top to accommodate

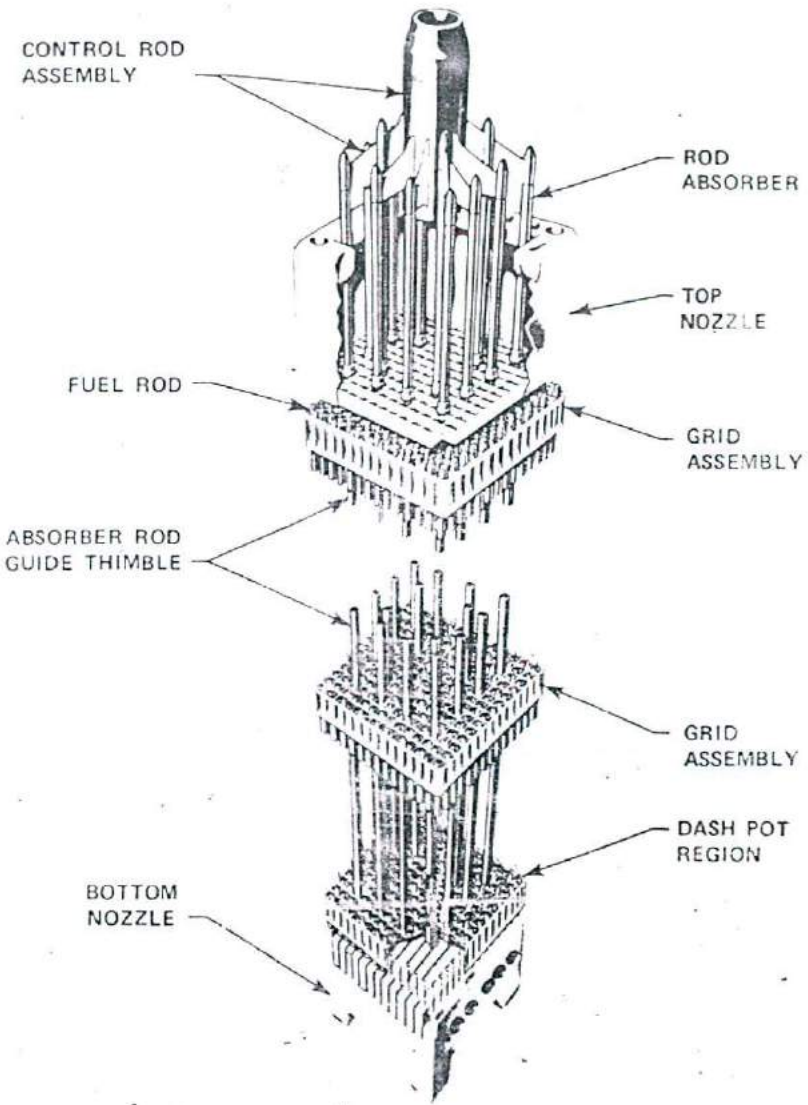


Figure 10-3 Typical control rod cluster fuel assembly. *(Courtesy Westinghouse Electric Corporation.)*

both gaseous fission products and fuel thermal expansion. A compression spring is placed within the void between the top plug and the top fuel pellet to prevent shifting of the fuel during shipment.

All fuel subassemblies are about 13.5 ft long with a 12-ft active fuel length, composed of a 14 × 14 array of fuel rods, and located on a square pitch. Each subassembly is supported axially by seven Inconel spring clip grids and bottom and top nozzles (Fig. 10-4). Five of the grids are mixing grids that help intermix coolant within the core and thus reduce temperature gradients. Each fuel rod is supported in two perpendicular directions by spring clips whose forces (11 to 14 lb_f) are opposed by two rigid dimples. This provides rigid support, reduces flow-induced vibrations of the fuel rods, and allows the rods to expand axially.

The fuel is loaded in three approximately equal-volume concentric regions in the core of 40, 40, and 41 fuel subassemblies (Fig. 10-5), with first-core fuel enrichment of 3.40, 3.03, and 2.27 percent in the outer, intermediate, and inner regions, respectively. Refueling takes place according to an inward loading schedule.

Each control-rod cluster is composed of 16 control rods that are inserted directly into 16 guide thimbles welded to the grids and top and bottom nozzles of the fuel subassemblies (Fig. 10-3). The control elements are fabricated of a silver (80 percent)-indium (15 percent)-cadmium (5 percent) alloy and are clad in stainless steel. Control-rod drives are of the magnetic-latch type. The latches are controlled by three magnetic coils that release the clusters upon loss of power, thus making them fall into the core by gravity to shut the reactor down. Some control-rod clusters, called the *control group*, are used to compensate for reactivity changes caused by variations in reactor operating conditions such as power or temperature. The rest of the control-rod clusters, called the *shutdown group*, are used to shut down the reactor in an emergency.

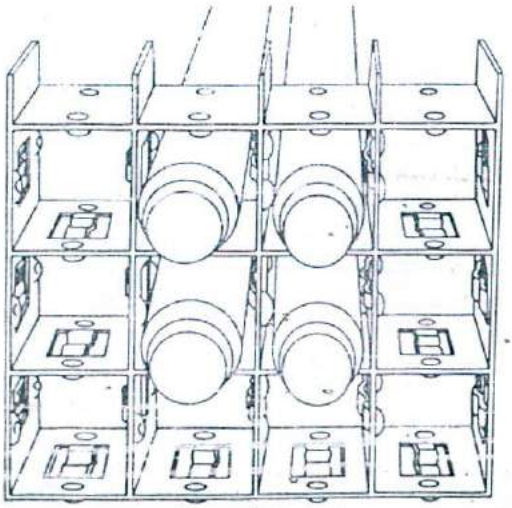


Figure 10-4 Fuel element spring clip grid detail. *(Courtesy Westinghouse Electric Corporation.)*

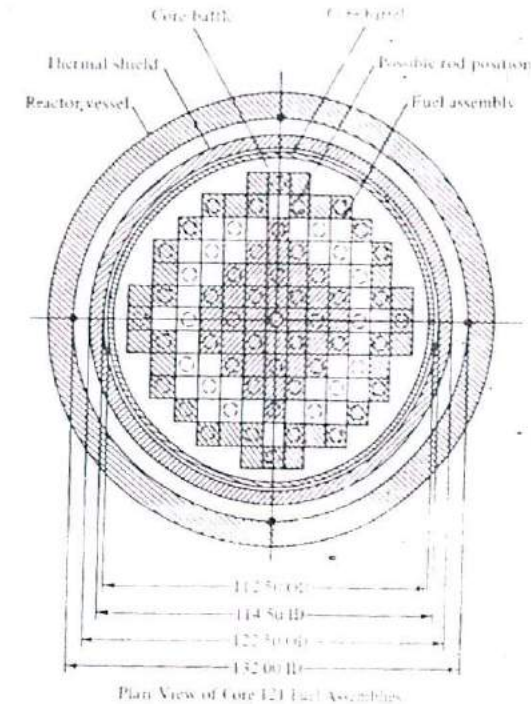


Figure 10-5 Typical PWR reactor cross section at the core

The core is surrounded by a form-fitting baffle (Fig. 10-5) that restricts the bulk of upward coolant flow to the fuel. The baffle is in turn surrounded by the core barrel. A small amount of coolant is allowed to flow between baffle and barrel. The coolant is diffused uniformly into the core by a perforated flow-mixture plate situated between the core support plate and the lower core plate. A thermal shield, supported by the core barrel, is provided to intercept core radiations and protect the pressure vessel.

The primary coolant enters the reactor vessel at about 552°F (289°C) via a number of inlet nozzles (two for 500 MW, four for 1000 MW and larger, three for intermediate) and flows downward through the annulus between the core barrel and reactor vessel wall (Fig. 10-5), thus cooling the thermal shield on both sides. It then enters a plenum at the bottom of the vessel, reverses direction, goes upward through the core where it picks up fission heat, and leaves through an equal number of exit nozzles at about 605°F (318°C). The maximum coolant temperature at the exit of the center fuel assemblies is about 650°F (343°C). The reactor coolant pressure is 2235 psig (155 bar), greater than the saturation pressure at 650°F.

10-3 THE PWR PRIMARY LOOP

The primary loop, also called the *nuclear steam supply system* (NSSS), consists of the reactor and a number of loops, depending upon reactor power, operating in parallel (Fig. 10-6). Each loop consists of a steam generator and a primary or main coolant pump. In addition, there is one pressurizer (Sec. 10-4) connected to one of the loops.

The coolant leaving the reactor enters the steam generators where it imparts its heat to the working fluid and leaves the steam generators to the main pumps where it is pumped back to the reactor. The steam generators can be of two common designs, shell-and-tube, with U-tube bundles (Fig. 10-7) or once-through (Fig. 10-8).

In the U-tube steam generator, the more common of the two, the hot coolant enters an inlet channel head at the bottom, flows through the U tubes, and reverses direction to an outlet at the bottom. The inlet and outlet channels are separated by a partition. The tubes are made of Inconel.

A typical U-tube steam generator has a capacity of over 250,000 kW, is about 67 ft high and 14 ft in diameter, and weighs about 330 tons. On the shell side, it consists of an evaporator section and an upper separator section. The working fluid

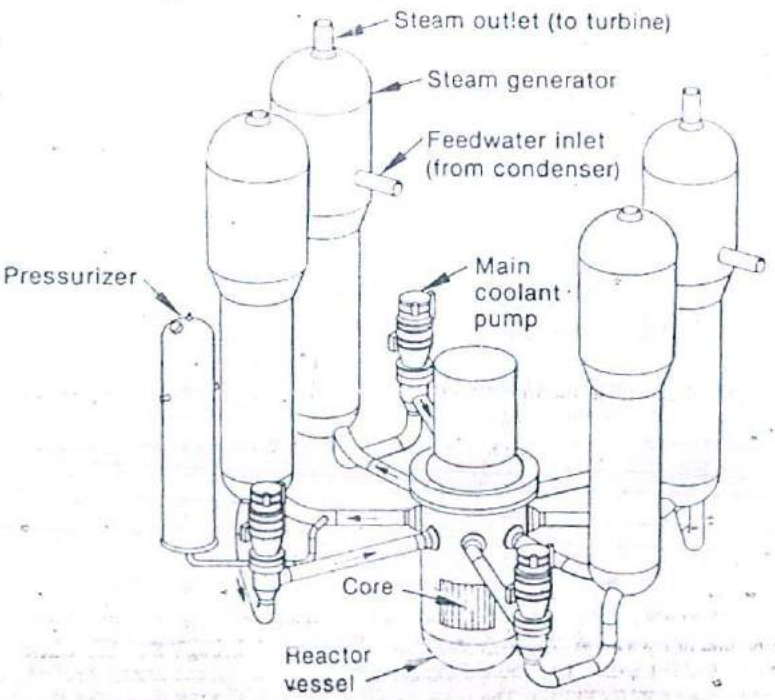


Figure 10-6 PWR nuclear steam supply system (NSSS).

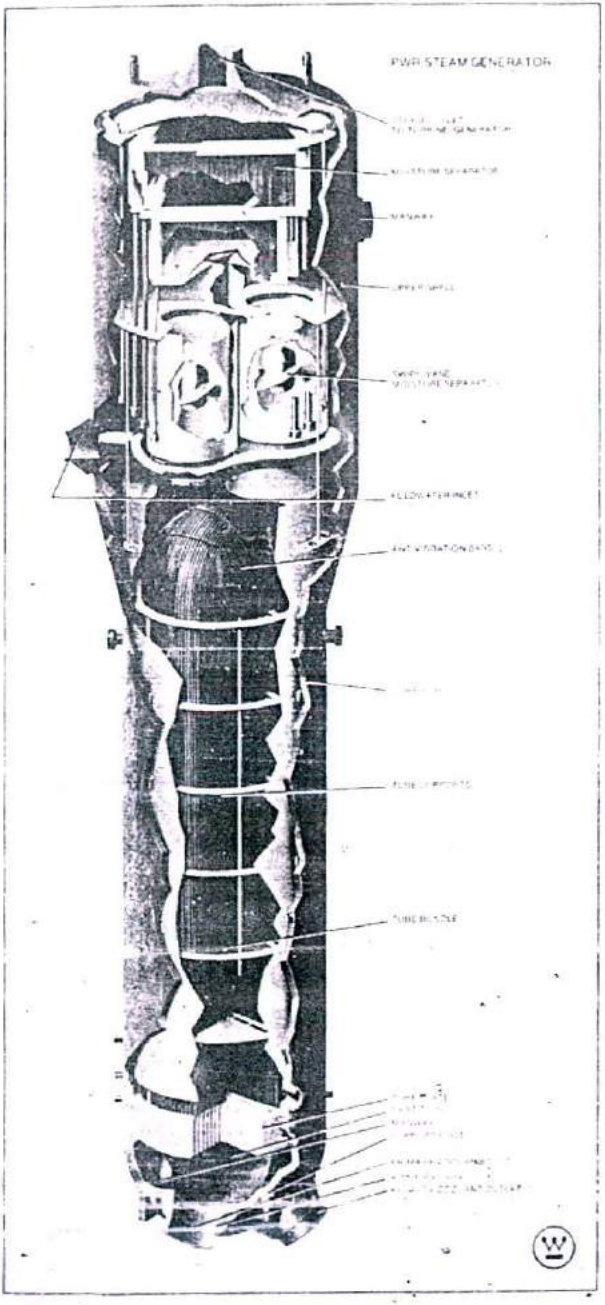


Figure 10-7 U-tube PWR steam generator. *(Courtesy Westinghouse Electric Corporation.)*

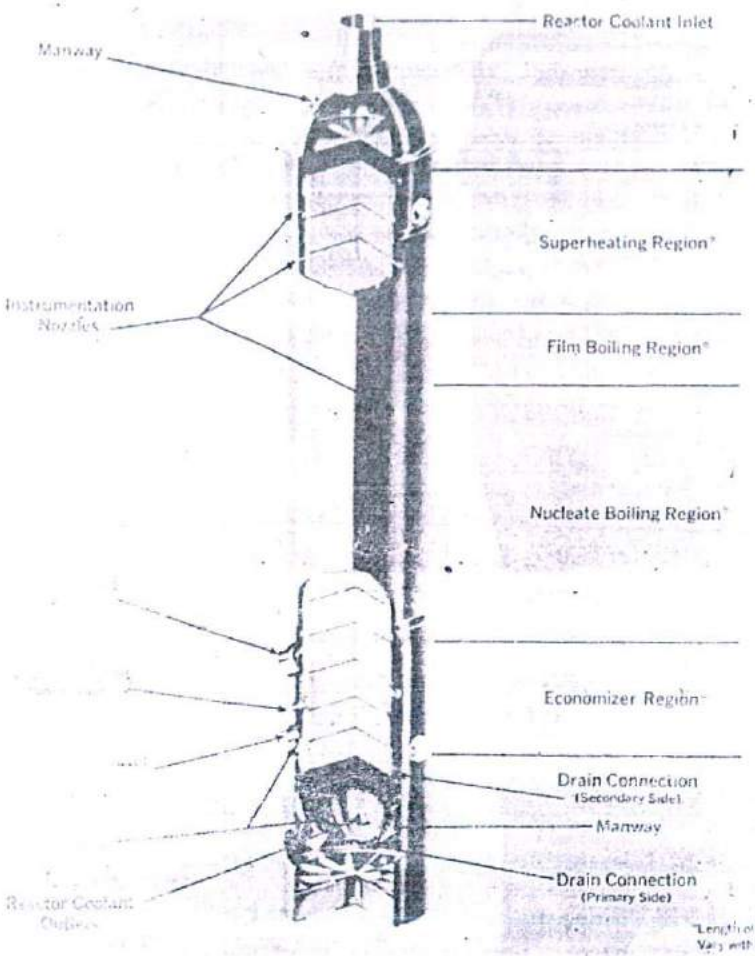


Figure 16-8 Cross through PWR steam generator. *(Courtesy Babcock and Wilcox Company.)*

Feedwater enters the generator through the feedwater inlet nozzle above the U-tube bundle and mixes with water that is recirculated from the moisture separators located near the top of the generator. The mixture then flows downward to the bottom of the shell via an annular downcomer between the lower shell and the tube bundle wrapper. It then flows upward by natural circulation through the tube bundle, where it partially boils into a steam-water mixture. The natural circulation driving force is caused by the density difference between the water in the downcomer and the two-phase mixture in the evaporator section.

The steam-water mixture is separated in the upper shell, first by swirl vane separators and finally by vane-type separators. Dry saturated steam discharges through the steam outlet nozzle at the top. The saturated water leaves the separators and mixes with feedwater before entering the downcomer.

Because boiling occurs in the same compartment with water, this type of steam generator can produce only saturated steam (or steam with very low moisture, about ¼ percent).

In the once-through steam generator (Fig. 10-8), the primary coolant enters at top, flows downward through the tubes, and exits at bottom to the main pumps. Feedwater is on the shell side moving in a general counterflow fashion to the primary water. Because of the once-through feature of this type, a dry or a low degree of superheat steam is possible and no separators are required.

Although the once-through generator results in somewhat better turbine efficiency, it has the disadvantage of containing a low water volume and, therefore, a reduced reserve in case of a reactor accident.

Steam is usually produced at about 1020 psia (70.3 bar). The main pumps are large vertical single-stage centrifugal shaft-seal pumps designed to handle large volumes of water at high pressure and temperature.

10-4 THE PRESSURIZER

It has been shown that, in PWR primary loops, the coolant is maintained at a pressure (around 2250 psia, 155 bar) greater than the saturation pressure corresponding to the maximum coolant temperature in the reactor. This avoids bulk boiling of the coolant and keeps it in the liquid phase throughout the loop. Because liquids are practically incompressible, small changes of volume—caused by changes in coolant temperatures because of normal load changes or accidental nuclear reactivity increases or caused by unforeseen expansions or contractions in the loop components—would cause severe or oscillatory pressure changes. These may be quite unsafe when the pressure drops, that is, when the pressures increase. They cause flashing into steam and consequent disruption of the reactor nuclear characteristics and possible structural damage to the fuel elements. They cause cavitation when they are negative, that is, when the pressures decrease. For these reasons, it is necessary to provide a chamber to accommodate coolant volume changes while maintaining pressure within acceptable limits. Such a chamber is called a *pressurizer*. There are two types of pressurizers in common use: vapor pressurizers and gas pressurizers. Pressurized-water reactors conveniently use vapor pressurizers because their coolant, water, can be used which results in a more compact pressurizer. Gas-type pressurizers are used in liquid-metal-cooled fast-breeder reactors.

A *vapor pressurizer* is essentially a small boiler (Fig. 10-9) in which liquid, the same as the primary coolant, is maintained by controlled electrical heating at a constant temperature and consequently a constant vapor pressure above its free surface. This pressure is the same as that of the primary coolant at the junction between the pressurizer and the hot leg of the primary loop. Thus the pressurizer temperature is higher than

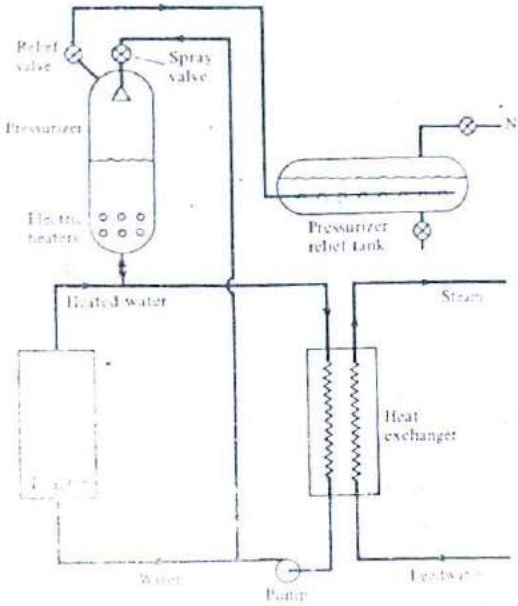


Figure 10-9 Simplified flow diagram of a PWR primary loop with a vapor-type pressurizer system.

the primary-coolant temperature because the latter is subcooled. For example, if the primary-loop pressure and temperature at the junction are 2250 psia and 605°F, the pressurizer temperature would be 653°F.

The heaters are of the electric immersion type, located in the lower section of the pressurizer vessel. These heaters are also used to heat the pressurizer and its contents at the desired rate during plant startup.

The bottom of the pressurizer is connected to the hot leg of the primary coolant system (Fig. 10-9). A spray nozzle located at the top of the pressurizer is connected, via control valves, to the cold leg of the primary coolant system after the pump. Under normal full-power operation, the pressurizer is about half full of water. The top half is full of vapor.

During a positive surge, the volume of the primary coolant increases and the vapor in the top half is compressed. Entry of the cooler primary coolant into the pressurizer condenses some of the vapor, thus limiting the pressure rise. In addition, the spray valves are power-actuated, and a cool spray (under pump pressure) enters the top, which helps condense vapor at a rapid rate and limits pressure rise. (The spray valves may also be manually operated.) A small continuous spray is usually provided to prevent excessive cooling of the spray piping and to maintain equal boron concentrations in the primary coolant and pressurizer water. Boron is used for chemical shim (Sec. 10-5).

A negative surge decreases the primary-coolant volume and expands the vapor in

the pressurizer, thus causing a momentary reduction in pressurizer pressure. The liquid in the pressurizer then partially flashes into vapor, and, assisted by further steam generation because of the automatic actuation of the electric heaters, the pressure is maintained above a minimum allowable limit.

A power-operated relief valve is attached to the top of the pressurizer to protect against pressure surges that are beyond the capacity of the pressurizer. The relief valve, in such a case, discharges steam into a pressurizer relief tank that is partly filled with water under a nitrogen blanket at near-room temperature and in which the vapor condenses. The condensate then goes to a waste-disposal system.

In a typical design of a 500-MW PWR, both the pressurizer and pressurizer relief tank had volumes of 800 ft^3 each, compared with 3700-ft^3 volume of the reactor pressure vessel. Figure 10-10 shows a photograph of a pressurizer for a PWR plant built by Westinghouse Electric Corporation.

The following is a treatment of the pressure change in an insurge (when a liquid expansion in the primary circuit causes a rush of primary coolant into the pressurizer) or, conversely, an outsurge. The situation is a nonsteady flow case. The general energy equation [Eq. (1-1)] in that case is written as

$$PE_1 + KE_1 + \Delta m_i h_i + m_{f,1}u_{f,1} + m_{g,1}u_{g,1} + \Delta Q = PE_2 + KE_2 + \Delta m_e h_e + m_{f,2}u_{f,2} + m_{g,2}u_{g,2} + \Delta W \quad (10\text{-}1)$$

where the subscripts 1 and 2 refer to conditions within the pressurizer before and after the insurge (or outsurge) and i and e refer to inlet and exit fluids. The changes in potential energy PE and kinetic energy KE as well as the work ΔW are zero. Equation (10-1) then becomes

$$\Delta m_i h_i + m_{f,1}u_{f,1} + m_{g,1}u_{g,1} + \Delta Q = \Delta m_e h_e + m_{f,2}u_{f,2} + m_{g,2}u_{g,2} \quad (10\text{-}2)$$

The above is an energy balance. A mass balance gives

$$\Delta m_i + m_{f,1} + m_{g,1} = \Delta m_e + m_{f,2} + m_{g,2} \quad (10\text{-}3)$$

A volume balance gives

$$m_{f,1}v_{f,1} + m_{g,1}v_{g,1} = m_{f,2}v_{f,2} + m_{g,2}v_{g,2} = V \quad (10\text{-}4)$$

where Δm = mass of water entering or leaving pressurizer

m_f, m_g = masses of water and steam within pressurizer

h = enthalpy of water entering or leaving pressurizer

u_f, u_g = internal energies of water and steam within pressurizer

v_f, v_g = specific volumes of water and steam within pressurizer

ΔQ = heat added from electric heaters, less heat lost to the ambient (relatively small)

V = total volume of pressurizer

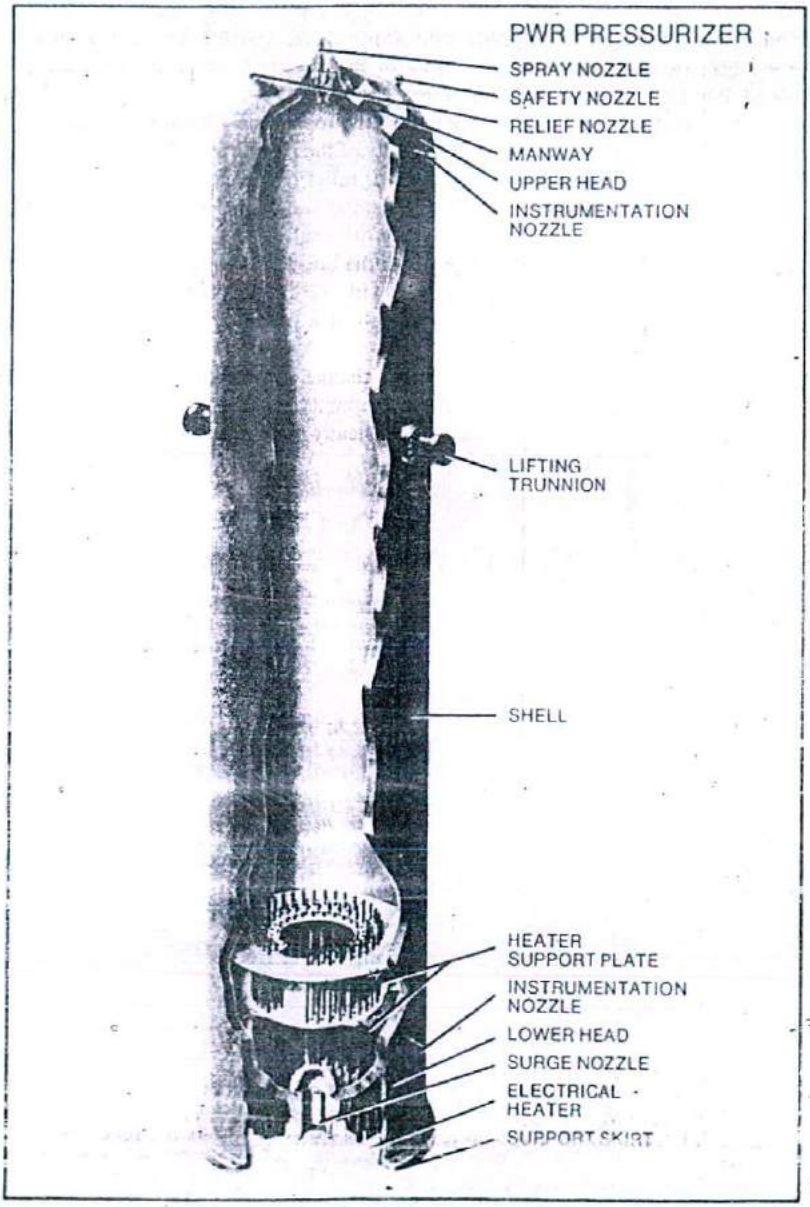


Figure 10-10 A PWR pressurizer. *(Courtesy Westinghouse Electric Corporation.)*

Noting that the pressurizer is large enough so that there will always be both water and steam in it,* the values of u and v above are for the saturated fluids at the corresponding pressures.

The above three equations are solved for the case of an insurge, where $\Delta m_i h_i$ is the sum of the incoming fluids from the hot leg and the spray from the cold leg, each times its specific enthalpy, and

$$\Delta m_e = 0 \tag{10-5}$$

or an outsurge, where

$$\Delta m_i = 0 \tag{10-6}$$

Solutions, for given primary-system conditions and permissible pressure fluctuations result in required total pressurizer volume. For given pressurizer volume and primary-system temperature fluctuations, the solutions (usually by trial and error) give resulting pressure fluctuations. As expected, the larger the pressurizer, the smaller the pressure surges.

Basic equations, for vapor-type pressurizers, that predict system transient pressure-time relationships during changes in reactor power level have been formulated [89] from energy, mass, and volume balances.

A *gas pressurizer* is simply a large volume of gas situated above the primary coolant that compresses or expands whenever the primary coolant expands or contracts, respectively. The gas, not miscible with the coolant, thus acts as a cushion to limit pressure changes in the primary system. Because of the absence of condensing or vaporizing of the gas, as occurs with the steam in a vapor pressurizer, a gas pressurizer is usually large in volume. It is limited in use to low-pressure systems such as the liquid metal fast breeder reactors. It is discussed here merely for comprehensiveness of coverage.

The pressure rise (or decrease) in a gas pressurizer is easily obtained from the gas laws $PV = mRT$ [Eq. (1-30a)] and $PV^n = C$ (Table 1-3) giving

$$\frac{\Delta P}{P_1} = \left(\frac{V_1}{V_1 + \Delta V}\right)^n - 1 \tag{10-7}$$

where ΔP = system pressure rise or decrease, lb_f/ft^2 or Pa

P_1 = initial system pressure, lb_f/ft^2 or Pa

V_1 = initial volume of gas, ft^3 or m^3

ΔV = volume change of gas

= negative of volume change of primary coolant, ft^3 or m^3

n = coefficient of polytropic exponent during gas compression or expansion

= 1 for an isothermal process

= ratio of specific heats of gas k for an adiabatic reversible process

* The pressurizer must never be allowed to be completely filled with water, a situation referred to as operating "solid," as it would lose all control over system pressure.

10-5 CHEMICAL-SHIM CONTROL

The term *chemical shim* refers to the use of a soluble neutron absorber, such as boric acid, that is dissolved in the primary reactor coolant. Control is then accomplished by varying the concentration of this absorber in the coolant. This, of course, is a slow process and is used only to control slowly varying reactivity effects in addition to conventional control rods.

Boric acid has good water solubility and has been used experimentally in both pressurized- and boiling-water reactors. For commercial power reactors, however, it is restricted to the former use. Since boron has no radioactive isotopes, no coolant radioactivity problems arise from it. (Boric acid has also been used as a shutdown device in many reactors.) The concentration of boric acid in the coolant is changed at startup and during the lifetime of a core to compensate for (1) changes in reactivity resulting from changes in moderator temperature from a cold shutdown condition to a hot operating, zero-power condition; (2) changes in reactivity caused by the buildup of neutron-absorbing xenon 135 and samarium 149 concentrations in the core; and (3) reactivity losses resulting from fuel depletion and the buildup of long-lived fission products other than zenon and samarium. Rapid reactor transients are handled by the usual control rods.

Boron concentration in the coolant may be adjusted by the *feed-and-bleed method*. By this method boron content is increased by feeding into the core some of a more concentrated boron solution than is in the core and decreased by feeding in some pure water or a less concentrated solution. Some coolant must necessarily be bled off to make room for the feed. This may be processed by distillation and the resulting concentrate and distillate reused for subsequent adjustments.

Since chemical shim permits a reduction in the amount of reactivity controlled by control rods, the number and/or size of rods may be reduced, the result being simplified design and reduced costs. Also, the rod blackness may be reduced; i.e., rod materials of lower neutron cross sections may be used. Note that because of strong water moderation a pressurized-water reactor normally needs a large number of control rods.

A chemical absorber does not by itself materially affect the spatial power distribution since it is uniformly distributed throughout the core except for a minor effect due to increased density in the lower half of the core because of cooler water temperatures there. The use of chemical shim results, however, in improvements in spatial power distribution and therefore increased average-to-maximum power density in the core because of reduced blackness, size, or degree of insertion of control rods.

The control rods that remain inserted in a chemically shimmed core will, as in any core, distort the axial power distribution (which is normally sinusoidal) so that peak power density is pushed from the core center to a point farther from the rod entrance (Fig. 10-11). The axial maximum-to-average power ratio is low when the rod is fully withdrawn because of the uniformity of the channel, increases with insertion up to a maximum, and then decreases as the rod approaches full insertion. Near full insertion the ratio increases again to the original value because the rod channel approaches uniform composition again. A hypothetical rod of zero reactivity worth (zero

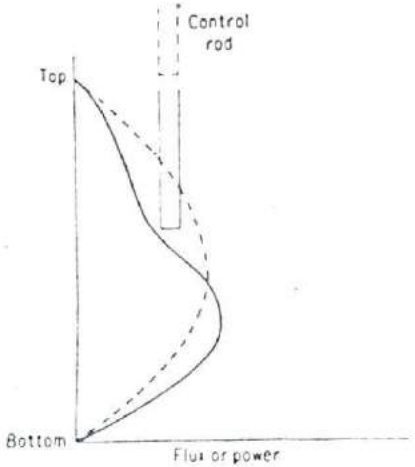


Figure 10-11 Axial neutron flux in a controlled reactor channel.

neutron absorption) would not perturb the channel at all. The degree of skewing is of course severer the higher the reactivity worth of the rod. These effects, which apply whether or not chemical shim is used, are shown by Fig. 10-12, which is representative of large pressurized-water reactors. It contains lines of different rod worths. It can be seen that the effects of partially inserted rods on peaking become severer the larger the reactivity and that, for a given rod reactivity, a more favorable axial distribution (lower axial maximum-to-average ratio) is obtained by the use of relatively low worths (and deeper insertion).

It can also be seen that rods of lower worths, obtainable when chemical shim supplements rods, are conducive to a more favorable axial power distribution.

At the beginning of core life, control rods are usually used to flatten the radial power distribution. Because of burnup, these rods are later moved out and the power distribution becomes less flat. When chemical shim is used, however, power can be flattened at the beginning of core life by spatial variations in fuel enrichment or core composition. This favorable power distribution is then maintained to a great degree by simply varying the concentration of the chemical absorber, while the rod positions are maintained. Large PWRs with chemical shim are now operated practically unrodded (i.e., rods almost completely withdrawn).

With favorable power distributions, the use of chemical shim results in increased fuel burnup for a given number of control rods. Note that without a chemical absorber, fuel burnup is limited by the number of control rods.

On the debit side, although chemical shim results in a decrease in number, size, or blackness of control rods, the effect is not linear. There actually is a slight decrease in rod worth with increasing concentrations of chemical absorber. The boron concentration necessary to ensure a minimum shutdown margin at room temperature is estimated at 3200 ppm. Figure 10-13 shows the boron worth as a function of boron

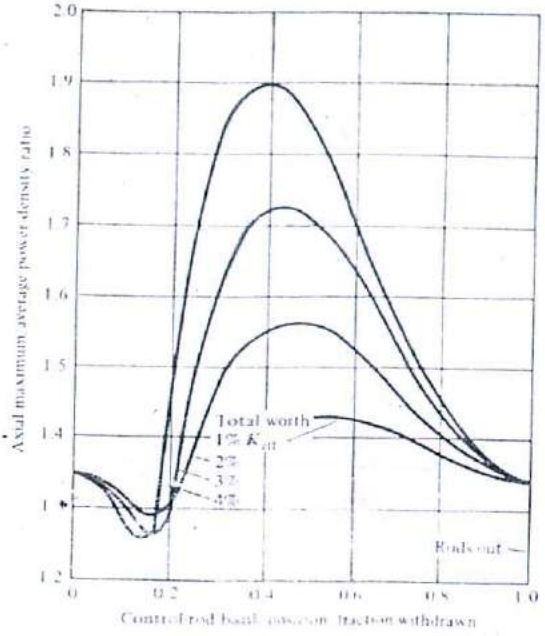


Figure 10-12 Effect of reactivity worth and depth of insertion of control rods on axial power density [90]

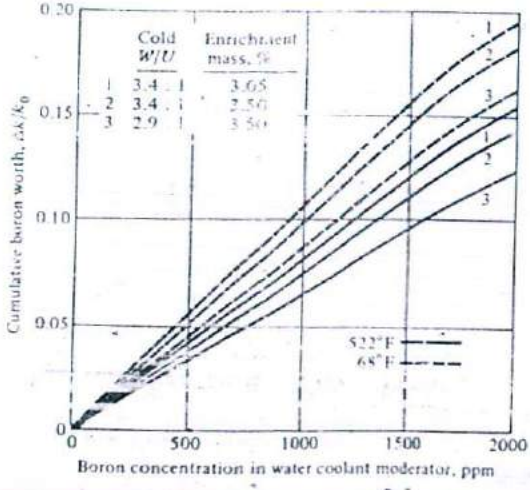


Figure 10-13 Calculated reactivity worth of dissolved boron [91]

Table 10-1 Typical reactivity requirements in a chemically shimmed PWR*

Reactivity requirement	Reactivity, %	
	Rods	Boron
Safety shutdown, cold to operating temperature change	3.0	
Doppler effect	2.2	2.0
Samarium poisoning	—	0.8
Xenon poisoning	—	2.2
Operating control	0.8	
Core lifetime (fuel depletion)	—	9.0

* Data from Ref 92.

concentration for two moderator-fuel ratios, two fuel enrichments, and two temperatures.

One problem with chemical shim is referred to as *hideout* and *plateout*. *Hideout* is defined as the precipitation of boron from solution onto solid surfaces or in deposits of corrosion products adhering to surfaces of the core and coolant system. It may later reenter the system as a result of changes in operation or water conditions. This is *plateout*. Hideout and plateout are of course undesirable because they would result in positive and negative reactivity drifts. But they represent the only significant safety problem resulting from the use of chemical shim. Tests have shown that boron deposition occurs to a limited extent onto corrosion products adhering to surfaces rather than on clean surfaces. Indications are also that it occurs on fouled surfaces when boiling occurs when the thickness of fouling is at least 0.3 to 0.4 mil. In this case deposition was proportional to the rate of evaporation. No rapid release of boron from a surface into the coolant has been observed. However, reactivity insertion from such rapid release is assumed in order to safely estimate the control rod requirements in a chemically shimmed reaction.

Typical reactivity requirements in a pressurized-water, chemically shimmed reactor are given in Table 10-1.

10-6 A PWR POWERPLANT

The secondary or working-fluid system of a twin-reactor, 1000-MW (total) powerplant [93] is shown in Fig. 10-14. It is composed on the shell side of the steam generators, a turbine-generator, condenser, two condensate pumps, five stages of feedwater heating, two feedwater pumps, and auxiliary equipment.

The turbine-generator system is designed to produce a guaranteed maximum of 502.841-MW gross and an expected maximum of about 523.7-MW gross. The 1800-

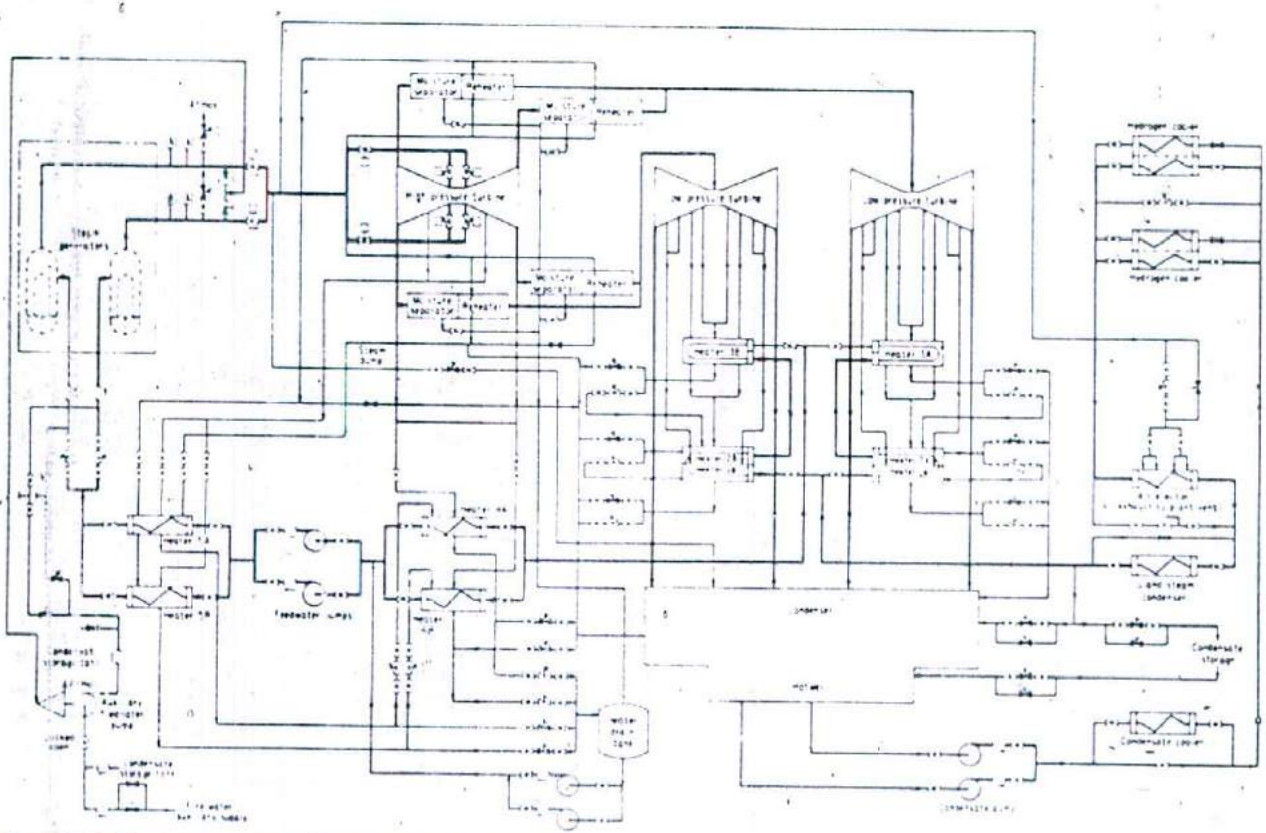


Figure 10-14 Flow diagram of the working fluid in a PWR powerplant [93].

r/min steam turbine is composed of one double-flow, high-pressure element in tandem with two double-flow, low-pressure elements.

There are four combination moisture separator-reheater assemblies between the high- and low-pressure units. Wet steam from the exhaust of the high-pressure elements enters each assembly at one end, is distributed by internal manifolds, and rises through a wire mesh where the moisture is removed. It then flows over the tubes in the reheater where it is heated by high-pressure steam from the steam generators. This enters the other end of each assembly, passes through the tubes and leaves as condensate to the high-pressure feedwater heater. The reheated steam flows back to the low-pressure turbines.

The AC generator, with rotating rectifier-exciter, is mounted on the turbine shaft. The generator is rated at 582,000 kVA and is hydrogen-cooled.

Turbine exhaust steam from four manifolds enters a radial-flow-type condenser. The condenser has a deaerating hotwell with sufficient storage for 3 min operation at full throttle and an equal free volume for surge flow. A steam-jet air ejector is provided. It has four first-stage elements and two second-stage elements mounted on shells of intermediate and after condensers. The ejector is driven with high-pressure steam from the steam-generator outlet.

Condensate from the condenser hotwell is pumped by two condensate pumps then normally passed through hydrogen coolers, air ejectors, a gland steam condenser, then through the first four feedwater-heater stages. It is then pumped by two feedwater pumps through the fifth and last feedwater-heater stage back to the steam generators. An auxiliary feedwater pump driven by a steam turbine is provided for decay-heat removal in case of loss of power. Steam for this turbine is produced from reactor decay heat. The normally closed steam valves to this turbine open automatically on loss of power, or manually. All feedwater heaters are of the closed type and are twin units operating in parallel. The lowest two stages are joined in a duplex arrangement. Steam for the heaters is obtained from five extraction points, one from the high-pressure turbine, one from high-pressure exhaust, and three from the low-pressure turbines. Drains from the high-pressure heater cascade to the second-stage heater and then to a drain tank. Heater drain pumps force water from the three lowest-pressure heaters to cascade to the condenser.

10-7 THE BOILING-WATER REACTOR (BWR)

The BWR has a function closely resembling that of the boiler in a conventional fossil-fuel steam powerplant and is basically simpler than it. In the boiler, heat is transmitted from the furnace to the water indirectly, partly by radiation, partly by convection, and partly by conduction, with combustion gases used as an intermediate agent or coolant. In the boiling-water reactor, the coolant is in direct contact with the heat-producing nuclear fuel and boils in the same compartment in which the fuel is located. It boils because the reactor pressure is maintained at about 1000 psia (about 70 bar), less than half that in a PWR, with the fuel temperatures roughly comparable. Because water

and vapor coexist in the core, a BWR produces saturated steam at about 545°F (285°C). The coolant thus serves the triple function of coolant, moderator, and working fluid.

In its simplest form (Fig. 10-15), a boiling-water-reactor powerplant consists of a reactor, a turbine generator, a condenser and associated equipment (air ejector, cooling system, etc.), and a feed pump. Slightly subcooled liquid enters the reactor core at the bottom, where it receives sensible heat to saturation plus some latent heat of vaporization. When it reaches the top of the core, it has been converted into a very wet mixture of liquid and vapor. The vapor separates from the liquid, flows to the turbine, does work, is condensed by the condenser, and is then pumped back to the reactor by the feedwater pump.

The saturated liquid that separates from the vapor at the top of the reactor or in a steam separator flows downward via downcomers within or outside the reactor and mixes with the return condensate. This recirculating coolant flows either naturally, by the density differential between the liquid in the downcomer and the two-phase mixture in the core, or by recirculating pumps in the downcomer (not shown in the figure). This is similar to what happens in modern large fossil-fueled steam generators. Modern large boiling-water reactors are of the internal, forced recirculation type.

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The ratio of the recirculation liquid to the saturated vapor produced is called the *recirculation ratio*. It is a function of the core average exit quality [Eq. (10-9), below]. Boiling-water core exit qualities are low, between 10 and 14 percent, so that recir-

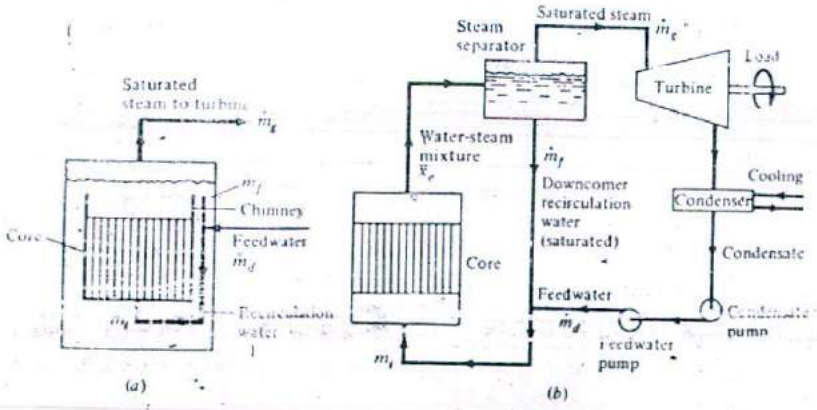


Figure 10-15 Schematic of a BWR system: (*a*) internal and (*b*) external recirculation.

culation ratios in the range of 6 to 10 are common. This is necessary to avoid large void fractions in the core, which would materially lower the moderating powers of the coolant and possibly result in low heat-transfer coefficients or vapor blanketing and burnout [2].

In either kind, a slightly subcooled liquid enters the core bottom at a rate of $\dot{m}_i$, mass per unit time. This liquid rises through the core and chimney, if any. The chimney is an unheated section above the core that helps to increase the driving pressure in natural circulation. The resulting vapor separates and proceeds to the powerplant at a mass flow rate of $\dot{m}_g$. The saturated recirculation liquid flows via the downcomer at mass flow rate of $\dot{m}_f$. There it mixes with the relatively cold return feedwater $\dot{m}_d$ from the power plant to form the slightly subcooled inlet liquid $\dot{m}_i$.

An overall mass balance in the rector core is given by

$$\dot{m}_d = \dot{m}_g \tag{10-8a}$$

$$\dot{m}_g + \dot{m}_f = \dot{m}_i \tag{10-8b}$$

The *average* exit quality of the entire core $\bar{x}_e$, that is, the quality of all the vapor-liquid mixture at the core exit, is given by

$$\bar{x}_e = \frac{\dot{m}_g}{\dot{m}_g + \dot{m}_f} = \frac{\dot{m}_d}{\dot{m}_d + \dot{m}_f} = \frac{\dot{m}_d}{\dot{m}_i} \tag{10-9}$$

The *recirculation ratio* R is the ratio of recirculation liquid to vapor produced. It is given by modifying Eq. (10-9) as follows

$$R = \frac{\dot{m}_f}{\dot{m}_g} = \frac{1 - \bar{x}_e}{\bar{x}_e} \tag{10-10}$$

Now, if the incoming feedwater has a specific enthalpy h_d Btu/lb$_m$ or kJ/kg and the recirculated liquid has a specific enthalpy h_f (at the system pressure), a heat balance is obtained, if we assume no heat losses to the outside (a good assumption) and neglect changes in kinetic and potential energies, as follows

$$\dot{m}_i h_i = \dot{m}_f h_f + \dot{m}_d h_d \tag{10-11}$$

where h_i is the specific enthalpy of the liquid at the reactor-core inlet. Equation (10-11) can be modified to

$$h_i = (1 - \bar{x}_e)h_f + \bar{x}_e h_d \tag{10-12}$$

Rearranging gives the following expression for $\bar{x}_e$

$$\bar{x}_e = \frac{h_f - h_i}{h_f - h_d} \tag{10-13}$$

The condition of the liquid entering the bottom of the core is given by the *enthalpy of subcooling*

$$\Delta h_{sub} = h_f - h_i = \bar{x}_e(h_f - h_d) \tag{10-14}$$

or by the *degree of subcooling*

$$\Delta t_{sub} = t_f - t_i \qquad (10\text{-}15)$$

where t_i is the core-inlet liquid temperature, corresponding to h_i.

The *total heat generation* Q_t can be obtained from a heat balance on the core as a system or on the reactor as a system. The two relationships, which yield identical results, are

$$Q_t = \dot{m}_t[(h_f + \bar{x}_e h_{fg}) - h_i] \qquad (10\text{-}16)$$

$$Q_t = \dot{m}_g(h_g - h_d) \qquad (10\text{-}17)$$

Shown diagramatically in its simplest form in Fig. 10-16, a boiling-water-reactor system supplies saturated (or 0.4 percent moisture) vapor directly to the turbine. After expansion through the turbine, the exhaust wet vapor is condensed and the condensate is pumped back to the reactor. The direct cycle has the advantages of simplicity and of relatively low capital costs. Because of the direct loop arrangement, there are no heat exchangers. Because of the large steam volume above the water, no pressurizer is needed.

A major drawback of the direct cycle is that the reactor is not load-following. To illustrate this, let us assume that the reactor is operating at some power level determined by its flow and control-rod setting. Let us also assume that a larger load is applied to the turbine, thus causing the turbine governor to increase the throttle valve opening, i.e., call for more steam. This reduces the flow resistance in the steam passages between the core and turbine. This in turn reduces the reactor pressure. A reduction in pressure causes flashing, i.e., increases steam voids in the core. This decreases moderation and decreases reactor power, which results in less steam generation,* opposite to the desired effect. A similar argument applies to a demand for reduced load.

* It can be shown that only above 500 psia (34.5 bar) and at very high qualities, far beyond those found in BWRs, the opposite effect takes place; i.e., a reduction in pressure causes a reduction in voids [3].

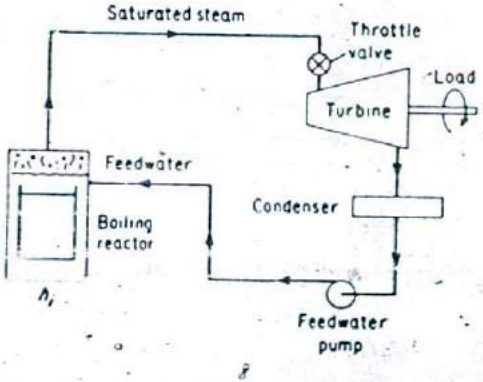


Figure 10-16 Schematic of a direct-cycle BWR plant.

This state of affairs has been corrected by one of several methods: by-pass control, a dual-pressure cycle, an overmoderated reactor in which voiding results in a reduced loss of neutrons due to water absorption and hence increased power, and recirculation control [3]. The last method is the one adopted in current designs. It is described in the next section.

10-8 BWR LOAD FOLLOWING CONTROL

The *recirculation control* method is based on a direct cycle but with variable recirculation flow in the downcomer. It is shown schematically in Fig. 10-17. Equation (10-17) is rewritten with the help of Eqs. (10-8*a*) and (10-9) as

$$Q_t = \bar{x}_e \dot{m}_i (h_g - h_d) \tag{10-18}$$

h_g, the saturated steam enthalpy at the system pressure, and h_d, the feedwater enthalpy, are both weak functions of load. Thus the plant load Q_t is therefore proportional to the product of $\bar{x}_e$ and m_i, the flow in the downcomer.

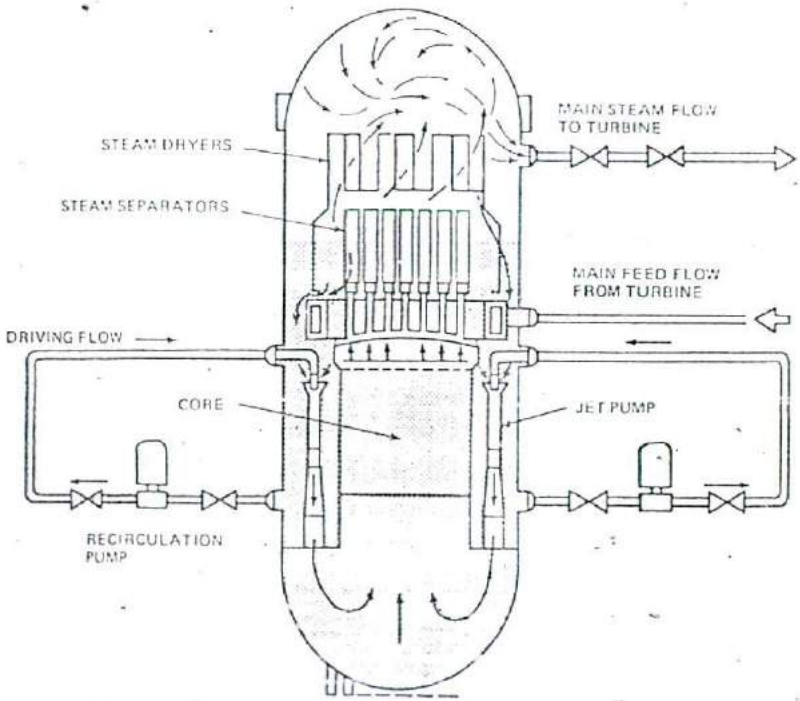


Figure 10-17 BWR reactor vessel internal flow paths.

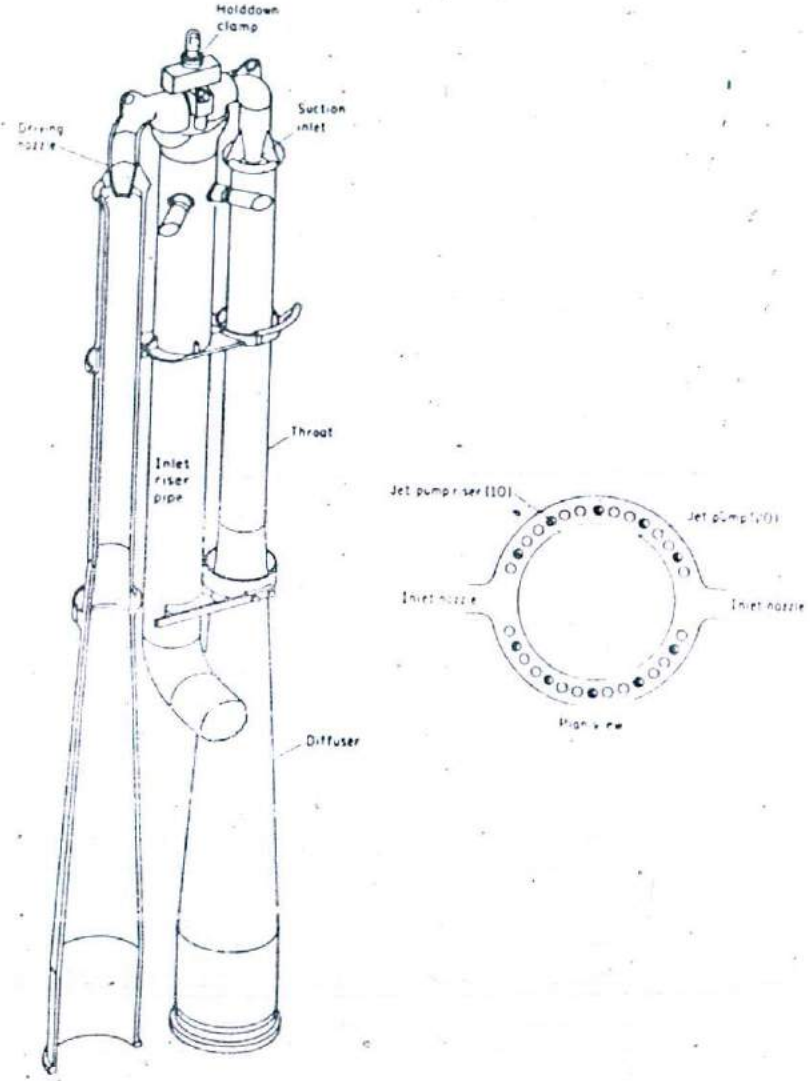


Figure 10-18 Jet-pump arrangement. *(Courtesy General Electric Company.)*

If the reactor power is to be increased by a certain percentage, $\dot{m}_t$ is increased by the same percentage. This momentarily decreases $\bar{x}_e$, as Q_t has not yet changed. $\bar{x}_e$ is a measure of the qualities in the core as a whole. When it decreases, the reactor power increases (due to improved moderation), in turn increasing $\bar{x}_e$ to the original value where nuclear equilibrium returns. We are now at the new $\dot{m}_t$, the original $\bar{x}_e$; thus Q_t is at the new desired value. The same holds true if a reduction in load is desired. In general then, except at very low loads, Q_t is essentially directly proportional to $\dot{m}_t$, hence the name recirculation control.

The recirculation system consists of 2 external recirculation pump loops and 20 internal jet pumps located inside the reactor vessel (Figs. 10-17 and 10-18). The two external recirculation pumps are single-stage vertical centrifugal units, with mechanical shaft seals, driven by variable-frequency motor-generator sets. Changing the pump speed changes recirculation flow, which in turn changes reactor power. One third of the total reactor recirculation water is pumped through the two external pumps. This flow leaves the reactor vessel through two outlet nozzles and returns to the jet-pump inlet-riser pipes to become the *driving flow* for the jet pumps. The pressure to which the driving fluid is raised in the external pumps is, however, higher than required in an all-pumped recirculation system (Fig. 10-19). This driving flow goes through the jet-pump nozzles and acquires high velocity and momentum. By a process of mo-

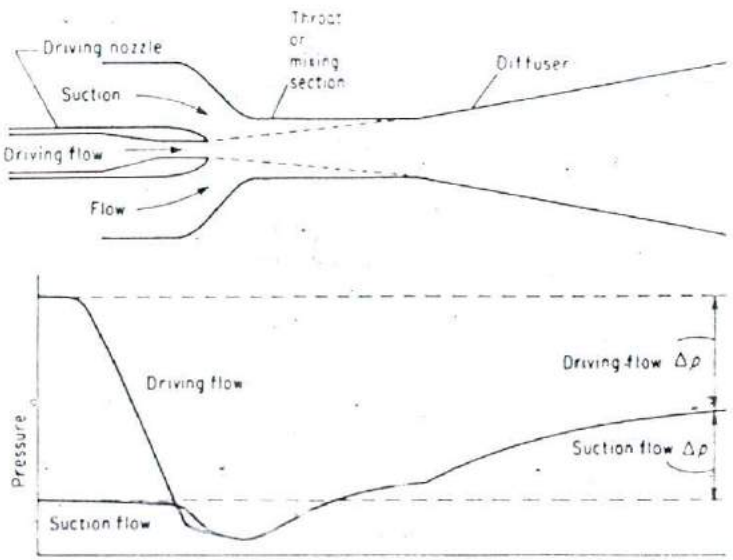


Figure 10-19 Jet-pump flows and pressure diagram (94).

mentum exchange, it entrains the remainder of the recirculation flow, called the *suction flow*, which is at a lower inlet pressure. The combined flow then enters the throat or mixing section of the jet pump where the momentum decreases and the pressure rises. Additional pressure recovery to the exit pressure occurs in the diffuser. The resultant suction flow Δp is sufficient to overcome losses through the reactor.

The jet-pump design has the advantages of fewer moving parts, lower probabilities of major line ruptures, capability to reflood the vessel in case of such rupture (loss-of-coolant accident, LOCA), improved natural circulation (lower pressure losses) in the event of power loss to the circulation pumps, and relative freedom from cavitation, a problem that arises in pumps where the liquid is near saturation conditions.

Figure 10-20 is a control "operation map." It shows the effect recirculation flow ratio $\dot{m}_r/\dot{m}_{r_o}$ on power ratio Q_r/Q_{r_o}, where the subscript o indicates conditions at full load. To understand this map it is important to note one distinguishing feature of BWRs. Unlike PWRs, which can be critical at one control-rod position and one

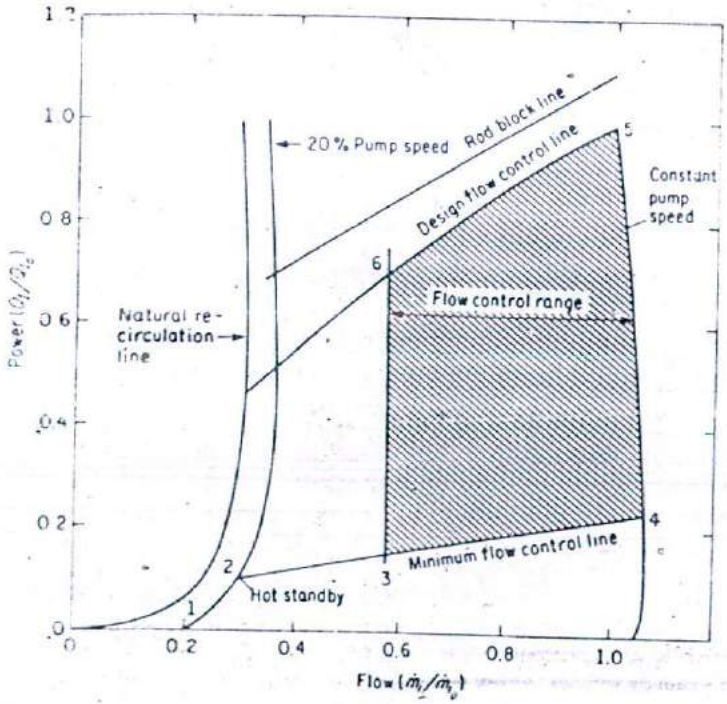


Figure 10-20 Operation map of BWR recirculation control [94].

chemical-shim concentration at a given time during the core lifetime and in which changes in these cause an increase or decrease in power until they are returned to original, BWRs can be critical at several control-rod positions because the resulting voids compensate for the change in reactivity due to the change in the rod positions.

Lines 3-4 and 5-6, and similar somewhat parallel lines in between, indicate recirculation control at various control-rod positions. They are nearly linear, as indicated by Eq. (10-18), and each indicates a control range of approximately 25 percent, without control-rod motion. Zero pump speed results in the natural circulation line at the left. Lines 3-6 and 4-5, and similar parallel lines in between, represent constant recirculation-pump speed lines.

The recirculation pumps are driven by motor-generator sets with adjustable-speed couplings that vary the frequency supply to the pump motors and hence their speed. To change reactor power, a demand signal from the operator, or a load-frequency error signal from the governor, is supplied to a master controller and compared with the actual generator speed. An error signal is used to adjust the speed of the recirculation pumps.

10-9 THE CURRENT BWR SYSTEM

Current BWR designs are of the direct-cycle, forced-internal-recirculation-type described above. The reactor (Fig. 10-21) is about 21 ft ID and 73 ft high for a 1000-MW plant. The fuel rods are similar to those of a PWR. They are Zircaloy-clad, 12 ft high in active length, and contain enriched UO_2 pellets. The fuel assemblies, however, are either 7×7 or 8×8 arrays and are enclosed in a Zircaloy-4 channel (Figs. 10-22 and 10-23). The control rods are cruciform in shape and occupy the space between four channels. They enter the reactor vessel from the bottom (vs. the top in a PWR). This is because the voids, mainly in the top part of the core, cause neutron peaking in the lower part where the control rods would be most effective. The blades of the cruciform contain sealed stainless steel tubes fitted with compacted boron carbide powder (Fig. 10-24). The reactor also contains temporary control curtains of borated stainless steel to supplement the control rods for the initial core.

The water-steam mixture leaves the top of the core and passes through steam separators and dryers located within the pressure vessel (Figs. 10-25 and 10-26). The separators are standpipes, each of which contains a centrifugal-type steam separator located at top. In each, the mixture impinges on vanes that impart a spin to separate water from steam. The separated water enters a pool surrounding the standpipes, from which it enters the downcomer annulus. Steam, still slightly wet, flows upward and outward through the dryers, where moisture is removed through troughs and tubes to a pool and back to the downcomer. The dry (or 0.4 percent wet) steam, separated from the wet steam by shrouds, finally leaves the reactor vessel through outlet nozzles and goes directly to the turbine.

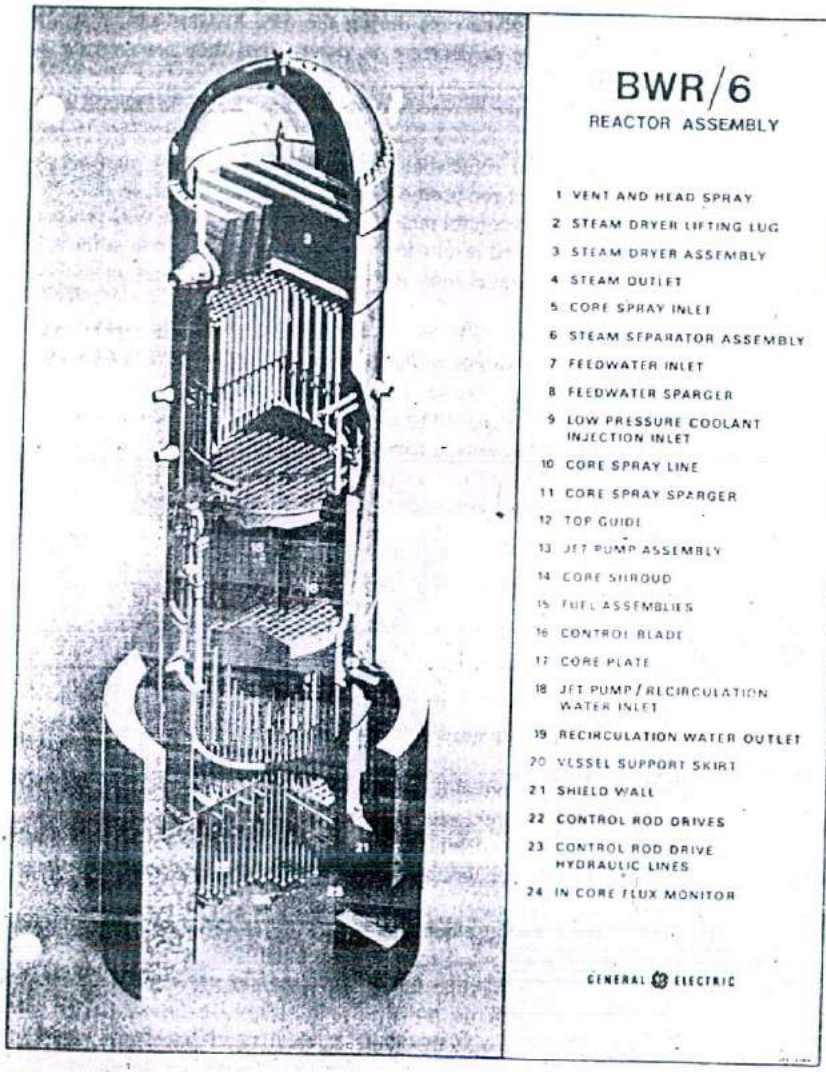


Figure 10-21 BWR reactor vessel and internals. *(Courtesy General Electric Company.)*

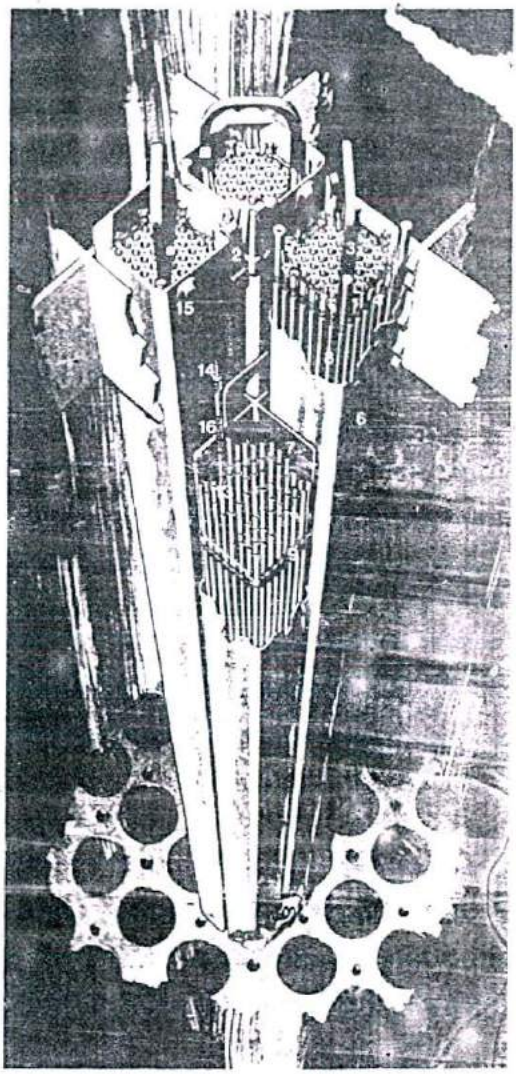


Figure 10-22 BWR fuel assemblies and control-rod arrangement. (*Courtesy General Electric Company.*)

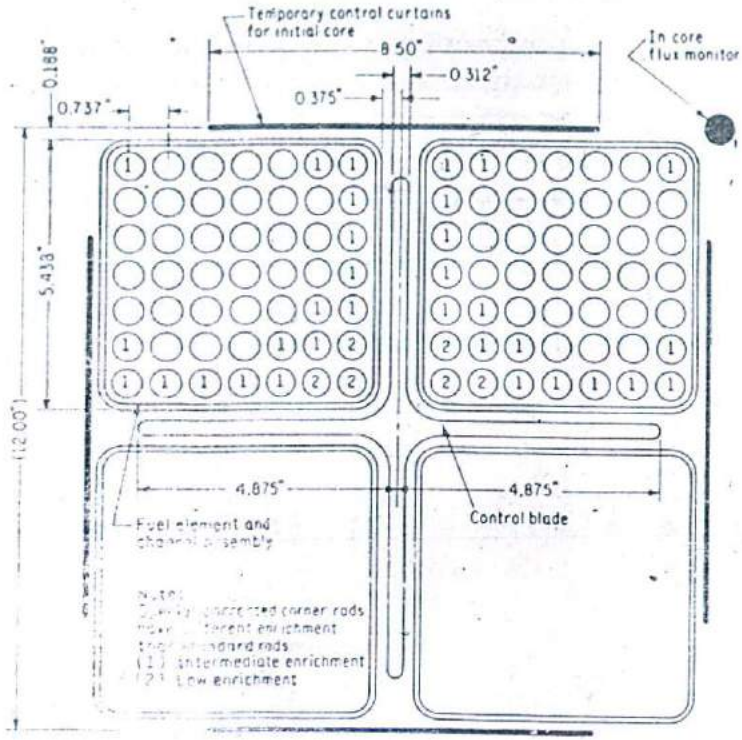


Figure [illegible] Active arrangement, cross section of Fig. 10-22.

10-10 A [illegible] POWERPLANT

Figure 10-[illegible] shows a typical flow diagram of a BWR powerplant [95]. It produces 780 MW [illegible] moisture steam at 965 psia (66.7 bar) at turbine inlet, condensing at [illegible] in.Hg absolute (0.067 bar). The turbine is a tandem-compound (with one high-pressure section and two low-pressure sections), four-flow (ducts to the condenser), 1800-r/min, nonreheat unit. The cycle has a moisture separator between the high-pressure and low-pressure turbines and five feedwater heaters, all with 5°F terminal temperature difference and 10°F drain-cooler temperature difference. The feedwater leaving the high-pressure feedwater heater to the reactor is at 381.7°F (194.3°C). The throttle steam flow is 9,610,000 lb_m/h (1210 kg/s). The flow to the condenser after moisture separation and bleeding steam for feedwater heating is 5,711,820 lb_m/h (719 kg/s). It has a "best point" heat rate of 10,287 Btu/kWh, corresponding to [illegible] percent thermal efficiency.

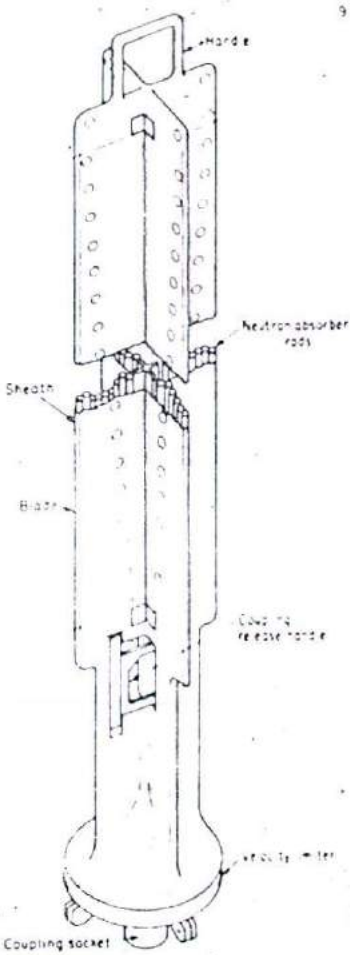
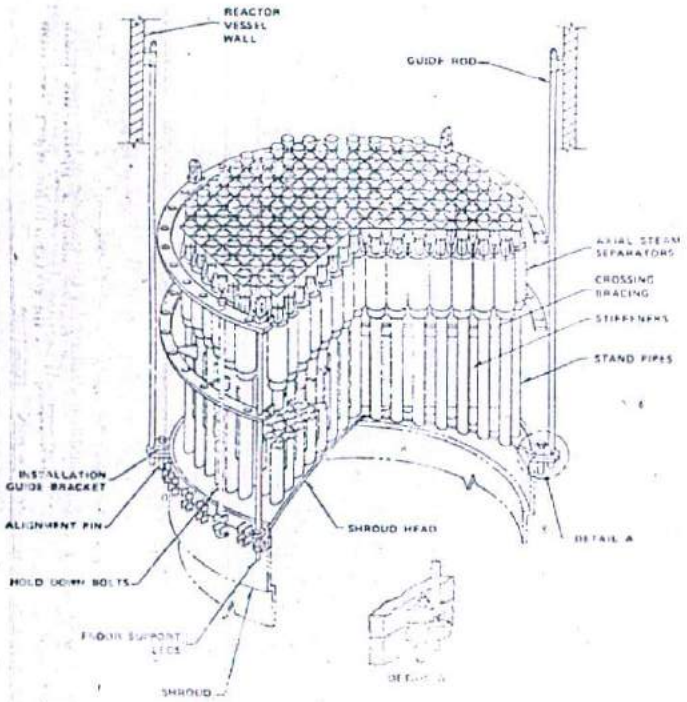
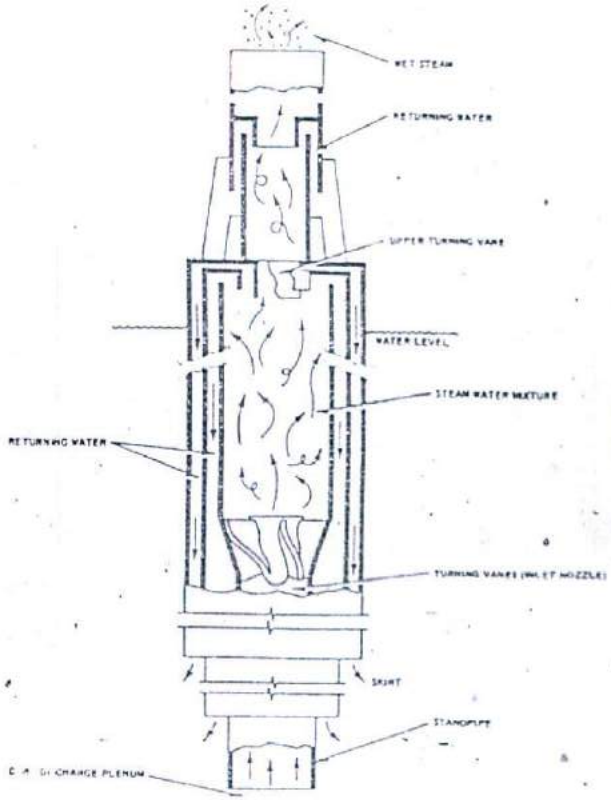


Figure 10-24 BWR control rod. *(Courtesy General Electric Company.)*

One final observation concerns radioactivity in the steam, since it is produced in the reactor and used in the balance of the plant. If mineral content of the coolant in [illegible] reactor systems is kept low enough (below about 1 ppm), the main radioactivity will be that due to the neutron capture by the oxygen in the water. Neutron capture by hydrogen converts it into nonradioactive deuterium. The most important of the oxygen reactions is the $O^{16}(n,p)N^{16}$ reaction



(a)



(b)

Figure 10-25 (a) PWR shroud head and steam separator assembly. (b) Steam separator detail.

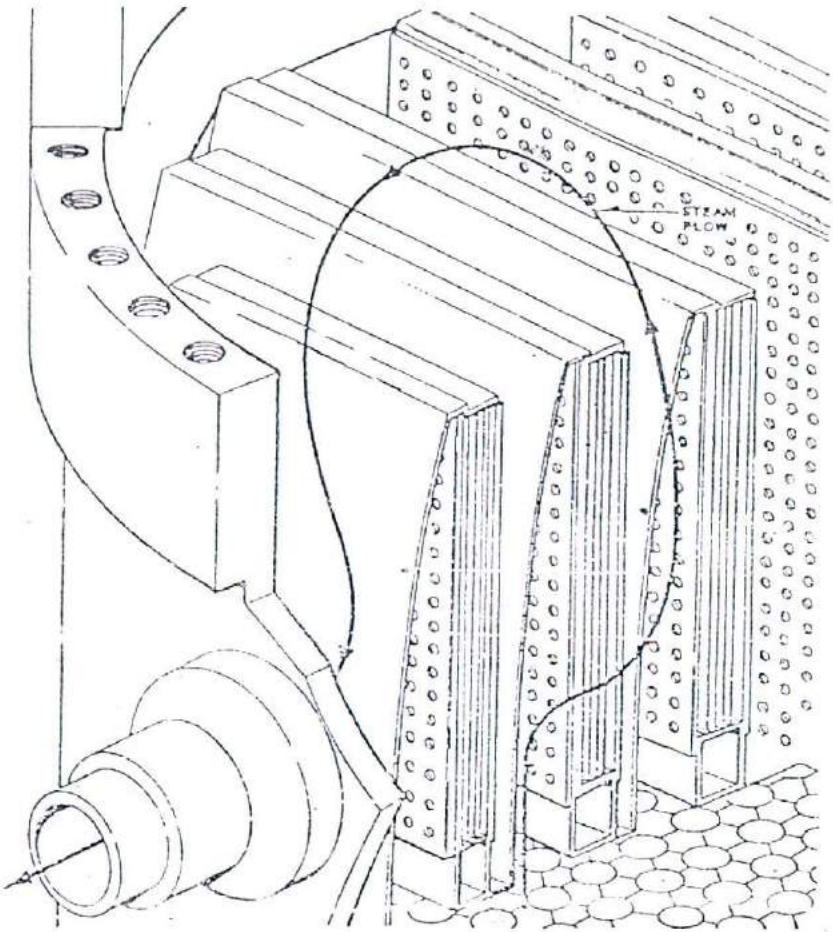


Figure 10-26 BWR steam dryer assembly.

$$ {}_8O^{16} + {}_0n^1 \rightarrow {}_7N^{16} + {}^{+}{}_1H^1 \qquad (10\text{-}19) $$

which has a microscopic cross section of 1.8×10^{-5} barn. Nitrogen 16 is a radioactive β and γ emitter (reverting to O^{16}) with a half-life of 7.2 s. The β rays are mainly of 3.8, 4.3, and 10.5 MeV energy. The γ rays are mainly of 6.13 and 7.10 MeV energy. The half-life of N^{16} is short enough so that only a small fraction of radioactivity remains when the steam reaches the turbine. Thus moisture remaining in contact with turbines and other equipment after shutdown is not dangerously radioactive, and maintenance work can usually be undertaken a short time after shutdown.

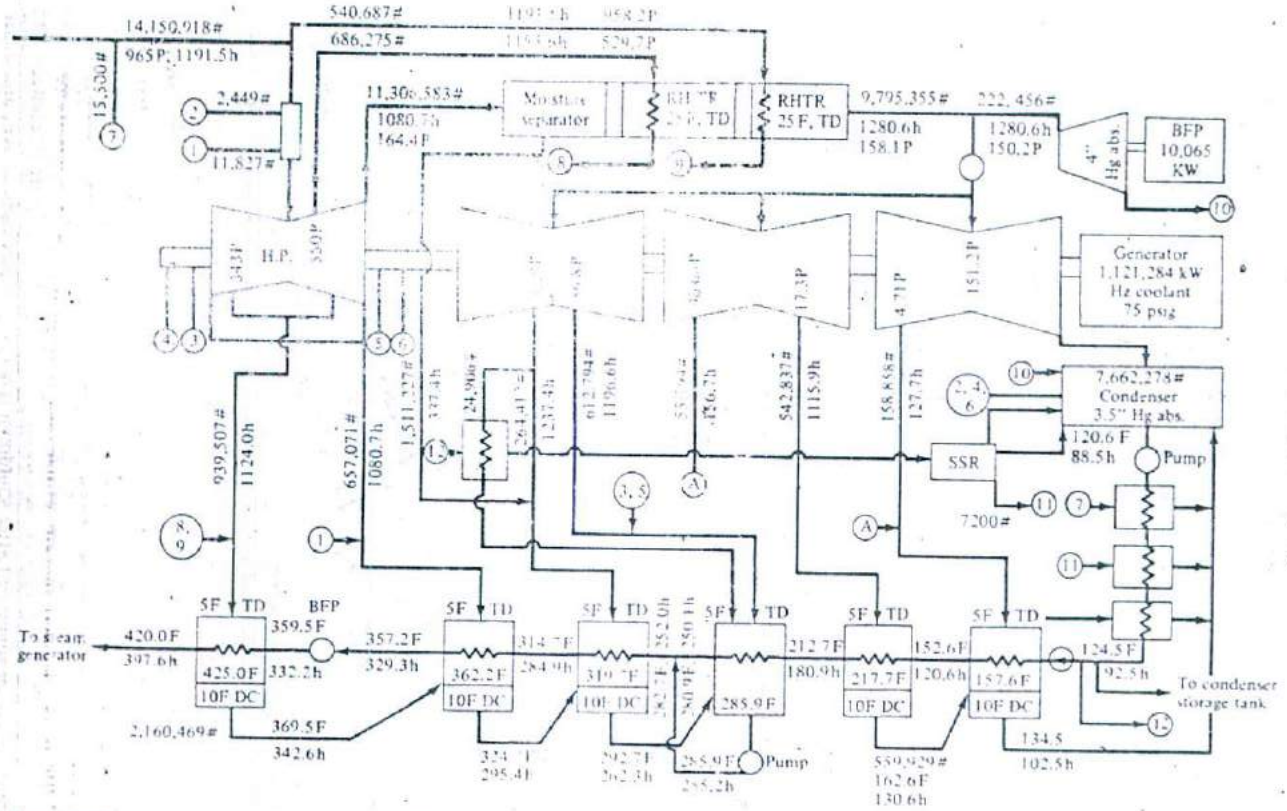


Figure 10-27 Flow diagram and mass and energy balances at rate load of twin-unit LaSalle BWR powerplant at Seneca, Ill. Capacity of each unit = 1121 MW gross, 1078 MW net.

Another reaction of somewhat less importance than the above is $O^{17}(n,p)N^{17}$. O^{17} is present to the extent of 0.037 percent of all oxygen. Nitrogen 17 is another β emitter of 4.16-s half-life. The microscopic cross section of this reaction is higher than the one above by a factor of about 10^3. This is outweighed, of course, by the smaller concentration of O^{17}. A third reaction of some importance is $O^{18}(n,\gamma)O^{19}$. Oxygen 19 is also a β emitter converting to stable fluorine 19, with a half-life of 29.0 s. This and other reactions that are not very important produce only a few weak radiations.

If the mineral content of the coolant is high, long-lived and strong radiations result. The radioactive particles may embed themselves in component parts, thus making maintenance difficult. Coolant treatment is thus an important feature in nuclear plants. Also, care should be taken in designing components to avoid pockets and crevices that may collect and retain radioactive particles.

10-11 THE GAS-COOLED REACTOR (GCR)

Gas-cooled reactors have received great attention, particularly in the United Kingdom and, to a lesser extent, in France, Germany, the United States, and the USSR. Both natural- and enriched-uranium fuels with CO_2 as coolant and graphite as moderator are used in the U.K. and France. In the United States and Germany, enriched fuels and helium coolant are used. Heavy-water-moderated, gas-cooled reactors have received some attention [96,97].

The attractiveness of gas cooling lies in the fact that, in general, gases are safe, are relatively easy to handle, have low macroscopic neutron cross sections, are plentiful and cheap (except helium), and may be operated at high temperatures without high pressurization.

The main disadvantages are the lower heat-transfer and heat-transport characteristics of gases, which require large contact surfaces and flow passages within the reactor and heat exchangers, and their high pumping requirements (between 8 to 20 percent of plant gross power), which necessitate careful attention to the problems of fluid flow, pressure drops, etc.

To partially overcome the inherent disadvantages of gas coolants and at the same time obtain attractive thermodynamic efficiencies, it is necessary to operate the fuel elements at as high temperatures as possible (commensurate with metallurgy) and permit a high gas-temperature rise in the reactor by reducing the gas mass-flow rate and pressurizing the gas. Because the fuel operates at high temperatures, fuel-element and cladding-material choice and fabrication in gas-cooled reactors present major problems, and the trend is toward using oxide and carbide fuel elements in such reactors. Also, because gas-cooled reactors are inherently large, they are particularly suited to large-capacity power plants. The reactor itself may impose structural and foundation problems. The size of the units can, of course, be reduced to a certain extent by fuel enrichment.

The British Program

One of the largest single programs per capita for civilian nuclear power is the British development and construction of a series of graphite-moderated, CO_2-cooled-reactor powerplants. This effort started in October 1956, when the first powerplant, using natural-metallic-uranium, magnesium-clad fuel, was put in operation at Calder Hall, England [98]. This powerplant type, originally known as "Calder Hall" but subsequently as "Magnox" because of magnesium cladding, was devised because of economic and military (production of Pu^{239}) necessity. It had the advantages of using familiar and economical materials and fuels that do not require enrichment. The coolant is cheap and plentiful. The steam temperatures and pressures are, however, rather low, being a few hundred degrees Fahrenheit and a few hundred psi, well below current, highly efficient fossil-fueled powerplants. Because of their low temperatures, the Magnox stations used a special dual-pressure cycle [3]. The Magnox program is composed of nine twin-reactor stations for a total of 4845 MW.

A second British program called the AGR (advanced gas-cooled reactor), based on the prototype at Windscale, was started in the late 1960s. Its objectives were to construct a nuclear steam supply system with steam conditions comparable to those in modern fossil-fueled power stations and with a degree of integrity that would permit siting nearer centers of population. AGR powerplants use enriched ceramic fuels clad in stainless steel but otherwise still use graphite moderators and CO_2 gas coolant. They operate at clad surface temperatures of about 1520°F (~825°C) with gas outlet temperatures of up to 1230°F (~665°C), resulting in steam temperatures of up to 1050°F (~565°C). The AGR stations are characterized by prestressed-concrete pressure vessels, double containment of all access penetrations, and provisions for refueling on load for high availability. The AGR program is composed of four twin-reactor stations and one four-reactor station for a total of 4750 MW.

An example of the AGR program is Hinkley Point B (Hinkley Point A was a Magnox), a twin-reactor station with a total output of 1250 MW net [99]. Figure 10-28 shows a cross section of one of the reactors in a building it shares with the other. The core is a 16-sided stack of graphite blocks, of which 308 form fuel channels and 81 are for the control rods. The vertical stacks are composed of 12 graphite blocks each, 10 of which house the fuel and the top and bottom ones act as neutron reflectors (Fig. 10-29). The core is surrounded by an annular graphite reflector, two blocks wide, and a top graphite shield. Each fuel channel contains eight fuel elements. Each is 40.8 in (1039 mm) long and consists of a 36-pin cluster of stainless steel–clad 0.57-in (14.5-mm)-diameter UO_2 pellets. The pins are 0.6 in (15.25 mm) in diameter. The total fuel is 122.5 tons of uranium and is enriched up to 2.6 percent U^{235}.

The CO_2 coolant enters at bottom at 590°F (310°C) and 617 psia (42.6 bar) and leaves at top at 1210°F (655°C) and 586 psia (40.4 bar). It then flows down 12 superheat-reheat boilers surrounding the core, within the cavity of a prestressed concrete reactor vessel (PCRV), leaving at 529°F (276°C) and 583 psia (40.2 bar). It is then pumped back to the reactor by eight constant-speed, electrically driven (28 MW) centrifugal circulators with variable-guide-vane flow control.

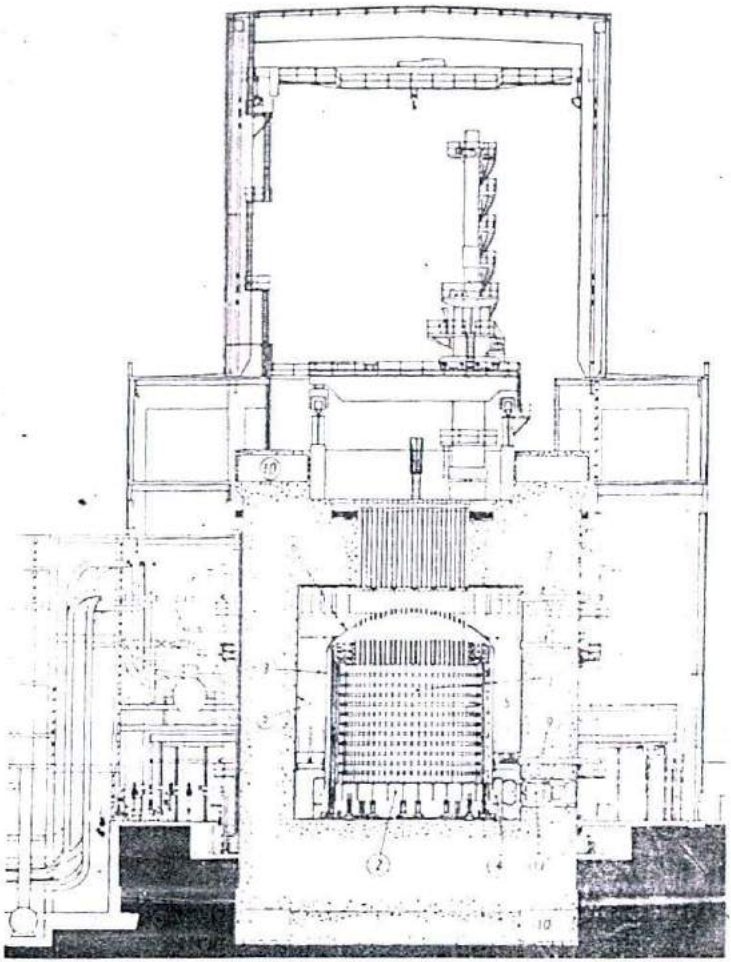


Figure 10-28 Hinkley Point B AGR building: (1) reactor core, (2) supporting grid, (3) gas baffle, (4) circulator outlet gas duct, (5) boiler, (6) thermal insulation, (7) reheat steam penetrations, (8) main steam penetrations, (9) boiler feed penetrations, (10) cable stressing gallery, (11) gas circulators. *(Courtesy The Nuclear Power Group, Limited.)*

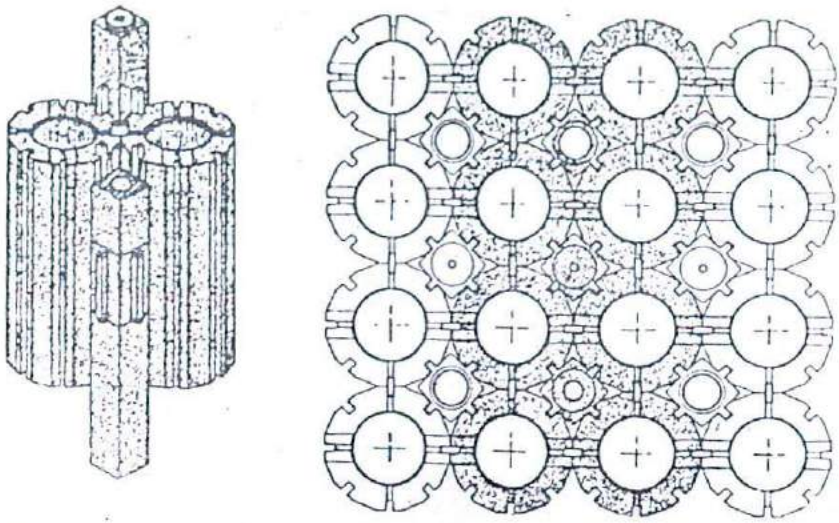


Figure 10-29 Cross section of Hinkley Point B core [99].

The steam is generated at 2418 psia (166.8 bar) and 1005°F (541°C). Reheat steam is at 590 psia (40.7 bar) and the same temperature. These conditions are comparable to modern fossil units. A plant thermal efficiency of 41.7 percent is claimed.

10-12 THE HIGH-TEMPERATURE GAS-COOLED REACTOR (HTGR)

In contrast to British development, ordinary-water reactors (PWR, BWR) constitute the bulk of the American effort in commercial nuclear power, and only a modest effort in gas cooling is under way. The American concept is usually referred to as the HTGR. In that concept, graphite is used for fuel particle coating (primary fission product barrier), fuel structural material, moderator, and coolant channel walls. Helium is used as coolant. The use of an all-ceramic fuel element results in low parasitic neutron capture in the core and therefore high conversion ratios (Sec. 11-2) and good fuel-cycle economics. The coated-fuel-particle design results in high specific powers and high fuel burnups.

Figure 10-30 shows coated fuel particles and the rod and graphite block in which they are contained [100]. The block is hexagonal, 14 in (35.6 cm) across sides, and 2.5 ft (76.2 cm) high. The coated particles are contained in over 200 fuel holes interspersed with a lesser number of coolant holes through which helium flows. The coated fuel particles are miniature spherical fuel elements containing UC_2 fuel and ThC_2 fertile material, with diameters of 200 ±50 micrometers (μm) for the fuel and 400 ±100 μm for the fertile material. They are clad with a three-layer coating of

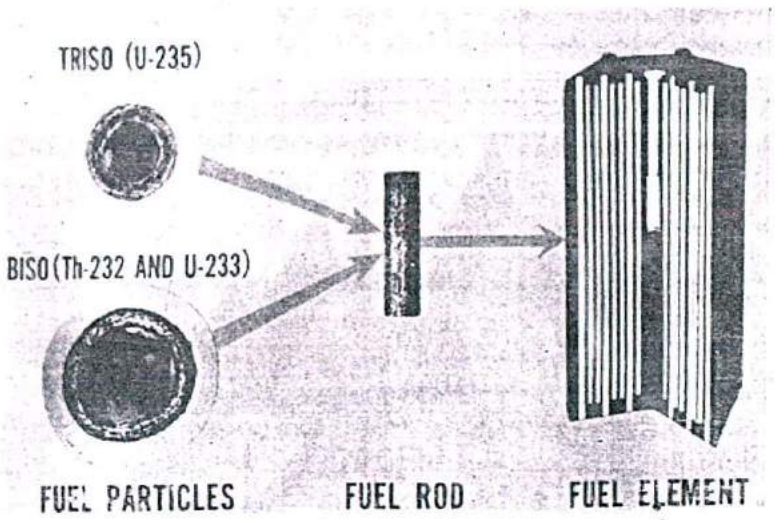


Figure 10-30 Coated particles, fuel element, and hexagonal block of HTGR. (*Courtesy GA Technologies, Inc.*)

pyrolytic carbon 100 μm thick for the fuel and 125 μm thick for the fertile material. This cladding here, as elsewhere, isolates the fuel and prevents chemical reactions and minimizes the release of fission products. The two sizes for fuel and fertile materials make it easy to separate the two for fuel recycle purposes. The Th^{232} is converted to fissionable U^{233}, which partly fissions in place and is partly processed for use in other reactors. With thorium plentiful in many parts of the world, this system can materially extend our fuel resources.

The first U.S. efforts in HTGR development were a 40-MW plant at Peach Bottom near Philadelphia (now decommissioned) and the 330-MW Fort St. Vrain plant near Platteville, Colorado. The latter plant had many engineering problems that discouraged many utilities that had once opted for the HTGR from proceeding with it despite its many favorable features. Research and development, however, are still continuing on large HTGRs in the hope that it will be revived in the future. The following is a description of a large HTGR.

The program envisions the building of units of 770- and 1160-MW capacity. The conceptual design for the latter is based on a helium-cooled, 3000-MW(t) reactor (Figs. 10-31 and 10-32). The reactor is housed in a vertical prestressed concrete reactor vessel (PCRV), which is prestressed vertically by individual cables and circumferentially by

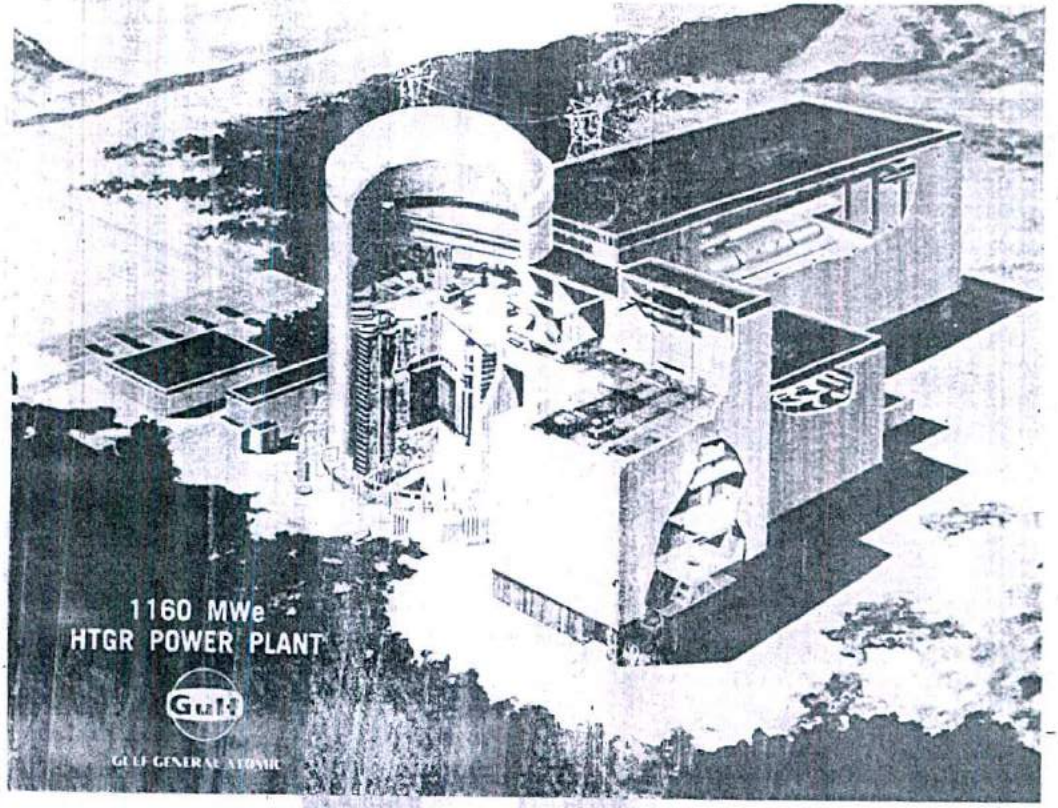


Figure 16-31 A proposed 1150-MW HTGR plant layout. *(Courtesy GA Technologies, Inc.)*

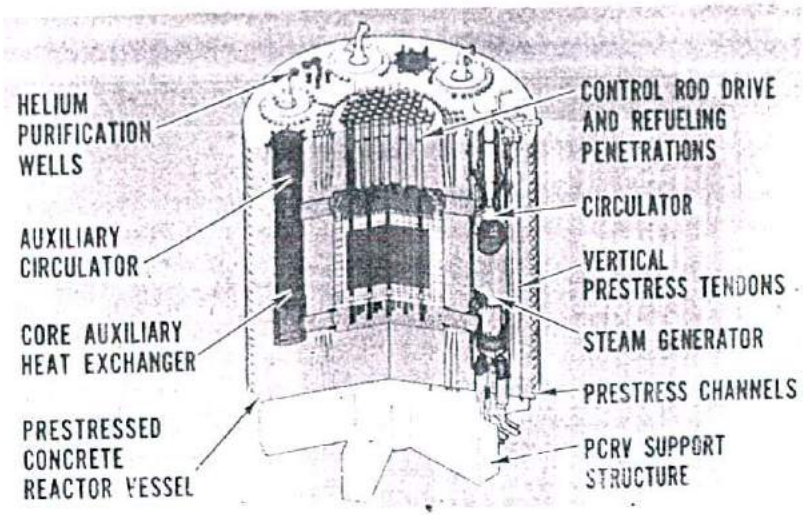


Figure 10-32 HTGR PCRV and nuclear steam supply system. *(Courtesy GA Technologies, Inc.)*

wrapped wires under tension. It is 29.4 m in diameter, 27.6 m high, and includes the reactor core and reflector, helium and nuclear steam supply systems, decay-heat removal system, control rods and their drives, and facilities. The reactor core and reflector are located in a vertical cylindrical center cavity that is 11 m in diameter and 14.5 m high. That cavity is surrounded by nine other cavities, six of which house steam generators and steam turbine driven circulators. Three other cavities, arranged between the six main cavities, house auxiliary cooling systems. This gives the reactor block a "telephone dial" appearance when viewed from the top. All cavities are provided with ceramic thermal insulation, gas-tight steel liners, and liner cooling systems. The PCRV penetrations are steel-lined and sealed by single covers with double gaskets preceded by flow resistors.

The reactor core is made up of the hexagonal graphite blocks with holes that contain the coated fuel particles (Figs. 10-30 and 10-33). In addition, each block has a central pickup hole, coolant channels, and three dowel pins for proper positioning with respect to each other. One block in seven has two additional large holes to accommodate control rods and one large hole to receive absorber granules (below). The blocks are stacked up vertically 14 to a column. Each column is composed of three upper graphite reflector blocks, eight fuel blocks, and three lower reflector blocks. There are 493 such adjacent columns for a total of 6902 blocks (1479 upper reflector,

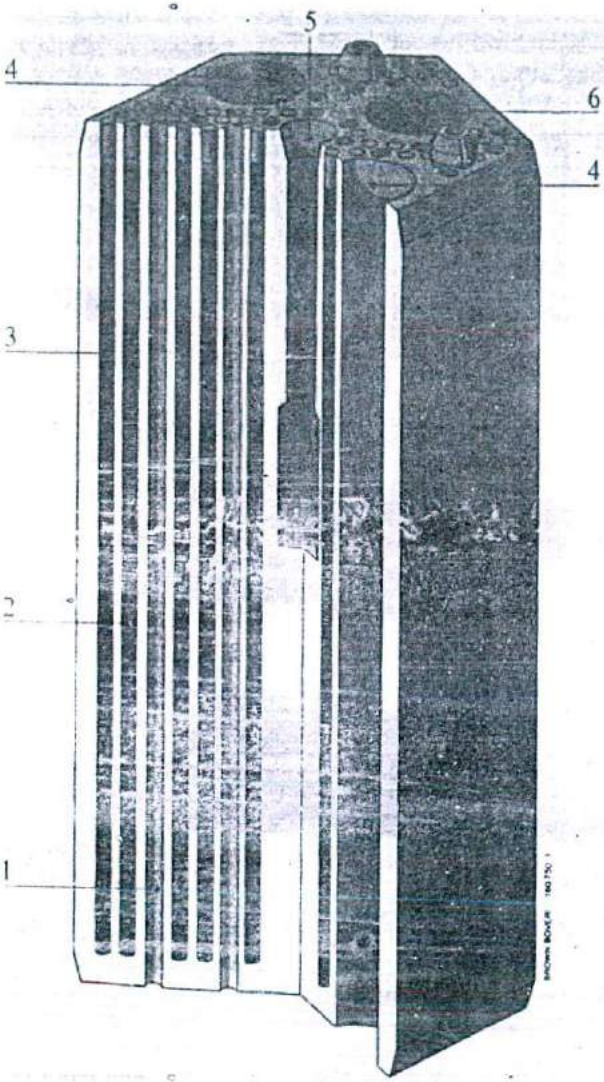


Figure 10-33 HTGR hexagonal graphite fuel block. (1) coolant channel, (2) fuel rod, (3) combustible poison rod, (4) control rod channel, (5) fuel handling pickup hole, (6) hole for absorber granules (reserve shutdown). *(Courtesy GA Technologies, Inc.)*

3944 fuel, and 1479 lower reflector) (Fig. 10-34). The fuel blocks are also surrounded by side reflectors. Groups of seven fuel columns make up a fuel region that is loaded and unloaded as a unit through a common refueling penetration in the top of the PCRV. The whole core fits within a cylindrical steel liner, the only nonceramic part of the core.

The control system consists of 146 control rods operated in pairs. The neutron absorber is boron carbide. Each pair travels in channels (item 4, Fig. 10-33) in the centermost fuel column of each of the seven-column regions. The rods are actuated by electric-motor drives that are located in standpipes in the top of the PCRV but can be made to drop under gravity for quick shutdown. A backup shutdown system consists

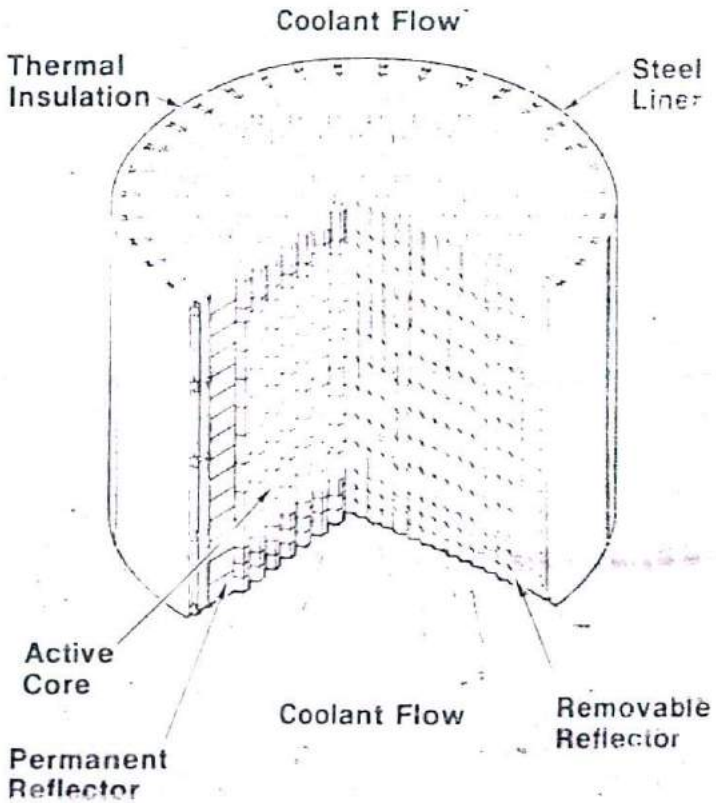


Figure 10-34 HTGR core arrangement. *(Courtesy GA Technologies, Inc.)*

of a neutron-absorbing granular material that drops into holes in the same fuel columns (item 6, Fig. 10-33).

The Primary Heat-Transfer System

Helium is circulated at an average pressure of about 48 bar (700 psi). It enters the reactor core at the top at about 240°C (644°F), flows downward through the coolant-holes in the graphite blocks (item 1, Fig. 10-33), and exits at 760°C (1,400°F) to a hot gas plenum at bottom. From there it goes radially to the bottom entrance of the six steam-generator cavities, enters the six circulators at the top of the cavities at its lowest temperature, and returns radially to a "cold" gas plenum at the top of the core.

The steam generators are shell-and-tube heat exchangers in which water and steam flow in tube coils and helium flows on the shell side. They are composed of a reheater at the bottom and an economizer-boiler-superheater at the top. The hot helium enters the steam generator above the reheater, flows downward through it, reverses direction and goes up a central pipe to the top of the steam generator, reverses direction again and flows downward through the superheater, boiler, and economizer, and then upward again through an outside annular space between the generator and its cavity to the helium circulator. The water-steam flow in the generator tubes is upward. The steam flow in the reheater tubes is downward.

The helium circulators are single-stage axial blowers and are driven by 14,500-hp, single-stage steam turbines at 6755 r/min. The circulator impeller and turbine wheel of each unit are mounted on the same shaft, which has water-lubricated bearings. A labyrinth and sealing system prevent water from entering the helium coolant and helium from entering the steam system.

The Steam Plant

Superheated steam from the six steam generators, a total of about 1,000 kg/s ($\sim 8 \times 10^6$ lb_m/h), combines into a single header and enters the high-pressure section of the turbine at 166 bar (2,400 psig) and 510°C (950°F) (Fig. 10-35). The full steam flow leaves that section, divides up to drive the six single-stage circulator turbines, then goes to the reheater in the steam generator. It leaves the reheater at 38 bar (550 psia) and 533°C (1000°F), recombines, and enters the intermediate section of the turbine and three double-flow low-pressure sections, exhausting through six ducts to a common condenser at 0.09 bar (1.3 psia).

Because steam in the HTGR is comparable in pressure and temperature to that in a fossil powerplant, the turbine is a standard 3600 r/min unit (PWR and BWR plant turbines run at 1800 r/min). All turbine sections are double-flow, resulting in six exhaust pipes to the common condenser. The condensed steam is pumped by three 50 percent flow condensate pumps (one on standby). It passes through three low-pressure feedwater heater stages, each consisting of two heaters in parallel, and a direct contact deaerating heater that has a 500-m^3 storage tank. The feedwater is then pumped by three feedwater pumps (one on standby) and enters the economizer section of the steam

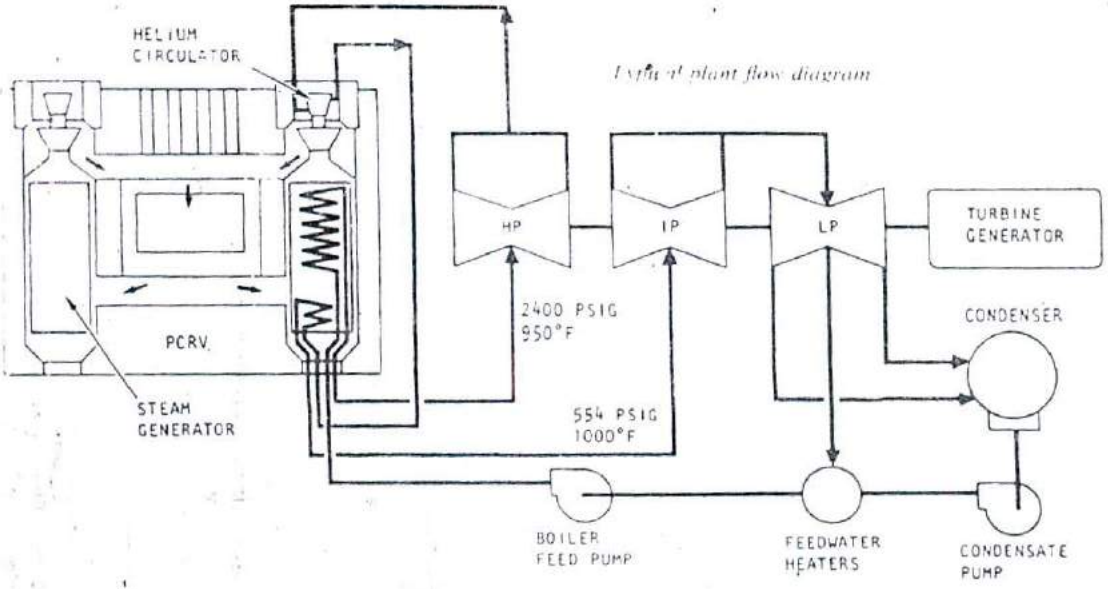


Figure 10-35 Schematic of HTGR plant flow diagram.

118°C (370°F). The feedwater pumps are driven by 16-MW condensing steam turbines whose condensate is fed back to the feedwater line.

The electric generator is a two-pole, three-phase machine, rated at 27 kV and 1170 MW. Both stator and rotor are water-cooled. The plant is started up and shut down with steam generated in auxiliary boilers or from other units on the plant site and fed into the exhaust line of the high-pressure turbine. From there it flows to drive the helium circulator turbines and, via reducing valves, to drive the feedwater pump turbines.

Decay-Heat Removal

Under normal shutdown conditions, the core decay heat (the heat that continues to be generated due to the radioactivity of the fission products) is initially sufficient to generate enough steam to operate the helium circulator turbines at the required reduced load. When decay heat is no longer sufficient to do this, the same auxiliary boilers used for plant startup take over and supply steam to these turbines. In addition, there are the three auxiliary heat exchangers located within the PCRV. These are helium-water heat exchangers with electrically driven helium circulators. The cooling water is in turn cooled in an air-blast heat exchanger.

It is to be noted that, because of the large mass and specific heat of graphite, the HTGR core has a very large heat capacity. It is estimated that it can absorb 20 times as much heat per unit temperature rise as other reactor types of comparable power. This results in a much slower temperature rise in the event of loss of coolant, thus allowing ample time for the initiation of emergency cooling.

10-13 PEBBLE-BED REACTORS

An interesting concept of a high-temperature gas-cooled reactor is the *pebble-bed reactor* under development in West Germany. The first effort resulted in a 15-MW reactor powerplant built at Julich by Arbeitsgemeinschaft Brown/Boveri/Krupp-Reaktorbau and called AVR. A second-generation plant, called the Thorium High-Temperature Reactor (THTR-300), is under construction near Hamm-Uentrop; commercial operation is expected in 1986.

A pebble-bed reactor is one in which the fuel and moderator are mixed together in the form of spherical pebbles that are randomly packed into a relatively simple vessel to form the reactor core. The coolant, helium, flows through the voids between the spheres.

The THTR-300

In the THTR-300 (Fig. 10-36), the core contains 675,000 60-mm-diameter fuel elements (Fig. 10-37). Each has a 5-mm-thick outer graphite shell surrounding a graphite matrix that contains (12.1 percent by volume) 40,000 400-μm-diameter biso-coated

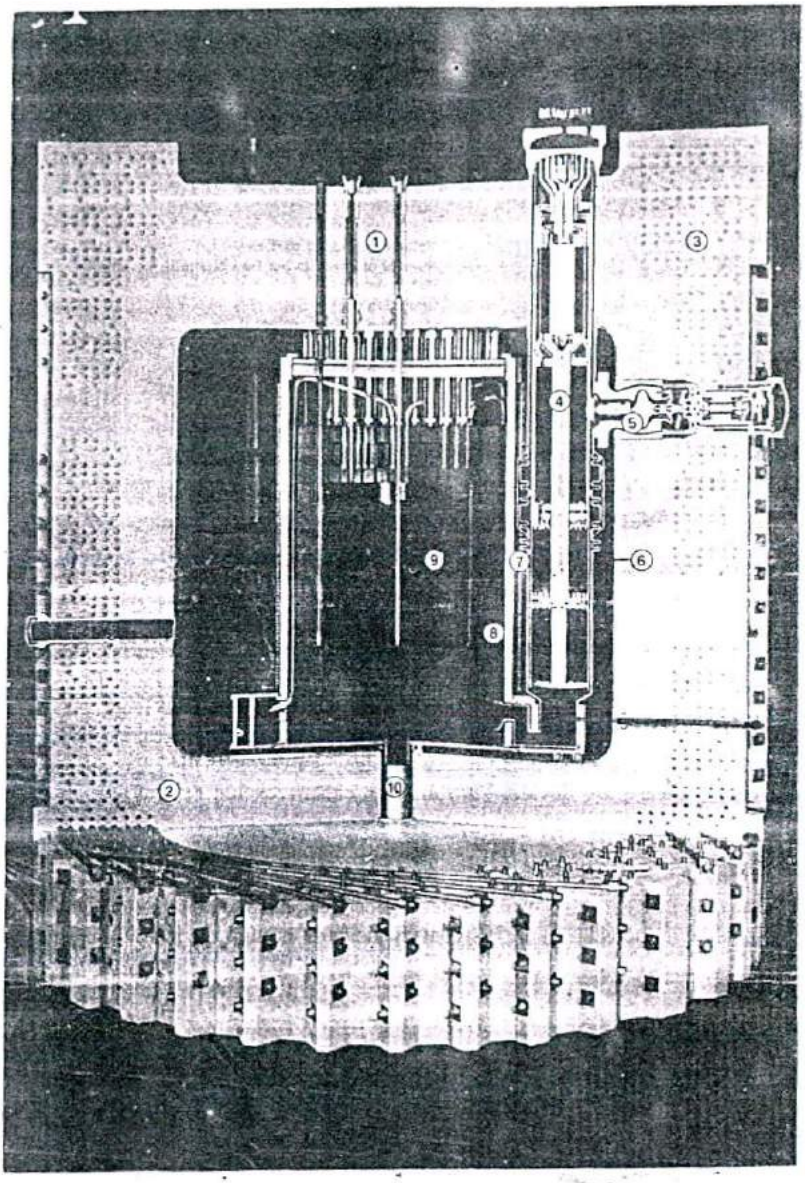
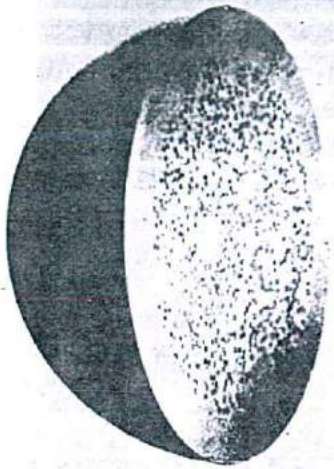


Figure 10-36 The THTR-300 pebble-bed reactor. (1) shutdown rod, (2) prestressed concrete vessel, (3) tendon rings, (4) steam generators, (5) helium circulators, (6) thermal insulation, (7) thermal shield, (8) side reflector, (9) core, (10) pebble-removal tube. *(Courtesy Hochtemperatur-Kernkraftwerk Gmbh, West Germany.)*



Diameter	60 mm
Fuel-free zone	5 mm
Volume loading of particles	12·1%
Number of particles per fuel element	40000
Matrix density	1·70 g cm^{-3}
Operation time	1100 full power days
Maximum surface temperature	1000°C
Maximum fuel temperature	1150°C
Maximum fast-neutron fluence ($E > 0{\cdot}1$ MeV)	$6{\cdot}4 \times 10^{21}$ cm^{-2}
Maximum burn-up	14·1% FIMA

Figure 10-37 Cross section of the THTR-300 spherical fuel element.

fuel particles, similar to those shown in Fig. 10-30. The particles in the THTR are not segregated by fuel but contain mixed UO_2-ThO_2.

One advantage of a pebble-bed reactor is the relative ease with which it continually charges and discharges fuel on load. Done pneumatically, the spherical elements leave the core through a central pipe, after which they are checked for breakage and sent to a burnup measuring device. They are then either recycled or sent to a discharge facility.

Helium at 39 bar (566 psia) enters the core at top at 270°C (518°F) and flows downward and leaves it at 750°C (1382°F). Surrounding the core, and within a PCRV (similar to HTGR), are six vertical steam generators and six helium circulators. The latter (unlike in the HTGR) are horizontally mounted through the PCRV. Superheated steam at 550°C (1022°F) is produced. The condenser water is cooled by a dry-cooling tower.

10-14 THE PRESSURIZED HEAVY-WATER REACTOR (PHWR)

Heavy water has almost the same physical, thermodynamic, and chemical characteristics as ordinary water (214.6°F versus 212°F normal boiling point, 38.9°F versus 32°F melting point, and 1.10 versus 1.0 g/cm^3 density at room temperature). Power reactors that use heavy water at high temperature must, as those using ordinary water, operate at high pressure. Heavy water, however, has markedly different nuclear characteristics. As a moderator, it requires a neutron to travel about twice as far as ordinary

water in order to lose the same energy fraction. On the other hand, heavy water absorbs practically no neutrons, which is not true for ordinary water.*

Because of near zero neutron absorption, heavy water can be used as coolant-moderator with natural-uranium fuels. Heavy water also results in good fuel burnups and conversion ratios. A further improvement is obtained by dividing the heavy water into a cooled moderator region and a hot coolant region in a *pressure-tube* design (below).

Other economic advantages of the PHWR are low-cost natural fuel because of the absence of the very costly enriching process and the absence of criticality hazards during fuel fabrication. The use of natural fuels can also be particularly attractive to those countries that do not possess enriching facilities. On the debit side, the production of heavy water is, in itself, a costly operation.

The economic incentives to develop heavy-water reactors thus vary in different countries and the economics depend on different methods of estimations and projections. In Canada, for example, it has been shown that natural-uranium heavy-water-moderated reactors can produce power at a lower cost than enriched-fuel reactors, and Canada has concentrated on developing this type of reactor. This view, plus other advantages already cited, has been shared elsewhere, and Canadian-type reactors (Candu) are now being built in such countries as Argentina, India, Korea, and Pakistan. In the United States, however, most studies have indicated cheaper power from enriched-fuel light-water reactors (BWR, PWR) and that, even if heavy water is used, power would be cheaper if enriched fuels are used. The overriding factors in this case are the higher capital costs and the large expensive heavy-water inventory, coupled with high fixed-charge rates. Hence, the United States effort in D_2O reactors is rather meager.

Because of the long neutron paths for moderation, heavy-water-moderated reactors require large moderator-to-fuel-volume ratios. Such reactors, therefore, require large-diameter pressure vessels. Large power reactors operating at high temperature and pressure, therefore, require larger, thicker, and costlier pressure vessels than ordinary-water reactors of comparable output. Both pressure vessel and pressure-tube designs have been used. The latter design [101] allows the use of a lower-pressure, less costly vessel but with the added expense of constructing a leak-tight *calandria* vessel free of differential expansion. It also results in the separation of coolant and moderator, and, as already mentioned, a cool moderator design can then be easily incorporated.

Some other problems associated with D_2O reactors are the loss by leakage of the expensive D_2O and the high activity associated with the decay of tritium formed in the reactor.

We will next present the Candu reactor of the pressurized heavy-water-moderated and cooled, pressure-tube variety. Other reactor types of interest that have seen various degrees of development are the D_2O-moderated, gas-cooled, organic-cooled, and ordinary-water-cooled reactors, all made possible by the pressure-tube concept. In the

* The ratio of slowing down to absorption is called the *moderating ratio*. It is about 90 times greater for heavy water than ordinary water.

latter concept the ordinary water may be allowed to boil, which results in the generation of steam within the pressure tubes.

The Candu Reactor Powerplant

The *Candu*-type reactors are designed by Atomic Energy of Canada, Limited. Their main features are a horizontal pressure-tube calandria reactor vessel, fuel elements

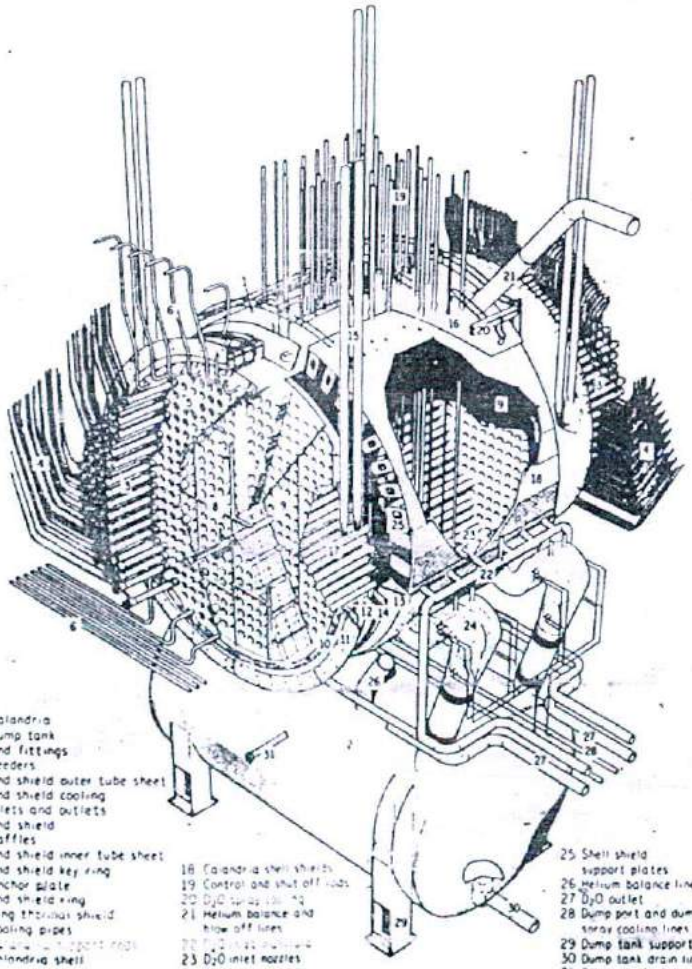


Figure 10-38 The Pickering heavy-water (Candu) reactor assembly. (*Courtesy Atomic Energy of Canada, Limited.*)

containing natural-uranium dioxide and housed in horizontal presure tubes, which also carry the pressurized heavy-water coolant, and a relatively cool heavy-water moderator on the shell side.

One such reactor (Fig. 10-38) has a stainless steel calandria shell (1) that has integral-steel end shields (7) cooled by ordinary water (6) and stainless steel circumferential shell thermal shields (13). The calandria contains 390 horizontal pressure tubes (17). Each pressure tube in turn contains a coolant tube that is supported in sliding bearings at the end shields of the calandria. The calandria and coolant tubes are separated by a sealed annulus containing CO_2 or nitrogen. There is a total of 276 tons of heavy water in the reactor. Below the calandria is a dump tank (2) with a heavy aggregate concrete vault to provide shielding from reactor radiation. The concrete is cooled by one layer of cooling coils embedded in it. The dump tank contains a helium atmosphere.

The fuel elements are hermetically sealed Zircaloy-2 rods that contain compacted and sintered naturally enriched UO_2 pellets. Twenty-eight such elements are attached mechanically at their ends to form a cylindrical fuel assembly (Fig. 10-39) 4.03 in in diameter and 19.5 in long. Spacers attached to the cladding maintain 0.05 in space between the elements.

The heavy-water coolant enters the fuel channels at 480°F (249°C) and leaves at 560°F (293°C) and 1300 psig (90.7 bar) via feeder pipes (4). Twelve U-shaped tube-in-shell heat exchangers generate a total of 6.46×10^6 lb_m/h (814 kg/s) of steam at 585 psia (40.3 bar) and 483.5°F (250.8°C) from the ordinary-water working fluid.

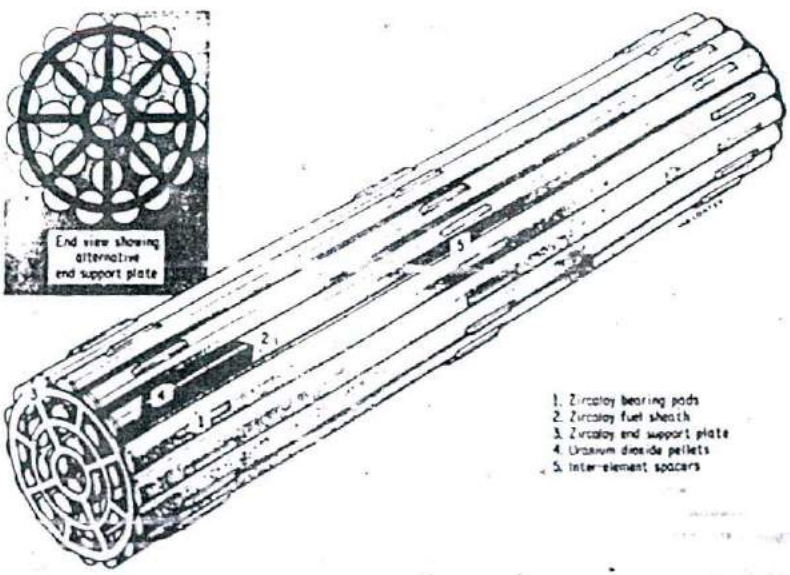


Figure 10-39 Candu 28-element fuel bundle.

The reactor is protected by 11 shutoff rods and by the moderator dump. The latter is made possible in this design by the fact that dumping the moderator would not starve the fuel from the coolant, a situation which would result in fuel meltdown. It is, however, resorted to only in case the shutoff rods fail to shut down the reactor.

The fuel is loaded and removed in the reactor *on power* by two coordinated fueling machines located at opposite ends of the reactor, each on the underside of a bridge. The bridges move vertically, whereas the machines move horizontally. In an on-power refueling operation, both machines locate, by remote control, onto an end fitting (3) of the selected channel. The machines pressurize themselves, then remove and store end closures and shield plugs. One machine then advances a fuel carrier containing the new fuel subassembly into the channel end fitting. Simultaneously the other machine extends an empty fuel carrier at the other end of the channel. The charge machine then rams the new fuel through a fuel latch into the channel, thus probing the spent fuel to enter the carrier in the discharge machine. The carriers are then withdrawn, the shield plugs and end closures are replaced, and the machines are detached from the end fittings. After the required number of bundles have thus been loaded, the discharge machine traverses to a spent-fuel port, where it jettisons the spent fuel to an underwater storage canal via a transfer mechanism and a conveyor.

PROBLEMS

10-1 A PWR powerplant pressurizer operates at a steady state at 2200 psia with a constant spray flow of 8 lb_m/min from the cold leg at 552°F. Calculate the amount of heat added by the electric heaters, in kilowatts, if the pressurizer heat losses to the ambient are 6144 Btu/h.

10-2 A pressurized water reactor has inlet and exit water at 290 and 320°C, respectively. It has a 30 m^3 vapor pressurizer which is normally 60 percent full of water at a pressure of 140 bar. A case of an insurge occurred during which 0.25 m^3 of water entered the pressurizer from the primary circuit hot leg, 0.05 m^3 entered through the spray, and 50 kWh was added by the electric heaters. Determine the internal energy of the pressurizer contents before and after the event, in kilojoules. Ignore heat losses to the ambient.

10-3 A pressurized-water reactor operating at 2000 psia has primary water entering at 550°F and leaving at 610°F. A 1000-ft^3 pressurizer is normally half full of water. During a transient the pressure rose to 2100 psia, 200 lb_m of spray water entered the pressurizer, and the pressurizer became 60 percent full of water. Ignoring heat losses to ambient, calculate the amount of heat added by the electric heaters, in kilowatt hours.

10-4 A PWR operates at 2000 psia and 600°F hot leg. It has a 1000 ft^3 pressurizer which is normally half full of water. The primary loop is 10,000 ft^3 in volume. The hot leg temperature suddenly rose to 605°F. Calculate (*a*) the final composition of the pressurizer, and (*b*) the final system pressure. Ignore spray flow and heat.

10-5 Repeat Prob. 10-4, except for the case when the hot-leg temperature dropped to 595°F.

10-6 It is desired to design a vapor pressurizer for a PWR that operates normally at 2200 psia. The primary loop has a volume of 8000 ft^3 and a hot-leg temperature of 610°F. The pressurizer is to be normally 60 percent full of water and designed to prevent a system pressure rise of more than 1 percent for a 10°F hot-leg temperature rise. Ignoring the spray and heat added or lost to the ambient, calculate the necessary pressurizer volume.

10-7 Derive Eq. (10-7) for gas pressurizers.

10-8 Repeat Prob. 10-6, but design a gas pressurizer. Assume two gases (*a*) argon and (*b*) helium, and calculate the pressurizer volume for each with the polytropic exponent corresponding to isothermal and to adiabatic reversible processes. Compare these volumes to that of the vapor pressurizer of Prob. 10-6.

10-9 A sodium-cooled reactor has 5000 ft^3 of primary coolant at an average temperature of 1400°R. A cover gas of argon acts as a pressurizer. Calculate the volume of argon necessary to limit the pressure rise to 1 percent for a coolant temperature rise of 100°F. Use a polytropic exponent corresponding to an (*a*) isothermal and (*b*) adiabatic reversible process.

10-10 A sodium-cooled reactor has 20,000 ft^3 of primary coolant at 600 psia and 1300°R average, and 8000 ft^3 of argon cover gas. Find the pressure rise, in pounds per square inch, if the coolant average temperature rose to 1400°R. Assume that the gas is compressed adiabatically and reversibly.

10-11 A PWR primary loop is 8000 ft^3 in volume and operates at an average temperature of 580°F. It has a 1000-ft^3 vapor pressurizer which normally contains 60 percent water by volume at 2200 psia. An accident occurred in which the relief valve suddenly stuck in an open position and fluid discharged to the relief tank. The system pressure steadily dropped to 1600 psia, during which time the electric heaters were fully activated to help slow down the rate of pressure drop. At 1600 psia, the primary loop average temperature was 550°F, the pressurizer was 95 percent full of steam, the heaters were turned off to protect them against overheating, and the emergency core cooling system (ECCS) was activated. This replenished the primary loop with water to prevent uncovering and damage to the fuel elements. The relief tank is assumed to remain at nearly atmospheric pressure, but there is a 15.3-psi pressure drop in the line connecting it to the pressurizer relief valve. Ignoring the effect of spray and heat losses to ambient, calculate (*a*) the initial mass composition of the pressurizer, in pounds mass, (*b*) the condition of the fluid leaving the relief valve at the instant it opened (pressure, temperature, and quality or degree of superheat), (*c*) the total loss of fluid from the primary loop (before ECCS) assuming for simplicity that its temperatures remained the same, in pounds mass, and (*d*) the condition of the fluid leaving the relief valve at the instant the ECCS came on the line.

10-12 A PWR powerplant producing 10^7 lb/h of 1100 psia saturated steam uses reheat from live steam (similar to that in Fig. 10-14). The high- and low-pressure turbines exhaust at 250 and 1 psia, respectively. For simplicity, assume that both turbines and pump are adiabatic reversible, that there are no feedwater heaters, and that the reheater drain is returned to the steam generator. Calculate (*a*) the fraction of live steam that is diverted to the reheater, if steam is reheated to 550°F, (*b*) the cycle net power, in megawatts, and (*c*) the cycle efficiency.

10-13 Calculate the gross station heat rate, in Btus per kilowatt-hour, and the thermal efficiency of the PWR powerplant shown in Fig. 10-14.

10-14 A chemically shimmed PWR uses 3.50 percent enriched fuel with a water-to-fuel ratio of 2.9. It has 4 percent worth control rods which are 30 percent inserted in the core. The boron concentration is 1600 ppm, hot conditions. Find (*a*) the axial to average power density ratio, (*b*) the total core reactivity, percent, and (*c*) the rod insertion and approximate boron concentration that would result in a 5 percent reduction in axial to average power density for the same total core reactivity as in *b*.

10-15 A chemically shimmed PWR uses 3.05 percent enriched fuel and a water-to-fuel ratio of 3.4. It is required to vary the reactivity of the core by 1 percent with rod insertions of 30 percent minimum and 60 percent maximum, without causing the maximum-to-average power density to exceed 1.72. Find (*a*) the total rod worth, percent, and (*b*) the boron concentration that would be required for a total core reactivity of 15 percent (hot conditions).

10-16 A boiling-water reactor operating at a pressure of 70 bar produces 1200 kg/s of saturated steam from feedwater at 200°C. The average core exit quality is 10 percent. Calculate (*a*) the recirculation ratio, (*b*) the core inlet enthalpy, in kilojoules per kilogram, and temperature, in degrees Celsius, (*c*) the degree of subcooling, in degrees Celsius, and (*d*) the heat generated in the reactor, in megawatts.

10-17 A 1000 MW boiling-water reactor powerplant with 33 percent efficiency was operating at a 75 percent of rated load with a steam mass flow rate of 1150 kg/s, a reactor core pressure of 70 bar, and an average exit quality of 13.6 percent. The plant uses recirculation control. Find (*a*) the feedwater temperature, in degrees Celsius, (*b*) the core degree of subcooling, in degrees Celsius, (*c*) the downcomer flow at 75 percent load, (*d*) the average exit quality immediately after initiation of load change to 80 percent, and when load has changed to 80 percent, and (*e*) the steam and downcomer flows, in kilograms per second after load change.

10-18 A forced-recirculation BWR core operating at 1000 psia generates 2500 MW. The core average exit quality is 10 percent. The average core and downcomer densities are 40 and 50 lb/ft^3, respectively. The

core is 12 ft high. Feedwater is added at top of downcomer. The total flow pressure losses are 6 psi. Find: (*a*) the core inlet temperature and enthalpy, (*b*) the steam generated, in pounds mass per hour, (*c*) the natural driving pressure, in pounds force per square inch, and (*d*) the pump power in kilowatts required to supplement the natural driving pressure if its efficiency is 0.75. (Note power = $\dot{m}\,\Delta P/\rho\eta$.)

10-19 Calculate the gross station heat rate, in Btus per kilowatt hour, and the thermal efficiency of the BWR powerplant shown in Fig. 10-27.

10-20 A pebble-bed reactor has a 14 ft diameter, 15-ft-high core containing 2.4-in-diameter fuel elements. The helium coolant flow is 1.36×10^6 lb$_m$/h at a pressure of 40 atm and core inlet and outlet temperatures of 520 and 1380°F, respectively. The blowers are 70 percent efficient and the pressure drop in the primary system is 50 percent greater than in the core alone. Find (*a*) the blower work, in megawatts, (*b*) the core heat generation, in megawatts, and (*c*) the steam generator capacity, in megawatts. (The pressure drop in a randomly packed pebble reactor in pounds force per square foot is given by

$$1.56 \times 10^{-6} (H/d^{1.27}) (\mu^{0.27}/\rho) G^{1.73}$$

where H = core height, ft
d = pebble diameter, ft
μ = gas viscosity, lb$_m$/ft h
ρ = gas density, lb$_m$/ft^3
G = apparent gas mass velocity, lb$_m$/h · ft^2, based on the total bed cross-sectional area)

CHAPTER

ELEVEN

FAST-BREEDER REACTORS AND POWERPLANTS

11-1 INTRODUCTION

The growth of nuclear power in the world's electric utility industry to date has primarily relied upon water-cooled and gas-cooled thermal neutron reactors. These are burners that in effect consume fissionable fuel with low conversion ratios (Sec. 11-2). The expected continued growth in the use of nuclear power and consequent consumption of available nuclear fuels have prompted the reevaluation of long-term goals. Ways of conserving uranium reserves and of keeping fuel-cycle costs down as fuel costs go up are therefore being sought. The fast-breeder reactor is the logical step in that direction.

Fast reactors are those whose neutrons are not slowed down to thermal energies by a moderator. Coolant and other reactor materials, however, moderate the neutrons to a certain extent in a fast reactor so that the neutron spectrum extends from fission energies that may be as high as 17 MeV but average about 2 MeV down to about 0.05 or 0.1 MeV. A *hard-spectrum* fast reactor is one in which the neutron density distribution extends over a narrow range nearer the high end of the energy spectrum. A *soft-spectrum* fast reactor, on the other hand, is one in which more moderation occurs, such as by liquid sodium and oxide fuels; hence the distribution extends to lower energies. A breeder reactor is one in which more fissionable fuel is produced than is consumed, or one that has a conversion, or breeding, ratio greater than 1.

Present-day burner reactors produce some Pu^{239} from the U^{238} in their fuel. Some of this plutonium is burned during the life of the core, but some remains. The remainder can be used to fuel initial fast-breeder reactors in a reactor "mix" until a self-supporting fast-breeder-reactor system is at hand. Such a system would convert the abundant U^{238}

(99.3 percent of all natural uranium) to fissionable Pu^{239} and hence extend our fuel reserves for centuries to come.

Table 11-1 lists the U.S. energy reserves as compiled by the Departments of Commerce and Energy in the late 1970s. The figures are best estimates and could be off by a factor of 2. This does not materially alter the conclusions, which are equally valid for the rest of the world.

A quad (Q) is a quadrillion, or 10^{15}, Btu, or roughly 10^{18} J. At present, U.S. energy consumption is about 60 Q/year, of which about 20 Q are consumed in the generation of about 8 Q of electricity. The balance is used for domestic services, industrial processing, and transportation. The "Years" column in Table 11-1 shows the time before depletion at present rates of consumption, if the particular fuel cited is used alone for *all* energy. "Present reactors" in the "Remarks" column refers to water-cooled thermal reactors (PWR and BWR). With breeding, our fuel resources are extended about 55 times from 50 to 2750 years. If an economical way is found to mine low-grade ores containing 1/55 as much uranium, we would have another factor of 10 to 12 to over 30,000 years. If the 0.003 ppm known to exist in sea water were extracted (the means are not yet available), the uranium reserves would be sufficient to supply the world for 1 or 2 million years. (Here would be a good use of solar energy to evaporate water from the sea.) These figures may seem overly optimistic, but they do dramatize the need to develop the fast-breeder reactor as a necessary power producer for the next century and probably beyond. Fusion has not yet been scientifically demonstrated and is not expected to make a big dent in that picture for some time to come.

Fast reactors have been built in the United States, the United Kingdom, the Soviet Union, France, Japan, and Germany, and Italian and Indian efforts are underway (Table 11-2). Some early reactors were cooled by NaK or mercury, but the majority are cooled by Na. Fuels varied from uranium and plutonium in the form of metal alloys, oxides, or carbides. The majority now favor the mixed-oxide fuel PuO_2 - UO_2.

The first fast reactor was built in the United States in 1946. Called Clementine,*

* Named after the ballad "In a cavern, in a canyon . . . Excavating for a mine . . . dwelt a miner, forty niner . . . and his daughter Clementine. . . ." 49 was the code number for plutonium in World War II, and the reactor was built in a cavern.

Table 11-1 Energy reserves in the United States

Fuel	Quads	Years	Remarks
Coal	5000	60	
Oil	500	6	
Gas	500	6	
U_3O_8	1600	20	No breeding
U_3O_8	4000	50	Present reactors
U_3O_8	220,000	2750	Breeding; high-grade ore
U_3O_8	>2,500,000	>30,000	Breeding; all economic ore

Table 11-2 The world's fast-breeder reactors

Country	Reactor	Power MW(t)	Power MW(e)	Coolant	Design	Initial fuel	Status*
USA	Clementine	0.025	—	Hg		Pu metal	1946,D
	LAMPRE-I	1	—	Na		Molten Pu	D
	EBR-I	1.4	0.2	NaK		U metal	1951,D
	EBR-II	62.5	16.0	Na	Pool	U metal	1963,O
	Enrico Fermi	200	62	Na	Loop	U metal	1966,D
	SEFOR	20	—	Na	Loop	PuO_2-UO_2	D
	FFTF	400	—	Na	Loop	PuO_2-UO_2	1980,O
	CRBR	975	350	Na	Loop	PuO_2-UO_2	1987,C
	CDS	2550	1000	Na	Loop	PuO_2-UO_2	1992,P
USSR	BR-1	0	—			PuU metal	D
	BR-2	0.1	—	Hg		Pu metal	D
	BR-5/10	10	—	Na	Loop	Pu metal	1959,O
	BOR-60	60	12	Na	Loop	PuO_2-UO_2	1969,O
	BN-350	1000	350	Na	Loop	PuO_2-UO_2	1972,O
	BN-600	1470	600	Na	Pool	PuO_2-UO_2	1980,O
	BN-1600-1	4200	1600	Na	Pool		1990,P
FRANCE	Rapsodie	40	—	Na	Loop	U metal	1967,O
	Phénix	563	250	Na	Pool	PuO_2-UO_2	1973,O
	Super Phénix I	3000	1200	Na	Pool	PuO_2-UO_2	1983,C
	Super Phénix 2&3	3750	1500	Na	Pool	PuO_2-UO_2	1990-91,P
	Super Phénix 3&4	3750	1500	Na	Pool	PuO_2-UO_2	1992-93,P
UK	Dounreay	60	13.5	NaK	Pool	U metal	1962,D
	PFR	600	250	Na	Pool	PuO_2-UO_2	1974,O
	CDFR-1	3250	1300	Na	Pool	PuU metal	1992,P
GERMANY	KNK-2	58	20	Na	Loop	UO_2	1977,O
(FRG)	SNR-300	762	327	Na	Loop	PuU metal	1985,C
	SNR-2	3250	1300	Na	Loop	PuU metal	1993,P
JAPAN	JOYO	75–100	—	Na	Loop	PuO_2-UO_2	1977,O
	MONJU	714	300	Na	Loop	PuO_2-UO_2	1987,C
	DFBR	2800	1000	Na	Pool	PuO_2-UO_2	1993,P
ITALY	PEC	135	—	Na	Loop	PuO_2-UO_2	1983,C
INDIA	Madras FBTR	42	17	Na	Loop	PuO_2-UO_2	1983,C

* Dates of initial or expected operation; C-under construction; D-decommissioned; O-operating, P-in the planning-design stage.

it was plutonium-fueled, produced 25 kW(t), was mercury-cooled (one of only two to date), and has long been shut down. The main initial effort was the Experimental Breeder Reactors EBR-I and EBR-II [102]. The 200 kW(e) EBR-I was the first reactor of any kind to produce electricity in the United States. Construction began in November 1947; it achieved criticality in August 1951 and produced electric power in December 1951. It was NaK-cooled and used a stainless steel–clad fully enriched 2 percent zirconium-uranium alloy for fuel and a stainless steel–clad natural-uranium external

blanket. It was discontinued after demonstrating the feasibility of fast reactors and breeding. EBR-II is a 62.5-MW(t), 20-MW(e), sodium-cooled reactor fueled with stainless steel–clad 49.5 percent enriched-uranium-alloy. The blanket is stainless steel–clad depleted uranium. EBR-II has an integrated fuel manufacturing and reprocessing facility and will operate on a fuel-recycle basis, ultimately with plutonium. These reactors were followed by Enrico Fermi [3] and the experimental reactors SEFOR [3] and FFTF. A 350-MW(e) demonstration plant, the Clinch River Breeder Reactor (CRBR), was under construction in the early 1980s (Sec. 11-4).

The British effort is centered around Dounreay [103] and PFR. The first is NaK-cooled with tubular fuel clad in vanadium on the inside and partly finned niobium on the outside. Natural uranium is used in external breeder blankets surrounding the core. Control is achieved by axially moving 12 fuel assemblies (of 10 fuel rods each), situated around the periphery of the core. These are divided into two safety, six control, and four shutoff assemblies. A large 1300-MW(e) reactor, CDFR-1, is in the planning-design stage.

The USSR effort includes three experimental reactors built since 1955. BR-1 is PuU-fueled and U-reflected, designed to carry out cross-sectional measurements at various neutron energies. BR-2, Pu-fueled and mercury-cooled, was used to test nuclear parameters, breeding ratios, and liquid metals. BR-5 (upgraded to BR-10) is UC-fueled, sodium-cooled, with a uranium-nickel blanket. The program includes three operating reactors: BOR-60; BN-350, a dual-purpose reactor that produces 150 MW(e) and process steam for a 120,000-m^3/day water desalination plant; and BN-600, a 600-MW(e) power reactor. A 1600-MW(e) plant, BN-1600-1, is in the planning-design stage.

France has what must be considered the most ambitious program. It started with Rapsodie, a 20-MW(t) sodium-cooled and plutonium-uranium-oxide-fueled reactor. A 600-MW(t), 250-MW(e) reactor, the Phénix,* has been in operation since 1973 and has given excellent service. Under construction and in the planning stages are the Super Phénix class of reactors (Sec. 11-5).

Gas-cooled fast-breeder reactors are receiving some attention in the United States and elsewhere (Sec. 11-6).

11-2 NUCLEAR REACTIONS, CONVERSION, AND BREEDING

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 1 eV. In a fast-breeder reactor, the objective is to maintain a chain reaction with fast neutrons that have an average energy of about 1 MeV by fission in U^{235} and Pu^{239}. It also must provide additional fast neutrons sufficient to convert U^{238} to Pu^{239}.

* Named after the mythical, beautiful, lone Egyptian bird which lived in the desert for 500 or 600 years and consumed itself in fire, then rose from the ashes to start another long life. It has become a symbol of immortality.

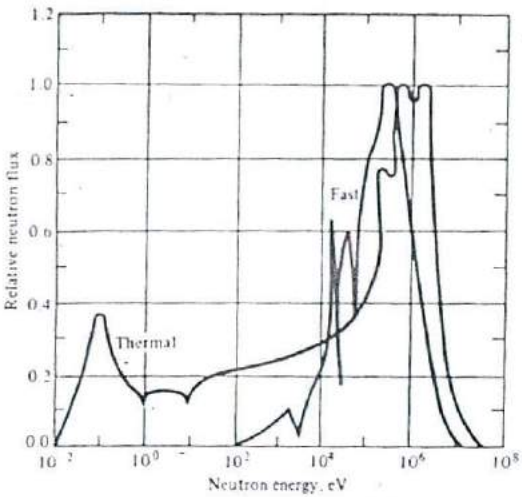


Figure 11-1 Typical neutron flux spectra for fast (LMFBR) and thermal (PWR) reactors.

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

$$_{92}U^{235} + {}_{0}n^{1} \rightarrow {}_{56}Ba^{137} + {}_{36}Kr^{97} + 2{}_{0}n^{1} \qquad (11\text{-}1)$$

$$_{92}U^{233} + {}_{0}n^{1} \rightarrow {}_{56}Ba^{136} + {}_{36}Kr^{96} + 2{}_{0}n^{1} \qquad (11\text{-}2)$$

and

$$_{94}Pu^{239} + {}_{0}n^{1} \rightarrow {}_{56}Ba^{137} + {}_{38}Sr^{100} + 3{}_{0}n^{1} \qquad (11\text{-}3)$$

A typical nonfission reaction is the one that occurs in the sodium coolant, which is composed of 100 percent Na^{23}

$$\left.\begin{aligned} {}_{11}Na^{23} + {}_{0}n^{1} &\rightarrow {}_{11}Na^{24} \\ {}_{11}Na^{24} &\xrightarrow[\beta]{14.8\ h} {}_{12}Mg^{24} + {}_{-1}e^{0} \end{aligned}\right\} \qquad (11\text{-}4)$$

where Na^{24} is a highly radioactive isotope that emits 2.76 MeV γ radiation and 1.3 MeV β decays with a 14.8-h half-life to stable Mg^{24}.

A fast neutron reaction with U^{238} results in a series of reactions that culminate in Pu^{239}, shown in Fig. 11-2 and given by

$$\left.\begin{aligned} {}_{92}U^{238} + {}_{0}n^{1} &\rightarrow {}_{92}U^{239} + \gamma \\ {}_{92}U^{239} &\xrightarrow[\beta]{23\ min} {}_{93}Np^{239} + {}_{-1}e^{0} \\ {}_{93}Np^{239} &\xrightarrow[\beta]{2.3\ days} {}_{94}Pu^{239} + {}_{-1}e^{0} \end{aligned}\right\} \qquad (11\text{-}5)$$

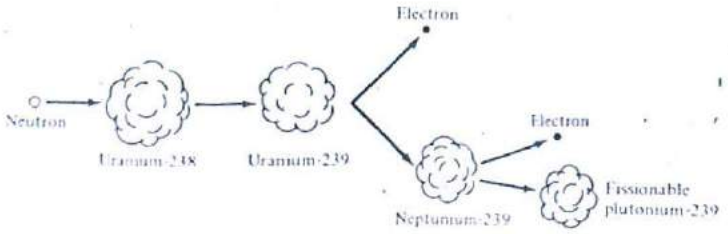


Figure 11-2 Schematic of the Pu^{239} breeding chain.

This is a *breeding* reaction that converts *fertile* U^{238} into *fissionable* Pu^{239}.* We will now define two important fuel parameters. These are:

ν = the average number of neutrons emitted per *fission*, as in Eqs. (11-1) to (11-3). ν 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). η is a strong function of neutron energy (Fig. 11-3).

The Conversion and Breeding Ratios

When a neutron is *absorbed* in the fuel, it produces η 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. The rest of the neutrons per neutron absorbed will be available for the breeding reaction, Eq. (11-5), and are called the *conversion* or *breeding* ratio C, or

$$C = \eta - 1 - L \tag{11-6}$$

The maximum possible C, C_{max}, is obtained if L were zero, or

$$C_{max} = \eta - 1 \tag{11-7}$$

Depending upon η 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 [104] that there is a maximum theoretical limit to the amount of fertile nuclei that can

* Some two billion years ago in what is now Gabon in Africa, large deposits of natural uranium ore, more enriched than now (about 3 percent), mixed with geologic water causing thermal fission. The low-power-density but huge natural reactors (at least six were identified) stayed critical for many centuries. Some U^{238} was converted to Pu^{239}, showing that not all plutonium is manufactured.

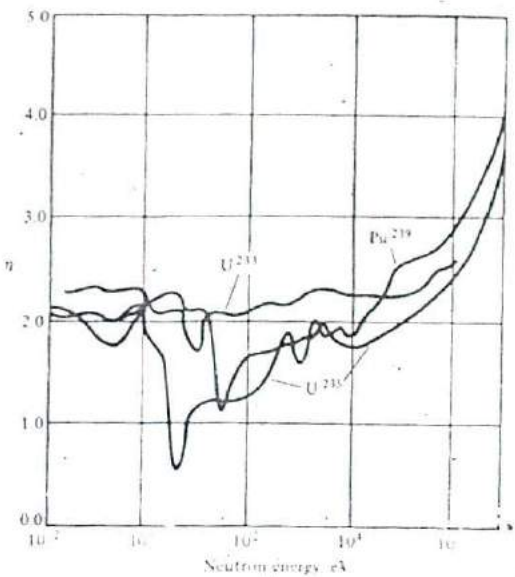


Figure 11-3 The variation of the fission factor η with neutron energy for three fissionable nuclei.

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 $C \geq 1$ 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

$$G = C - 1 = \eta - 2 - L \tag{11-8}$$

and a maximum gain, G_{max}, is

$$G_{max} = C_{max} - 1 = \eta - 2 \tag{11-9}$$

Average values of ν, η, and G_{max} are given in Table 11-3 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 occur in fast reactors fueled with Pu^{239} (and Pu^{241}, which is not readily available), followed by U^{233} and 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 11-3 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 θ, ΔN_b nuclei/cm^3, may be given by

$$\Delta N_b = \Delta N_{ff} G = \Delta N_{ff}(\eta - 2 - L) \tag{11-10}$$

but

$$\Delta N_{ff} = F_c (N_0)_{ff} (\sigma_a)_{ff} \bar{\phi}\, \theta \tag{11-11}$$

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, neutrons/s · cm^2

θ_d, called the *simple doubling time*, by definition, occurs when $\Delta N_b = (N_0)_{ff}$. Thus

$$\theta_d = \frac{1}{F_c (\sigma_a)_{ff} \bar{\phi} (\eta - 2 - L)} \tag{11-12}$$

Note that θ_d is shortened by operating at high power levels (high $\bar{\phi}$) and that it is also inversely proportional to the breeding gain ($\eta - 2 - L$). In practice θ_d slightly increases with 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 Pu^{239} 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*, θ_{dc}, related to θ_d by

$$\theta_{dc} = \frac{0.6931}{\ln [1 + (1/\theta_d)]} \tag{11-13}$$

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

11-3 LMFBR PLANT ARRANGEMENTS

Because sodium and other liquid metals suffer from high induced radioactivities (Sec. 11-4), and are generally chemically active, intermediate coolant loops are used between the primary radioactive coolant and the steam cycle (Fig. 11-4). 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.

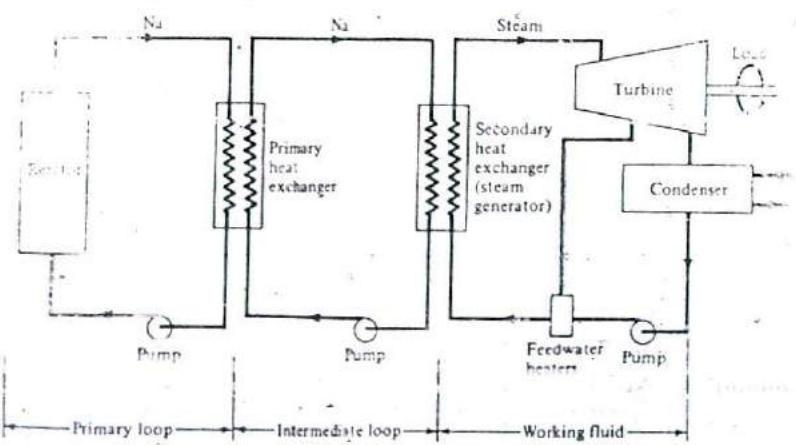


Figure 11-4 Schematic arrangement of a liquid-metal fast-breeder reactor (LMFBR) powerplant.

by the strong γ 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. 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 fires 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 pipeways, equipment cells, and of the large and complex structure resulting from the spread of the components. 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 filled 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.

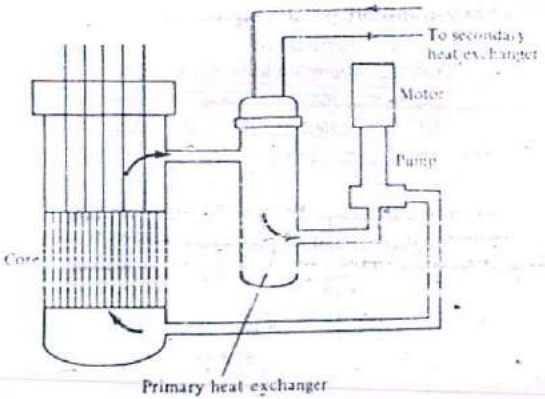


Figure 11-5 Schematic of a loop-type LMFBR.

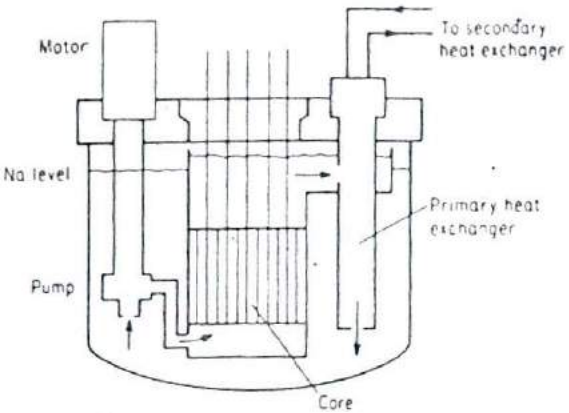


Figure 11-6 Schematic of a pool-type LMFBR.

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 Phénix, a French commercial-size plant of the pool design.

11-4 THE CLINCH RIVER BREEDER REACTOR PROJECT

The Clinch River Breeder Reactor (CRBR) project is a demonstration liquid-metal fast-breeder reactor plant. It is owned by DOE with Westinghouse as primary contractor with responsibility for reactor and primary heat-transfer system manufacturing. General Electric is responsible for the intermediate heat transport and steam-generation systems and turbine-generator manufacturing, and Rockwell International is responsible for refueling and auxiliary maintenance systems. Several subcontractors are entrusted with various component design and manufacturing. Burns and Roe is the architect-engineer, and Stone and Webster is the constructor. CRBR is situated on 100 acres of a 1364-acre TVA site on the Clinch River near Oak Ridge, Tenn. Figure 11-7 shows a general layout of the plant [105].

CRBR has been facing economic and political delays because of cost overruns and governmental caution against the universal production of plutonium and its implications regarding nuclear weapons proliferation. This argument may be answered by the fact that plutonium production can be accomplished easier, faster, and cheaper by means other than a complex LMFBR. Because of the need for fast-breeder reactors

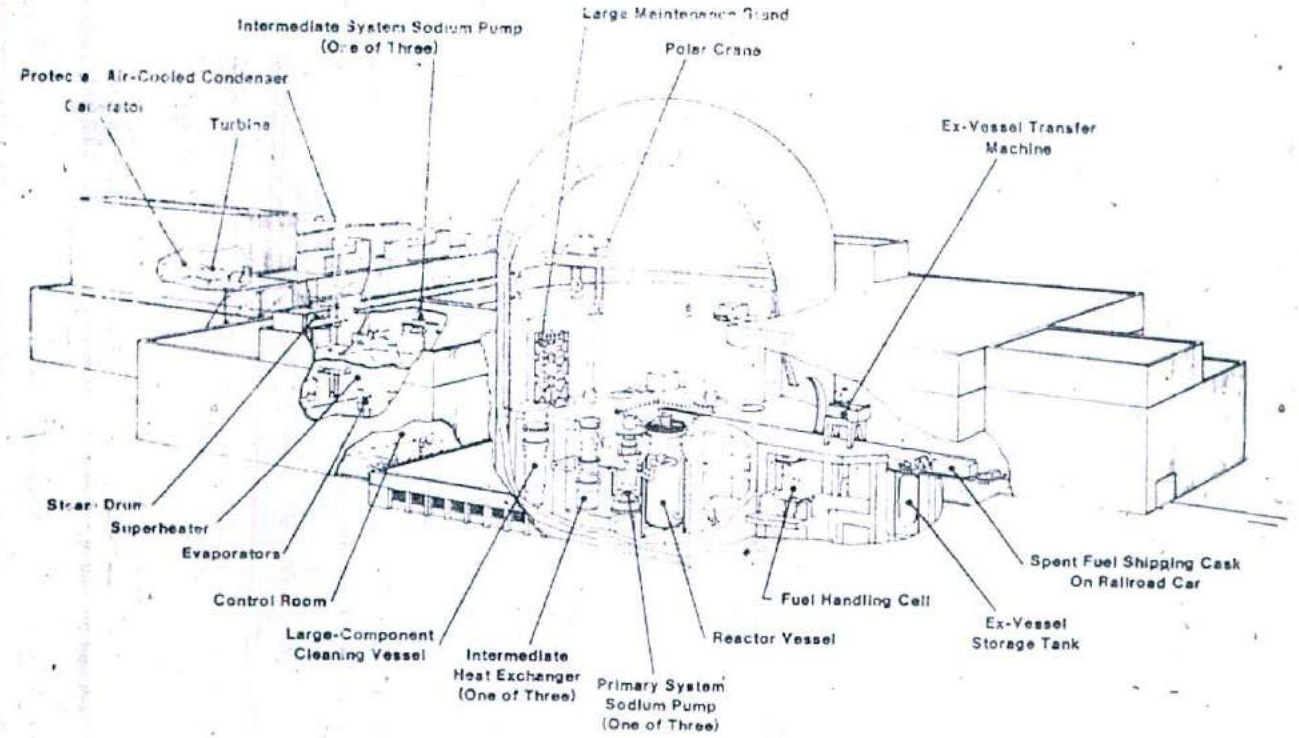


Figure 11-7 Layout of the Clinch River Breeder Reactor (CPBR) powerplant. *(Figures 11-7 through 11-18 courtesy Project Management Corporation.)*

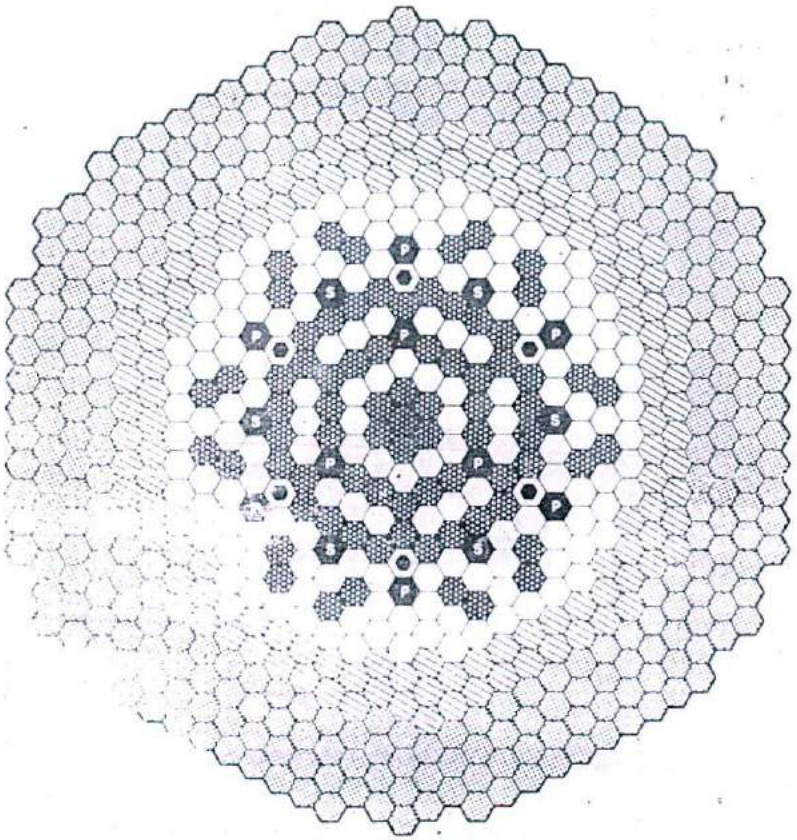
to expand our dwindling energy supplies, there is optimism that CRBR will eventually be completed. Initial startup is now planned for the late 1980s.

CRBR is a loop design, so chosen to utilize experience gained by Westinghouse from the Fast Flux Test Facility (FFTF), also a loop design. It is designed to produce 375 MW(e) gross from 975-MW(t) reactor thermal power. Table 11-4 lists some design data of CRBR.

Reactor core Figure 11-8 shows a cross section of the CRBR reactor core. It is composed of 156 fuel assemblies containing a mixed oxide fuel (PuO_2-UO_2), interspersed with 76 inner blanket assemblies, 6 assemblies that are used alternately for fuel or inner blanket, and 15 control assemblies. This mix is surrounded by 132 radial blanket assemblies which is turn are surrounded by 306 removable radial shield assemblies. All assemblies are contained within hexagonal ducts having the same external dimensions. Refueling is planned annually.

Table 11-4 Design data of the Clinch River breeder-reactor plant

Power	375 MW(e) gross, 975 MW(t)
Fuel	PuO_2-UO_2 mixed oxide, 1511 kg fissile Pu at beginning of life
Fuel elements	0.23-in (5.8-mm)-diameter rods, SS-316 clad, pitch-diameter ratio 1.25 spaced on a triangular array by spiral wire wraps, 217 rods per assembly, 156 assemblies
Blankets	Depleted UO_2, 76 inner blanket and 132 radial blanket assemblies
Control rods	Nine primary worth ~$22, six secondary worth ~$13
Core	6.2-ft (1.89-m) diameter, 3.0 ft (0.91 m) high
Maximum neutron flux	5.5×10^{15}
Power peaking factor	1.28 axial, 1.18 radial at beginning of cycle
Linear power rating	16 kW/ft (52.5 kW/m) peak, 14.3 kW/ft (46.9 kW/m) average
Fuel burnup	80,000 MW · day/ton initial core, 110,000 peak
Breeding ratio	1.24, 30-year doubling time
Core temperatures	Fuel cladding wall, max. 1215°F (657°C); sodium 730°F (388°C) [illegible] 995°F (535°C) outlet
Primary pumps (3)	33,700 gal/min (2.13 m^3/s) each, 458 ft (139.6 m) Na head, [illegible] drop 123 psi
Intermediate systems (3)	Cold leg 651°F (344°C), hot leg 936°F (502°C); pumps 29,500 gal/min (1.86 m^3/s) each; 330 ft (100.6 m) Na head
Steam systems (3)	Two evaporators and one superheater per loop, 3.53 × 10^6 lb$_m$/h [illegible] s) each, superheater outlet 906°F (486°C), 1535 psig (105.8 bar) feedwater 468°F (242°C), to steam separator drum, 548°F (287°C) evaporators
Turbine	3600 r/min tandem-compound, nonreheat, six extractions to feedheaters, total extracted steam 1.15 × 10^6 lb$_m$/h (145 kg/s)
Generator	Synchronous 3600 r/min, 485 MVA self-excited three-phase wye connected 60 Hz (6.17 m), 22,000 V_y 0.9 power factor
Dimensions	Reactor vessel 20.25-ft (6.17-m) diameter, 54.67 ft (16.66 m) high, intermediate heat exchangers, 8.8-ft (2.68-m) diameter, [illegible] m) high; steam generators 4.33-ft (1.32-m) diameter, 65 ft [illegible] high; containment 186 ft (56.7 m) diameter
Gross thermal efficiency	39%



Symbol	Component Name	Number
	Fuel Assembly	156
	Inner Blanket Assembly	78
	Radial Blanket Assembly	132
	Alternate Fuel/Blanket Assembly	6
	Primary Control Assembly	9
	Secondary Control Assembly	6
	Removable Radial Shield Assemblies	306

Figure 11-8 Cross section of the CRBR core.

The fuel-rod design is based on that of the FFTF. Each fuel assembly (Fig. 11-9) contains 217 0.23-in OD rods arranged in a triangular array and spaced by wire wraps of 12-in spiral pitch along the rod length, giving a rod pitch-to-diameter ratio of 1.26. Each fuel rod contains a 36-in-long stack of fuel pellets and two 14-in-long stacks of depleted UO_2 blanket pellets above and below the fuel pellets. The rods are 316 stainless steel and are capped at top and bottom, with a fission gas plenum at the top. An axial spring prevents pellet movement during handling, preirradiation, and shipping.

The inner and radial blanket assemblies (Fig. 11-10) contain 61 rods each, also on a triangular pitch and spaced by wire wraps. Each rod contains a 64-in-long stack of 0.470-in-diameter depleted UO_2 pellets. This length is equal to the total of fuel and upper and lower blankets in the fuel rods. The cladding is 0.506 in OD and 0.015 in thick. All blanket pellets breed plutonium from the depleted UO_2, generate some of the power, and provide some of the shielding.

The removable radial shield assemblies contain 19 rods each. Their principal function is to provide neutron shielding and to limit radiation damage to such reactor structures as the core barrel and pressure vessel. In addition, there is a fixed stainless steel segmented-ring radial shield located between the removable assemblies and the core barrel. All shields are designed to ensure a 10 percent residual ductility for a 30-year design life.

The control rod system is composed of nine primary and six secondary hexagonal assemblies. The primary assemblies (Fig. 11-11) contain 37 pins each. The neutron absorber is boron carbide, B_4C, 92 percent enriched in B^{10} and contained in a 36-in stack of pellets in each pin. The primary control rods are used for primary shutdown (scram) as commanded by a plant protection system and for reactivity control of the reactor. Their drive mechanisms are mounted on the intermediate rotating plug of the reactor closure head. The secondary control rods have the sole function of providing an independent backup shutdown for the primary rods. The two systems are designed to be independent and are not subject to the same mode failure. Either one can achieve reactor shutdown with the other completely inoperable and with one of its own control rods inoperative.

The core is supported by a 24-in-thick type 304 stainless steel plate that is welded to the core barrel and through which 61 lower sodium inlet modules are positioned (Fig. 11-12). Each module supports seven of the fuel, control, inner and radial blanket, and some of the removable radial shield assemblies, for a total of 427. The modules have seven different configurations to control the sodium flow distribution to the different assemblies. In addition, flow-control orifice cartridges provide intermediate-pressure sodium to the remaining shielding components and low-pressure sodium to the annulus between reactor vessel and core barrel.

Located directly above the core is a welded internal structure made of 316 stainless steel with local Alloy 718 protection. Its principal functions are to [illegible] core hold-down, for positioning instruments, mixing the core exit sodium and directing it to the upper region of the reactor vessel outlet plenum, and providing guidance and cross-flow protection for control-rod drivelines. A core restraint system, among other things,

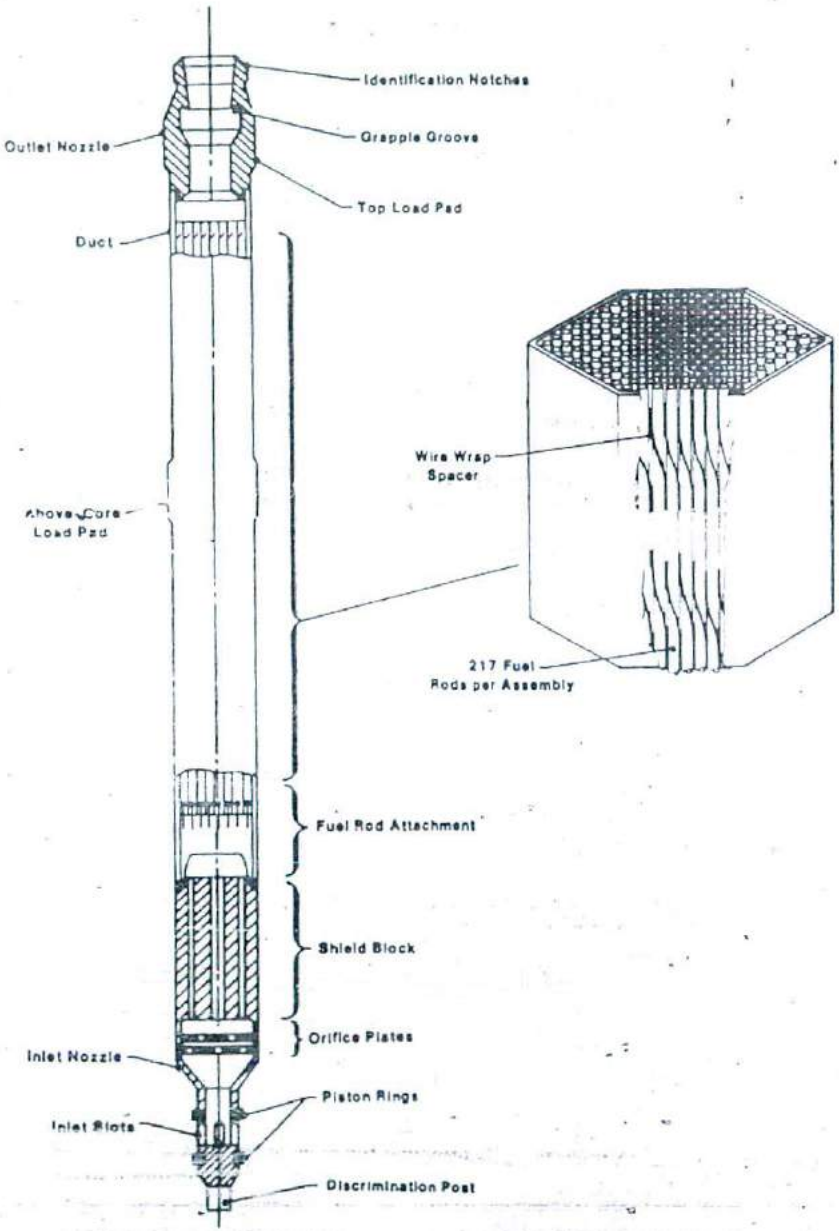


Figure 11-9 CRBR fuel rod and assembly.

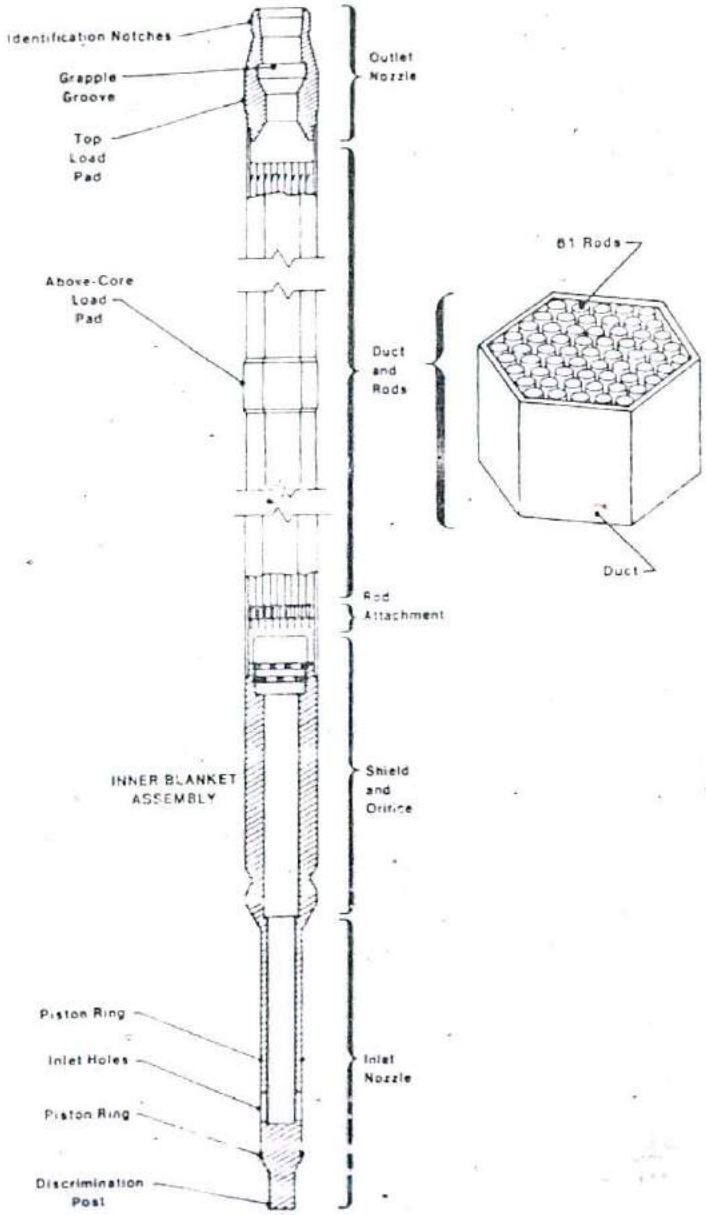


Figure 11-10 CRBR inner blanket assembly.

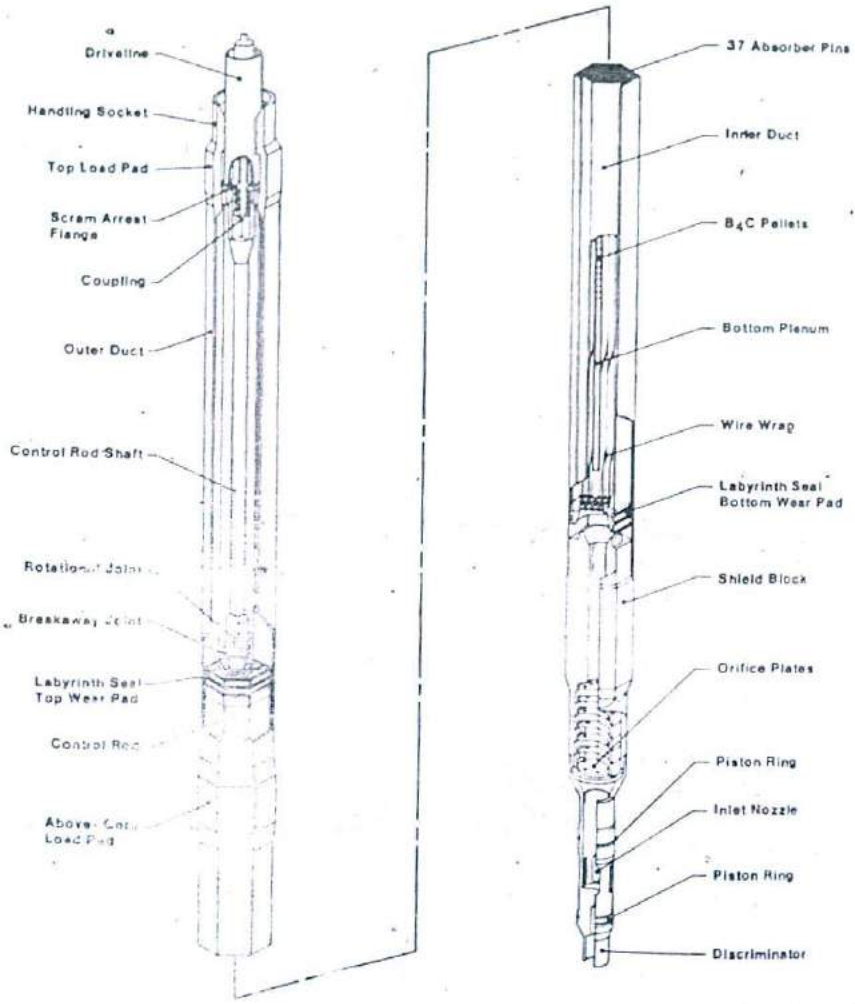


Figure 11-11 CRBR primary control-rod assembly.

maintains clearances between core assemblies that are subject to creep and swelling caused by fast neutron irradiation.

The reactor The reactor vessel (Fig. 11-12) is 59 ft (18 m) high, almost 27 ft (8.23 m) in diameter at its widest, and weighs 505 tons. Sodium enters through three 2-ft-diameter pipes and exits through three 3-ft-diameter pipes. Four smaller nozzles in

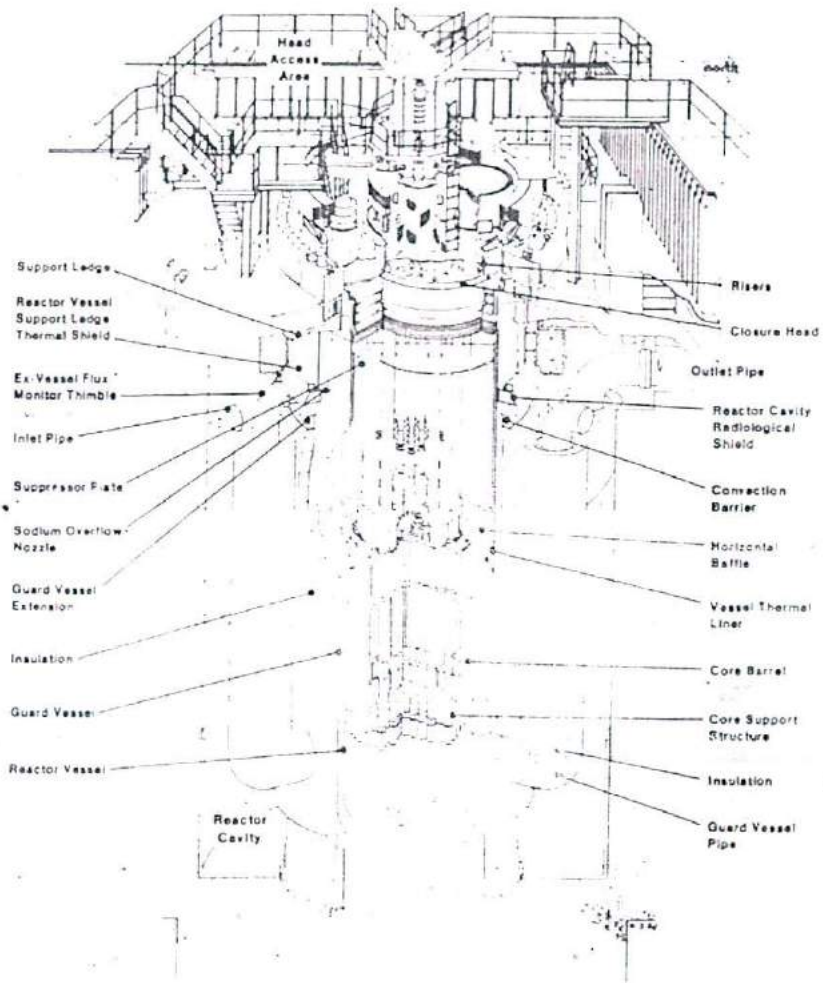


Figure 11-12 The CRBR reactor system

the upper regions are for entry and exit of cover gas, sodium overflow, and makeup flow. The vessel wall is kept below 900°F by directing 2 percent of sodium flow into the annulus between the vessel and a 20-ft-diameter, 25-ft-long stainless steel thermal liner. A 22-ft-diameter, 47-ft-high guard vessel surrounds the liner and provides secondary containment, which ensures that the core is always covered by sodium and that undamaged pipes can remove heat from it in the unlikely event of a leak from

the reactor vessel or piping. The whole reactor system is situated in a 40-ft-diameter reactor cavity.

The upper closure head assembly (Fig. 11-13) contains various penetrations such as those for the control rod assemblies, as well as three independently rotatable plugs for positioning components of the refueling system. The smallest plug is nested within and eccentric to the intermediate plug, which, in turn, is nested inside and eccentric to the large plug. The plugs rotations in respect to the reactor vessel and one another provide straight pull access to fuel and other removable components.

The heat-transfer system The reactor is connected to three primary loops, each in turn connected to one of three independent intermediate loops (Fig. 11-14). In each primary loop, sodium enters the reactor vessel at 730°F (388°C) and exits at 995°F (535°C). A primary sodium pump, located in the hot leg, pumps the hot sodium to an intermediate heat exchanger (IHX) at the rate of 33,700 gal/min (2.126 m^3/s) and a pressure of 175 psi (12 bar). The primary sodium leaves the IHX and returns to the reactor via a check valve and a flow meter. The primary piping is 1/2-in-thick welded stainless steel and has a 3-ft diameter between reactor and pump and 2-ft diameter between pump and IHX and between IHX and reactor. The primary pumps, IHXs, and some associated piping are surrounded with guard vessels for the same reasons

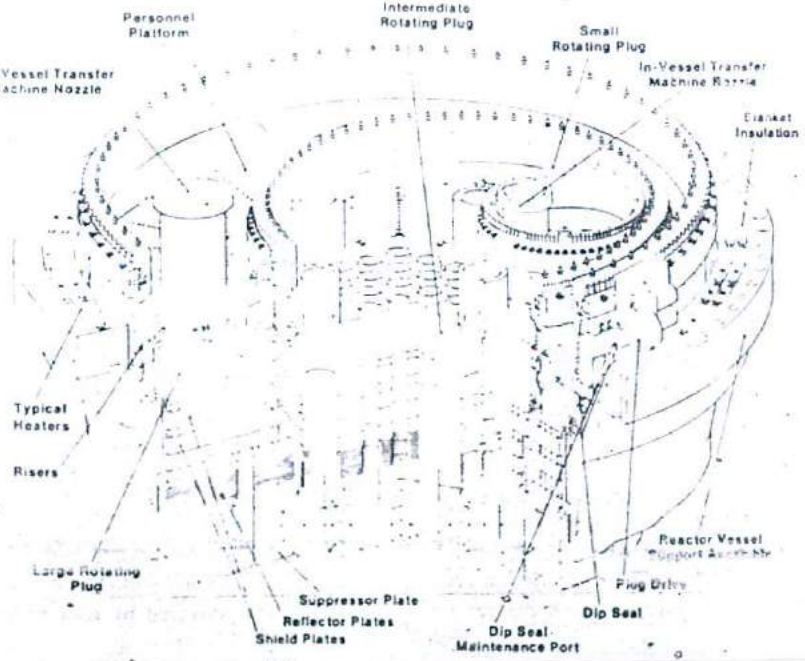


Figure 11-13 The CRBR reactor closure head assembly.

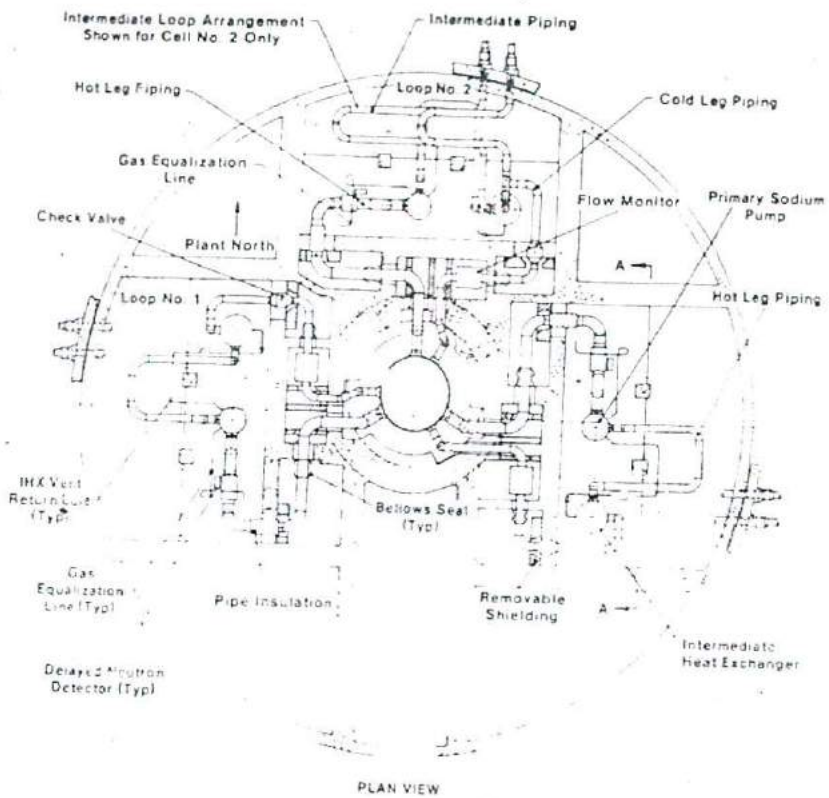


Figure 11-14 Plan view of the CRBR primary heat-transport system.

as the reactor vessel. The whole reactor system is situated in a 40-ft-diameter reactor cavity. Argon is used as cover gas for sodium in the reactor and other components. (Argon is also used as cover gas for NaK, fuel handling cell atmosphere and other places, while nitrogen is provided for inerted cell atmospheres and other functions. An elaborate gas receiving, processing, and decontamination system is provided.)

The sodium in each intermediate loop enters the IHX at 651°F (344°C) and exits at 936°F (502°C) and 180 psi (12.4 bar) to go to the superheater, which it leaves at 885°F (474°C). It then divides into two paths, each of which enters an evaporator. Sodium leaving both evaporators is combined and pumped by an intermediate cold-leg pump back to the IHX via a flow meter at 29,500 gallons (111.6 m³/s) and 195 psi (14.5 bar). Thus there are two pumps, one superheater, two evaporators, and a steam drum (below) for each of the three loops. The higher pressure of the intermediate sodium in the IHX ensures that any leakage would be from the intermediate to the primary side, thus preventing contamination by the highly radioactive primary sodium. All intermediate-loop piping is 1/2-in-thick welded stainless steel 2 ft in diameter.

The total mass of operating sodium in the plant is 2,368,000 lb_m (1,074,107 kg). This compares with 3,500,000 kg of sodium in the primary system plus 1,500,000 kg in the secondary system of the pool-type Super Phénix (Sec. 10-6).

The primary- and intermediate-loop components are situated so that each component is above the one preceding it in the loop. This provides natural circulation of the sodium to remove decay heat from the core in the event of pump failure.

The intermediate heat exchangers (IHX) The three intermediate heat exchangers (Fig. 11-15) are vertical, counterflow, and shell-and-tube units. They are rated at 325 MW(t) each, are about 52 ft (15.88 m) high and 8.75 ft (2.667 m) in diameter, and weigh 115 tons, dry. Each contains 2850 tubes that are 0.875 in (2.225 cm) OD, 0.045 in (1.143 mm) thick, 25.8 ft (7.86 m) in active length, and made of 304 and 316 stainless steel.

Hot primary sodium enters through a nozzle near the middle and is directed by a bypass seal assembly upward through a distribution cylinder at the top of the tube bundle. There it reverses direction, flows down on the shell side (outside the tubes), and exits at bottom center back to the reactor. Baffles are spaced along the path to distribute the flow and act as lateral tube restraints.

Intermediate sodium, pumped from two evaporators, enters at the top, flows down a central downcomer to the bottom, reverses direction and flows upward through the tubes to the upper plenum, and leaves through a side exit nozzle near the top to the superheater.

The steam-generating system Three steam loops (Fig. 11-16) are used. In each, a steam drum receives heated feedwater at 468°F (242°C) and 2032 psi (140.1 bar) from the steam power cycle, as well as saturated steam-water mixture from two evaporators. In the drum, steam separates from water and goes to the superheater. The water remaining is force-recirculated to the two evaporators by a recirculating pump (not unlike a forced external recirculation BWR, Sec. 10-7) and enters them at 548°F (287°C) and 2034 psi (140.2 bar). There, it receives heat from the intermediate sodium and leaves the drum as a mixture of steam and water at 628°F (331°C) and 1896 psi (130.7 bar) of 50 percent quality at full load, corresponding to a recirculation ratio of 2:1.

The separated saturated steam enters the superheater at 625°F (329°C) and 1851 psi (127.6 bar) where it receives heat from the hot intermediate sodium and leaves at 906°F (486°C) and 1450 psi (100 bar).

The evaporators and superheaters All evaporators and superheaters are essentially identical in design to minimize cost and spare parts (Fig. 11-17). They are shell-and-tube counterflow "hockey stick" design heat exchangers that are vertical except for a 90° bend at the top. This provides for differential thermal expansion between the tube bundle and shell. Each is 65 ft (19.8 m) high and 4.33 ft (1.32 m) in diameter and weighs 115 tons, dry. Each contains 757 tubes that are 5/8 in (1.5875 cm) OD, 0.109 in (2.7686 mm) thick, 46 ft (14.02 m) in active length, and made of Cr-Mo steel.

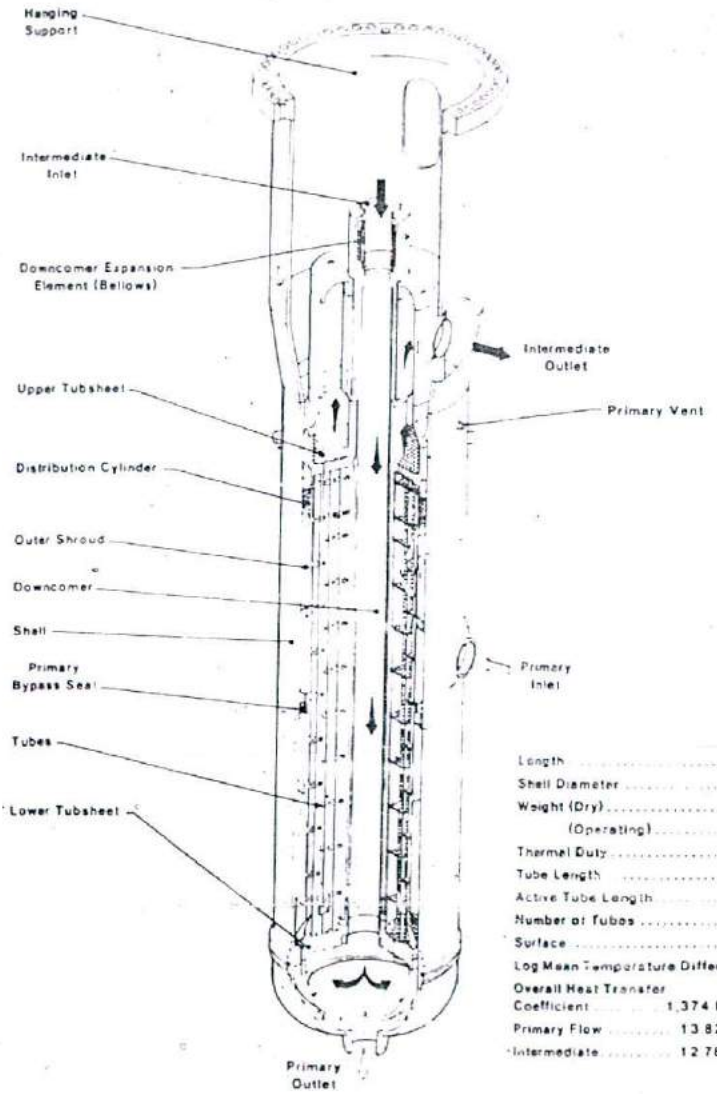


Figure 11-15 The CRBR intermediate heat exchanger (IHX).

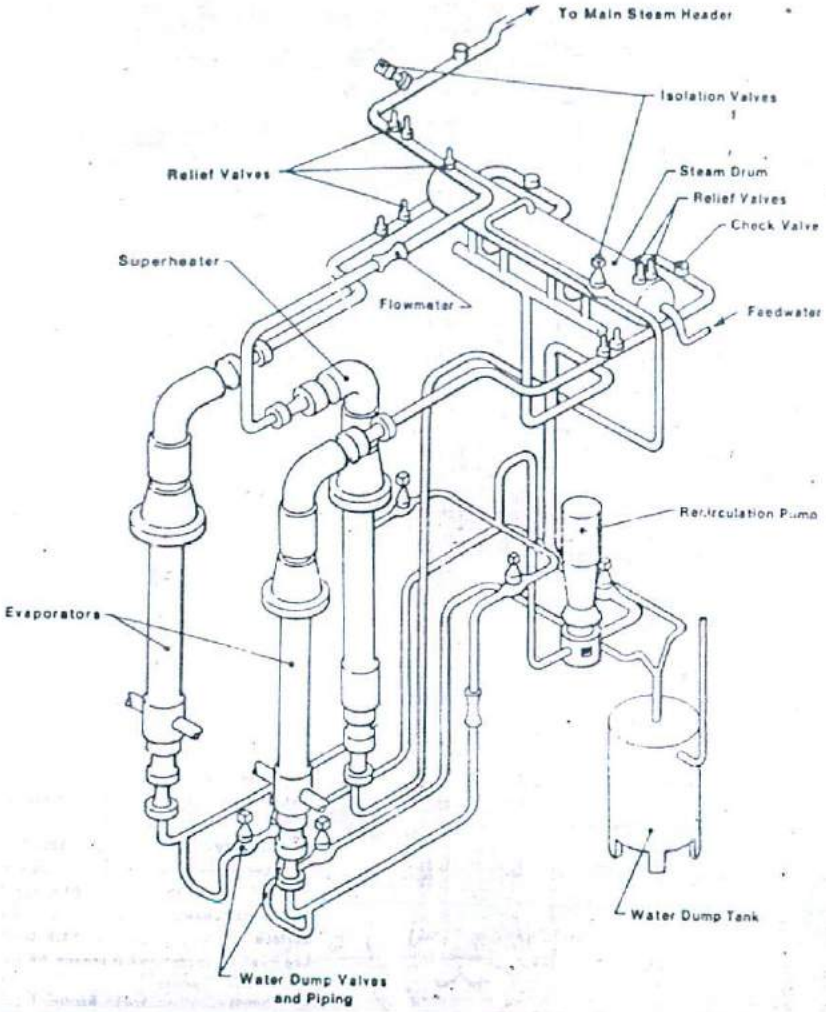


Figure 11-16 CRBR steam-generation loop (one of three).

Intermediate sodium from the IHX first enters the superheater, and sodium leaving the superheater enters each of the two evaporators, through a nozzle just below the bend, flows down on the shell side, and exits from two nozzles near the bottom. As in the IHX, baffles are spaced in that path to distribute the sodium and support the tubes.

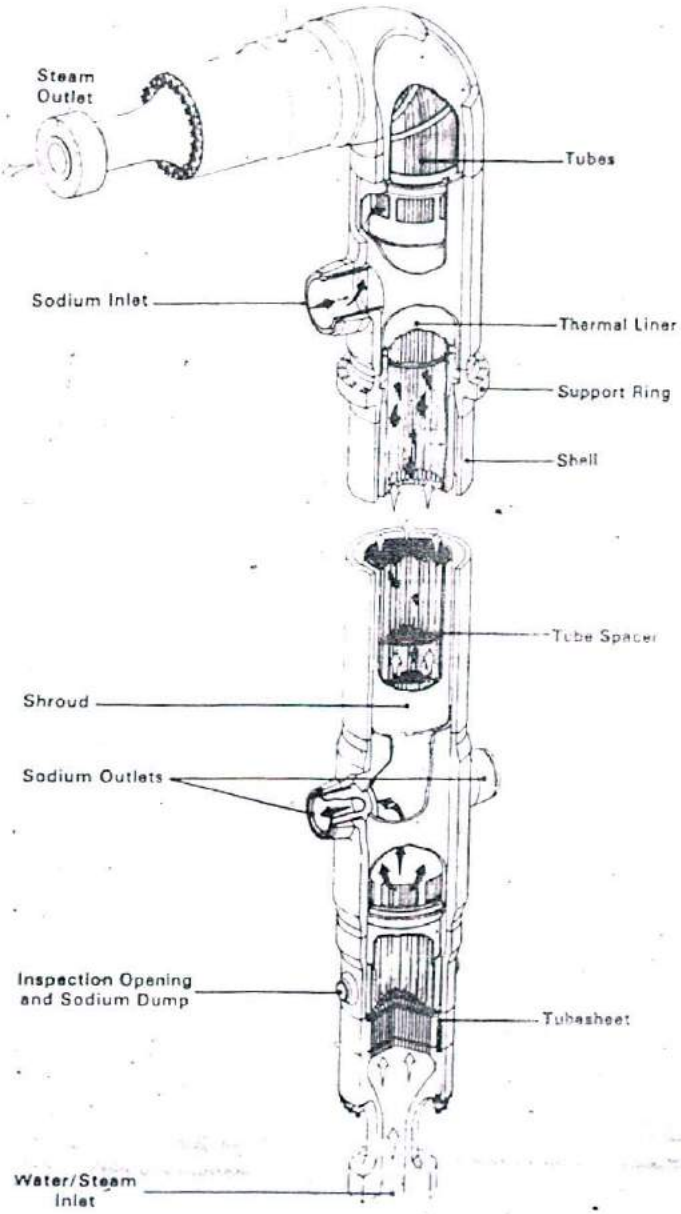


Figure 11-17 CRBR module for both steam evaporators and superheaters.

Saturated steam enters the superheater, and water enters the evaporators, at bottom center; they flow upward through the tubes and exit horizontally at the top.

The pumps The primary and intermediate sodium pumps are identical motor-driven, vertically mounted, single-stage, centrifugal 316 stainless steel units with double suction impellers. They have an overall height, less motor, of 23 ft (7 m) and a maximum diameter of 8.5 ft (2.6 m) and have dry masses of 82 tons (primary) and 64.5 tons (intermediate). The impeller is supported above and below by two sodium-lubricated hydrostatic bearings that are fed by sodium from the pump discharge. Both suction and discharge occur at the bottom of assembly. Pressurized argon cover gas fills the space between the top of the sodium level (about halfway up the driveshaft) and a thermal shield.

The drive motors are supported by one thrust bearing and two radial bearings each. They are variable-speed 5000-hp (3773.5-kW) units powered by variable-frequency AC motor-generator sets. This allows a variable speed range that permits load following between 40 and 100 percent of full power. Independently-powered, constant-speed 75-hp (60-kW) pony motors provide power for sodium circulation during start-up, shutdown, and decay-heat removal.

The steam plant A flow diagram of the steam plant at rated power is shown in Fig. 11-18. The 3600-r/min turbine is composed of one high-pressure and three low-pressure sections. The total turbine inlet steam flow from the three superheaters is about 3,320,000 lb_m/h (418.3 kg/s) at full power. A deaerating single-pass condenser is located directly below and parallel to the turbine. Condensate at 101°F is pumped from the condenser hot well to a demineralizer consisting of three parallel mixed-bed ion exchangers (one always on standby) that provide full-flow demineralization. It then flows through three low-pressure closed feedwater heaters and one tray-type open deaerating heater, which also serves as a storage tank. Three boiler feedwater pumps (one always on standby) then pump the water to 2160 psi (148.9 bar) through three high-pressure closed feedwater heaters. The last one, called a tapping heater, uses blowdown from the steam drums to heat the feedwater instead of bleed steam from the turbine as usual. Feedwater leaves the tapping heater at 468°F and goes to the three steam drums via regulating valves. A total of 1,150,000 lb_m/h (145 kg/s) of steam is bled from the turbine for the six feedwater heaters. This is about 34.6 percent of the steam entering the turbine high-pressure stage.

The turbine drives a 416-MW, 22,000-V, 60-Hz, 0.9 power-factor generator. The generator is a totally enclosed three-phase wye-connected single winding machine with a hydrogen-cooled rotor and water-cooled stator.

Heat sink Heat rejection from the main condenser and miscellaneous component cooling water systems is accomplished by three pumps that deliver 185,200 gal/min (11.68 m^3/s) of cooling water. This water is in turn cooled by a forced-draft, counterflow wet-cooling tower capable of handling full turbine and component cooling

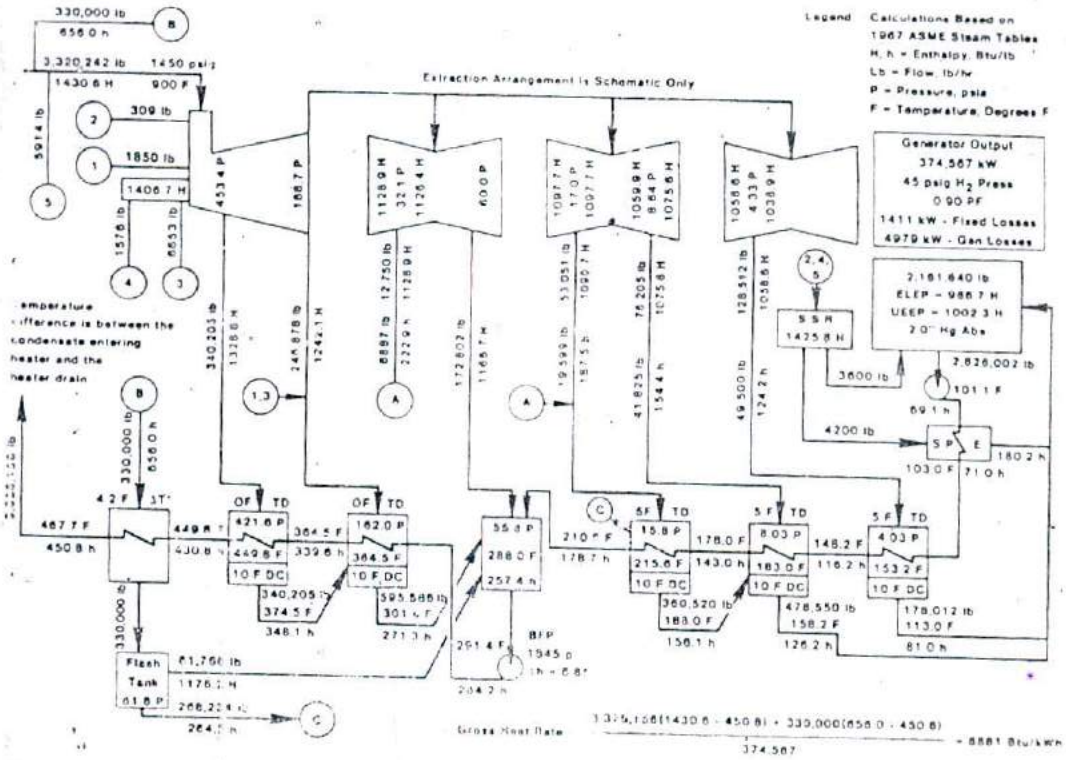


Figure 11-18 CRBR steam powerplant flow diagram, heat rate at full power

loads with 76°F wet-bulb temperature air yielding a turbine back pressure of less than 5 inHg. Makeup due to evaporation, drift, and blowdown in the cooling tower is provided by approximately 5790 gal/min (0.365 m^3/s) of Clinch River water drawn in via two submerged perforated pipes about 75 ft (23 m) from shore. There is continuous cooling-tower blowdown to the river at about 2210 gal/min (0.14 m^3/s) to control chemical levels in the cooling water while meeting environmental standards on discharge to the river.

Residual heat removal There are three modes of removing reactor decay heat in the case of plant malfunctions. These are (1) steam bypass, (2) steam-generator heat removal, and (3) reactor heat removal.

Steam bypass This is the case of turbine trip, nonoperation, or malfunction, but power is available to condensate and feedwater pumps. Steam is diverted around the turbine directly to the condenser by closing the turbine throttle valve and opening a bypass valve in a bypass line between the main steam line and the condenser. The steam, which becomes superheated upon throttling by the bypass valve, is desuperheated, condensed in the condenser, and then recycled to the feedwater line.

Steam generator heat removal This is the case where power is not available to the condensate and feedwater pumps. Initially steam is dumped directly to the atmosphere via steam atmospheric dump valves after the superheaters. Makeup water to the steam drums to compensate for this steam loss is supplied from a 72,000-gal (273 m^3) storage tank by three auxiliary feedwater pumps, one steam driven and two electrically driven. This continues until sufficient heat is removed and cooling can be handled by air-cooled condensers and no makeup water is required. These condensers are situated above the steam drum so that natural circulation between them can take place.

Reactor heat removal This is an independent system that removes heat directly from the reactor during steam-generator maintenance or severe plant malfunction. It uses a reactor-coolant makeup system that receives sodium overflow and supplies sodium makeup via a bypass valve during normal operation. To remove heat directly this bypass valve is closed and the primary sodium is pumped in a closed loop to a heat exchanger that is normally isolated from the reactor by before-and-after isolation valves. These now open and pumping is done by electromagnetic pumps. In the heat exchanger, primary sodium on the tube side gives its heat to NaK on the shell side, which is also pumped in a closed loop by electromagnetic pumps to two air-blast heat exchangers where heat is rejected to the atmosphere. NaK remains liquid at room temperatures and thus requires no auxiliary heating to remain fluid. The electromagnetic pumps and other components in this system are powered by the plant emergency diesel generators.

The last two systems are situated in hardened buildings and expected to operate under all postulated accidents.

11-5 THE SUPER PHÉNIX LMFBR

Among the industrialized countries, France is one of the poorest in energy resources. In the 1970s she was importing more than three quarters of her energy. She has, therefore, embarked on an aggressive nuclear program that is expected to reduce energy imports to 50 percent by the year 2000. The program calls for more than 40,000 MW by 1985, 60,000 MW by 1990, and 100,000 MW by 2000.

The proven uranium reserve inside France, and where it has controlling interest, is estimated at 160,000 tons. If used solely for light-water reactors (PWRs and BWRs), it would suffice for only 32,000 MW for 30 years, or a total of 960,000 MW-years. If used in fast-breeder reactors, however, it could produce 50,000,000 MW-years, hence the urgent need in France to develop a breeder program [106].

Development of the liquid-metal fast-breeder reactor (LMFBR) in France started in the early 1960s and can be represented so far by three main reactor types. The first is Rapsodie, a 20- to 40-MW(t) experimental reactor that went into operation in 1967. It produced no electricity and was used to check technical features and operations for use in the design of succeeding LMFBRs. The next is Phénix, a 250-MW demonstration plant whose construction started in 1964 and which went critical in 1973 and began on-line operation in July 1974. Phénix is a pool-type reactor with three secondary loops and modular steam generators. Early in its life it had some minor fuel problems and needed some redesign of its intermediate heat exchangers but otherwise has had an excellent operational record since 1978.

The third and latest major step in the French breeder program is the Super Phénix, a class of reactors which is an extrapolation of Phénix. The first of this class, a near-commercial plant called Super Phénix Mark I, is under construction at Creys-Malville on the Rhône River, east of Lyons, in cooperation with four other European countries. It is a 1200-MW plant that retains the pool design that was originally chosen for safety but also proved stable and manageable with Phénix.

The future, while somewhat clouded by a reassessment of energy future needs in 1981, is bright. A study of Super Phénix Mark II is already underway with contracts expected to be awarded in 1983 for possibly a twin-reactor station with operation in 1990. Following this, several identical twin reactor plants are considered on a reasonable time schedule. Based on the use of French-produced plutonium in various reactors, forecasts call for 16- to 23-GW breeders in operation by the year 2000.

General arrangement Figures 11-19 and 20 show the general layout and Table 11-5 lists some design data of Super Phénix Mark I [107] (with data in parentheses pertaining to Phénix.) The containment structure is a 64-m ID, 80-m-high reinforced concrete circular building that houses the reactor and its auxiliary circuits, fuel, and other active handling equipment, the primary heat-transfer loop; part of the intermediate heat-transfer loop; and temporary storage of radioactive wastes.

There are four buildings situated symmetrically around the containment structure, each of which houses a 750-MW steam generator and auxiliary equipment. This design maximizes physical separation between the intermediate loops. Interspersed between

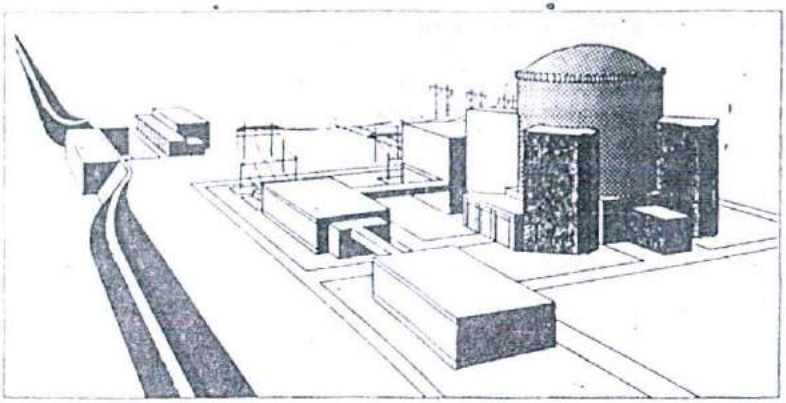


Figure 11-19 General layout of Super Phénix powerplant at Creys-Malville [107].

these are four shorter buildings, one for liquid-waste processing and three nuclear-service buildings that accommodate cooling-water circuits, sets powering the primary pumps, ventilation equipment, and subsidiary circuits for steam, organic liquids, nitrogen, argon, etc. Upstream of these buildings, and separated from them, are two turbine halls, each housing one 600-MW turboalternator. Closer to the river, and connected to the reactor and turbine buildings by underground and sky passages, is the electrical sections building. It houses the control room in its center and duplicate auxiliary electrical units in the two outer wings, as well as the diesel generators. Other buildings on the premises include one to the west for housing sodium storage tanks; one to the north used as an engineering and assembly shop for large components; one to the east for condenser cooling water intake, filtration, and pumping from the Rhône; and one, downstream of the latter, for cooling-water discharge to the river.

Reactor and fuel Figure 11-21 shows the reactor block that houses the nuclear steam supply system (NSSS) and other components. The main reactor vessel contains the sodium pool and the entire primary sodium system consisting of the core with its fuel and fertile and shielding assemblies at the center, surrounded by four primary sodium pumps and eight intermediate heat exchangers (IHX). In addition it contains handling machines for loading and unloading assemblies and control rods. The main vessel is made of 316L stainless steel and is a 21-m-diameter and 18-m-high cylinder with a torispherical base. Its thickness varies between 2.5 and 6.0 cm. An argon blanket is provided over the sodium at a pressure of about 100 millibars, gage.

The main vessel is surrounded at a distance of 70 cm by a 304 stainless steel safety vessel which is 22.5 m in diameter and 19 m deep and has a torispherical base. The safety vessel serves to recover sodium in the unlikely event of a leak in the main vessel and thus keeps the core flooded and the sodium free to circulate normally to remove decay heat after reactor shutdown. It is provided with a recovery tank and a

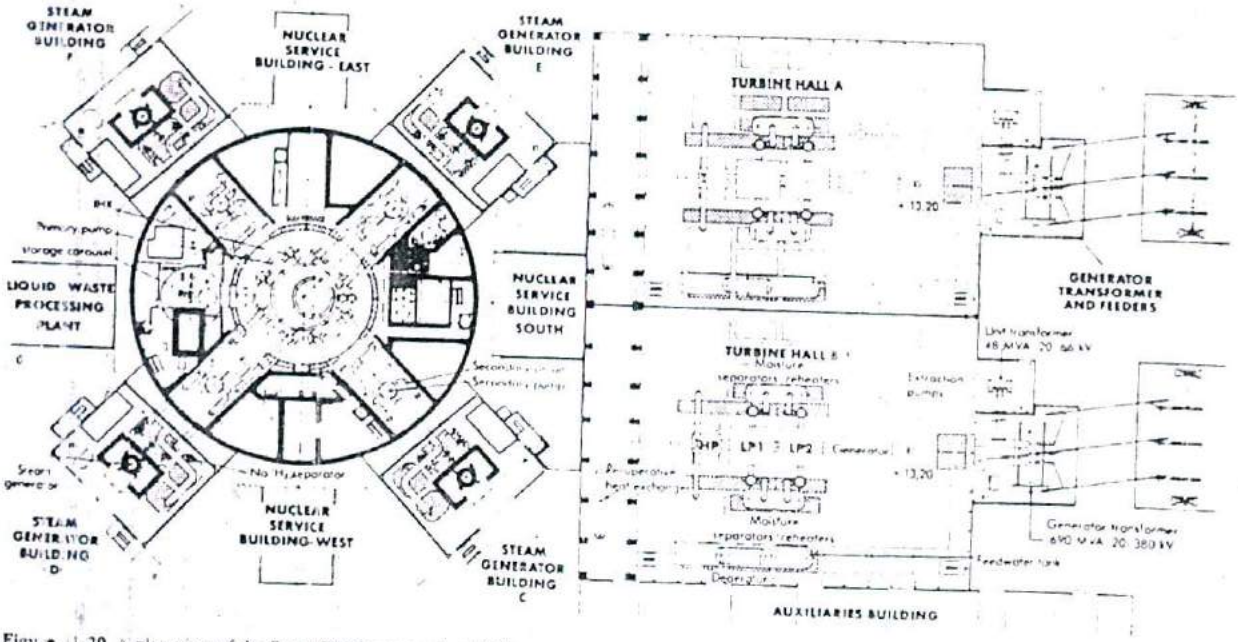


Figure 1-20 A plan view of the Super Phénix powerplant [107]

Table 11-5 Design data for the Super Phénix Mark I reactor plant

Power	3000 (590)* MW(t), 1240 (264) MW(e) gross
Fuel	PuO_2 UO_2 mixed oxide, enrichment 15.12% Pu^{239} equivalent
Fuel elements	364 assemblies, 5.4-m-long stainless steel clad rods, 8.65 (6.6) mm OD, 2.7 m long, 271 rods per assembly
Radial blanket	233 assemblies, 5.4 m (4.3) long, stainless steel clad, 1.950 m long, 91 rods per assembly
Control rods	Main system: 21 assemblies, 31 1.3-m-long rods each; back up shutdown system: three assemblies, three rods each; all stainless steel clad
Core	10.820 (1.227) m^3 volume
Reactor vessel	Stainless steel cylindrical with torispherical bottom, 21 m ID, 19.5 m high, contains 3500 tons Na
Maximum neutron flux	6.2×10^{15} (7.2×10^{15})
Linear power rating	450 (430) W/cm maximum
Fuel burnup	70,000–100,000 (50,000) MW · day/ton
Breeding ratio	1.24 (1.12)
Core temperatures	Fuel cladding wall max. 620 (650)°C; sodium 395°C inlet, 545°C outlet
Primary systems (4)	One pump and two intermediate heat exchangers per system arranged symmetrically around the core within the reactor vessel
Intermediate systems (4)	Na IHX inlet 345°C, outlet 525°C, nominal flow 3.27 ton/s each; total Na 1500 tons
Steam generators (4)	750 MW each, inlet 235°C, 210 bar, outlet 487°C, 177 bar, nominal flow 340 kg/s each
Turboalternators (2)	600 MW each, 3000 r/min, connected in parallel
Gross thermal efficiency	41.5% (44.75%)

*Numbers in parentheses refer to the Phénix reactor plant.

sodium detector at its lowest point to detect leaks. Two cooling circuits are placed outside the safety vessel, on the concrete side of the biological shield, to maintain acceptable concrete temperatures during normal operation.

Within the main vessel the core has a support structure that supports and positions various inner components such as the core diagrid, the lateral neutron shielding, the inner vessel, baffles, pipes and spheres connecting the primary pumps to the diagrid, the loading and unloading mechanism, and a core catcher. The core catcher is a stepped conical structure below the core diagrid designed to receive molten fuel in a noncritical configuration in the unlikely event of a core meltdown.

The core (Fig. 11-22) is made up of 364 fuel assemblies surrounded by 233 fertile blanket assemblies, which in turn are surrounded by 197 steel reflector assemblies that serve as neutron-flux attenuators. On the outside are 1076 nonremovable steel protective lateral neutron shields. Interspersed within the core are 21 main regulating control-rod assemblies and 3 supplementary shutdown assemblies.

All assemblies are hexagonal in cross section and are held to the core support structure on a 179-mm pitch. They are supplied with high-pressure (5 bar) sodium in the support structure through a large enough number of ports in their bases to minimize the possibility of blockage. Feed is made radially so that pressure on the foot of the assembly prevents liftoff. Orificing to adjust flow to power is situated within the base of each assembly.

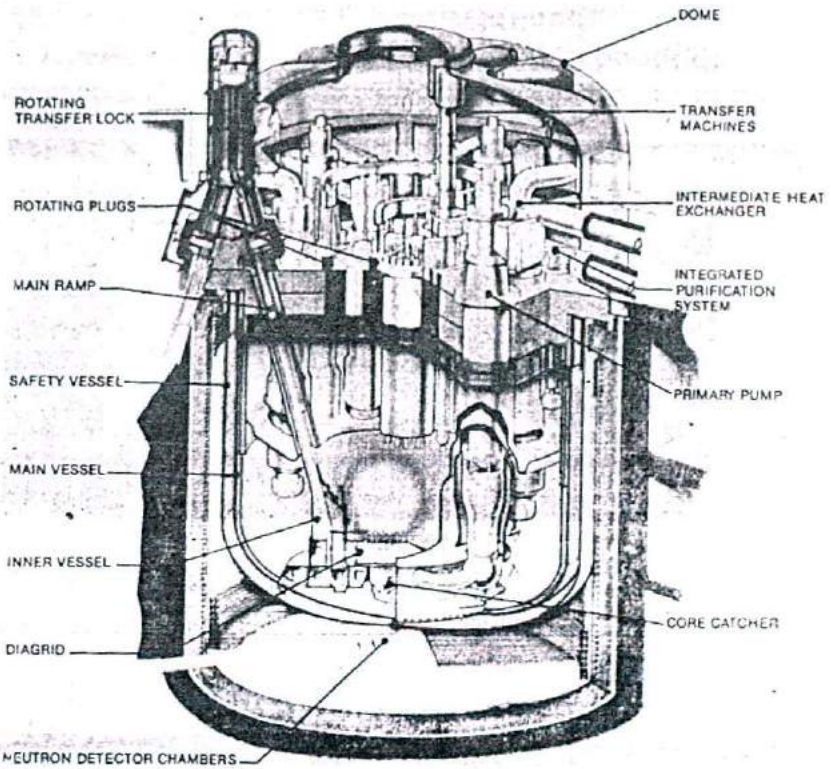


Figure 11-21 Super Phénix reactor block [107].

The fuel assemblies (Fig. 11-23), are cold-worked 316 stainless steel, are 5.4 m long and 173 mm across the flats, and contain 271 fuel rods each. There are 193 assemblies in an inner and 171 in an outer zone, with $PuO_2/(PuO_2 + UO_2)$ enrichments of 15 and 18.8 percent, respectively. Orificing of the fuel assemblies results in six flow rates: 44.5, 42.0, and 39.0 kg/s for the inner zone and 40.3, 36.9, and 36.0 kg/s for the outer zone.

Each fuel assembly has a hollow hexagonal steel block at the top with a centrally-drilled hole for sodium exit. It acts as an integral upper shield that, while lengthening the assemblies, makes it unnecessary to have special shielding for nearby components within the pool, such as the IHXs, core cover, pumps, etc.

The fuel rods are 8.5 mm OD and 2.7 m long. The cladding is cold-worked 316 stainless steel that contains a 0.162-m-long retaining spring at the top, followed by a 0.3 m zone of fertile UO_2 pellets, 1.0 m of the mixed-oxide fuel pellets, another 0.3 m of fertile UO_2 pellets and, at the bottom, a 0.85-m fission gas expansion chamber.

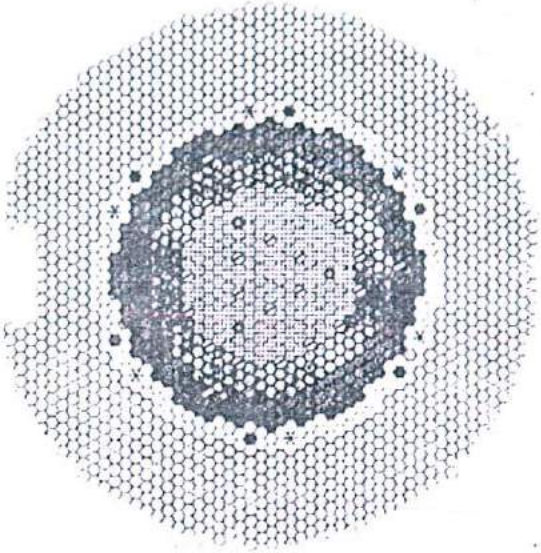


Figure 11-22 Cross section of the Super Phenix reactor core.

Helically wound around the rods are 1.20-mm wires that, besides correctly spacing the pins, minimize pin vibration, increase sodium flow turbulence, and permit some fuel swelling in relation to the hexagonal channel.

Nominal fuel pellet linear power is 450 W/cm and nominal cladding temperature is 620°C. Fuel burnup is planned at 70,000 MW · day/ton, which results in refueling every 2 years, after which fuel is sent for reprocessing. (Some Phénix assemblies have reached 65,000 MW · day/ton, and some experimental rods in Rapsodie have exceeded 160,000 MW · day/ton.) The limit on burnup is more due to steel irradiation damage, in particular swelling, than to oxide fuel damage.

The fertile assemblies In addition to the fertile material above and below the fuel in the fuel rods, the fertile assemblies contain fertile material, also of depleted UO_2. They are similar in design to the fuel assemblies but contain 91 rods of 15.8 mm OD each. Each rod is 1.944 m long, 1.6 m of which is the fertile section, equal to total

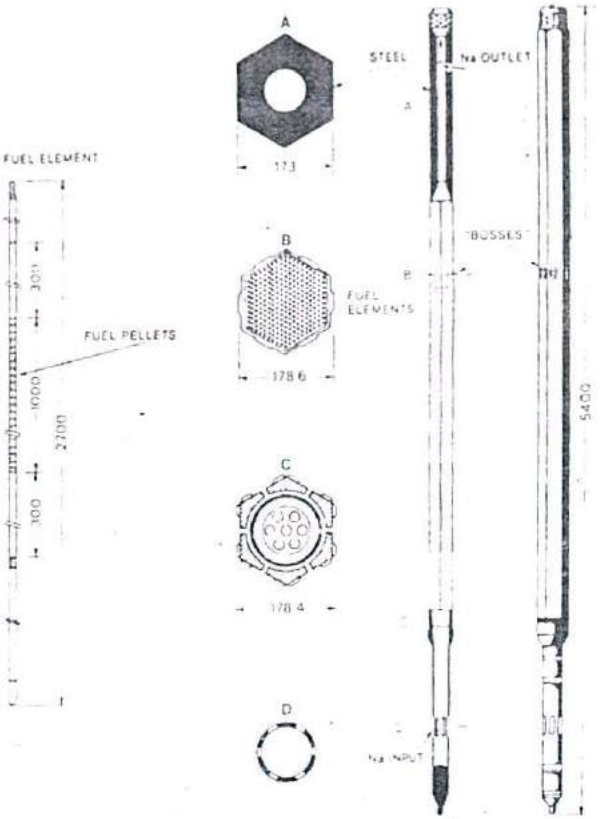


Figure 11-23 Super Phénix fuel rod and assembly.

length of fuel and fertile pellets in the fuel rods. The rest is a gas expansion chamber. The pellets here are thicker than the fuel pellets and the expansion chamber is smaller because the specific power is lower. As with the fuel rods, the fertile rods have 0.9-mm wires helically wound around them and hollow hexagonal steel upper neutron shields.

Orificing allows three flow rates in the fertile assemblies, depending upon their position and residence time in the reactor: 9.3 kg/s (3 years), 3.5 kg/s (4 years), and 1.2 kg/s (5 years). These rates are designed for cooling at the end of life when plutonium loading and hence power are greatest.

The control and shutdown rods. There are 21 main control assemblies that consist of one group of five situated on an inner circle and one group of six situated on an outer circle within the core (Fig. 11-22). They have the multiple functions of shutdown, load, and temperature compensation and control. They are always partially inserted

in the core and are progressively withdrawn as fuel burnup increases. The outer sleeve of each assembly is identical to that of the fuel but contains a 149-mm diameter, 1.3-m-long cylinder that, in turn, contains 31 rods, 21 mm in diameter each. The rods contain a 1.1-m section of B_4C absorber pellets, 90 percent enriched in B^{10}, enclosed in 316 stainless steel cans. Vents allow helium from the n,α reaction to escape to an expansion chamber, a design used successfully in Rapsodie and Phénix. The assemblies are sodium-cooled.

There is also a shutdown system that consists of three assemblies situated between the inner and outer main control groups. They also have B_4C pellets but a different interior. They have only two positions in the core, fully withdrawn and fully inserted, and therefore perform no control or compensation functions. Actuated by electromagnets, they are inserted on an emergency command to the main control system by a self-actuated system or if the sodium temperature exceeds a preset value, between 650 and 700°C.

Loading and unloading of removable assemblies such as the fuel, control, fertile, and lateral shielding assemblies is to a loading and unloading chamber that is fixed to the core support structure. This is accomplished by the two off-center rotating plugs (Fig. 11-24) using two transfer machines attached to the small rotating plug (Fig. 11-21). The assemblies are then removed to storage outside the reactor by means of an inclined ram and a rotating transfer lock. A core cover plug is housed in the small rotating plug above the core with its bottom just above the assembly heads but sub-

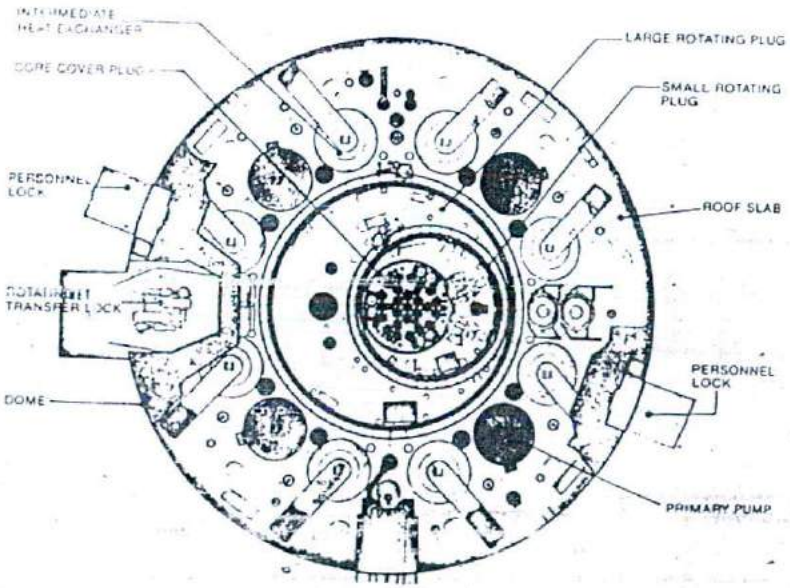


Figure 11-24 Plan view of Super Phénix reactor block.

merged in sodium. Its functions are to deflect sodium flow from the assemblies and to properly position the rod mechanisms and core instrumentation.

The heat-transfer systems As with all sodium-cooled reactor plants, a primary sodium system transfers heat from the reactor to an intermediate sodium loop, which in turn transfers heat to the steam cycle. In this pool design, the entire primary system is inside the main vessel. It consists of four pumps and eight intermediate heat exchangers (IHX) arranged symmetrically in groups of three (one pump and two exchangers) on a 16.2-m-diameter circle slightly off-center to the core (Fig. 11-24).

Unlike CRBR, Super Phénix uses "cold leg" primary pumps. Each draws 4100 kg/s of cool (392°C) sodium from the IHX outlet region through a skirt, which ensures uniform feed to its inlet diffuser (Fig. 11-25). It then delivers sodium axially to a

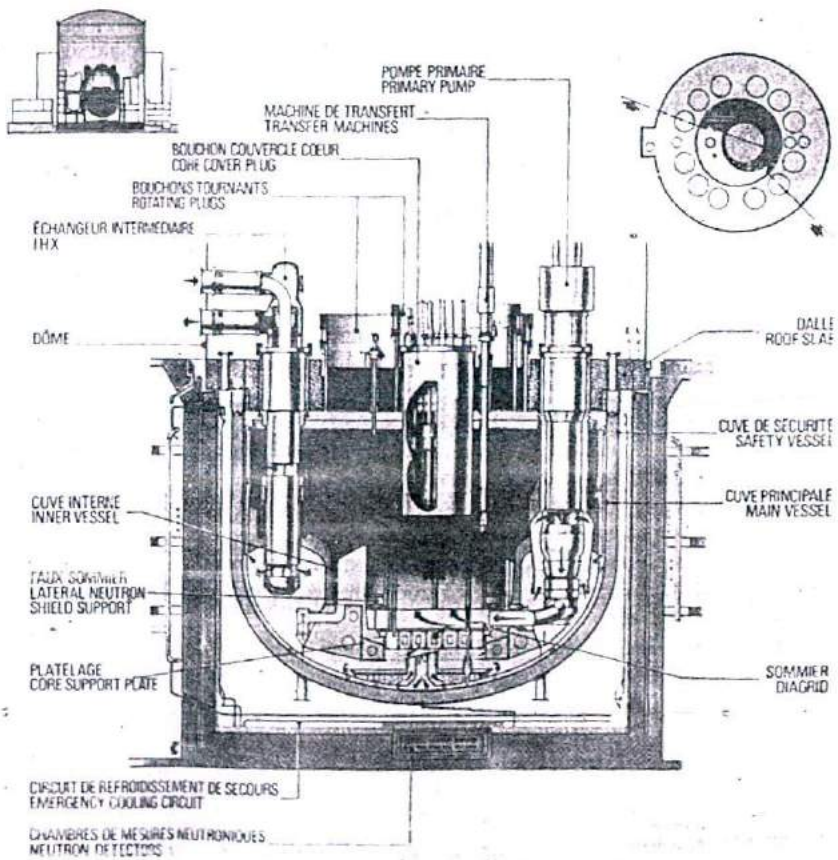


Figure 11-25 Cross section of Super Phénix reactor block showing primary coolant circuit. *(Courtesy Novatome, Le Plessis-Robinson, France.)*

plenum below the core diagrid with a head of 65-m Na. A one-way valve in the delivery pipe prevents flowback in case of individual pump shutdowns. The pumps are 2.5 m OD and 12 m long and weigh 125 tons each. They are driven by 3300-kW asynchronous motors that are powered by variable voltage and variable frequency current. This current is provided by a variable-speed alternator that is connected to a constant-speed asynchronous motor by a hydraulic coupler. This allows a pump speed variation from 15 to 100 percent of its 460 r/min nominal speed. In addition, each pump can be powered by a pony motor that runs at about 15 percent of nominal speed, and that receives its power from the emergency diesels.

Primary sodium enters and leaves the core at 395 and 545°C, respectively. It then enters the shell-and-tube IHXs at 542°C via inlet ports at top, flows outside the tubes and leaves at 392°C through outlet ports at bottom to the "cold" IHX outlet region in the pool, and then flows back to the pump inlets. Each IHX is 2.5 m OD and 19 m high, contains 1300 m^2 of heat-transfer area, weighs 52 tons, and is rated at 375 MW.

There are four independent intermediate loops (Fig. 11-26). Intermediate sodium at 345°C enters the IHXs at the top via a center tube, flows down to a lower plenum, reverses direction, and goes up through the tubes to an annular space at the top and leaves laterally at 525°C. Sodium leaving each pair of IHXs goes to one steam generator, leaves it at 345°C, and enters an intermediate (secondary) pump that is installed in a free-level expansion tank. Sodium leaves the pump laterally through two 0.7-m pipes to the two associate IHXs. Each intermediate pump is 2 m OD, weighs 35 tons, delivers 3300 kg/s of sodium at a head of 30 m Na, and consumes 1300 kW.

In the steam generator (Fig. 11-27), hot intermediate sodium enters laterally through two 0.7-m pipes at the top, mixes in a distribution chamber, flows outside the tube bundle, and leaves at 345°C via one central 1-m-diameter pipe at the bottom to the expansion tank. There are free sodium levels in the upper parts of the expansion tank and steam generator.

The steam plant The steam generator combines both boiler and superheater functions. Feedwater at 235°C and 210 bar enters each steam generator at the rate of 340 kg/s through four water chambers at the lower end. It flows in a once-through fashion through four sets of tubes from the water chambers and leaves as superheated steam through four corresponding steam chests that combine into two exit pipes. There are a total of 357 20-mm ID tubes wrapped around the central pipe in 17 concentric layers for a total of 2700-m^2 heat-transfer area. Each individual tube has a total length of 92 m. Each steam generator is 3 m OD and 25 m high and weighs 175 tons.

Steam enters two 3000 r/min turboalternators at 487°C and 177 bar (909°F and 2567 psia). Each turbine has its own reheat steam plant. The turbines are tandem-compound with one high-pressure section and two low-pressure sections. Steam leaving the high-pressure section at 6 bar is reheated by steam bled from it at 90 bar before entering the low-pressure sections. Each plant has one condenser that is cooled by a once-through system from the Rhône river. The condensate goes through a full-flow water treatment unit (to avoid fouling and corrosion in the steam generator). It then goes to two twin low-pressure feedwater heaters, a deaerating heater with storage tank,

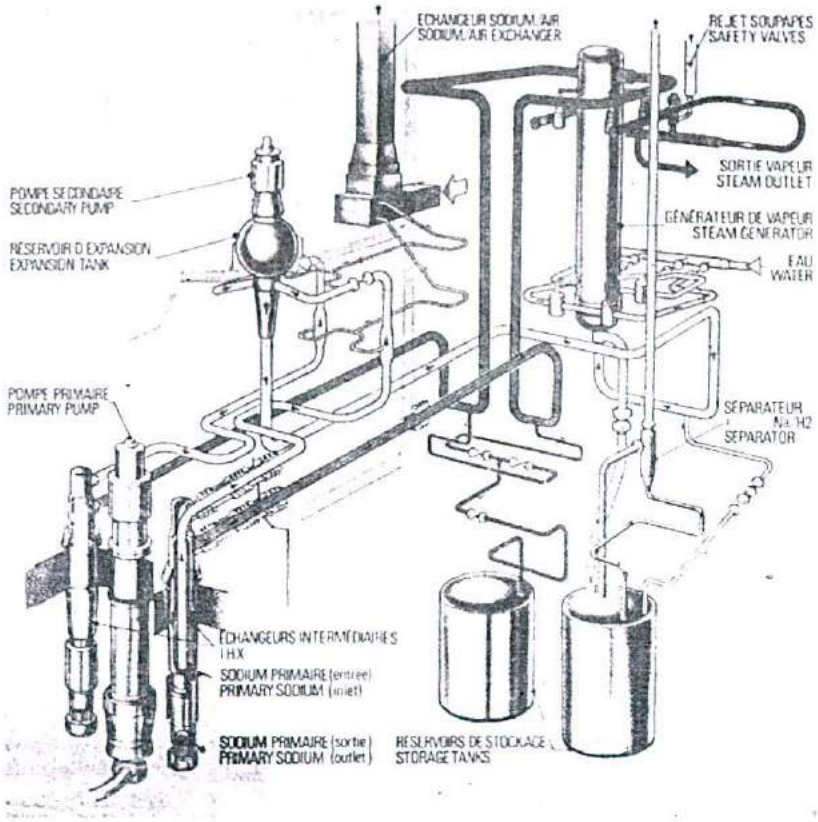


Figure 11-26 Super Phénix intermediate sodium loop (one of four). *(Courtesy Novatome.)*

three boiler feed pumps, one steam-driven and full-flow, the other two electrically driven for start-ups and other special operations. These are followed by two twin high-pressure feedwater heaters.

To avoid sodium freezing (about 100°C) in steam generators during shutdown or transients, water in the storage tank is maintained at 150°C by a variety of steam inlets.

Decay-heat removal In normal plant shutdown the reactor decay heat is channeled as usual by the primary and intermediate sodium loops and the steam generators. The resulting steam is shunted to the two turbine condensers and a special water loop called the shutdown–start-up circuit. In the event of the steam loop being inoperative, a backup system of sodium-air heat exchangers connected to each of the four intermediate sodium loops is used. These exchangers are cooled by forced convection with air

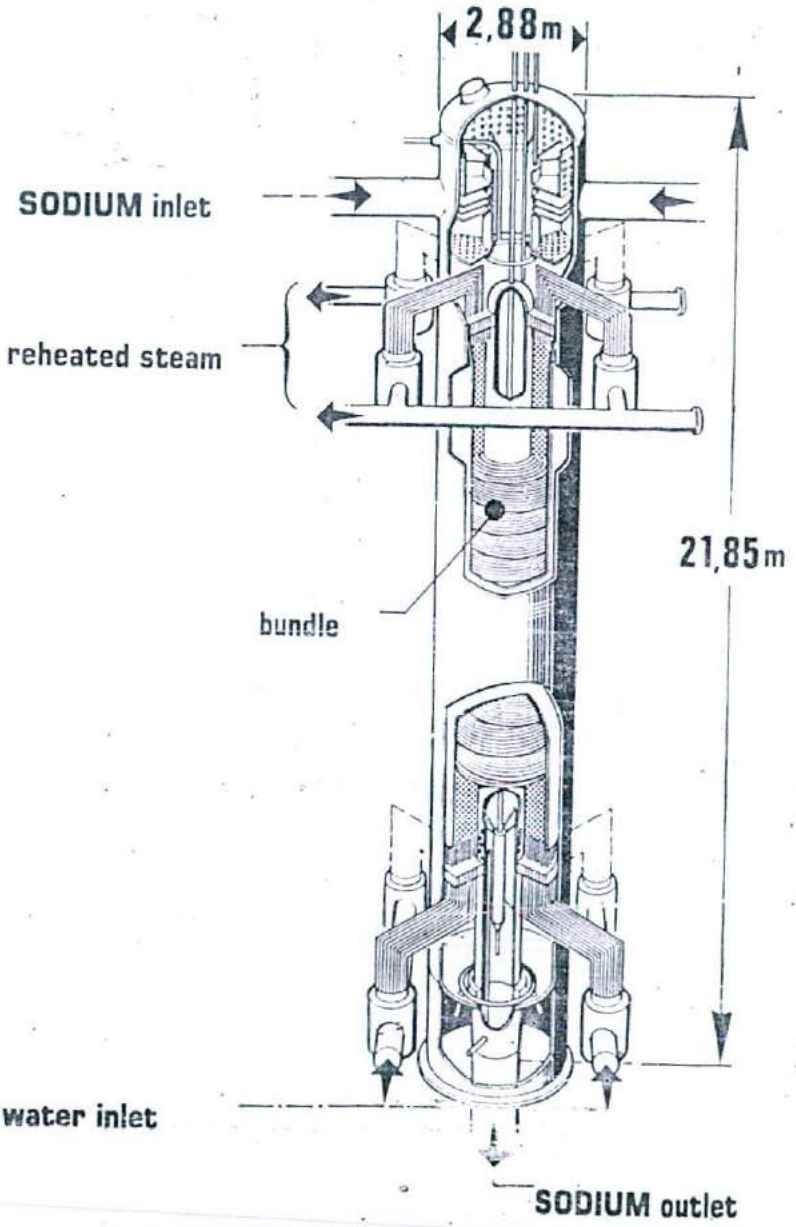


Figure 11-27 Super-Phénix steam generator. *(Courtesy Novatome.)*

blowers but are capable of removing the decay heat by natural convection alone in case of power failure. Additional backup systems operate in the event of failure of all four intermediate loops. They are composed of two independent water circuits in the reactor cavity and four sodium-to-sodium heat exchangers in the reactor vessel which connect to sodium in heat exchangers situated on top of the steam-generator building.

11-6 THE GAS-COOLED FAST-BREEDER REACTOR (GCFBR)

Research and development for the gas-cooled fast-breeder reactor (GCFBR) started in 1962 by Gulf General Atomic Corporation with some utility and, somewhat later, European participation. Studies were made on fuel development, systems, safety, physics, heat transfer, and fluid flow. Preliminary designs were made for a reactor experiment, a 300-MW demonstration plant, and a 1000-MW commercial plant. The studies also included the possible future uses of the GCFBR with a direct gas-turbine cycle for power production and with a high-temperature gas generator for industrial applications.

Figures 11-28 and 11-29 show the proposed 300-MW demonstration plant and its nuclear steam supply system. As were the thermal neutron AGR and HTGR gas-cooled reactors (Chap. 10), it is contained in a cylindrical prestressed-concrete reactor vessel

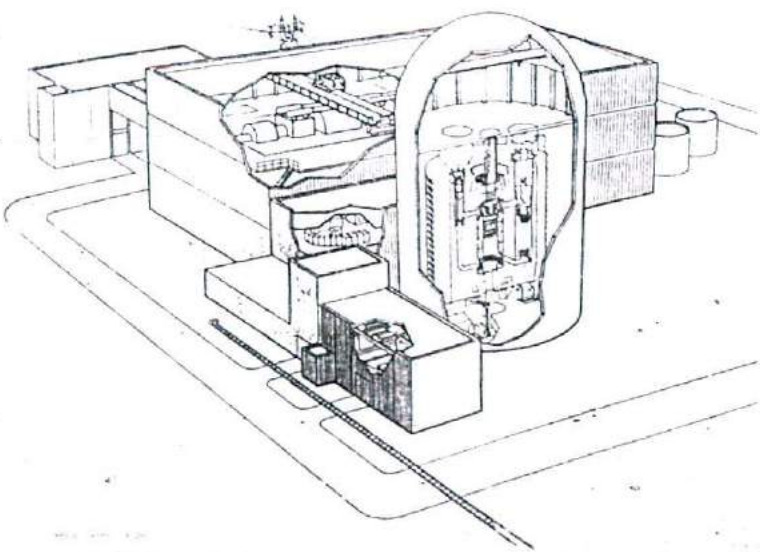


Figure 11-28 Layout of the 300-MW gas-cooled fast-breeder reactor (GCFBR) demonstration plant. *(Courtesy GA Technologies, Inc.)*

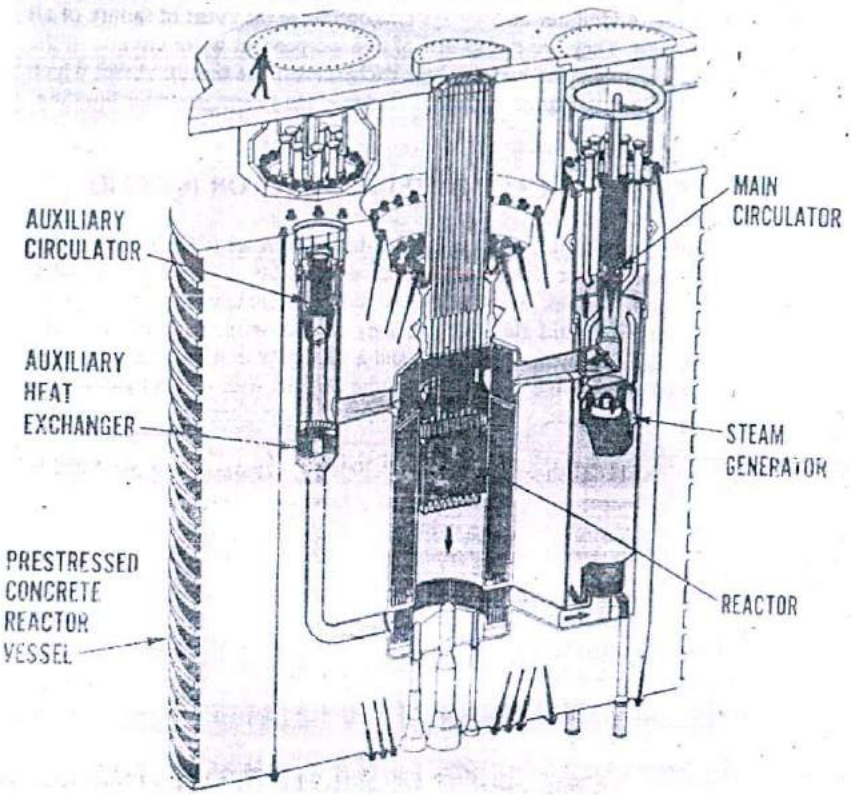


Figure 11-29 The GCFBR PCRV containing the nuclear steam supply system.

(PCRV) that has linear tendons extending through from top to bottom and circumferential steel cables wrapped around the outside under tension.

The PCRV in this case is 84 ft (25.6 m) in diameter and 71 ft (21.6 m) high and contains seven steel-lined interconnected cavities. The central cavity contains the reactor and is surrounded by three main cavities that contain the steam generators and helium circulators and, alternating with them, three that contain auxiliary cooling equipment. Helium coolant at 1250 psia (86 bar) passes downward through the core (as in the HTGR) then flows laterally to the bottom of the three main cavities. In each it goes up through a steam generator, through a 22,300-hp single-stage, axial, steam-driven main helium circulator, and then back laterally to the top of the core. Although

these three heat-transfer loops are in interconnected cavities, they are designed to operate independently of one another, thus increasing the reliability of the system.

The PCRV is housed in a secondary containment building. The space between the two contains atmospheric air and is accessible during plant operation for inspection and maintenance of equipment outside the PCRV.

The reactor core and fuel The reactor core is composed of 211 hexagonal fuel and radial blanket assemblies. The assemblies, 10 ft (3.05 m) high and 6.5 in (16.5 cm) across the flats, are supported by clamping to a top grid plate where they are at their coolest. The grid plate, 11 ft (3.35 m) in diameter and 2 in (5.08 cm) thick, is in turn supported by a cylindrical structure connected to the liner of the central penetration in the PCRV. It contains closely spaced 6-in (15.24-cm) diameter holes that accommodate the circular top extensions of the fuel assemblies.

The assemblies are spaced about 1/4 in (6.3 mm) apart and are to be rotated during reloading to accommodate and reduce swelling (Sec. 9-13). Reloading is done by changing one-third of the core approximately annually and requires shutdown and depressurization.

The fuel assemblies (Fig. 11-30) contain 271 fuel rods except 27 that are used as control assemblies and contain 234 rods. The rods are the same as those used in LMFBRs except that their cladding surface is roughened to increase the gas-coolant heat-transfer coefficient. They are composed of annular pellets of mixed-oxide fuel (PuO_2-UO_2) stacked in 0.25-in (0.63-cm) OD, 20-mil (0.5-mm) thick 316 stainless steel cladding. An axial blanket of depleted UO_2 pellets is provided above and below the fuel within each rod.

The radial blanket assemblies occupy two rows outside the fuel assemblies and are of the same external dimensions. They, however, contain 127 larger rods containing depleted UO_2 pellets.

Each fuel rod has a fission product trap at the top. Fission gases are allowed to vent to an annular trap in the assembly and to a helium purification system. This system equalizes pressures of the fission gases inside and the coolant outside the rods, thus relieving mechanical stresses on the cladding. It also limits the release of radioactivity from failed elements into the coolant. Radioactivity monitors on the vent lines of separate groups of assemblies are used to detect and locate failed cladding.

The power plant Figure 11-31 shows a flow diagram of the plant. Helium at 1250 psia (86 bar) enters the reactor top at 595°F (313°C) and leaves at 1010°F (543°C). It then enters the three steam generators and steam-driven helium circulators (one shown) and goes back to the reactor. There are no intermediate loops as in a LMFBR as there are no problems of primary-coolant radioactivity or coolant-water chemical reactions as with sodium.

The steam generators are once-through helical-coiled units with reheaters. Feedwater leaving the last feedwater heater divides into three paths. Each enters a steam generator and leaves as superheated steam at 2900 psia (200 bar) and 875°F (468°C). That steam partially expands as it is made to drive the helium circulator turbine, after

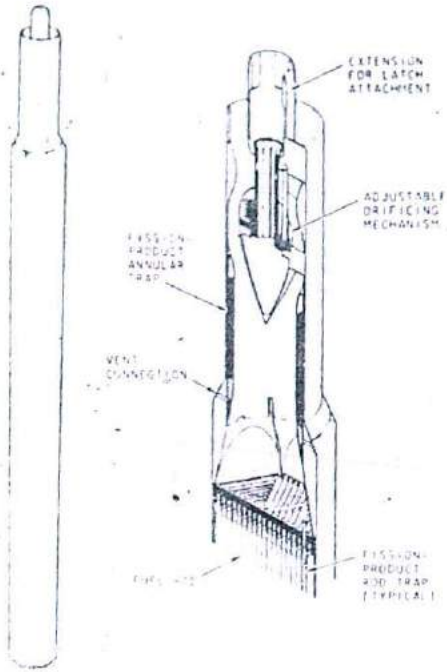


Figure 11-30 The GCFBR fuel assembly.

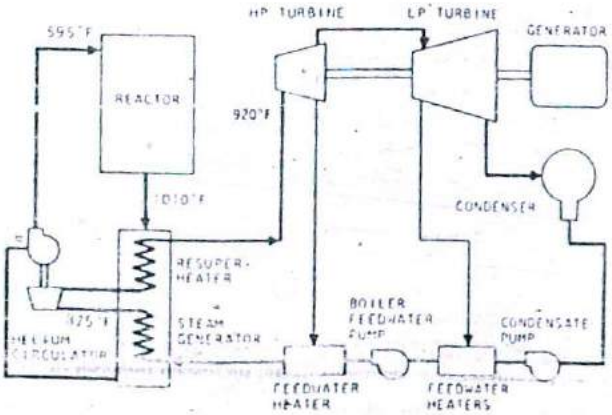


Figure 11-31 Simplified flow diagram of the GCFBR demonstration plant.

Table 11-6 Design parameters of the GCFBR

GCFBR	Demonstration	Commercial
Power	310 MW(e)	1000 MW(e)
Core: height	40 in (101 cm)	54 in (1.36 m)
diameter	80 in (2.01 cm)	107 in (2.71 m)
Fuel	Mixed oxide	Mixed oxide, 16% fissile
Fuel pins	0.25 in (6.3 mm) OD	0.25 in (6.3 mm) OD
Cladding	316 stainless steel, 0.48 mm thick	316 stainless steel, 0.286 mm thick
Fuel fissile rating	0.6 MW(t)/kg	1.1 MW(t)/kg
Max fuel linear rating	12.5 kW/ft (410 W/cm)	15 kW/ft (490 W/cm)
Cladding surface hot spot	1275°F (690°C)	1382°F (594°C) outlet
Coolant	Helium	Helium
Reactor coolant conditions	1250 psia (86.2 bar), 595°F (313°C) inlet, 1010°F (543°C) outlet	1250 psia (86.2 bar), 602°F (317°C) inlet, 1100°F (594°C) outlet
Throttle steam conditions	1225 psia (84.5 bar), 920°F (493°C)	1250 psia (86.2 bar), 925°F (496°C)
Plant thermal efficiency	37.6%	38%
Breeding ratio, doubling time	1.33	1.5 (~8 years)
Maximum fuel burnup	—	100,000 MW · day/ton

which it is admitted to the reheater. It then combines with the other two paths to enter the high-pressure turbine at 1225 psia (84.5 bar) and 920°F (492°C). The balance of the steam plant is essentially identical to a fossil powerplant because it uses superheated steam of similar properties. Note the difference between this steam cycle and that of the HTGR (Sec. 10-12). In the latter, the steam leaving the main steam generator expands in the high-pressure turbine and then drives the helium-circulator turbine before being reheated and sent back to the lower-pressure section of the main turbine.

The three main cooling loops are also used to remove decay heat from the core after shutdown. As in the HTGR, decay heat initially generates enough steam to drive the circulator turbines to help remove that decay heat, an advantage of the steam cycle design that uses the main steam flow into the circulator turbines. A half hour after shutdown, small auxiliary boilers, also used for plant start-up, come on line to supply steam to the circulator turbines. In addition, there are the three separate electric motor-driven auxiliary cooling loops (Fig. 11-30).

Table 11-6 includes data for the GCFBR demonstration plant as well as for a commercial-size extrapolation of GCFBR. The large plant is expected to have thinner cladding, higher cladding temperature, higher fuel linear heat rating, and a higher breeding ratio. These improved parameters are the results of the increased freedom of selecting fuel-fertile and fuel:coolant ratios in a large GCFBR.

PROBLEMS

11-1 A large commercial gas-cooled fast-breeder reactor using a plutonium-uranium mixed oxide fuel has a breeding ratio of 1.5 and a compound doubling time of 6 years. Estimate the average neutron flux in the reactor. Assume that the fuel in the core represents 80 percent of the reactor fuel inventory.

11-2 A large sodium-cooled fast-breeder reactor uses a plutonium-uranium oxide fuel mix. It has an average neutron flux of 3×10^{15}. The number of neutrons lost by leakage and parasitic absorption is 0.13 per neutron absorbed. Calculate (*a*) the reactor simple and doubling times, in years, based on the fuel in the core only, and (*b*) the neutron losses per neutron born in fission that would render the reactor a nonbreeder.

11-3 A fast-breeder reactor generates 3000 MW of heat. The fuel is composed of 20% $Pu^{239}O_2$, 80% $U^{238}O_2$ by mass. The average neutron flux is 10^{16}. Estimate the total mass of the fuel material in the core. Ignore fast fission in U^{238} and take neutron losses by leakage and parasitic absorption as 0.25 per neutron absorbed.

11-4 Estimate the maximum fraction (zero losses) of all neutrons that is available for breeding for the three fuels U^{233}, U^{235}, and Pu^{239} if they existed in monoenergetic neutron fluxes at 1, 10^2, 10^4, and 10^6 eV.

11-5 A fast-breeder reactor core that generates 2800 MW is fueled with $Pu^{239}O_2$-$U^{238}O_2$. The mass of $Pu^{239}O_2$ is 4 tons (metric). Ignoring fast fission in U^{238}, estimate (*a*) the maximum theoretical breeding ratio and gain, and (*b*) the corresponding minimum simple and compound doubling times, in years. Assume that the fuel in the core represents 75 percent of the reactor fuel cycle inventory.

11-6 A fast breeder reactor powerplant generates 1000 MW with a 39 percent efficiency. The core is fueled with $Pu^{239}O_2$-$U^{238}O_2$ with Pu-U nuclear number ratio of 1 : 4. The core average neutron flux is 10^{16}. Estimate (*a*) the number and mass of Pu^{239} nuclei originally in the core, (*b*) the number and mass of the Pu^{239} nuclei consumed per day, and (*c*) the number and mass of Pu^{239} nuclei bred and gained per day. Assume neutron losses by leakage and parasitic absorption to be 0.25 per neutron absorbed in the fuel.

11-7 A thermal-neutron breeder reactor using a U^{233}-Th^{232} breeding cycle is considered. The fuel is composed of 15 percent $U^{233}O_2$ by mass. The average neutron flux is 10^{14}. The reactor generates 2000 MW. Find (*a*) the total mass of fuel in the core, (*b*) the breeding ratio and gain if the neutron losses due to leakage and parasitic capture are 0.2 per neutron absorbed, (*c*) the simple and compound doubling times, in years, if the fuel in the core is 80 percent of the total reactor fuel inventory, and (*d*) the mass of $U^{233}O_2$ bred per day.

11-8 From the data of the Clinch River Breeder Reactor calculate the overall heat transfer coefficients, in Btus per hour per square foot per degree Fahrenheit for (*a*) the intermediate heat exchangers, (*b*) the evaporators, and (*c*) the superheaters.

11-9 From the data of Super Phénix, calculate the overall heat transfer coefficients in watts per square meter per kelvin and Btus per hour per square foot per degree Fahrenheit of (*a*) the intermediate heat exchanger, and (*b*) the economizer, evaporator, and superheater of the steam generator. Use for the specific heat of sodium 1.256 kJ/kg · K.

11-10 Assume that a gas-cooled fast-breeder reactor powerplant operates on a similar cycle and has the same power and efficiency and helium and steam pressures and temperatures as the demonstration GCFBR (Sec. 11-6). Assume further that the feedwater enters the steam generator at 468°F and that 35 MW of steam power is consumed in driving the helium circulators. Draw a temperature-path-length diagram of the steam generator and calculate (*a*) the helium mass flow rate, in pounds mass per hour, (*b*) the steam mass flow rate, in pounds mass per hour, and (*c*) the heat transferred, in Btus per hour in the economizer, evaporator, and first and second superheaters, respectively.

CHAPTER

TWELVE

GEOTHERMAL ENERGY

12-1 INTRODUCTION

With this chapter we begin a series of four chapters on the so-called *renewable energy resources*,* which are defined as those resources that draw on the natural energy flows of the earth. In this book, the ones that we are concerned with are those that arise from the earth's interior heat (this chapter), the sun (Chap. 13), the wind (Chap. 14), and the oceans (Chap. 15). Another, biomass, was covered briefly in Sec. 4-12.

Renewable energy resources are so named because they recur, are seemingly inexhaustible, and are free for the taking. The recurrence is often periodic, ranging from daily (the sun) to a few short years (biomass). Their main disadvantages are in their intermittency, lack of dependability, and their usually extreme low energy densities. Despite this, they were the predominant energy forms used by humankind during its early developmental millenia. When the earth's populations grew and nature alone could no longer support life on earth, human beings, as often is the case, discovered solutions when they were needed. The solutions came in the form of fossil fuels that were concentrated in certain pockets of the earth and that delivered much higher energy densities; they eventually brought on the industrial revolution and thus the modern way of life as we know it today. To be sure, fossil fuels are also renewable, but on

* Another term that is often used interchangeably with renewable energy is *alternative energy*. This term seems to mean different things to different people. Some include nuclear fuel used in fission, especially when its use is extended by breeder reactions (Sec. 11-2), and/or fusion. To others, the term precludes all established forms of energy. Others limit it to solar and solar-induced energies such as the wind and the rise and fall of waters. Still others make no distinction between the two terms.

a geologic time scale of hundreds of millions of years. They, however, are consumed much faster than they are renewed, so for all practical purposes, they have to be considered as finite. The same conclusion could be applied to nuclear fuels, which were formed with the earth. Their use, however, is believed to extend far beyond fossil fuels because of breeding, a reality today, and fusion, when it becomes a workable system.

With high energy density come high temperatures and therefore higher efficiencies. Advances in metallugy had the multiplier effect of further increasing temperatures and efficiency. Together with abundant and cheap fuels, these effects relegated renewable energy systems to doing odds and ends here and there forever, or so it seemed. Then came the 1970s and the great economic squeeze of the oil producers and resultant soaring prices; the era of cheap energy came to an end. This, and the environmental concerns of the same decade, reopened the era of the renewables (or alternatives). Goaded by public fears and pressures, a rush was on for broad technological solutions of the problems of the renewable systems. Before they were technologically and economically demonstrated, many people were inclined to overstate the case for them, which led to further public confusion about their true nature and their possible contributions to the total electric-energy picture. (Most renewable resources are aimed at electric production as their main contribution.) With time, however, and probably inevitably, cooler heads prevailed and a more sensible reexamination began taking place. This is what we will do in these four chapters: reexamine the renewables and follow up on modern solutions to their old problems.

We will begin our examination with geothermal energy, the one renewable resource that has practically no intermittency, has the highest energy density, and is economically not far removed from conventional technologies. Geothermal energy is classified as renewable because the earth's interior is and will continue in the process of cooling for the indefinite future. Hence, geothermal energy from the earth's interior is almost as inexhaustible as solar or wind energy, so long as its sources are actively sought and economically tapped.

12-2 PAST, PRESENT, AND FUTURE

Geothermal energy is primarily energy from the earth's own interior. The natural heat in the earth has manifested itself for thousands of years in the form of volcanoes, lava flows, hot springs, and geysers. These were mostly picturesque, often awesome proofs that vast heat stores lie beneath the earth's crust. In earlier times, natural steam that spouted from the earth was used only therapeutically. Roman documents, more than 2000-years-old, tell of a steam field that is now Larderello, south of Florence, a site that was to become history's first geothermal electric-generating station.

In the United States, geothermal fields were first discovered in 1847 by [illegible] Bell Elliot, an explorer-surveyor who was hiking in the mountains between Cloverdale and Calistoga, California, in search of grizzly bears. He discovered steam seeping out of the ground along a quarter of a mile on the steep slope of a canyon near Colb Mountain, an extinct volcano, now known as the Geysers. Telling friends that he came

upon the gates of hell, the word spread and the area became something of a tourist attraction.

The Geysers is really a misnamed field, as a geyser, like Old Faithful in Yellowstone National Park, periodically and dramatically spews jets of water and steam. In the Geysers, however, steam is continuously vented through fissures in the ground. These vents are called *fumaroles*.

Historically, the first applications of geothermal energy were for space heating, cooking, and medicinal purposes. The earliest record of space heating dates back to 1300 in Iceland. In the early 1800s, geothermal energy was used on what was then a large scale by the Conte Francesco de Laderel to recover boric acid. The first mechanical conversion was in 1897 when the steam of the field at Larderello, Italy, was used to heat a boiler producing steam which drove a small steam engine. The first attempt to produce electricity (our main concern in this book) also took place at Larderello in 1904 with an electric generator that powered four light bulbs. This was followed in 1912 by a condensing turbine; and by 1914, 8.5 MW of electricity was being produced. By 1944 Larderello was producing 127 MW. The plant was destroyed near the end of World War II, but was fortunately rebuilt and expanded and eventually reached 360 MW in 1981.

In the United States, the first attempt at developing the Geysers field was made in 1922. Steam was successfully tapped, but the pipes and turbines of the time were unable to cope with the corrosive and abrasive steam. The effort was not revived until 1956 when two companies, Magma Power and Thermal Power, tapped the area for steam and sold it to Pacific Gas and Electric Company. By that time stainless steel alloys were developed that could withstand the corrosive steam, and the first electric-generating unit of 11-MW capacity began operation in 1960. Since then 13 generally progressively larger units have been added to the system. The latest, No. 17 (Fig. 12-1), is a 109-MW unit that began operation in September 1982 and which brought the Geysers total capacity to 909 MW. Two more units are under construction and four more are planned, which will bring the total capacity to 1514 MW by the late 1980s.

Other electric-generating fields of note are in New Zealand (where the main activity at Wairakei dates back to 1958), Japan, Mexico (at Cerro Prieto), the Phillipines, the Soviet Union, and Iceland (a large space-heating program). These and other electric-generating fields are listed in Table 12-1.

Future world projections for geothermal electric production, based on the decade of the 1970s, are 7 percent per year. In the last four years of that decade, however, the growth rate was 19 percent per year (Fig. 12-2). In the United States, the projections are for growth between 13.5 and 22 percent per year through the 1980s, which is 2.5 to 4 times the 5.3 percent per year growth rate of the total electric-generating capacity. This includes the steam field at the Geysers and other fields of different types (Sec. 12-3).

The U.S. Geological Survey [109] predicts a U.S. potential from currently identified sources to be around 23,000 MW of electric power and around 40×10^{15} Btu (about 42×10^{12} kJ) of space and process heat for 30 years with existing technology, and 72,000 to 127,000 MW of electricity and 144 to 294×10^{15} Btu of heat from unidentified sources. Areas of geothermal potential in the North American continent

Figure 12-4 The 109 MW unit No. 14 of the Geysers. (Courtesy Pacific Gas and Electric Co.)

Table 12-1* World geothermal-energy utilization as of December 1979

Country	Electricity, MW		Space and process heat, MW
	Installed	Under construction	
USA	773	641	30
Italy	421	—	1
New Zealand	203	150	70
Japan	166	100	37
Mexico	150	30	
El Salvador	60	35	
Iceland	33	30	475
USSR	6	58	300
Phillipines	60	605	
Turkey	0.5	—	
Hungary	—	—	363
France	—	—	5
Total	1872	1649	1281

*From Ref. 108.

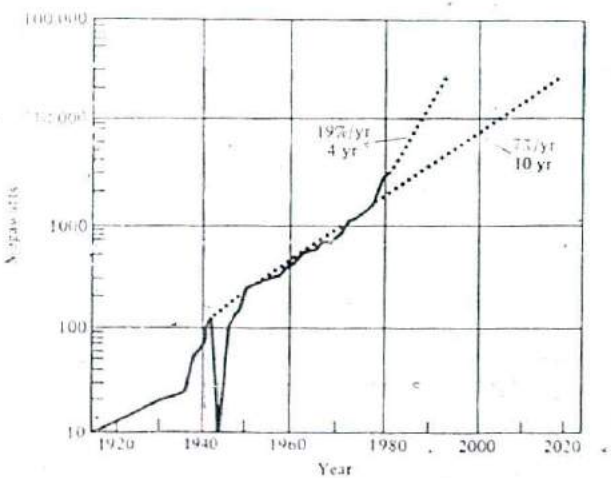


Figure 12-2 Worldwide installed geothermal electric capacity and future projections. The dip in 1942 represents the destruction of the Larderello plant [109].

are mainly west of the Great Plains from Canada to Mexico, with a geopressured zone (described in the next section) extending along the Gulf Coast and a low-temperature zone extending down the eastern seaboard. These include about 1.8 million acres of land of known sources in the western states and additional 96 million acres of prospective value. Between 800 and 1000 acres are needed for 100 MW of production for 30 years.

It can be seen that while geothermal energy is not the sought-after sole and long-range solution to our energy problems (the U.S. total installed electric capacity in 1982 is nearly 500 million MW), it nevertheless represents a not insignificant factor if its resources are developed in a careful and efficient manner.

12-3 ORIGIN AND TYPES OF GEOTHERMAL ENERGY

As indicated earlier, geothermal energy is heat transported from the interior of the earth. It is recoverable in some form such as steam or hot water.

The earth is said to have been created as a mass of liquids and gases, 5 to 10 percent of which was steam. As the fluids cooled, by losing heat at the surface, an outer solid crust formed and the steam condensed to form oceans and lakes in depressions of that crust. The crust now averages about 20 mi (32 km) in thickness. Below that crust, the molten mass, called *magma*, is still in the process of cooling.

Earth tremors in the early Cenozoic period* caused the magma to come close to the earth's surface in certain places and crust fissures to open up. The hot magma near the surface thus causes active volcanoes and hot springs and geysers where water exists. It also causes steam to vent through the fissures (fumaroles).

Figure 12-3 shows a typical geothermal field. The hot magma near the surface (A) solidifies into igneous rock† (B). (Igneous rock found at the surface is called volcanic rock.) The heat of the magma is conducted upward to this igneous rock. Ground water that finds its way down to this rock through fissures in it will be heated by the heat of the rock or by mixing with hot gases and steam emanating from the magma. The heated water will then rise convectively upward and into a porous and permeable reservoir (C) above the igneous rock. This reservoir is capped by a layer of impermeable solid rock (D) that traps the hot water in the reservoir. The solid rock, however, has fissures (E) that act as vents of the giant underground boiler. The vents show up at the surface as geysers, fumaroles (F), or hot springs (G). A well (H) taps steam from the fissure for use in a geothermal powerplant.

It can be seen that geothermal steam is of two kinds: that originating from the magma itself, called *magmatic steam*, and that from ground water heated by the magma, called *meteoritic steam*. The latter is the largest source of geothermal steam.

* The Cenozoic period is a geologic era that started some 60 million years ago following the Mesozoic period and includes the present. It is characterized by the appearance and development of mammals.

† From the Latin *igneus* meaning "of fire" or "fiery," from *ignus*, "fire"; specifically formed by volcanic action or great heat.

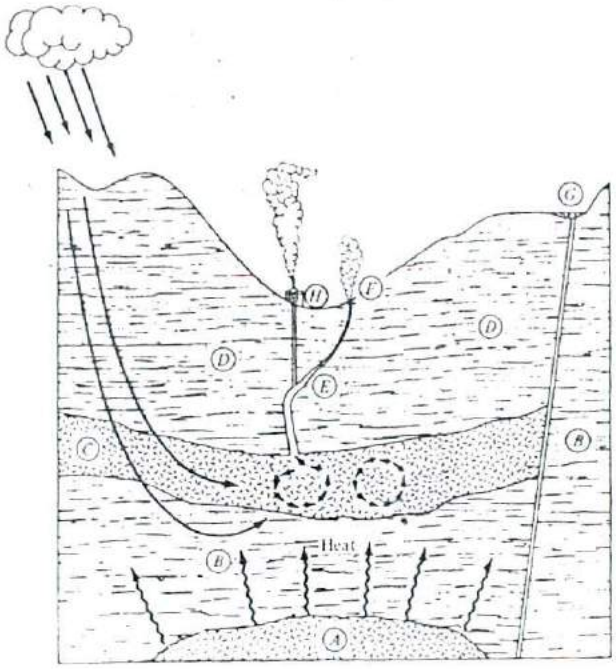


Figure 12-3 A typical geothermal field.

Not all geothermal sources produce steam as described above. Some are lower in temperature so that there is only hot water. Some receive no ground water at all and contain only hot rock. Geothermal sources are therefore of three basic kinds: (1) *hydrothermal*, (2) *geopressured*, and (3) *petrothermal*. These are explained below.

Hydrothermal Systems

Hydrothermal systems are those in which water is heated by contact with the hot rock, as explained above. Hydrothermal systems are in turn subdivided into (1) *vapor-dominated* and (2) *liquid-dominated* systems.

Vapor-dominated systems In these systems the water is vaporized into steam that reaches the surface in a relatively dry condition at about 400°F (205°C) and rarely above 100 psig (8 bar). This steam is the most suitable for use in turboelectric power plants, with the least cost. It does, however, suffer problems similar to those encountered by all geothermal systems, namely, the presence of corrosive gases and erosive material and environmental problems (see below). Vapor-dominated systems,

however, are a rarity; there are only five known sites in the world to date. These systems account for about 5 percent of all U.S. geothermal resources. The Geysers plant in the United States, the largest in the world today, and Larderello in Italy, are both vapor-dominated systems.

Liquid-dominated systems In these systems the hot water circulating and trapped underground is at a temperature range of 350 to 600°F (174 to 315°C). When tapped by wells drilled in the right places and to the right depths, the water flows either naturally to the surface or is pumped up to it. The drop in pressure, usually to 100 psig (8 bar) or less, causes it to partially flash to a two-phase mixture of low quality, i.e., liquid-dominated. It contains relatively large concentrations of dissolved solids ranging between 3000 to 25,000 ppm and sometimes higher. Power production is adversely affected by these solids because they precipitate and cause scaling in pipes and heat-exchange surfaces, thus reducing flow and heat transfer. Liquid-dominated systems, however, are much more plentiful than vapor-dominated systems and, next to them, require the least extension of technology. The U.S. Geological Survey [109] shows from 900 to 1400 quads (Q) (1Q = 10^{15} Btu, about 10^{18} J) of energy available from liquid-dominated systems with liquid above 300°F (150°C).

The hydrothermal systems, of both kinds, are the only ones in commercial operation today. Figure 12-4 shows the major high-temperature hydrothermal areas of the world. The next two systems are under study, but mainly in the preliminary stages at this time (1982).

Geopressured Systems

Geopressured systems are sources of water, or brine, that has been heated in a manner similar to hydrothermal water, except that geopressured water is trapped in much deeper underground acquifers,* at depths between 8000 to 30,000 ft. (about 2400 to 9100 m). This water is thought to be at the relatively low temperature of about 325°F (160°C) and is under very high pressure, from the overlying formations above, of about 15,000 psia (more than 1000 bar). It has a relatively high salinity of 4 to 10 percent and is often referred to as brine. In addition, it is saturated with natural gas, mostly methane CH_4, thought to be the result of decomposition of organic matter.

Such water is thought to have thermal and mechanical potential to generate electricity. The temperature, however, is not high enough and the depth so great that there is little economic justification of drilling for this water for its thermal potential alone. What is drawing attention, however, is the amount of recoverable methane in solution that can be used for electric generation. The U.S. Geological Survey estimates 100 Q of electricity from the thermal content of geopressured water and 500 Q of energy in the gas. Studies have been under way to determine the economic feasibility of generating electricity by a combined cycle, one that involves the combustion of the methane as well as heat from the thermal content of the water.

* An *aquifer* is a water-bearing stratum of permeable rock, gravel, or sand.

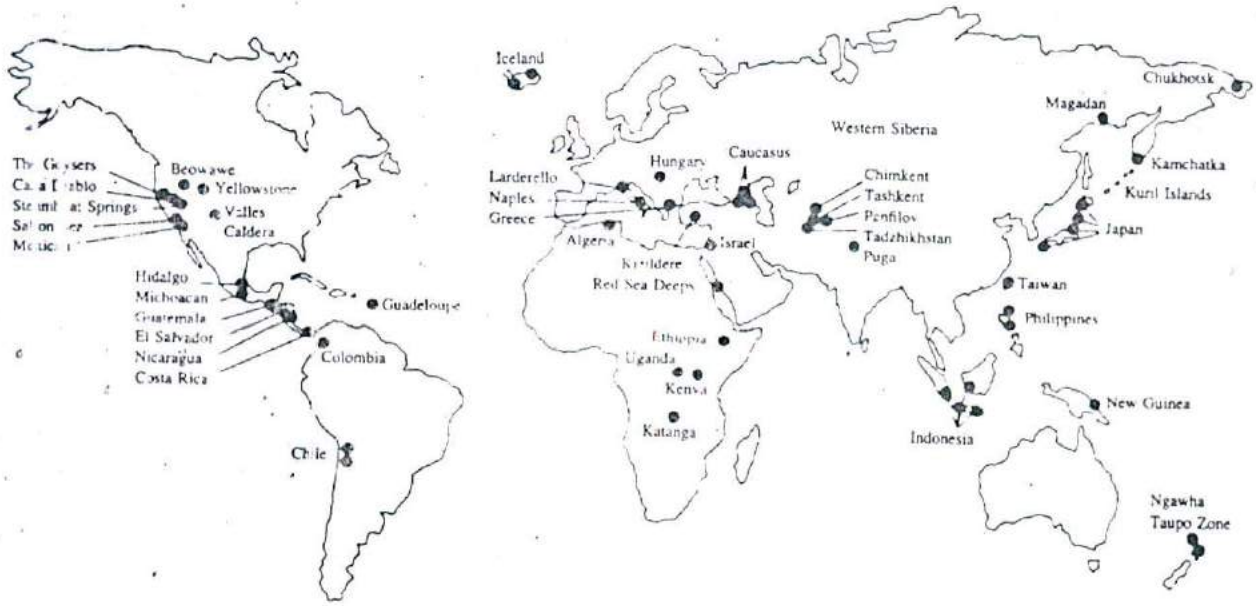


Figure 12-4 Major high-temperature hydrothermal areas of the world [110].

There are some 20 prospective geopressured sites along the Texas and Louisiana Gulf coasts in the United States. Work to determine the extent and quality of geopressurized energy has been undertaken with the drilling of some test wells. The initial results were not as encouraging as had been hoped for, however. A test well, called the General Crude-DOE Pleasant Bayou No. 2, was drilled in 1979 in Brazilia County, Texas, to a depth of 16,500 ft (5030 m) and tested at a flow rate of 2500 bbl/day (~300 m^3/day) and a pressure of 4570 psig (316 bar). Initial data indicated a potential flow rate of 30,000 bbl/day (~3575 m^3/day). The gas content was 20 to 25 ft^3/bbl of water, or about 4.75 to 6 gas-to-liquid volume ratio. Economic studies to determine if the cost of drilling and spent brine reinjection are recoverable from the energy content in the water and gas are yet inconclusive. It is estimated that a minimum yield of 40,000 to 50,000 bbl/day is necessary to make a well worth considering from an economic point of view.

Further work is continuing, however, with the possibility of building a pilot plant in the late 1980s. A study by the Southwest Research Institute for the Electric Power Research Institute (EPRI) optimistically predicts 1100 MW of geopressured capacity could be on line by the end of the century.

Petrothermal Systems

Magma lying relatively close to the earth's surface heats overlying rock as previously explained. When no underground water exists, there is simply hot, dry rock (HDR). The known temperatures of HDR vary between 300 and 550°F (~150 to 290°C). This energy, called petrothermal energy, represents by far the largest resource of geothermal energy of any type, as it accounts for about 85 percent of the geothermal resource base of the United States. Other estimates put the ratio of steam:hot water:HDR at 1:10:1000 [111].

Much of the HDR occurs at moderate depths, but it is largely impermeable. In order to extract thermal energy out of it, water (or other fluid, but water most likely) will have to be pumped into it and back out to the surface. It is necessary for the heat-transport mechanism that a way be found to render the impermeable rock into a permeable structure with a large heat-transfer surface. A large surface is particularly necessary because of the low thermal conductivity of the rock. Rendering the rock permeable is to be done by fracturing it. Fracturing methods that have been considered involve drilling wells into the rock and then fracturing by (1) high-pressure water or (2) nuclear explosives.

High-pressure water Fracturing by high-pressure water is done by injecting water into HDR at very high pressure. This water widens existing fractures and creates new ones through rock displacement. This method is successfully used by the oil industry to facilitate the path of underground oil. The oil-bearing stratum is sedimentary rock that is softer than HDR. The cost to the oil companies is thus lower and, in addition justified by the additional oil it produces. The method is under study by Los Alamos Scientific Laboratory (LASL) with support from the Department of Energy (DOE), Japan, and West Germany.

Nuclear explosives Fracturing by nuclear explosives is a scheme that has been considered as part of a program for using such explosives for peaceful uses, such as natural gas and oil stimulation, creating cavities for gas storage, canal, and harbor construction, and many other applications [112]. In the United States the program is called *Plowshare*.* Fracturing by this method would require digging in shafts suitable for introducing and sealing nuclear explosives and the detonation of several such devices for each 200-MW plant operating for 30 years. Initial studies revealed that large explosions, about 5 megatons (TNT equivalent), at substantial depths would be required before the scheme became economic. The principal hazards associated with this are the ground shocks, the danger of radioactivity releases to the environment, and the radioactive material that would surface with the heated water and steam.

A variation of the above concept would be to generate heat by the nuclear explosions themselves in deep salt formations. This would create an underground pool of molten salt that may be exploited for many years. Both schemes have many problems that are difficult to assess without actual experimentation. Not much progress has been made beyond the study stage.

12-4 OPERATIONAL AND ENVIRONMENTAL PROBLEMS

Steam and water from both hydrothermal systems contain, besides the dissolved solids in the water, entrained solid particles and noncondensable gases. The entrained solids must be removed as much as possible, usually by centrifugal separators at the well head, before they enter plant equipment, and by strainers, usually before turbine entry.

The noncondensable gas content varies from 0.2 to 4.0 percent, depending upon the particular well and its age. The younger the well, the higher the percentage, as the noncondensables tend to vent out a bit faster than the H_2O. The noncondensables themselves are mostly CO_2 (about 80 percent) plus varying amounts of methane CH_4, hydrogen H_2, nitrogen N_2, ammonia NH_3, and hydrogen sulfide H_2S. Besides finding their way with the fluid into the plant equipment, the noncondensables also partly escape to the atmosphere via the particle centrifugal separators, the condensor ejectors, and in some cases the cooling towers.

The presence of the noncondensable gases has several effects. First, the large quantity of these gases, relative to noncondensables in conventional steam systems, necessitates the careful design of adequate gas ejectors to maintain vacuum in the condenser. Second, although the presence of acid-forming gases causes no particular problems in dry steam lines that are made of ordinary carbon steels, their corrosive effect in wet conditions necessitates the use of stainless steel in all equipment exposed to wet steam or condensate. Such equipment includes turbine erosion shields and shaft seals, exhaust duct lining, condenser lining, condensate lines and pumps, and metal parts in cooling towers. (Condensers in geothermal plants may be of the direct-contact

* "And he shall judge among the nations, and shall rebuke many people, and they shall beat their swords into plowshares and their spears into pruning hooks. Nation shall not lift sword against nation, neither shall they learn war anymore." (Isiah 2:4)

type, and hence cooling towers are exposed to geothermal condensate.) In the turbine, steam nozzles and blades subjected to dry or high-quality steam are usually made of 11 to 13 percent chrome steel. The nozzles are usually designed with large throat areas on a wide pitch to minimize scaling. Because nickel is particularly sensitive to H_2S corrosion, it is not recommended for use in the rotor. The cooling towers are usually designed with plastic fill and concrete shells, the latter coated with coal-tar epoxy. Aluminum is recommended for condensate pipes and valves that are made large enough to allow low velocities and hence erosion. Aluminum is also recommended for switchyard structures that are in the open but in a generally corrosive atmosphere.

Another effect of H_2S is that it is corrosive to bare copper, particularly in the humid atmosphere around geothermal plants. Unprotected copper is to be avoided in plant electrical equipment that requires special attention. Electrical relays, motor control equipment, excitation gear, switchgear, and others are often kept in "clean rooms" under positive gauge pressure to isolate them from the corrosive atmosphere outside. Static-type exciters, instead of copper-commutator, motor-driven exciters, are used. Auxiliaries are usually motor-driven to avoid the additional corrosion of steam that occurs with turbine drives.

A further effect of the noncondensables is that they are environmentally undesirable because they partly escape into the atmosphere. Most are corrosive in the normally damp atmosphere of the plant site and are noxious and toxic and hence major air pollutants. The most objectionable are H_2S and, to a lesser extent, NH_3.

Another environmental problem caused by geothermal plants is land *surface subsidence*. This occurs because of the extraction of large quantities of underground fluids, though this is partly alleviated by reinjecting the spent brine or condensate into the ground, a procedure widely used in the oil industry. Reinjection also minimizes surface pollution. Large extractions and reinjections also pose the possibility of seismic disturbances.

Noise pollution is another problem. Exhausts, blowdowns, and centrifugal separation are some of the sources of noise that necessitate the installation of silencers on some equipment.

Geopressured water, in addition to the above problems, is thought to carry large quantities of sand, especially at the high flows required. The result is increased erosion and scaling problems.

12-5 VAPOR-DOMINATED SYSTEMS

As indicated previously, vapor-dominated systems are the rarest form of geothermal energy but the most suitable for electricity generation and the most developed of all geothermal systems. They have the lowest cost and the least number of serious problems.

Figures 12-5 and 12-6 show a schematic and T-s diagram of a vapor-dominated power system. Dry steam from the well (1) at perhaps 400°F (200°C) is used. It is nearly saturated at the bottom of the well and may have a shut-off pressure up to 500 psia (~35 bar). Pressure drops through the well causes it to slightly superheat at the

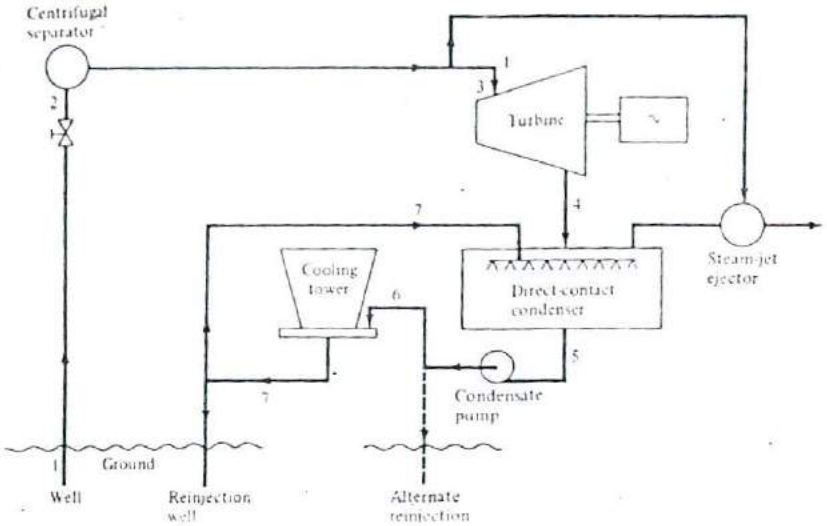


Figure 12-5 Schematic of a vapor-dominated powerplant.

well head (2). The pressure there rarely exceeds 100 psia (~7 bar). It then goes through a centrifugal separation to remove particulate matter and then enters the turbine after an additional pressure drop (3). Processes 1-2 and 2-3 are essentially throttling processes with constant enthalpy. The steam expands through the turbine and enters the condenser at 4.

Because turbine flow is not returned to the cycle but reinjected back into the earth

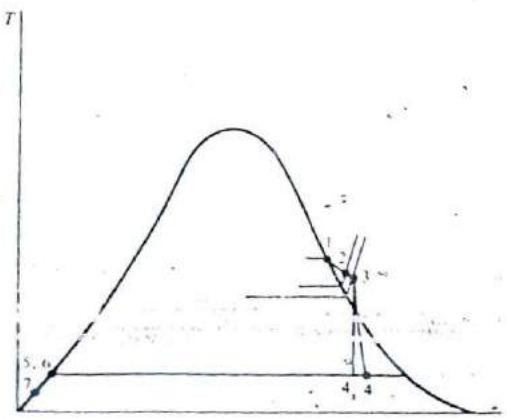


Figure 12-6 *T-s* diagram of the cycle shown in Fig. 12-5.

(Mother Nature is our boiler), a direct-contact condenser of the barometric or low-level type (Sec. 6-2) may be used. Direct-contact condensers are more effective and less expensive than surface-type condensers. (The latter, however, are used in some new units with H_2S removal systems, below.) The turbine exhaust steam at 4 mixes with the cooling water (7) that comes from a cooling tower. The mixture of 7 and 4 is saturated water (5) that is pumped to the cooling tower (6). The greater part of the cooled water at 7 is recirculated to the condenser. The balance, which would normally be returned to the cycle in a conventional plant, is reinjected into the ground either before or after the cooling tower. The mass-flow rate of the reinjected water is less than that originating from the well because of losses in the centrifugal separator, steam-jet ejector (SJE), evaporation, drift and blowdown in the cooling tower, and other losses. No makeup water is necessary.

A relatively large SJE (Sec. 6-3) is used to rid the condenser of the relatively large content of noncondensable gases and to minimize their corrosive effect on the condensate system.

Examples of vapor-dominated systems are the plants at the Geysers in the United States, Larderello in Italy, and Matsukawa, Japan. A view of one of the 110-MW units, No. 14 at the Geysers, was shown in Figure 12-1. Note the large number of mechanical draft towers, which are necessitated by the relatively large amount of heat rejected. Geothermal plants use much lower temperature and pressure steam and higher condenser pressures than conventional plants, and hence they are much lower in efficiency, having heat rates some 2 to 3 times those of the best fossil-fueled plants. Other differences are the large-diameter steam piping as a result of the large specific volume of the low-pressure steam, the large SJE, and because of the low efficiency, large turbines, condensers, and plant auxiliaries. In order to reduce the amount of cooling water needed, and therefore cooling-tower flow, the turbines are usually operated at relatively high back pressures, about 4 inHg absolute (~2 psia, 0.135 bar) or higher.

Example 12-1 A 100-MW vapor-dominated system as shown in Figs. 12-5 and 12-6 uses saturated steam from a well with a shut-off pressure of 400 psia. Steam enters the turbine at 80 psia and condenses at 2 psia. The turbine polytropic efficiency is 0.82 and the turbine-generator combined mechanical and electrical efficiency is 0.9. The cooling-tower exit is at 70°F. Calculate the necessary steam flow, lb_m/h and ft^3/min; the cooling-water flow, lb_m/h; and the plant efficiency and heat rate, Btu/kWh, if reinjection occurs prior to the cooling tower.

SOLUTION Refer to Figs. 12-5 and 12-6 and to the steam tables, App. A.

$$h_1 = h_g \text{ at } 400 \text{ psia} = 1204.6 \text{ Btu/lb}_m$$

and

$$h_3 \text{ at } 80 \text{ psia} = h_4 = 1204.6 \text{ Btu/lb}_m$$

Thus

$$T_3 = 350°\text{F (38°F superheat)}$$

$$s_3 = 1.6473 \text{ Btu/(lb}_m \cdot °\text{R)}$$

$$v_3 = 5.801 \text{ ft}^3/\text{lb}_m$$

$$s_{4,s} \text{ at 2 psia} = s_3 = 1.6473 = 0.1750 + x_{4,s}(1.7450)$$

Therefore

$$x_{4,s} = 0.8437$$

$$h_{4,s} = 94.03 + 0.8437(1022.1) = 956.4 \text{ Btu/lb}_m$$

$$\text{Isentropic turbine work} = h_3 - h_{4,s} = 1204.6 - 956.4$$

$$= 248.2 \text{ Btu/lb}_m$$

$$\text{Actual turbine work} = 0.82 \times 248.2 = 203.5 \text{ Btu/lb}_m$$

$$h_4 = 1204.6 - 203.5 = 1001.1 \text{ Btu/lb}_m$$

$h_{5,a}$ (ignoring pump work) $= 94.03$ Btu/lb$_m$

$$h_7 = h_f \text{ at } 70°\text{F} = 38.05 \text{ Btu/lb}_m$$

$$\text{Turbine steam flow} = \frac{100 \times 3.412 \times 10^6}{203.5 \times 0.9} = 1.863 \times 10^6 \text{ lb}_m/\text{h}$$

$$\text{Turbine volume flow} = \frac{1.863 \times 10^6 \times v_3}{60} = 1.8 \times 10^5 \text{ ft}^3/\text{min}$$

$$\text{Cooling-water flow } \dot{m}_7. \quad \dot{m}_7(h_5 - h_7) = \dot{m}_4(h_4 - h_5)$$

Therefore

$$\dot{m}_7 = \frac{1001.1 - 94.03}{94.03 - 38.05}\dot{m}_4 = 16.2\dot{m}_4 = 16.2 \times 1.863 \times 10^6$$

$$= 30.187 \times 10^6 \text{ lb}_m/\text{h}$$

$$\text{Heat added} = h_1 - h_6 = 1204.6 - 94.03 = 1110.57 \text{ Btu/lb}_m$$

$$\text{Plant efficiency} = \frac{203.5 \times 0.9}{1110.57} = 0.1649 = 16.49\%$$

$$\text{Plant heat rate} = \frac{3412}{0.1649} = 20{,}690 \text{ Btu/kWh}$$

H_2S Removal

H_2S is found in the Geysers steam at concentrations around 200 ppm. It is toxic, noxious, and poses major air quality problems. Because of recent environmental regulations on its release to the atmosphere, the latest Geysers units (13 through 17) use conventional shell-and-tube surface-type condensers so that the cooling water does not mix with the turbine exhaust until after the noncondensable gases have been removed. This reduces plant efficiency somewhat because surface condensers are less effective than direct-contact ones. In addition, a process called the *Stretford process* is used to remove H_2S. This process was originally developed for the coal industry and is said to achieve more than 90 percent H_2S abatement when used in conjunction with a surface condenser, and commercial-grade sulfur is produced as a by-product [111]. Figure 12-7 shows unit No. 15 of the Geysers, a 59-MW unit that uses the Stretford process (shown to the left in the picture). The system has not always proven satisfactory in operation, however.

Pacific Gas and Electric Co., DOE, and EPRI are testing methods of upstream H_2S abatement. One approach that has proven technically feasible in small-scale experiments operates on steam upstream of the turbine. That steam is cooled and condensed in a vessel at a temperature where H_2S and other undesirable gases do not condense and are removed. The purer water is then reevaporated and sent to the turbine. The loss of heat in the condensing process is reduced by a regenerative-type heat exchanger that is placed so that the incoming steam reevaporates the condensed steam.

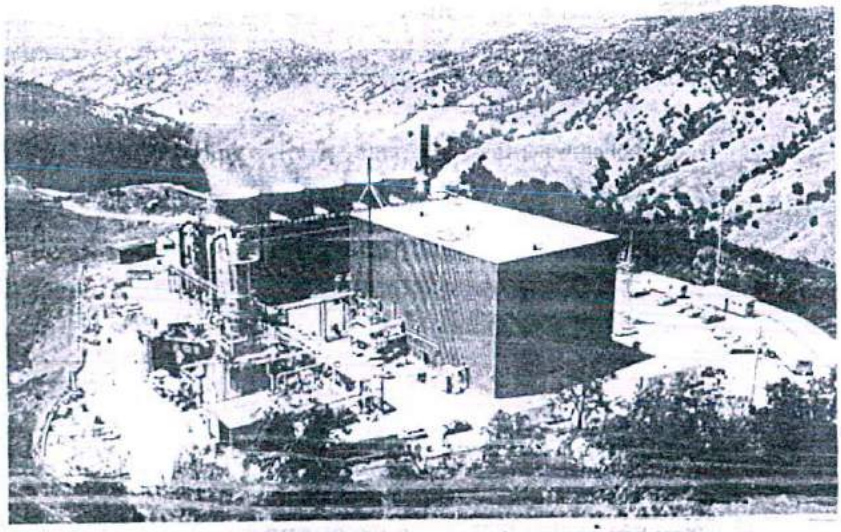


Figure 12-7 59-MW unit No. 15 of the Geysers, with H_2S removal by the Stretford method shown to the left. *(Courtesy Pacific Gas and Electric Co.)*

Some loss of availability, however, does occur, thus posing a further penalty on cycle efficiency. Studies have shown this to be of the order of a few percent. The concept is simple, economical, and has the advantage of removing the gases before they reach the turbine. The experiments showed a 94 percent H_2S abatement. A scaled-up pilot unit is planned for the mid-1980s.

12-6 LIQUID-DOMINATED SYSTEMS: FLASHED STEAM

Although the largest geothermal power generation to date (1982) comes from vapor-dominated systems, these systems are rare, and the natural expansion of generation must come from liquid-dominated systems, which are much more abundant, though not so much as geopressured or petrothermal systems. However, as indicated earlier, liquid-dominated systems require the least extension of technology. The known resources show that water is available above 300°F (150°C), with some up to 600°F (315°C). When tapped, the water can flow naturally under its own pressure or be pumped to the surface. The drop in pressure causes it to partially flash into steam and arrive at the well head as a low-quality, i.e., liquid-dominated, two-phase mixture.

The water comes with various degrees of salinity, ranging from 3000 to 280,000 ppm of dissolved solids, and at various temperatures. There are, therefore, various systems for converting liquid-dominated systems into useful work that depend upon these variables. Two methods stand out: (1) the *flashed-steam system*, suitable for water in the higher-temperature range, and covered in this section, and (2) the *binary-cycle system*, suitable for water at moderate temperatures (Sec. 12-7). A third method, called the *total-flow system*, awaits further development (Sec. 12-8).

The Flashed-Steam System

This system, reserved for water in the higher-temperature range, is illustrated by the flow and *T-s* diagrams of Figs. 12-8 and 12-9. Water from the underground reservoir at 1 reaches the well head at 2 at a lower pressure. Process 1-2 is essentially a constant enthalpy throttling process that results in a two-phase mixture of low quality at 2. This is throttled further in a flash separator resulting in a still low but slightly higher quality at 3. This mixture is now separated into dry saturated steam at 4 and saturated brine at 5. The latter is reinjected into the ground.

The dry steam, a small fraction of the total well discharge (because of the low quality at 3), and usually at pressures below 100 psig (8 bar), is expanded in a turbine to 6 and mixed with cooling water in a direct-contact condenser with the mixture at 7 going to a cooling tower in the same fashion as the vapor-dominated system. The balance of the condensate after the cooling water is recirculated to the condenser is reinjected into the ground.

Example 2-2 A flashed-steam system such as that shown in Fig. 12-8 uses a hot-water reservoir that contains water at 460°F and 160 psia. The separator pressure

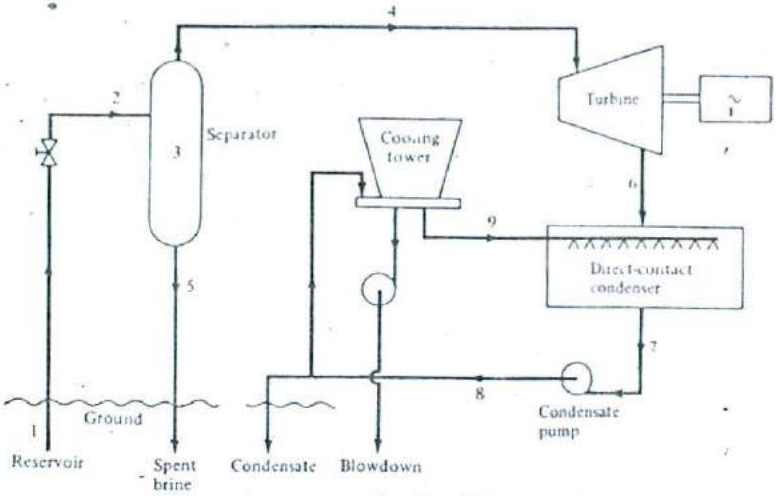


Figure 12-8 Schematic of a liquid-dominated single-flash steam system.

is 100 psia. Find (1) the mass-flow rate of water from the well and of reinjected brine per unit mass-flow rate of steam into the turbine and (2) the ratio of enthalpies of spent brine to steam.

SOLUTION

$$h_1 \approx h_f \text{ at } 460°\text{F} = 441.5 \text{ Btu/lb}_m$$

$$h_3 = h_1 = (h_f + x_3 h_{fg})_{100 \text{ psia}}$$

$$441.5 = 298.5 + x_3(888.6)$$

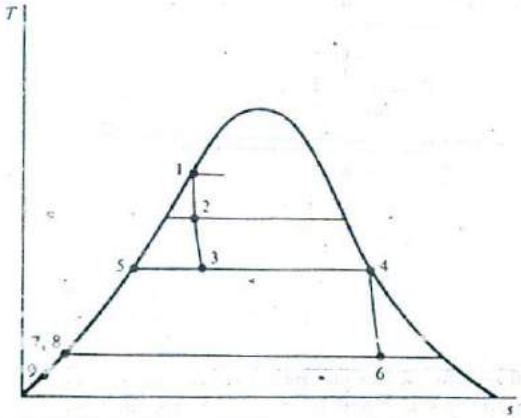


Figure 12-9 *T-s* diagram of the cycle shown in Fig. 12-8.

Therefore

$$x_3 = 0.161$$

1. Mass of water from well per unit mass of steam $= 1/x_3 = 6.21$.
   Mass of reinjected brine per unit mass of steam $= 6.21 - 1 = 5.21$.
2. Ratio of enthalpy at 5 to enthalpy at 4 $= 5.21(h_5/h_4) = 5.21 \times (298.54/1187.2) = 1.31$.

The flashed-steam system is a more difficult proposition than the vapor-dominated system for several reasons: (1) much larger total mass-flow rates through the well, as shown by the above example; (2) a greater degree of ground surface subsidence as a result of such large flows; (3) a greater degree of precipitation of minerals from the brine, resulting in the necessity for design of valves, pumps, separator internals, and other equipment for operation under scaling conditions; and (4) greater corrosion of piping, well casing, and other conduits.

Flashed-steam systems have been widely used in Japan, New Zealand, Italy, Mexico, and elsewhere. An example is the 10-MW Onuma plant in Akita Prefecture in northern Honshu, Japan, which has been in operation since 1973. In this plant steam enters the turbine at 127°C (661°F) and 2.45 bar (35.6 psia). Another is the 75-MW Cerro Prieto plant located in the Mexicali-Imperial Rift Valley in Mexico, 35 km south of the U.S. border. It has been in operation since 1973. Additional units are being built there with a potential between 400 and 1000 MW. In the United States, development has lagged because of the availability of lower-cost energy sources. With the energy crisis, however, activity in this field began with a 10-MW pilot plant built by the Southern California Edison Company and Union Oil Company of California that went into operation in 1981 in Brawley, California. Several plants, in the 20- to 50-MW range are being planned in California, Nevada, New Mexico, and Utah.

Improvements in the Flashed-Steam System

The spent brine leaving the separator at 5 (Fig. 12-8) has a large mass-flow rate and a large total energy compared with that in the steam used to drive the turbine at 4. In Example 12-2, the ratio of the brine enthalpy to the steam enthalpy was found to be 1.31:1. Improvements in the cycle would therefore use some of this otherwise lost energy in the cycle. Two methods are being developed:

1. *Double flash.* Depending upon the original water conditions, the brine at 5 is admitted to a second, lower-pressure separator, where it flashes to a lower-pressure steam that would be admitted to a low-pressure stage in the turbine. The new lower-pressure brine carries less energy with it and represents a reduced energy loss to the cycle. Figures 12-10 and 12-11 show a schematic flow and T-s diagram of a double-flash steam system. The saturated brine from the first-stage flash separator at 5 is reflashed in a second-stage separator at lower pressure to 6. The lower-pressure steam from that separator is admitted to the admission turbine at a lower-pressure stage. The remaining spent brine at 8 is reinjected into the ground. An example of the double-flash system is the 50-MW Hatchobaru plant built on the island of Kyushu in Japan.

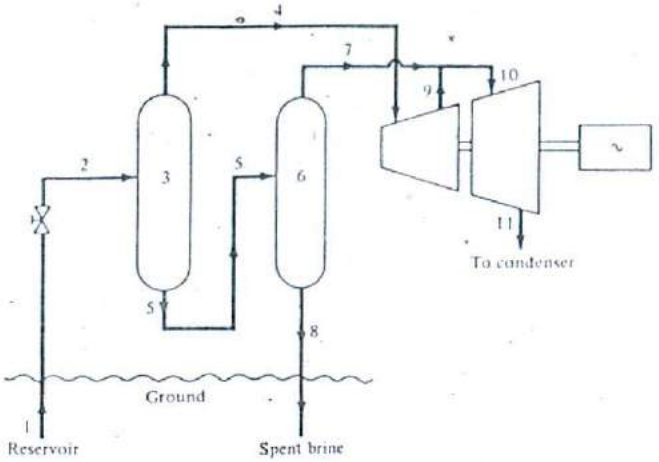


Figure 12-10 Schematic of a liquid-dominated, double-flash steam system. The cycle below the turbine is the same as in Fig. 12-8.

It uses an innovative steam condenser and gas extraction system and a dual-admission, double-flow steam turbine.

2. *Water turbine.* Here the spent brine at 5, still at high pressure, is used instead to drive a water turbine and an additional electric generator operating in parallel with the steam-turbine generator. A variation of this principle, being development under an EPRI contract, uses a so-called *rotary separator turbine* (RST). In this system, the

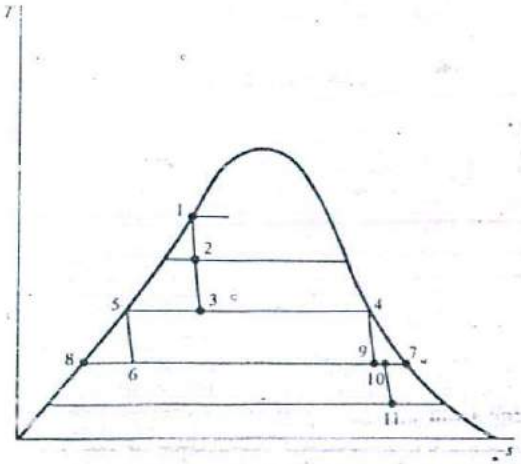


Figure 12-11 *T-s* diagram of the cycle shown in Fig. 12-10.

fluid leaving the well head as a two-phase mixture of steam and water is partially expanded in a nozzle. This increases the steam fraction (as was done in the separator of Fig. 12-8) but also imparts a higher kinetic energy to the denser water, thus facilitating separation of the two phases. This separation takes place in a rotating drum by centrifugal acceleration. The steam is admitted to a steam turbine in the usual fashion. The water at high kinetic energy is used in a special liquid turbine, after which it is reinjected into the ground. A 20-kW unit based on this system is undergoing tests at Roosevelt Hot Springs, Utah. Initial results show a 15 to 20 percent improvement in utilization of the original water energy over a single flashed-steam system. Larger RST units are being developed for possible commercial use in the late 1980s [113].

12-7 LIQUID-DOMINATED SYSTEMS: BINARY CYCLE

About 50 percent of hydrothermal water is in the moderate temperature range of 300 to 400°F (~150 to 205°C).* If used in a flashed-steam system, it would have to be throttled down to such low pressures that result in excessively large specific volume flows as well as even poorer cycle efficiencies. Instead this water is used as a heat source for a closed cycle that uses another working fluid that has suitable pressure-temperature-volume characteristics. This is likely to be an organic with a low boiling point, such as isobutane (2-methyl propane) C_4H_{10} (normal boiling point at one atm pressure: 14°F, −10°C), Freon 12 (normal boiling point: −21.6°F, −29.8°C) (App. B), ammonia (App. C), or propane (App. D). The working fluid would operate at higher pressures, corresponding to the source-water and heat-sink temperatures.

Figure 12-12 shows a schematic flow diagram of a binary-cycle system. Hot water or brine from the underground reservoir circulates through a heat exchanger and is pumped back to the ground. In the heat exchanger it transfers its heat to the organic fluid thus converting it to a superheated vapor that is used in a standard closed Rankine cycle. The vapor drives the turbine and is condensed in a surface condenser; the condensate is pumped back to the heat exchanger. The condenser is cooled by water from a natural source, if available, or a cooling-tower circulation system. The blowdown from the tower may be returned to the ground with the cooled brine. Makeup of the cooling-tower water must be provided, however.

In the binary cycle there are no problems of corrosion or scaling in the working cycle components, such as the turbine and condenser. Such problems are confined only to the well casing and the heat exchanger. The heat exchanger is a shell-and-tube unit so that no contact between brine and working fluid takes place.

The first binary cycle was installed in the Soviet Union on the Kamchatka peninsula in 1967. It had a gross output of 680 kW using a low-temperature water reservoir at 80°C (176°F) and Freon 12 as working fluid. The in-plant power consumption (pump-

* Water in lower temperature ranges is unsuitable for power production. It is however suitable for direct utilization for domestic and industrial process heating.

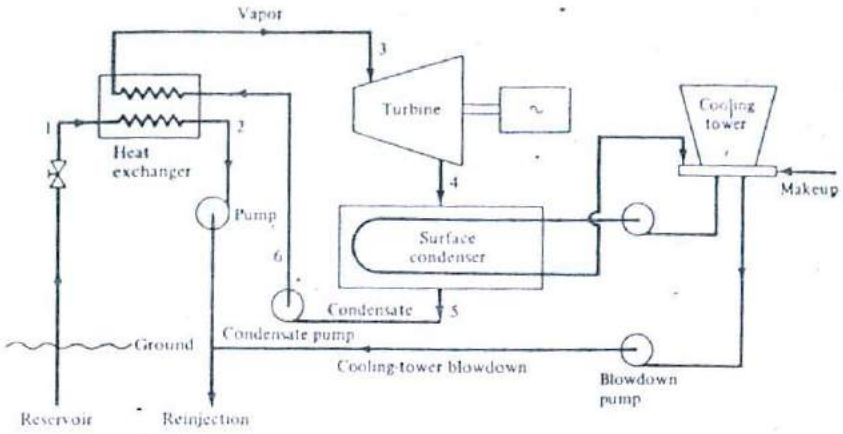


Figure 12-12 Schematic of liquid-dominated binary-cycle system.

ing, etc.) was 35 percent; the net output was 440 kW. The first binary cycle to be built in the United States is an 11-MW plant built by the Magma Company at East Mesa, in the Imperial Valley site in California. The site has a potential of 10,000 MW, but a more conservative projection is about 1500 MW, including a 260-MW unit at East Mesa as well as others at Heber, Westmoreland, Brawley, and Salton Sea.

The $14 million unit at East Mesa has production wells at depths of 5200 to 7500 ft (1585 to 2290 m) which receive 360°F (182°C) pressurized brine to 11 shell-and-tube heat exchangers. The heat exchangers are of the one-pass, counterflow type with very long tubes and no baffles to minimize scaling. The brine leaves the heat exchangers at 180°F and is reinjected into the ground. The brine has 9000 to 10,000 ppm of dissolved solids (mainly calcium chloride and salt), another reason for using a binary cycle. The working fluid is isobutane. A parallel loop using propane is provided. Isobutane was chosen in preference to Freon-12, primarily because of its lower cost. The Magma Company contracted with San Diego Gas and Electric Company to sell it East Mesa power at 25 mill/kWh.

The circulating-water system used in the isobutane condenser consists of two cooling ponds, one of which contains sprays used only at night to take advantage of the cool night-air to cool the warm water. The cooled water goes to the other pond, which has sufficient capacity for cooling the plant during daytime.

A second U.S. binary-cycle plant is being built at Raft River, Idaho. It is a 10-MW plant that uses water at 300°F (~150°C). A full scale $122 million, 45-MW demonstration plant is planned for construction at Heber in the Imperial Valley. It is co-sponsored by EPRI, DOE, San Diego Gas and Electric Company, and various other agencies.

12-8 LIQUID-DOMINATED SYSTEMS: TOTAL-FLOW CONCEPT

In the flashed-steam system, some useful energy is discarded with the separated brine regardless of how many stages of separation are used. Thermodynamically, therefore, direct expansion of the fluid from the well head to the condenser has the potential of converting the greatest fraction of available energy in the fluid to mechanical work. This means that the total well-head flow is to be expanded to the condenser pressure, hence the name *total-flow concept* [114]. In principle this concept is simple, as can be seen from the flow and T-s diagrams of Figs. 12-13 and 12-14. Again, hot brine from the well at 1 is throttled to 2, where it becomes a two-phase mixture of low quality. Instead of separating the two phases at this point, the full flow is expanded to 3, condensed to 4, and reinjected into the ground at 5. Comparing the T-s diagrams of Figs. 12-9 and 12-14 shows that the throttling process 2-3 that occurs in the flash separator in the former is no longer necessary and that, considering equal pressures at 2, the full available energy at 2 is used in the latter, while part of it is destroyed in the former as a result of throttling. In addition the flow in the flashed-steam turbine and hence the work per unit flow from the well head is only a small fraction, equal to the quality in the flash separator, x_3, of the total flow and work that would occur in the latter.

Flashed-steam systems, and for that matter vapor-dominated systems, rely on axial-flow multistage steam turbines similar to those used in conventional powerplants except that they are designed for much lower pressures. These turbines use relatively clean high-quality or even superheated steam. The total-flow concept, on the other

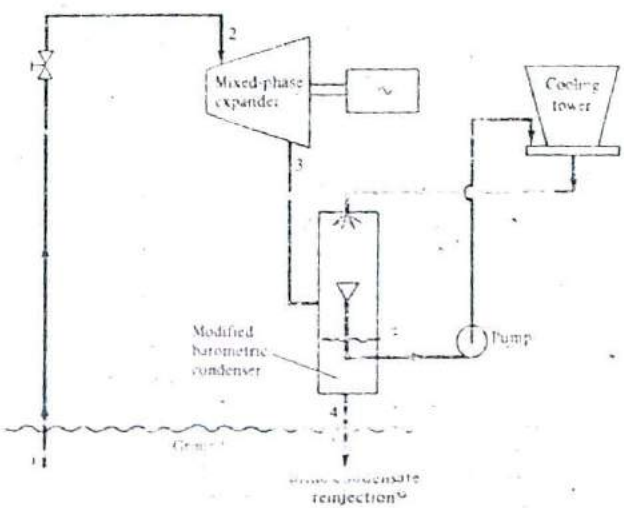


Figure 12-13 Schematic of a liquid-dominated total-flow concept.

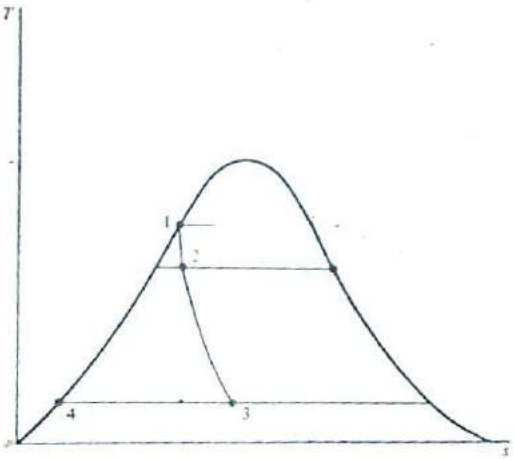


Figure 12-14 T-s diagram of cycle in Fig. 12-13.

hand, requires the use of a *mixed-phase expander* powered by a two-phase mixture of low quality (2, Fig. 12-14). Such expanders must be able to overcome the losses associated with the impingement of liquid droplets on blades (turbines operate less efficiently as the quality decreases). They must also be able to withstand the corrosive and erosive effects of the significant quantities of dissolved solids in the brine. Such expanders have not yet been developed, although experimental and analytical work at Lawrence Livermore Laboratory, California, has taken place [113].

The considerations mentioned above point to an expander that is of simple and clean design, with minimum contact surface and minimum number of moving parts. It must also be easy to service and maintain and have long-term reliability. It appears from these requirements and from a study of a variety of expanders, including impulse, reaction, axial and radial flow, and positive displacement (helical screw and oscillating vane), that an impulse, single-stage (De Laval) turbine would be most suitable, as it has the advantage of mechanical simplicity and small size (although impulse staging may work in some cases). Recall from Sec. 5-3 that in the De Laval turbine full expansion of the well-head fluid from 2 to 3 would occur in a converging-diverging nozzle that converts the fluid enthalpy drop to kinetic energy at the turbine back pressure. The kinetic energy is then converted to mechanical work by the impulse blading.

Problems to be solved before the total-flow concept can be a commercial success include brine management; inhibition of precipitation and scale; turbine material selection for maximum erosion resistance; the handling of the condensate, which is in the form of a slurry (solids in suspension rather than in solution in a liquid); and the carryover into the coolant system. The condenser, for example, would be a modified direct-contact barometric condenser (Sec. 6-2) designed to separate the vapor fraction from the brine fraction to prevent fouling of the circulating-water system. No water

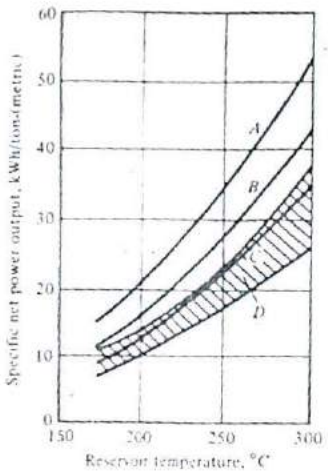


Figure 12-15 Comparison of various liquid-dominated systems: (*A*) total-flow concept, (*B*) two-stage flash, (*C*) single-stage flash, (*D*) binary-cycle range [114].

makeup in the cooling tower is required as that can be supplied from the condensed steam before reinjection into the ground.

Figure 12-15 shows the results of an analytical comparison of various liquid-dominated systems. The curves indicate that the total-flow concept produces the highest specific power per unit mass-flow rate at the well head. The calculations were based on the same condenser temperatures of 120°F, the same turbine efficiencies of 85 percent, and the same in-plant power requirements of 30 percent of the gross. Differences in these would naturally change the differences between the curves.

12-9 PETROTHERMAL SYSTEMS

As indicated earlier (Sec. 12-3), petrothermal systems are those that are composed of hot dry rock (HDR) but no underground water. They represent by far the largest geothermal resource available. The rock, occurring at moderate depths, has very low permeability and needs to be fractured to increase its heat-transfer surface.

The thermal energy of the HDR is extracted by pumping water (or other fluid) through a well that has been drilled to the lower part of the fractured rock. The water moves through the fractures, picking up heat. It then travels up a second well that has been drilled to the upper part of the rock and finally back to the surface. There, it is used in a powerplant to produce electricity.

A feature of this scheme is that, as the reservoir heat is depleted with time, temperature differences within the rock result in stresses that cause the original fractures to propagate, thereby unlocking more HDR surface to the water and resulting in a pancake-shaped fracture zone.

Figure 12-16 shows the petrothermal concept as investigated by the Los Alamos Scientific Laboratory (LASL). It envisages a fracture zone in crystalline rock with bottom pipe at about 7500 ft (~2300 m) below the surface, water in at 1610 psia (111 bar) and 65°C (150°F) and out at 2000 psia (138 bar) and 280°C (540°F), and sufficient flow for a 150-MW heat exchanger.

It is believed that HDR systems offer more flexibility in operation and design than other geothermal systems. For example, the designer can have a choice of water flow rates and temperatures by drilling to various depths, and the operator can change pumping pressure and hence flow rates to suit load conditions.

Problems that are faced by developers include leakage of water (or other fluid) underground and the necessity of makeup for it from resources above ground, the effect of the water or fluid on rock composition, material carryover with the fluid, and

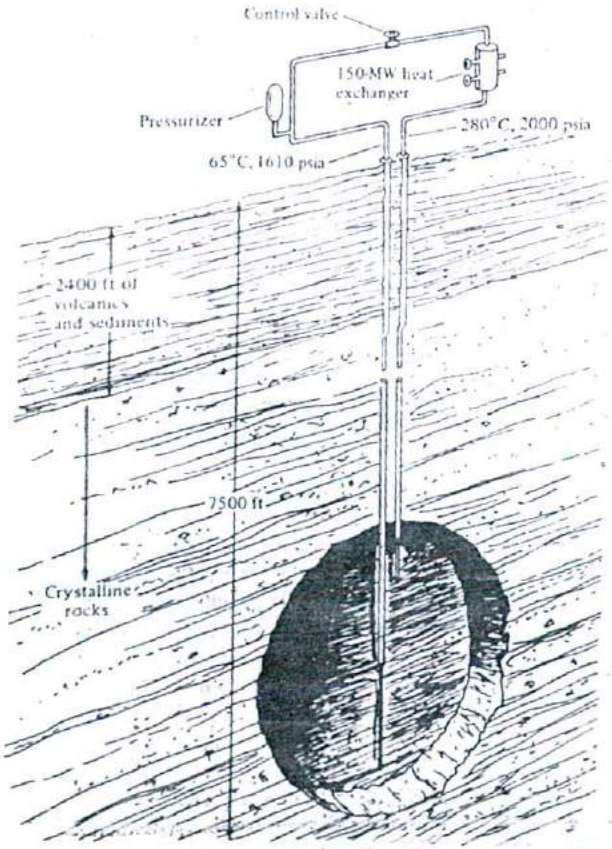


Figure 12-16 Petrothermal energy extraction (LASL).

cost. It should be noted that two wells are to be drilled instead of the one for hydrothermal energy and that these wells are drilled deeper and in much harder rock. This is expected to make petrothermal exploitation very costly, unless the underground rock being developed is very hot.

Many more studies on the mechanical, thermodynamic, and economic aspects of petrothermal systems are necessary before commercial exploitation becomes feasible. So far (1982), LASL has drilled several test wells at a facility near Fenton Hill, New Mexico. Based on these tests, consideration is being given to constructing a 40- to 50-MW plant for start-up, tentatively, in the late 1980s.

12-10 HYBRID GEOTHERMAL-FOSSIL SYSTEMS

The concept of hybrid geothermal-fossil-fuel systems utilizes the relatively low-temperature heat of geothermal sources in the low-temperature end of a conventional cycle and the high-temperature heat from fossil-fuel combustion in the high-temperature end of that cycle. The concept thus combines the high-efficiency of a high-temperature cycle with a natural source of heat for part of the heat addition, thus reducing the consumption of the expensive and nonrenewable fossil fuel.

There are two possible arrangements for hybrid plants [115]. These are (1) *geothermal preheat*, suitable for low-temperature liquid-dominated systems, and (2) *fossil superheat*, suitable for vapor-dominated and high-temperature liquid-dominated systems.

Geothermal-Preheat Hydrid Systems

In these systems the low-temperature geothermal energy is used for feedwater heating of an otherwise conventional fossil-fueled steam plant. Geothermal heat replaces some or all, of the feedwater heaters, depending upon its temperature. A cycle operating on this principle is illustrated in Fig. 12-17. As shown, geothermal heat heats the feedwater throughout the low-temperature end prior to an open-type deaerating heater (DA) (Sec. 2-8). The DA is followed by a boiler feed pump and three closed-type feedwater heaters with drains cascaded backward (Sec. 2-9). These receive heat from steam bled from higher-pressure stages of the turbine. No steam is bled from the lower-pressure stages because geothermal brine fulfills this function.

Fossil-Superheat Hybrid Systems

In these systems, the vapor-dominated steam, or the vapor obtained from a flash separator in a high-temperature liquid-dominated system, is superheated in a fossil-fired superheater.

Figures 12-18 and 12-19 show schematic flow and T-s diagrams of a system proposed in [115]. It comprises a double-flash geothermal steam system. Steam produced at 4 in the first-stage flash separator is preheated from 4 to 5 in a regenerator by exhaust steam from the high-pressure turbine at 7. It is then superheated by a fossil

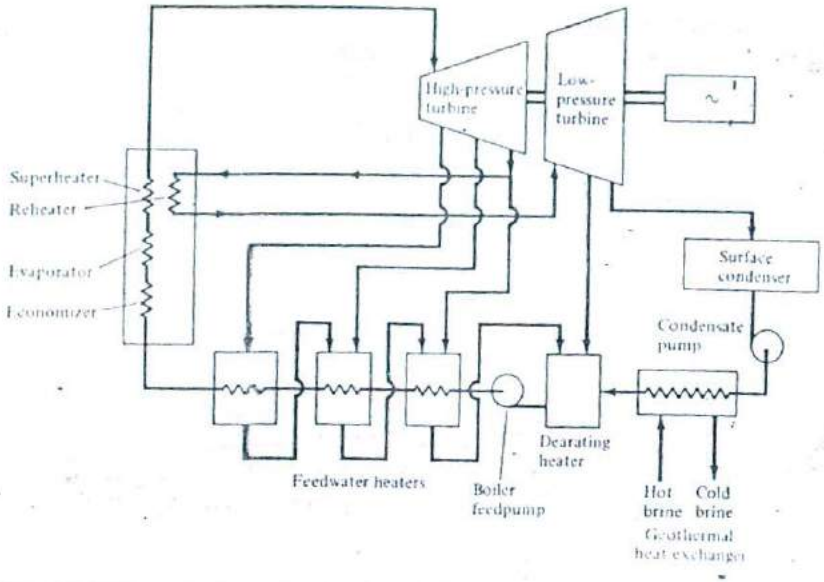


Figure 12-17 Schematic of a geothermal-preheat hybrid system.

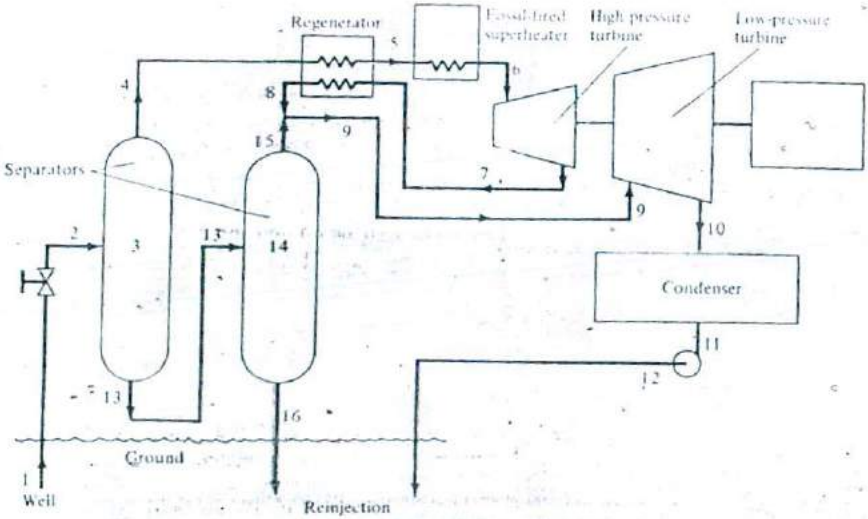


Figure 12-18 Schematic of a fossil-superheat hybrid system with two-stage flash evaporation, regenerator, and fossil-fired superheater.

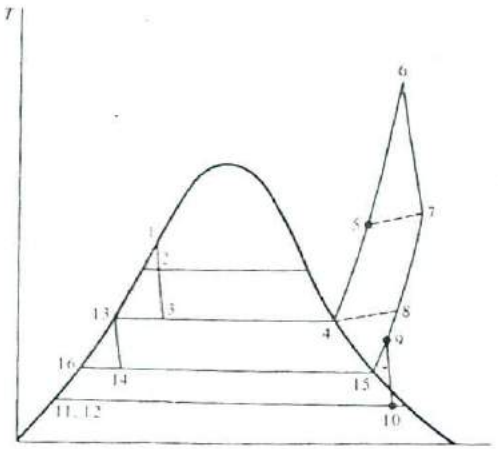


Figure 12-19 *T-s* diagram of cycle in Fig. 12-18

fuel-fired superheater to 6 and expands in the high-pressure turbine to 7 at a pressure near that of the second-stage steam separator. It then enters the regenerator, leaves it at 8, where it mixes with the lower-pressure steam produced in the second-stage flash separator at 15, and produces steam at 9, which expands in the lower-pressure turbine to 10. The condensate at 11 is pumped and reinjected into the ground at 12. The spent brine from the second-stage evaporator is also reinjected into the ground at 16.

PROBLEMS

12-1 Using a *T-s* diagram, (*a*) compare the range of temperatures and/or steam qualities or superheats of liquid-dominated and vapor-dominated hydrothermal systems at ground level, and (*b*) calculate the range of well mass flow rates necessary to produce a unit dry-steam mass flow rate at turbine inlet.

12-2 A geopressured well produces water at the surface at a temperature of 240°F, a salinity of 6 mass percent of pure water, and a methane gas content of 25 ft^3 per barrel (U.S. liquid) of water. The water flow rate is 50,000 bbl/day. Calculate (*a*) the mass of solids to be removed, in pounds mass per day, (*b*) the available thermal power in pure water, in Btus per hour and megawatts, and (*c*) the available chemical power, in Btus per hour and megawatts. Take water specific heat = 1 $Btu/lb_m \cdot °R$, and methane higher heating value = 23,560 Btu/lb_m.

12-3 A 250-MW vapor-dominated hydrothermal powerplant uses well steam that is saturated at 450 psia at shutoff. The steam is throttled to a turbine inlet pressure of 140 psia. A direct-contact condenser operates at a pressure of 5 psia with a cooling-water inlet temperature of 48°F. The turbine polytropic efficiency is 0.80, and the turbine-generator combined mechanical-electrical efficiency is 0.90. Calculate (*a*) the steam mass flow rate, in pounds mass per hour, (*b*) the condenser cooling water mass flow rate, in pounds mass per hour, (*c*) the cycle thermal efficiency and heat rate, in Btus per kilowatt hour, and (*d*) the difference between well flow and disposal flow, in pounds mass per hour, if the cooling tower losses by evaporation and drift are 2 percent.

12-4 A vapor-dominated hydrothermal field is capable of supplying 2500 kg/s of underground steam at 215°C. The turbines operate with throttle pressures of 6 bar and have a polytropic efficiency of 0.8. The turbine generators have a combined mechanical-electrical efficiency of 0.875. The condensers are of the surface type (for H_2S removal) and operate at 0.4 bar with cooling water at 26°C, and a terminal temperature difference of 10°C. Determine (*a*) the total net capacity of the field, in megawatts, if the steam jet ejectors use 1 percent of the steam, and the auxiliaries are estimated to consume 25 percent of the gross output, (*b*) the gross and net plant efficiencies and (*c*) the necessary total cooling water flow, in kilograms per second and cubic meters per second.

12-5 A 50-MW turbine receives hydrothermal steam at 100 psia and 350°F and exhausts to a surface condenser at 5 psia. The turbine polytropic efficiency is 80 percent. Estimate the mass of hydrogen sulfide that can be removed from the condenser, in pounds mass per hour.

12-6 Consider a hydrogen sulfide abatement method from hydrothermal steam upstream of the turbine. Steam comes into the system at 100 psia and 350°F, passes through a heat exchanger to reevaporate condensed steam, and then to a second heat exchanger where it is cooled further, leaving as saturated condensate at 324°F. After H_2S and other noncondensables are removed, the condensate flows back through the first heat exchanger, where it evaporates to saturated steam at 90 psia. This steam now expands isentropically in a turbine to 5 psia. Draw the flow diagram of the system and calculate (*a*) the heat transferred in the two heat exchangers, in Btus per pound mass, and (*b*) the loss in isentropic work due to this process, in Btus per pound mass.

12-7 A liquid-dominated geothermal plant with a single flash separator receives water at 400°F. The separator pressure is 150 psia. A direct-contact condenser operates at 5 psia. The turbine has a polytropic efficiency of 0.75. For a cycle output of 10 MW, find (*a*) the steam mass flow rate, in pounds mass per hour, (*b*) the well water mass flow rate, in pounds mass per hour, (*c*) the reinjected brine mass flow rate, in pounds mass per hour, (*d*) the cycle efficiency, and (*e*) the cooling water mass flow rate if such water is available at 80°F. Ignore the pump work.

12-8 A liquid-dominated geothermal plant with double flash separators receives water at 440°F. The pressures in the separators are 150 and 80 psia. A direct contact condenser operates at 5 psia. The two turbine sections have polytropic efficiencies of 0.75. For a cycle output of 10 MW calculate, (*a*) the high- and low-pressure steam mass flow rates, in pounds mass per hour, (*b*) the well water mass flow rate, in pounds mass per hour, (*c*) the reinjected brine mass flow rate, in pounds mass per hour, and (*d*) the cycle efficiency. Compare these answers with those of Prob. 12-7, which has the same data but a single flash system. Ignore the pump work.

12-9 A geothermal plant uses a liquid-dominated heat source and a binary cycle. 5×10^6 lb_m/h of underground brine enter a shell-and-tube heat exchanger at 280°F and leave at 110°F to be reinjected into the ground. The working fluid, Freon-12, leaves the heat exchanger as saturated vapor at 230°F. It expands in a turbine that has a polytropic efficiency of 0.70 to a surface condenser at 128.24 psia. The mechanical-electrical efficiency of the turbine-generator is 0.9. The condensate pump has a polytropic efficiency of 0.65. Calculate (*a*) the Freon-12 mass flow rate, in pounds mass per hour, (*b*) the plant power, in megawatts, and (*c*) the plant efficiency and heat rate, in Btus per kilowatt-hour.

12-10 A 10-MW binary cycle geothermal powerplant uses ammonia as working fluid. Brine from the ground enters and leaves the vapor generator at 320°F and 120°F, respectively. Ammonia vapor is generated at 200 psia and 300°F and expands in the turbine to 120 psia with a polytropic efficiency of 0.72. Calculate (*a*) the turbine and net cycle work, in Btus per pounds mass of NH_3, (*b*) the mass flow rate of ammonia, in pounds mass per hour, if the combined mechanical-electrical efficiency of the turbine generator is 0.80, (*c*) the mass flow rate of underground water, in pounds mass per hour, and (*d*) the cycle efficiency.

12-11 A hot-water geothermal plant of the total-flow type receives water at 440°F. The pressure at turbine inlet is 150 psia. The plant uses a direct contact condenser that operates at 5.0 psia. The turbine has a polytropic efficiency of 0.65. For a cycle net output of 10 MW, calculate (*a*) the hot water flow, in pounds mass per hour, (*b*) the condenser cooling water flow, in pounds mass per hour, if such water is available at 80°F, and (*c*) the cycle efficiency. Ignore the pump work.

12-12 A hybrid geothermal-fossil powerplant of the type shown schematically in Fig. 12-18 receives underground water at 440°F. The pressures in the separators are 150 and 80 psia. A direct-contact condenser

operates at 5 psia. The regenerator is 80 percent effective. The high pressure steam is superheated in the fossil-fired superheater to 500°F. Because of operations with superheated steam, the high- and low-pressure turbine sections have polytropic efficiencies of 0.88 and 0.85, respectively. For a cycle output of 10 MW, calculate (*a*) the well water mass flow rate, in pounds mass per hour, (*b*) the high- and low-pressure steam mass flow rates, in pounds mass per hour, (*c*) the heat added in the superheater, in Btus per hour, and (*d*) the cycle efficiency. Ignore the pump work.