Nuclear Fuel Cycle

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Chapter 18
Nuclear Power Plants

Currently, about half of all nuclear power plants are located in the US. There are many different kinds of nuclear power plants, and we will discuss a few important designs in this text. A nuclear power plant harnesses the energy inside atoms themselves and converts this to electricity. All of us use this electricity. In Sect. 18.1 of this chapter we show you should the idea of the fission process and how it works. A nuclear power plant uses controlled nuclear fission. In this chapter, we will explore how a nuclear power plant operates and the manner in which nuclear reactions are controlled. There are several different designs for nuclear reactors. Most of them have the same basic function, but one's implementation of this function separates it from another. There is several classification systems used to distinguish between reactor types. Below is a list of common reactor types and classification systems found throughout the world and they are briefly explained down below according to three types of classification either; (1) Classified by Moderator Material or (2) Classified by Coolant Material and (3) Classified by Reaction Type.

18.1 Fission Energy Generation

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 power plant saves more than 50,000 barrels of oil per day. At $ 30–40 per barrel (1982), such a power plant 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 kilowatt-hour 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 power

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plants 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.

Nuclear fission is the process of splitting atoms, or fissioning them. This page will explain

18.2 The First Chain Reaction

Early in World War II the scientific community in the United States, including those Europeans now calling the US their safe home, pursued the idea that uranium fission and the production of excess neutrons could be the source of extraordinary new weapons. They knew Lisa Meitner's interpretation, in Sweden, of Hahn's experiments would likely be known in Germany. Clearly there might now be a race commencing for the development and production of a new, super weapon based on the fission of ${}^{235}U_{92}$ or ${}^{239}Pu_{94}$.

By early 1942, it was known that the two naturally occurring isotopes of uranium reacted with neutrons as follows:

$$ {}^{235}U_{92} + {}^{1}n_0 \quad \rightarrow \text{fission products} + (2.5)^{1}\, n_0 + 200 \text{ MeV Energy} $$

$$ {}^{238}U_{92} + {}^{1}n_0 \rightarrow {}^{239}\, U_{92} $$

$$ {}^{239}U_{92} \quad \rightarrow {}^{239}\, Np_{93} + ß^{-1} \qquad t_{1/2} = 23.5 \text{ min.} $$

$$ {}^{239}Np_{93} \quad \rightarrow {}^{239}\, Pu_{94} + ß^{-1} \qquad t_{1/2} = 2.33 \text{ days} $$

Each U-235 that undergoes fission produces an average of 2.5 neutrons. In contrast, some U-238 nuclei capture neutrons, become U-239, and subsequently emit two beta particles to produce Pu-239. The plutonium was fissile also and would produce energy by the same mechanism as the uranium. A flow sheet for uranium fission is shown in Fig. 18.1 below.

The answers to two questions were critical to the production of plutonium for atomic bombs:

1. Is it possible, using natural uranium (99.3 % U-238 and 0.7 % U-235), to achieve a controlled chain reaction on a large scale? If so, some of the excess neutrons produced by the fission of U-235 would be absorbed by U-238 and produce fissile Pu-239.

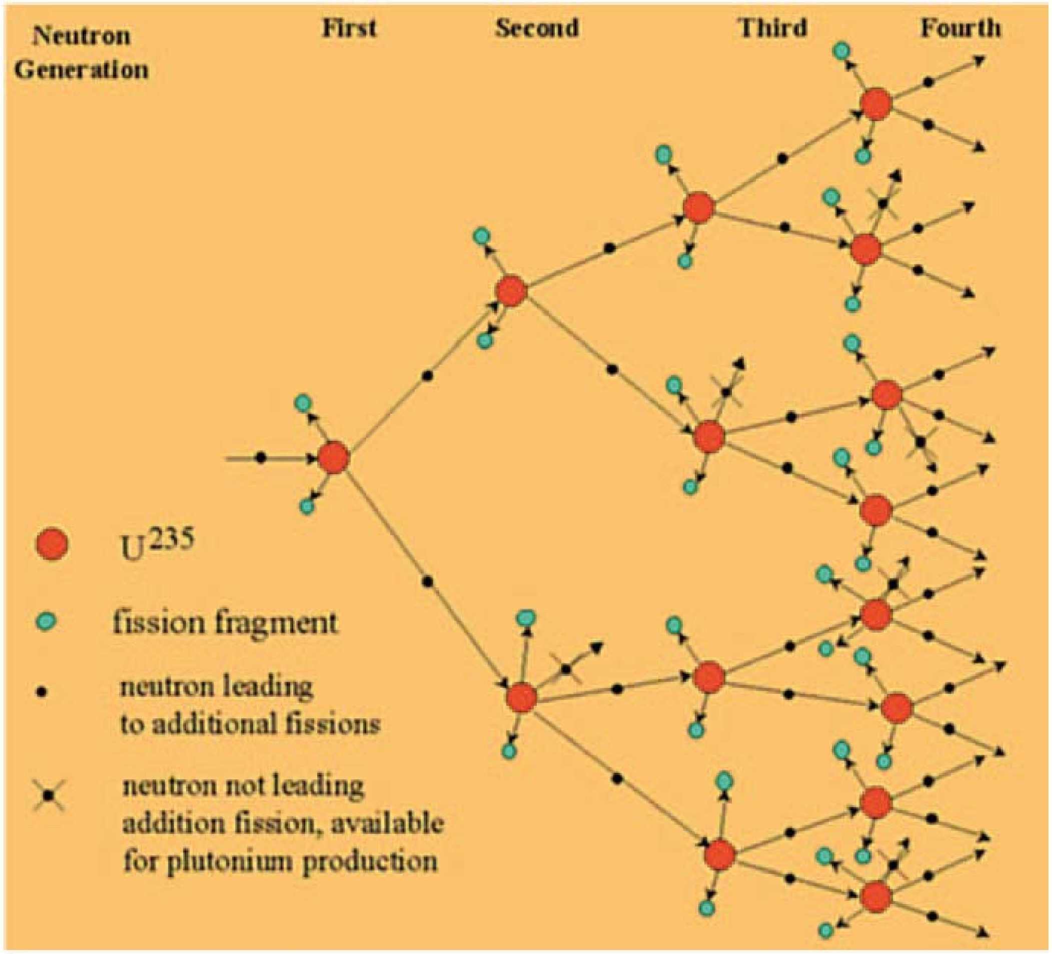


Fig. 18.1 The first generations of a nuclear chain reaction

2. How can we separate (in a reasonable period of time) the relatively small quantities of Pu-239 from the unreacted uranium and the highly radioactive fission-product elements?

Although fission had been observed on a small scale in many laboratories, no one had carried out a controlled chain reaction that would provide continuous production of plutonium for isolation.

Enrico Fermi thought that he could achieve a controlled chain reaction using natural uranium. He had started this work with Leo Szilard at Columbia University, but moved to the University of Chicago in early 1942.

The first nuclear reactor, called a pile, was a daring and sophisticated experiment that required nearly 50 t of machined and shaped uranium and uranium oxide pellets along with 385 t—the equivalent of four railroad coal hoppers—of graphite blocks, machined on site.

The pile itself was assembled in a squash court under the football field at the University of Chicago from the layered graphite blocks and uranium and uranium oxide lumps (Fermi's term) arranged roughly in a sphere with an anticipated

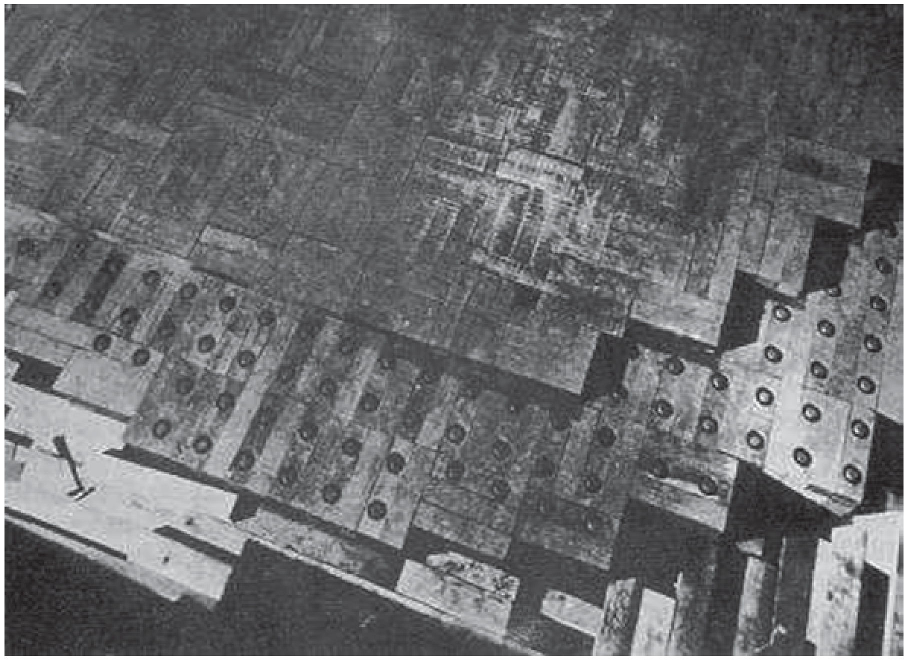


Fig. 18.2 CP-1-Graphite blocks with 3 inch diameter uranium cylinders inserted—part of a layer of CP-1, the first nuclear reactor. A layer of graphite blocks without inserted uranium is seen covering the active layer

13 foot radius. Neutron absorbing, cadmium coated control rods were inserted in the pile. By slowly withdrawing the rods, neutron activity within the pile was expected to increase and at some point, Fermi predicted, there would be one neutron produced for each neutron absorbed in either producing fission or by the control rods (Fig. 18.2).

On December 2, 1942, with 57 of the anticipated 75 layers in place, Fermi began the first controlled nuclear chain reaction occurred. At around 3:20 p.m. the reactor went critical; that is, it produced one neutron for every neutron absorbed by the uranium nuclei. Fermi allowed the reaction to continue for the next 27 min before inserting the neutron-absorbing control rods. The energy releasing nuclear chain reaction stopped as Fermi predicted it would.

In addition to excess neutrons and energy, the pile also produced a small amount of Pu-239, the other known fissile material (Fig. 18.3).

The achievement of the first sustained nuclear reaction was the beginning of a new age in nuclear physics and the study of the atom. Humankind could now use the tremendous potential energy contained in the nucleus of the atom. However, while a controlled chain reaction was achieved with natural uranium, and could produce plutonium, it would be necessary to separate U-235 from U-238 to build a uranium bomb.

On December 28, 1942, upon reviewing a report from his advisors, President Franklin Roosevelt recommended building full-scale plants to produce both U-235 and Pu-239.

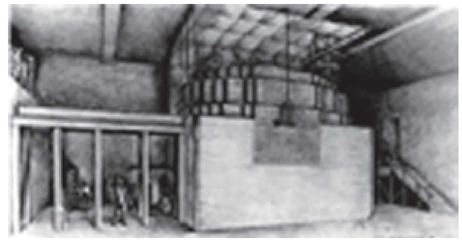


Fig. 18.3 The first controlled chain reaction, Stagg Field, Chicago, December 2, 1942. (Courtesy of the Argonne National Laboratory)

This changed the effort to develop nuclear weapons from experimental work in academic laboratories administered by the U.S. Office of Scientific Research and Development to a huge effort by private industry. This work, supervised by the U.S. Army Corps of Engineers, was codenamed the Manhattan Project. It spread throughout the entire United States, with the facilities for uranium and plutonium production being located at Oak Ridge, Tennessee, and Hanford, Washington, respectively. Work on plutonium production continued at the University of Chicago, at what became known as the Metallurgical Laboratory or Met Lab. A new laboratory at Los Alamos, New Mexico, became the focal point for development of the uranium and plutonium bombs.

18.3 Concepts in Nuclear Criticality

A nuclear reactor works on the principle of a chain reaction. An initial neutron is absorbed by a fissile nuclide and during the process of fission; additional neutrons are released to replace the neutron that was consumed. If more neutrons are produced than are consumed, then the neutron population grows. If fewer neutrons are produced than are consumed, the neutron population shrinks. The number of fissions caused by the neutron population determines the energy released.

In order to quantify this concept let us define a multiplication factor k. We will define k as the ratio of the production to consumption of neutrons.

$$k = \frac{Production}{Consumption} \tag{18.1}$$

18.4 Fundamental of Fission Nuclear Reactors

Today many nations are considering an expanded role for nuclear power in their energy portfolios. This expansion is driven by concerns about global warming, growth in energy demand, and relative costs of alternative energy sources. In 2008, 435 nuclear reactors in 30 countries provided 16 % of the world's electricity. In January 2009, 43 reactors were under construction in 11 countries, with several hundred more projected to come on line globally by 2030.

Concerns over energy resource availability, climate change, air quality, and energy security suggest an important role for nuclear power in future energy supplies. While the current Generation II and III nuclear power plant designs provide a secure and low-cost electricity supply in many markets, further advances in nuclear energy system design can broaden the opportunities for the use of nuclear energy. To explore these opportunities, the U.S. Department of Energy's Office of Nuclear Energy has engaged governments, industry, and the research community worldwide in a wide ranging discussion on the development of next generation nuclear energy systems known as "Generation IV." See Sect. 18.4 of this Chapter for more information on New Generation of Power Plant know as Gen IV (Fig. 18.4).

Nuclear reactors produce energy through a controlled fission chain reaction (See Sect. 1.1 in above: The First Chain Reaction). While most reactors generate electric power, some can also produce plutonium for weapons and reactor fuel. Power reactors use the heat from fission to produce steam, which turns turbines to generate electricity. In this respect, they are similar to plants fueled by coal and natural gas. The components common to all nuclear reactors include a fuel assembly, control rods, a coolant, a pressure vessel, a containment structure, and an external cooling facility.

In a nuclear reactor neutrons interact with the nuclei of the surrounding atoms. For some nuclei (e.g. U-235) an interaction with a neutron can lead to fission: the nucleus is split into two parts, giving rise to two new nuclei (the so-called fission products), energy and a number of new highly energetic neutrons. Other possible interactions are absorption (the neutron is removed from the system), and simple collisions, where the incident neutron transfers energy to the nucleus, either elastically (hard sphere collision) or inelastically.

The speed of the neutrons in the chain reaction determines the reactor type (See Fig. 16.5). Thermal reactors use slow neutrons to maintain the reaction. These reactors require a moderator to reduce the speed of neutrons produced by fission. Fast

Fig. 18.4 A nuclear power plant. (Courtesy of R2 Controls)

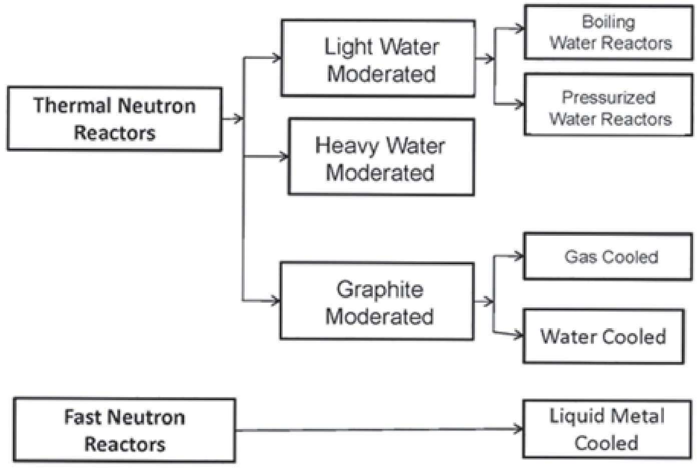


Fig. 18.5 Types of nuclear reactors. (Courtesy of Chem Cases)

neutron reactors, also known as Fast Breeder Reactors (FBR), use high speed, un-moderated neutrons to sustain the chain reaction (Fig. 18.5).

Thermal reactors operate on the principle that uranium-235 undergoes fission more readily with slow neutrons than with fast ones. Light water (H^2O), Heavy water (D_2O), and carbon in the form of graphite are the most common moderators. Since slow neutron reactors are highly efficient in producing fission in uranium-235, they use fuel assemblies containing either natural uranium (0.7 % U-235) or slightly enriched uranium (0.9–2.0 % U-235) fuel. Rods composed of neutron-absorbing material such as cadmium or boron are inserted into the fuel assembly. The position of these control rods in the reactor core determines the rate of the fission chain reaction. The coolant is a liquid or gas that removes the heat from the core and produces steam to drive the turbines. In reactors using either light water or heavy water, the coolant also serves as the moderator. Reactors employing gaseous coolants (CO_2 or He) use graphite as the moderator. The pressure vessel, made of heavy-duty steel, holds the reactor core containing the fuel assembly, control rods, moderator, and coolant. The containment structure, composed of thick concrete and steel, inhibits the release of radiation in case of an accident and also secures components of the reactor from potential intruders. Finally, the most obvious components of many nuclear power plants are the cooling towers, the external components, which provide cool water for condensing the steam to water for recycling into the containment structure. Cooling towers are also employed with coal and natural gas plants.

18.5 Reactor Fundamentals

It is important to realize that while the U-235 in the fuel assembly of a thermal reactor is undergoing fission, some of the fertile U-238 present in the assembly is also absorbing neutrons to produce fissile Pu-239. Approximately one third of the energy produced by a thermal power reactor comes from fission of this plutonium. Power reactors and those used to produce plutonium for weapons operate in dif-

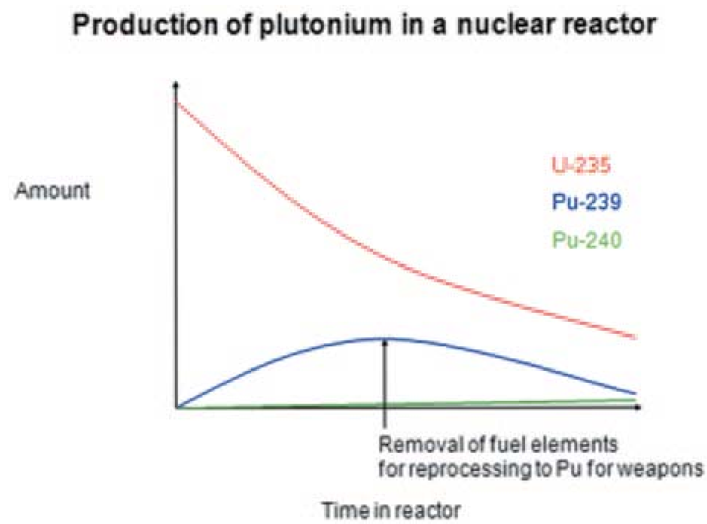


Fig. 18.6 The fate of plutonium in a thermal reactor. (Courtesy of Chem Cases)

ferent ways to achieve their goals. Production reactors produce less energy and thus consume less fuel than power reactors. The removal of fuel assemblies from a production reactor is timed to maximize the amount of plutonium in the spent fuel (See Fig. 15.6). Fuel rods are removed from production reactors after only several months in order to recover the maximum amount of plutonium-239. The fuel assemblies remain in the core of a power reactors for up to 3 years to maximize the energy produced. However, it is possible to recover some plutonium from the spent fuel assemblies of a power reactor (Fig. 18.6).

The power output or capacity of a reactor used to generate electricity is measured in megawatts of electricity, MW(e). However, due to the inefficiency of converting heat into electricity, this represents only about one third of the total thermal energy, MW(t), produced by the reactor. Plutonium production is related to MW(t). A production reactor operating at 100 MW(t) can produce 100 g of plutonium per day or enough for one weapon every 2 months.

Another important property of a reactor is its capacity factor. This is the ratio of its actual output of electricity for a period of time to its output if it had been operated at its full capacity. The capacity factor is affected by the time required for maintenance and repair and for the removal and replacement of fuel assemblies. The average capacity factor for U.S. reactors has increased from 50% in the early 1970s to over 90% today. This increase in production from existing reactors has kept electricity affordable.

18.6 Thermal Reactors

Currently the majority of nuclear power plants in the world are water-moderated, thermal reactors. They are categorized as either light water or heavy water reactors. Light water reactors use purified natural water (H^2O) as the coolant/moderator, while heavy water reactors employ heavy water, deuterium oxide (D^2O). In light water reactors, the water is either pressured to keep it in superheated form (in a pressurized water reactors, PWR) or allowed to vaporize, forming a mixture of water and steam (in a boiling water reactors, BWR). In a PWR (Fig. 16.10), superheated water flowing through tubes in the reactor core transfers the heat generated by fission to a heat exchanger, which produces steam in a secondary loop to generate electricity. None of the water flowing through the reactor core leaves the containment structure. In a BWR (Fig. 16.12), the water flowing through the core is converted to directly to steam and leaves the containment structure to drive the turbines. Light water reactors use low enriched Uranium as fuel. Enriched fuel is required because natural water absorbs some of the neutrons, reducing the number of nuclear fissions. All of the 103 nuclear power plants in the United States are light water reactors; 69 are PWRs and 34 are BWRs.

18.7 Nuclear Power Plants and Their Classifications

A nuclear power plant uses controlled nuclear fission. In this section, we will explore how a nuclear power plant operates and the manner in which, nuclear reactions are controlled. There are several different designs for nuclear reactors. Most of them have the same basic function, but one's implementation of this function separates it from another. There is several classification systems used to distinguish between reactor types. Below is a list of common reactor types and classification systems found throughout the world and they are briefly explained down below according to three types of classification either;

1. Classified by Moderator Material [i.e. Light Water Reactor, or Graphite Moderated Reactor, and Heavy Water Reactor].
2. Classified by Coolant Material [i.e. Pressurized Water Reactor, or Boiling Water Reactor, and Gas Cooled Reactor].
3. Classified by Reaction Type [i.e. Fast Neutron Reactor, or Thermal Neutron Reactor, and Liquid Metal Fast Breeder Reactor].

18.8 Classified by Moderator Material

These types of reactors and their general description are presented below;

18.8.1 Light Water Reactors (LWR)

A light water reactor is a type of thermal reactor that uses "light water" (plain water) as a neutron moderator or coolant instead of using deuterium oxide (^2H_2O); light water reactors are the most commonly used among thermal reactors. Light water reactors are contained in highly pressurized steel vessels called reactor vessels. Heat

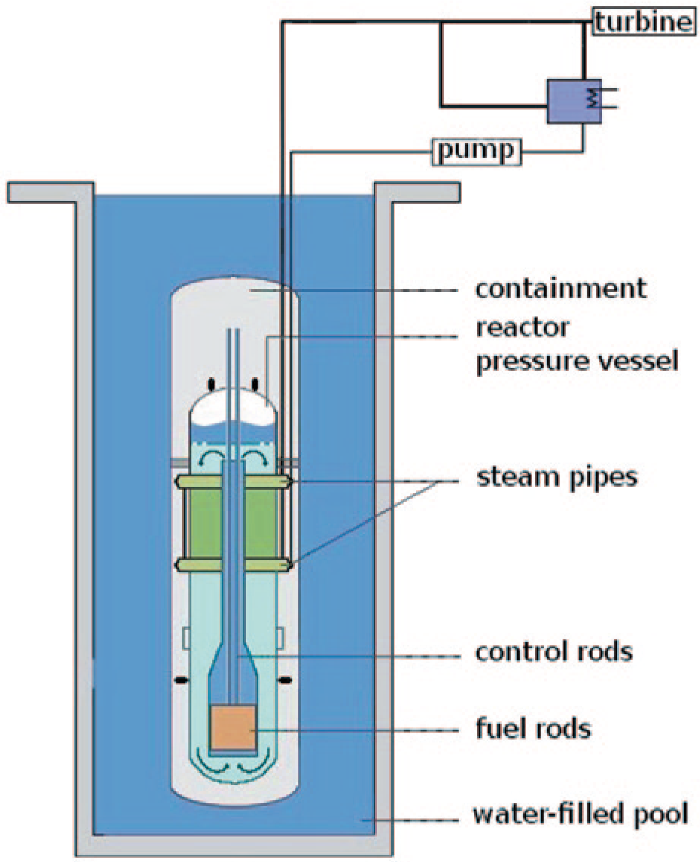


Fig. 18.7 A pumpless light water reactor

is generated by means of nuclear fission within the core of the reactor. The hundreds into a "fuel assembly" about 12 ft in length and about as thin as a pencil, group the nuclear fuel rods, each. Each fuel rod contains pellets of an oxidized form of Uranium (UO_2). A light water fuel reactor uses ordinary water to keep the system cool. The water is circulated past the core of the reactor to absorb the generated heat. The heated water then travels away from the reactor where it leaves the system as nothing more than water vapor. This is the method used in all LWRs except the BWR for in that specific system water is boiled directly by the reactor core (Fig. 18.7).

18.8.2 Graphite Moderated Reactors (GMR)

A Graphite Moderated Reactor (GMR) is a type of reactor that is moderated with graphite. The first ongoing nuclear reaction carried out by Enrico Fermi at The University of Chicago was of this type, as well as the reactor associated with the Chernobyl accident. GMRs share a valuable property with heavy water reactors, in that natural un-enriched Uranium may be used. Another highlight for the GMR is a low power density, which is ideal if power were to suddenly stop; this would not waste as much power/fuel. The common criticisms for this design are a lack

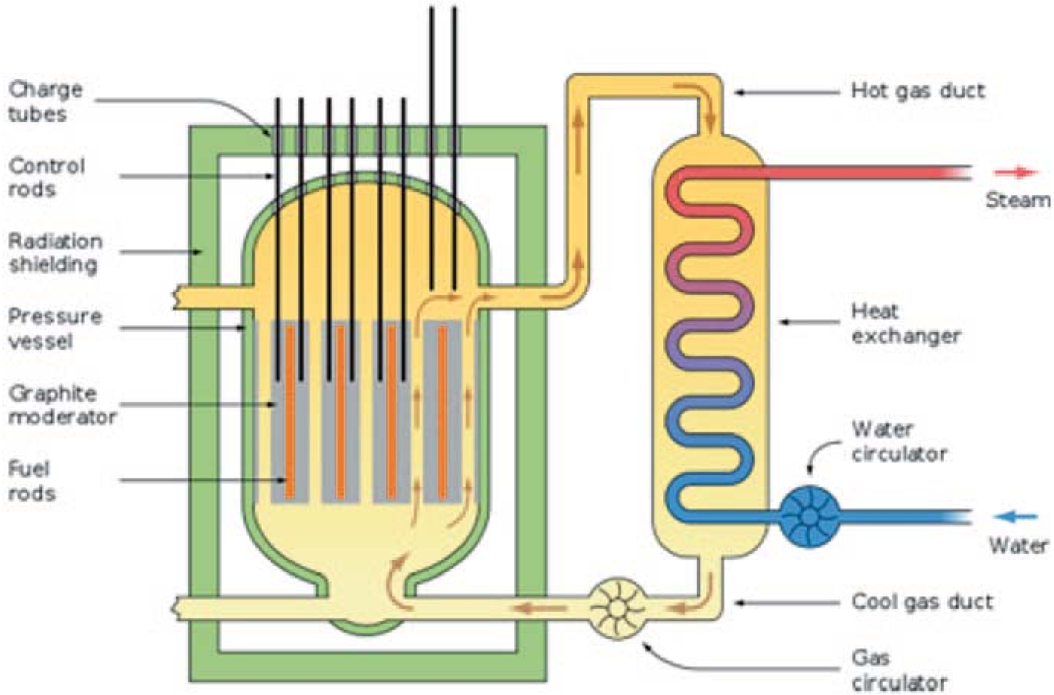


Fig. 18.8 A typical core layout of graphite moderated Reactor. (Courtesy Osterreichisches Ökologie-Institut)

of room for steam suppression and the limited safety precautions available to the design (Fig. 18.8).

18.8.3 Heavy Water Reactors (HWR)

Heavy water reactors (HWR) are a class of fission reactor that uses heavy water as a *neutron moderator*. Heavy water is deuterium oxide, D_2O. Neutrons in a nuclear reactor that uses uranium must be slowed down so that they are more likely to split other atoms and get more neutrons released to split other atoms. Light water can be used, as in a light water reactor (LWR), but since it absorbs neutrons, the uranium must be enriched for criticality to be possible. The most common pressurized heavy water reactor (PHWR) is the CANDU reactor.

Usually the heavy water is also used as the coolant but as example, the Lucens reactor was gas cooled. Advantages of this type reactor are that they can operate with unenriched uranium fuel. Although the opponents of heavy water reactors suggest that such reactors pose a much greater risk of nuclear proliferation because of two characteristics:

1. They use unenriched uranium as fuel, the acquisition of which is free from supervision of international institutions on uranium enrichment.
2. They produce more plutonium and tritium as by-products than light water reactors, these are hazardous radioactive substances that can be used in the produc-

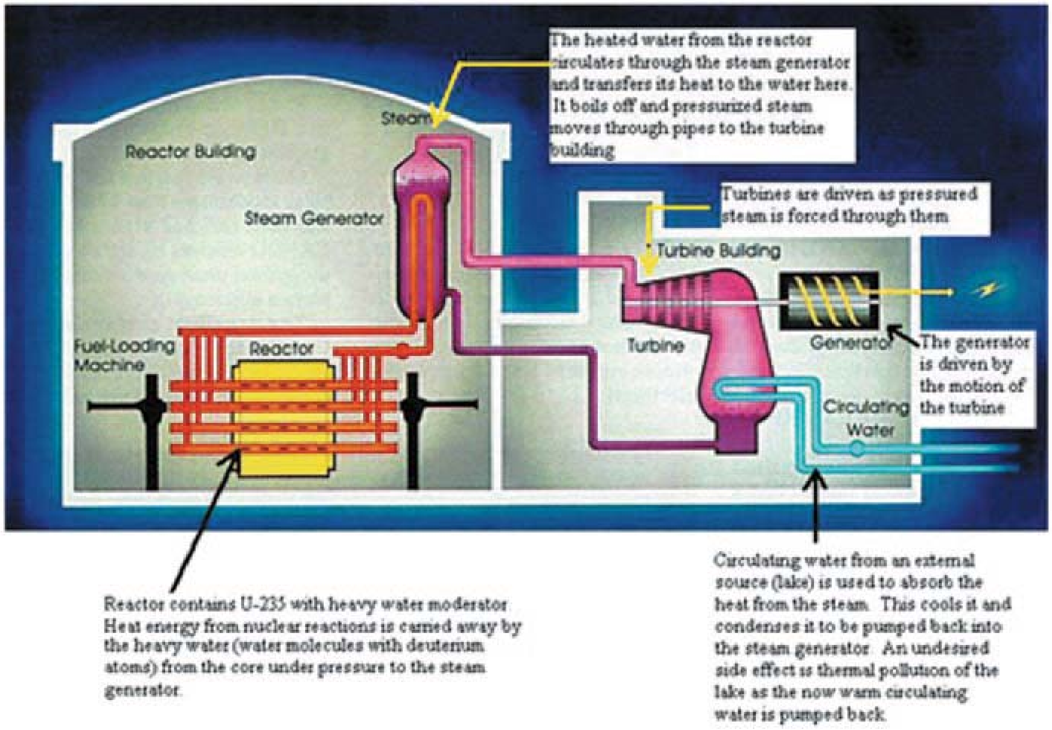


Fig. 18.9 A typical outline layout of heavy water reactor (Courtesy of Atomic Energy of Canada Limited)

tion of modern nuclear weapons such as fission, boosted fission, and neutron bombs as well as the primary stages of thermonuclear weapons. For instance, India produced its plutonium for Operation Smiling Buddha, its first nuclear weapon test, by extraction from the spent fuel of a heavy water research reactor known as "CIRUS (Canada India Research Utility Services)". It is advocated that safeguards need to be established to prevent exploitation of heavy water reactors in such a fashion.

In heavy water reactors, both the moderator and coolant are heavy water (D_2O). A great disadvantage of this type comes from this fact: heavy water is one of the most expensive liquids. However, it is worth its price: this is the best moderator. Therefore, the fuel of HWRs can be slightly (1–2 %) enriched or even natural uranium. Heavy water is not allowed to boil, so in the primary circuit very high pressure, similar to that of PWRs, exists (Fig. 18.9).

The main representative of the heavy water type is the Canadian CANDU reactor. In these reactors, the moderator and coolant are spatially separated: the moderator is in a large tank (calandria), in which there are pressure tubes surrounding the fuel assemblies. The coolant flows in these tubes only.

The advantage of this construction is that the whole tank need not be kept under high pressure; it is sufficient to pressurize the coolant flowing in the tubes. This arrangement is called pressurized tube reactor. Warming up of the moderator is much less than that of the coolant; it is simply lost for heat generation or steam production. The high temperature and high-pressure coolant, similarly to PWRs, goes to the steam generator where it boils the secondary side light water. Another advantage of this type is that fuel can be replaced during operation and thus there is no need for outages.

The other type of heavy water reactor is the Pressurized Heavy Water Reactor (PHWR). In this type, the moderator and coolant are the same and the reactor pressure vessel has to stand the high pressure.

Heavy water reactors produce cca. 6 % of the total NPP power of the world; however 13.2 % of the under construction nuclear power plant capacity is accounted for by this type. One reason for this is the safety of the type; another is the high conversion factor, which means that during operation a large amount of fissile material is produced from U-238 by neutron capture.

18.9 Classified by Coolant Material

The descriptions of these types of reactors are as follow;

18.9.1 Pressurized Water Reactors (PWR)

A Pressurized Water Reactor (PWR) is Westinghouse Bettis Atomic Power Laboratory has used a type of light water reactor for decades in designs for military ship applications, now the primary manufacturers are Framatome-ANP and Westinghouse for present day power plant reactors. The pressurized water reactor is unique in that although water passes through the reactor core to act as moderator and coolant it does not flow in to the turbine. Instead of the conventional flow cycle, the water passes into a pressurized primary loop. This step in the PWR cycle produces steam in a secondary loop that drives the turbine. Advantages of the PWR include zero fuel leaks of radioactive material into the turbine or environment, and the ability to with stand higher pressures and temperatures to higher the Carnot efficiency. Disadvantages include complex reactor designs and costs. This reactor type accounts for the majority of reactors located in the U.S (Fig. 18.10).

Pressurized Water Reactor (PWR) is a type of a nuclear power reactor that uses enriched Uranium as a fuel which in turn heats the light water used for producing steam. The main feature which differentiates it from a BWR nuclear reactor is that a PWR has a separate arrangement to make steam in the form of a heat exchanger

18.9.1.1 The Arrangement of PWR

A Pressurized Water Reactor (PWR) is a type of power plant reactor consisting of two basic circuits having light water as the working fluid. In one of the circuits water is heated to a high temperature and kept at high pressure as well, so that it does

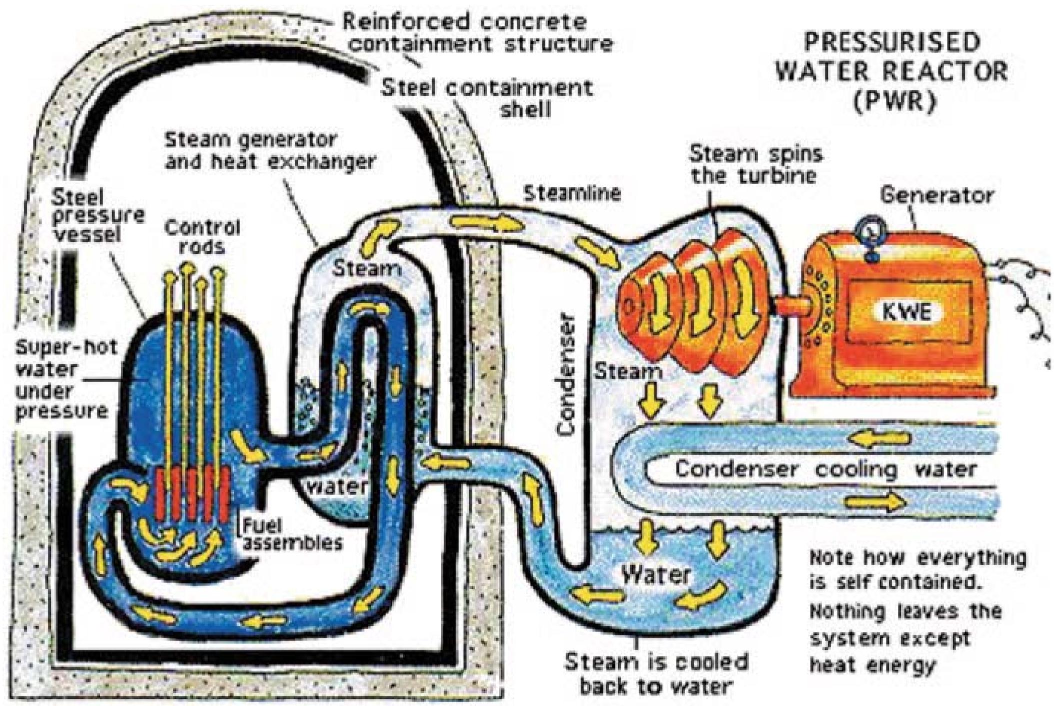


Fig. 18.10 A typical pressurized water reactor. (Courtesy of the Uranium Information Centre)

not get converted into a gaseous state. This superheated water is used as a coolant and a moderator for the nuclear reactor core hence the name PWR or pressurized water reactor.

The secondary circuit consists of water at high pressure in the gaseous state i.e. steam which is used to run the turbine-alternator arrangement. The point of interaction between these two circuits is the heat exchanger or the boiler wherein heat from the superheated high-pressure water converts the water in the secondary circuit to steam.

18.9.1.2 Advantages of PWR

- Much fewer control rods are required in a PWR. In fact, for a typical 1000 MW plant just around five dozen control rods are sufficient.
- Since the two circuits are independent of each other, it makes it very easy for the maintenance staff to inspect the components of the secondary circuit without having to shut down the power plant entirely.
- A PWR has got a high power density and this, combined with the fact that enriched Uranium is used as fuel instead of normal Uranium, leads to the construction of very compact core size for a given power output.
- One feature, which makes a PWR reactor very suitable for practical applications, is its positive demand coefficient, which serves to increase the output as a direct proportion to demand of power.
- The water used in the primary circuit is different from that used in the secondary circuit and there is no intermixing between the two, except for heat transfer, which takes place in the boiler or heat exchanger. This means that the water used

in the turbine side is free from radioactive steam hence the piping on that side is not required to be clad with special shielding materials.

18.9.1.3 Drawbacks of PWR

- The primary circuit consists of high temperature, high pressure water which accelerates corrosion. This means that the vessel should be constructed of very strong material such as stainless steel, which adds to construction costs of PWR.
- PWR fuel charging requires the plant to be shut down and this certainly requires a long time period of the order of at least a couple of months.
- The pressure in the secondary circuit is relatively quite low as compared to the primary circuit hence the thermodynamic efficiency of PWR reactors is quite low of the order of 20

18.9.1.4 Pressuriser

One important point to note here is that despite the changing loads the pressure in the primary circuit needs to be maintained at a constant value. This is achieved by installing a device known as pressuriser in the primary circuit. It basically, consists of a dome shaped structure which has heating coils which, are used to increase or decrease pressure as and when required depending on varied load conditions.

Note that in the Pressurized Water Reactor (PWR), the water, which passes over the reactor core to act as moderator and coolant, does not flow to the turbine, but is contained in a pressurized primary loop. The primary loop water produces steam in the secondary loop, which drives the turbine. The obvious advantage to this is that a fuel leak in the core would not pass any radioactive contaminants to the turbine and condenser (Fig. 18.11).

Another advantage is that the PWR can operate at higher pressure and temperature, about 160 atmospheres and about 315 C. This provides a higher *Carnot efficiency* than the BWR, but the reactor is more complicated and more costly to construct. Most of the U.S. reactors are pressurized water reactors.

18.9.2 Boiling Water Reactor (BWR)

The Boiling Water Reactor (BWR) date back to their General Electric introduction in the 1950s. The distinguishing feature in the BWR is the boiling method for steam. In this type of reactor, water passes over the core as a coolant to expand and become steam source for a turbine placed directly above. Advantages of this design type include a simpler reactor design, a smaller reactor system, and lower costs. Disadvantages found are the increase of radioactive materials in the turbine and a greater chance for fuel to burn out as water quickly evaporates to expose fuel rods to an atmosphere absent of a coolant. BWRs have found fame all over the world due to the cheap simple design.

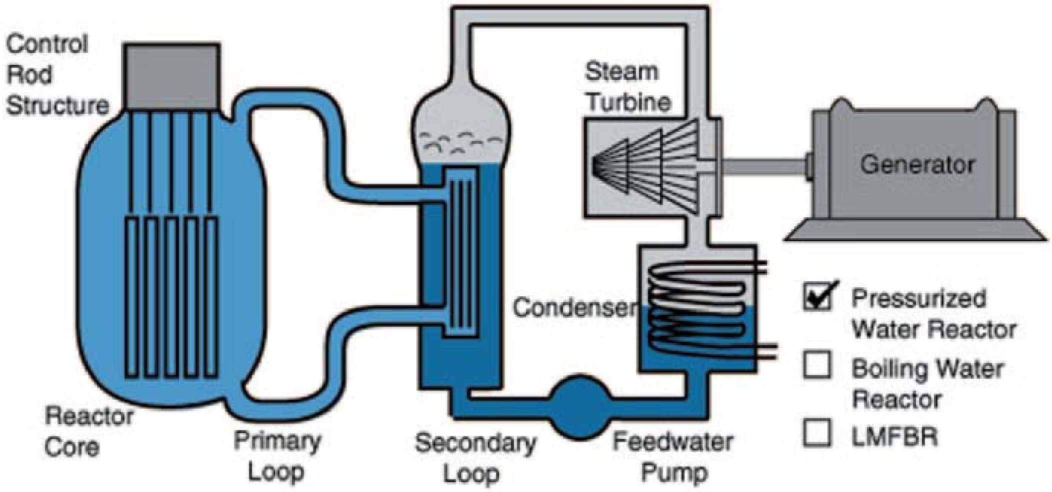


Fig. 18.11 A typical outline of pressurized water reactor

In Fig. 18.12 (1) the core inside the reactor vessel creates heat, (2) a steam-water mixture is produced when very pure water (reactor coolant) moves upward through the core, absorbing heat, (3) the steam-water mixture leaves the top of the core and enters the two stages of moisture separation where water droplets are removed before the steam is allowed to enter the steam line, and (4) the steam line directs the steam to the main turbine, causing it to turn the turbine generator, which produces electricity.

Note that in the Boiling Water reactor (BWR), the water, which, passes over the reactor core to act as moderator and coolant, is also the steam source for the turbine. The disadvantage of this is that any fuel leak might make the water radioactive and that radioactivity would reach the turbine and the rest of the loop (Fig. 18.13).

A typical operating pressure for such reactors is about 70 atmospheres at which pressure the water boils at about 285 °C. This operating temperature gives a *Carnot efficiency* of only 42 % with a practical operating efficiency of around 32 %, somewhat less than the PWR.

18.9.3 Gas Cooled Reactors (GCR)

The Gas Cooled Reactor (GCR) or the gas-graphite reactors operate using graphite as moderator and some gas (mostly CO_2, lately helium) as coolant. This belongs to the oldest reactor types. The first GGR was the Calder Hall power plant reactor, which was built in 1955 in England. This type is called MAGNOX after the special magnesium alloy (Magnox), of which the fuel cladding was made. The fuel is natural uranium. These reactors account for 1.1 % of the total NPP power of the world and are not built any more (Fig. 18.14).

The Advanced Gas cooled Reactor (AGR) is a development from MAGNOX: the cladding is not Magnox and the fuel is slightly enriched. The moderator is also graphite and the coolant is CO_2. Contribution to total world capacity is 2.5 %. This type is not manufactured any longer (Fig. 18.15).

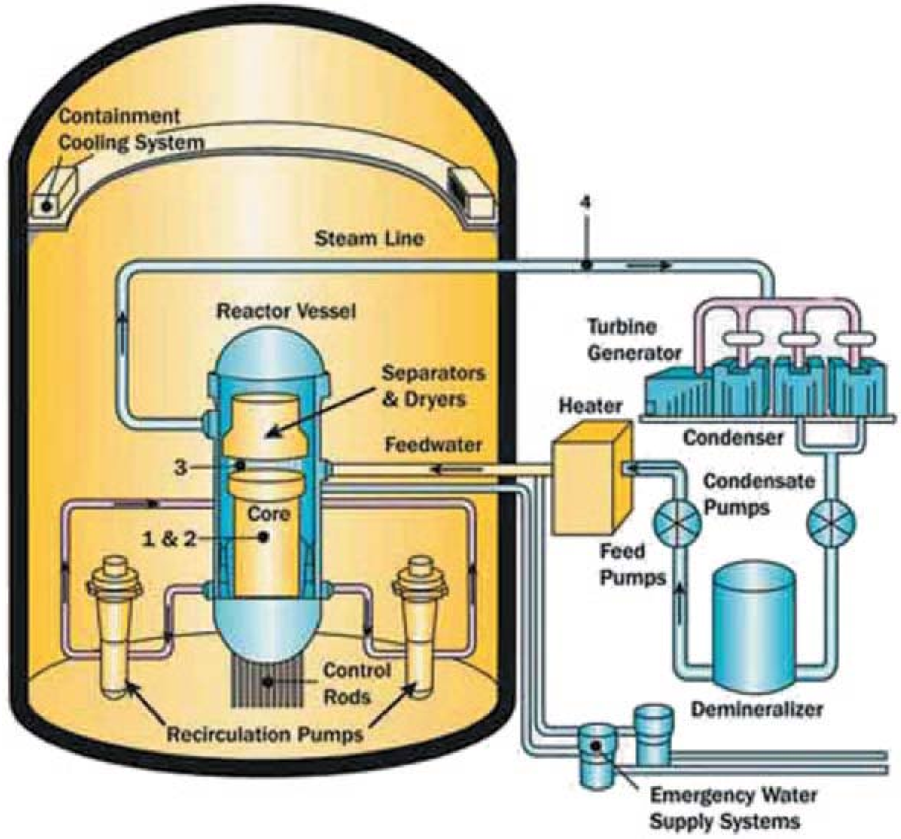


Fig. 18.12 A typical boiling water reactor. (Courtesy of the U.S. Nuclear Regulatory Commission)

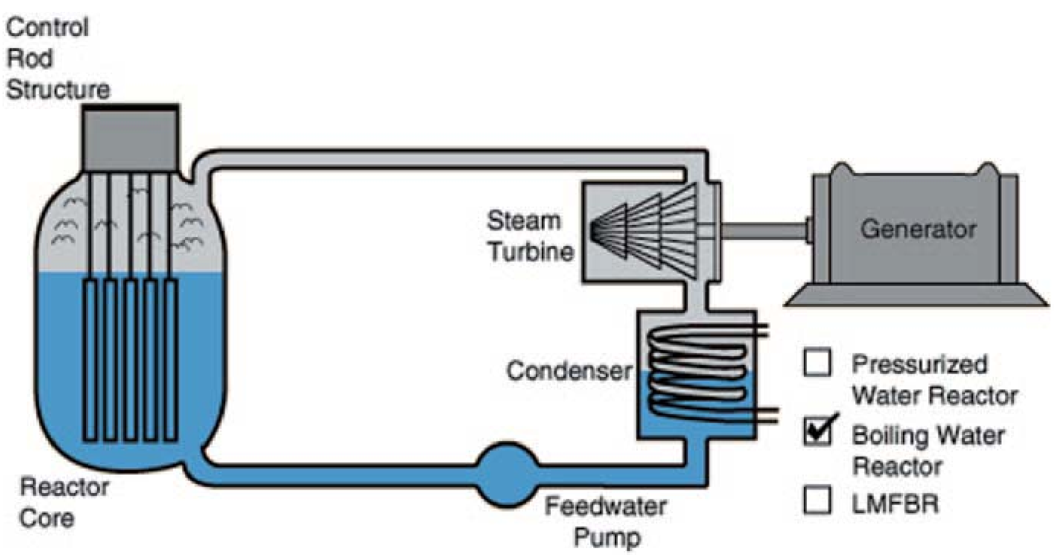
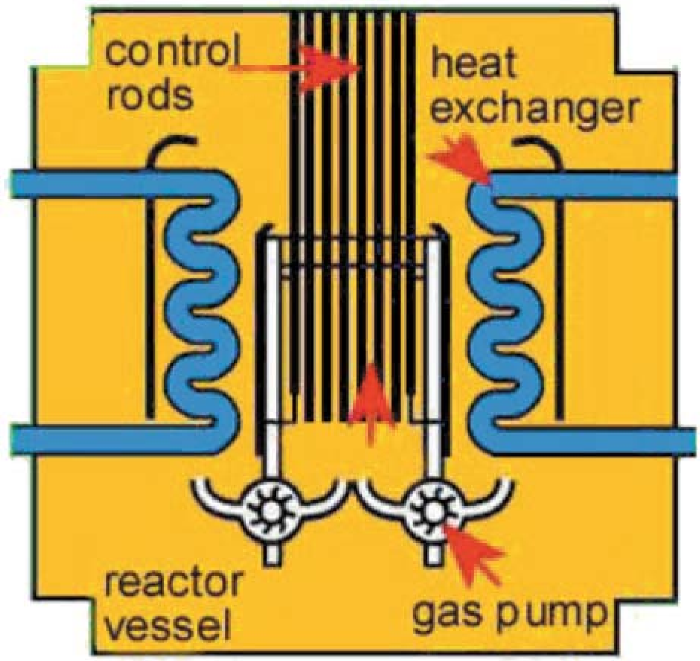


Fig. 18.13 A typical layout of boiling water reactor

Fig. 18.14 A typical core layout of gas cooled reactor



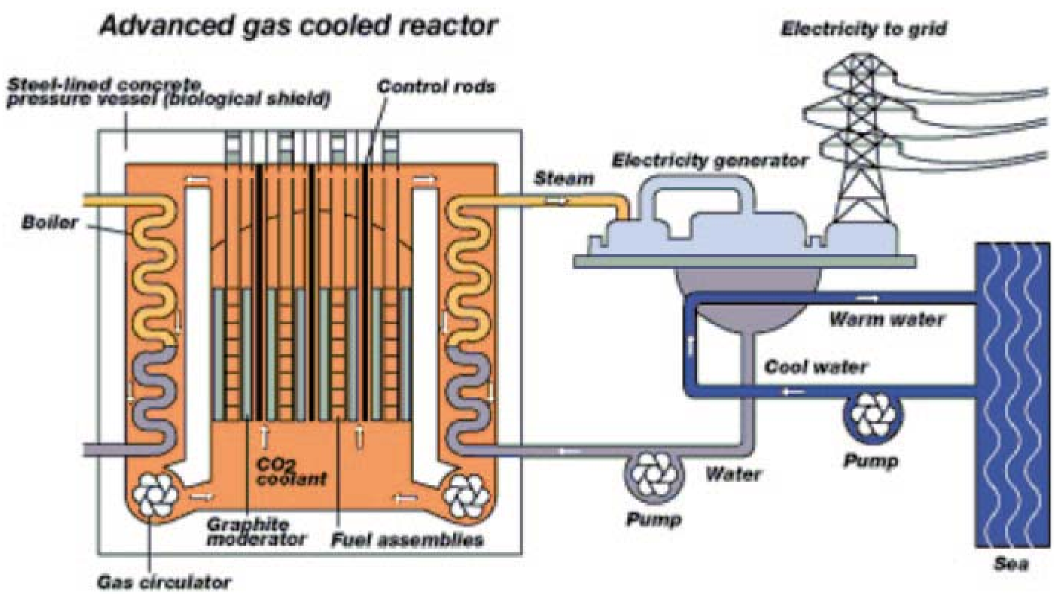


Fig. 18.15 A typical outline layout of gas cooled reactor

The newest gas cooled reactor type is the High Temperature Gas cooled Reactor (HTGR), which is cooled by helium and moderated by graphite. In this reactor as high as 950 °C coolant temperature can be achieved. The efficiency of a newly developed type, the Gas Turbine Modular Helium Reactor (GT-MHR) might be as high as almost 50 %.

Gas Cooled Reactors (GCR) and Advanced Gas Cooled Reactors (AGR) use carbon dioxide as the coolant to carry the heat to the turbine, and graphite as the moderator. Like heavy water, a graphite moderator allows natural uranium (GCR) or slightly enriched uranium (AGR) to be used as fuel.

18.10 Classified by Reaction Type

The descriptions of each of these reactors are given as follows;

18.10.1 Fast Neutron Reactor (FNR)

Fast Neutron Reactors (FNR), also known as Fast Breeder Reactors (FBR), use depleted nuclear waste as a form of energy. Uranium, which is composed of 0.7 % Uranium-235 and 99.3 % Uranium-238, is processed in the fast neutron reactors into isotopes of usable plutonium of plutonium 239 and 241. Fast neutron reactors are 60 % more efficient than normal reactors; a fast neutron reactor uses liquid metal as its coolant as opposed to water, which makes the reactor safer to use and its fuel is metallic, which keeps the reactors under control more easily. Some cons of fast neutron reactors though are that they are very unpredictable, making them more tedious to use. Bubbles are more present in processes, so fast neutron reactors tend to heat up more rather than cool down and the coolant that it requires is much more exotic, such liquid sodium and bismuth eutectic.

Several countries have research and development programs for improved Fast Breeder Reactors (FBR), which are a type of Fast Neutron Reactors. These use the uranium-238 in reactor fuel as well as the fissile U-235 isotope used in most reactors.

Natural uranium contains about 0.7 % U-235 and 99.3 % U-238. In any reactor, the U-238 component is turned into several isotopes of plutonium during its operation. Two of these, Pu 239 and Pu 241, then undergo fission in the same way as U 235 to produce heat. In a fast neutron reactor this process is optimized so that it can 'breed' fuel, often using a depleted uranium blanket around the core. FBRs can utilize uranium at least 60 times more efficiently than a normal reactor.

Fast-neutron reactors could extract much more energy from recycled nuclear fuel, minimize the risks of weapons proliferation and markedly reduce the time nuclear waste must be isolated.

If developed sensibly, nuclear power could be truly sustainable and essentially inexhaustible and could operate without contributing to climate change. In particular, a relatively new form of nuclear technology could overcome the principal drawbacks of current methods—namely, worries about reactor accidents, the potential for diversion of nuclear fuel into highly destructive weapons, the management of dangerous, long-lived radioactive waste, and the depletion of global reserves of economically available uranium. This nuclear fuel cycle would combine two innovations: pyrometallurgical processing (a high-temperature method of recycling reactor waste into fuel) and advanced fast-neutron reactors capable of burning that fuel. With this approach, the radioactivity from the generated waste could drop to safe levels in a few hundred years, thereby eliminating the need to segregate waste for tens of thousands of years.

Fast Reactor Technology: A Path to Long-Term Energy Sustainability Position Statement November 2005

"The American Nuclear Society believes that the development and deployment of advanced nuclear reactors based on fast-neutron fission technology is important to the sustainability, reliability, and security of the world's long-term energy supply. Of the known and proven energy technologies, only nuclear fission can provide the large quantities of energy required by industrial societies in a sustainable and environmentally acceptable manner".

"Natural uranium mined from the earth's crust is composed primarily of two isotopes: 99.3 % is U-238, and 0.7 % is the fissile U-235. Nearly all current power reactors are of the "thermal neutron" design, and their capability to extract the potential energy in the uranium fuel is limited to less than 1 % of that available. The remainder of the potential energy is left unused in the spent fuel and in the uranium, depleted in U-235 that remains from the process of enriching the natural uranium in the isotope U-235 for use in thermal reactors. With known fast reactor technology, this unutilized energy can be harvested, thereby extending by a hundred-fold the amount of energy extracted from the same amount of mined uranium".

"Fast reactors can convert U-238 into fissile material at rates faster than it is consumed making it economically feasible to utilize ores with very low uranium concentrations and potentially even uranium found in the oceans [1–3]. A suitable technology has already been proven on a small scale [4]. Used fuel from thermal reactors and the depleted uranium from the enrichment process can be utilized in fast reactors, and that energy alone would be sufficient to supply the nation's needs for several hundred years".

"Fast reactors in conjunction with fuel recycling can diminish the cost and duration of storing and managing reactor waste with an offsetting increase in the fuel cycle cost due to reprocessing and fuel prefabrications. Virtually all long-lived heavy elements are eliminated during fast reactor operation, leaving a small amount of fission product waste that requires assured isolation from the environment for less than 500 years [5].

"Although fast reactors do not eliminate the need for international proliferation safeguards, they make the task easier by segregating and consuming the plutonium as it is created. The use of onsite reprocessing makes illicit diversion from within the process highly impractical. The combination of fast reactors and reprocessing is a promising option for reasons of safety, resource utilization, and proliferation resistance [5].

"Reaping the full benefits of fast reactor technology will take a decade or more for a demonstration reactor, followed by buildup of a fleet of operating power stations. For now and in the intermediate-term future, the looming short-term energy shortage must be met by building improved, proven thermal-reactor power plants. To assure longer-term energy sustainability and security, the American Nuclear Society sees a need for cooperative international efforts with the goal of building a fast reactor demonstration unit with onsite reprocessing of spent fuel."

18.10.2 Thermal Neutron Reactor

Thermal reactors go through the same process as fast neutron reactors, but in a thermal reactor the process of obtaining plutonium is slower. These types of reactors use a neutron moderator to slow neutrons until they approach the average kinetic energy of the surrounding particles, that is, to reduce the speed of the neutrons to low velocity thermal neutrons. The nuclear cross section of uranium-235 for slow thermal neutrons is about 1000 barns. For fast neutrons, it is in the order of 1 barn. In a thermal reactor, the neutrons that undergo the reaction process have significantly lower electron-volt energy, so the neutrons are considered to be slower. A neutron's speed will determine its chances to interact with the nucleus of an atom; the slower its speed the bigger its fission cross section becomes and thus the higher its chance of interacting with the nucleus becomes (Fig. 18.16).

This figure gives the value of the fission cross section for some fissile isotopes. Note that both axes are logarithmic. The thermal and fast energy regions are indicated. For thermal energies, the fission cross section equals several thousand barn, at high energies the fission cross section is of the order of 1–10 barn.

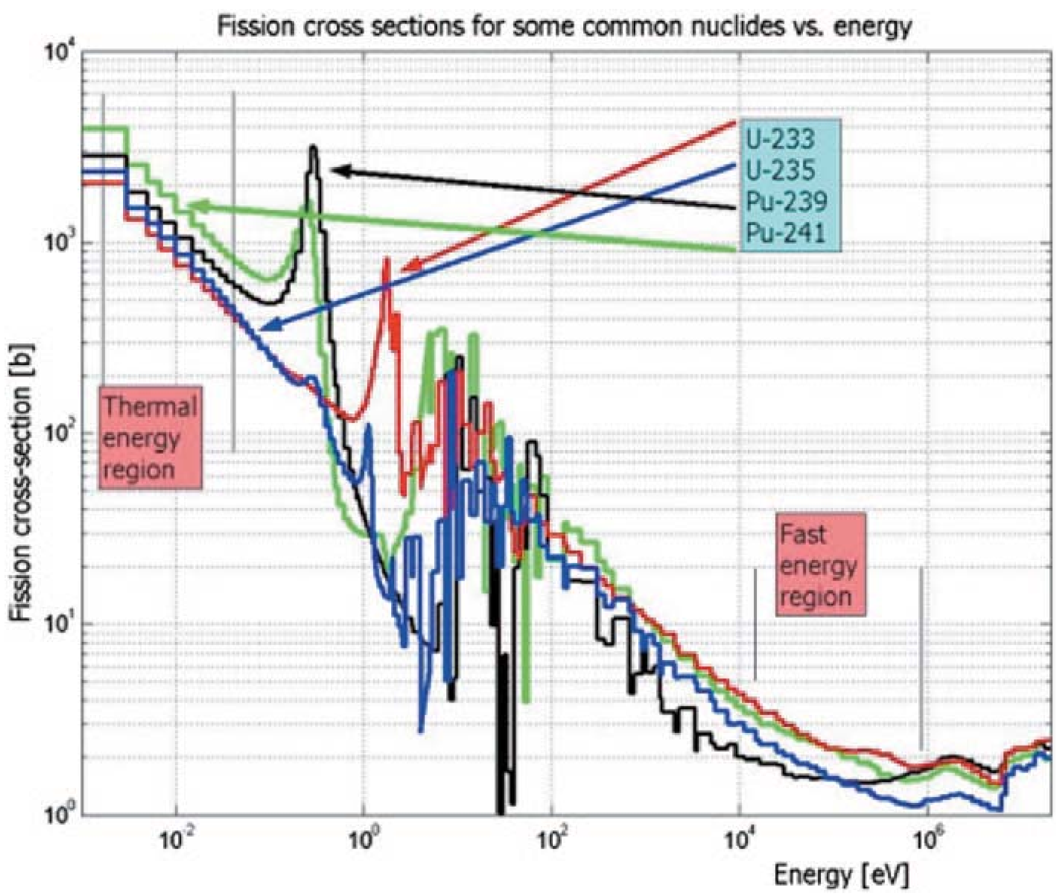


Fig. 18.16 Fission cross section for some common nuclides vs. energy. (Courtesy of TUDelft)

The fact that the fission cross section is rather large for low-energy neutrons has an important effect on the design of a nuclear reactor: in a reactor where the neutrons have a low energy, not much fissile material is required, because the probability of an interaction is very large. The lowest energy a neutron can have in a nuclear reactor is the energy at which it is in equilibrium with its environment. The movement of the neutron is then identical to the thermal movement of the atoms that constitute the reactor. The neutrons have slowed down from the high energy (2 MeV) where they are born to this equilibrium energy are called 'thermal neutrons'. The average energy of a neutron in thermal equilibrium is 0.025 eV—the neutron is slowed down over nine decades, more than a billion times. Reactors in which, most fissions are induced by thermal neutrons are called thermal reactors. Thermal reactors are by far the most widely used reactors in the world today. Most reactors use water, heavy water or graphite as moderator. The reason for the choice of thermal reactors is a simple one: a thermal reactor requires a small amount of fuel to become critical, and thus the fuel is cheap.

18.10.3 Liquid Metal Fast Breeder Reactors (LMFBR)

The plutonium-239 breeder reactor is commonly called a fast breeder reactor, and the cooling and a liquid metal does heat transfer. The metals, which can accomplish this, are sodium and lithium, with sodium being the most abundant and most commonly used. The construction of the fast breeder requires a higher enrichment of U-235 than a light-water reactor, typically 15–30 %. The reactor fuel is surrounded by a "blanket" of non-fissionable U-238. No moderator is used in the breeder reactor since fast neutrons are more efficient in transmuting U-238 to Pu-239. At this concentration of U-235, the cross-section for fission with fast neutrons is sufficient to sustain the chain-reaction. Using water as coolant would slow down the neutrons, but the use of liquid sodium avoids that moderation and provides a very efficient heat transfer medium (Fig. 18.17).

The Super-Phenix was the first large-scale breeder reactor. It was put into service in France in 1984. It ceased operation as a commercial power plant in 1997. Such a reactor can produce about 20 % more fuel than it consumes by the breeding reaction. Enough excess fuel is produced over about 20 years to fuel another such reactor. Optimum breeding allows about 75 % of the energy of the natural uranium to be used compared to 1 % in the standard light water reactor (Fig. 18.18).

Under appropriate operating conditions, the neutrons given off by fission reactions can "breed" more fuel from otherwise non-fissionable isotopes. The most common *breeding reaction* is that of plutonium-239 from non-fissile uranium-238. The term "fast breeder" refers to the types of configurations, which can actually produce more fissionable fuel than they use, such as the LMFBR. This scenario is possible because the non-fissile uranium-238 is 140 times more abundant than the fissionable U-235 and can be efficiently converted into Pu-239 by the neutrons from a fission chain reaction.

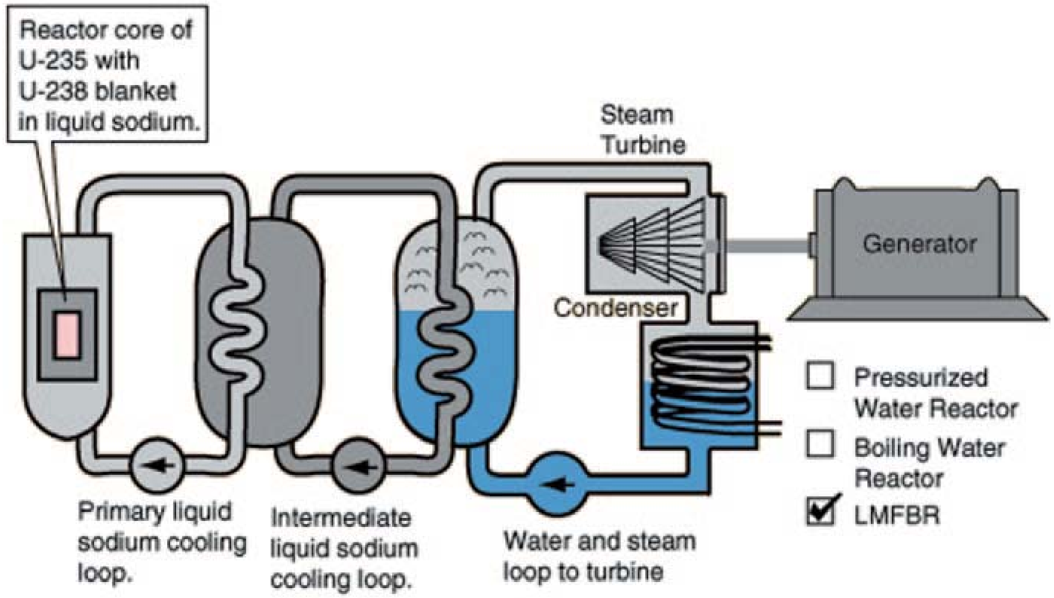
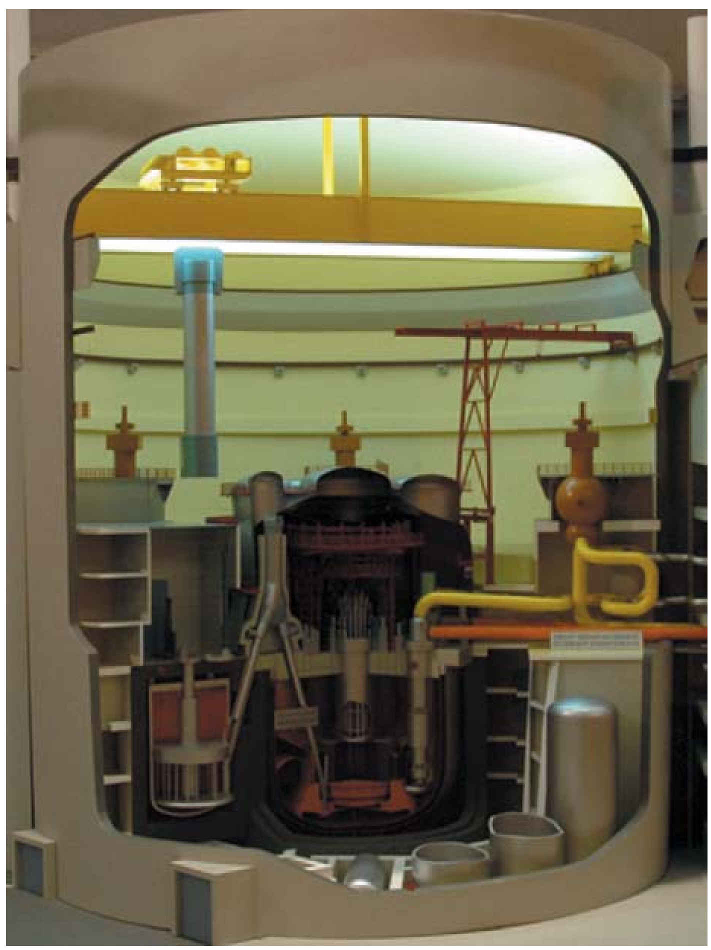
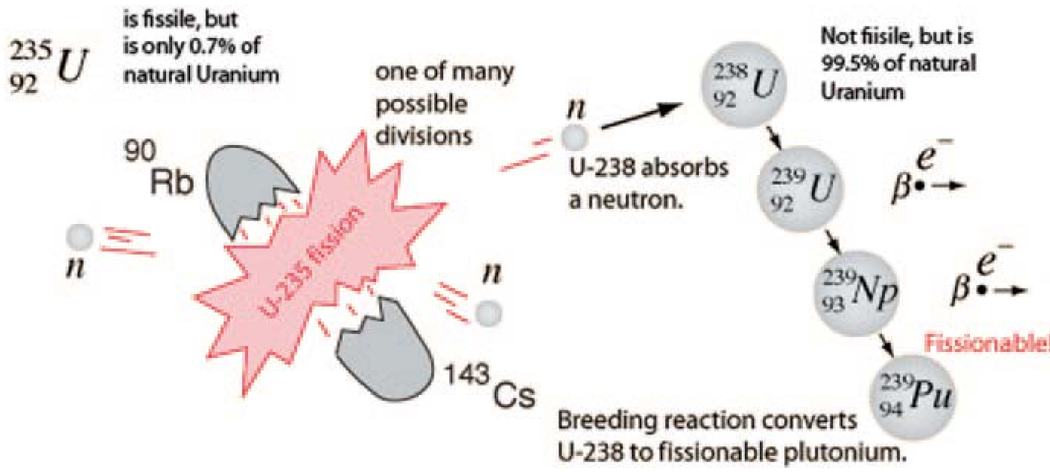


Fig. 18.17 A typical layout of Liquid metal fast breeder reactor

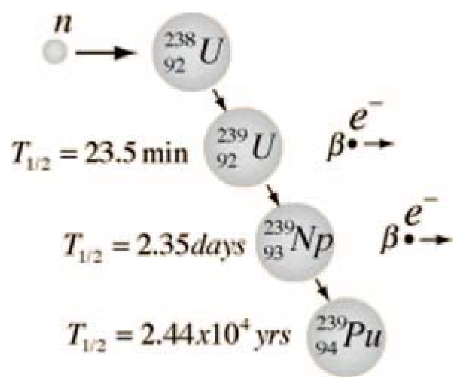
Fig. 18.18 This is a photo of a model of the containment vessel of the Super-Phenix. It is displayed at the National Museum of Nuclear Science and Technology in Albuquerque, NM





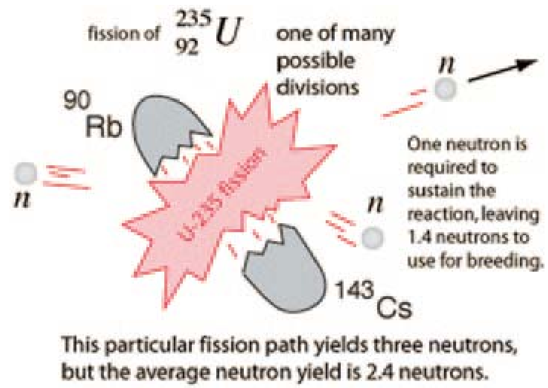
France has made the largest implementation of breeder reactors with its large Super-Phenix reactor (today is not in production line) and an intermediate Russian scale reactor (BN-600) on the Caspian Sea for electric power and desalinization.

Breeding Plutonium-239 can be accomplished from non-fissile uranium-238 by the reaction illustrated.



The bombardment of Uranium-238 with neutrons triggers two successive *beta decays* with the production of plutonium. The amount of plutonium produced depends on the *breeding ratio.*

The concept of breading ratio of Plutonium-239 can be defined in following. In the breeding of plutonium fuel in breeder reactors, an important concept is the breeding ratio, the amount of fissile plutonium-239 produced compared to the amount of fissile fuel (like U-235) used to produce it. In the liquid-metal, fast-breeder reactor (LMFBR), the target-breeding ratio is 1.4 but the results achieved have been about 1.2. This is based on 2.4 neutrons produced per U-235 fission, with one neutron used to sustain the reaction.



The time required for a breeder reactor to produce enough material to fuel a second reactor is called its doubling time, and present design plans target about 10 years as a doubling time. A reactor could use the heat of the reaction to produce energy for 10 years, and at the end of that time have enough fuel to fuel another reactor for 10 years.

Liquid sodium is used as the coolant and heat-transfer medium in the LMFBR reactor. That immediately raised the question of safety since sodium metal is an extremely reactive chemical and burns on contact with air or water (sometimes explosively on contact with water). It is true that the liquid sodium must be protected from contact with air or water at all times, kept in a sealed system. However, it has been found that the safety issues are not significantly greater than those with high-pressure water and steam in the light-water reactors.

Sodium is a solid at room temperature but liquefies at 98 °C. It has a wide working temperature since it does not boil until 892 °C. That brackets the range of operating temperatures for the reactor so that it does not need to be pressurized as does a water-steam coolant system. It has a large *specific heat* so that it is an efficient heat-transfer fluid.

In practice, those reactors, which have used liquid metal coolants, have been fast-neutron reactors. The liquid metal coolant has a major advantage there because water as a coolant also moderates or slows down the neutrons. Such fast-neutron reactors require a higher degree of enrichment of the uranium fuel than do the water-moderated reactors.

18.11 Nuclear Fission Power Generation

Nuclear fission energy is today a competitive and mature low-carbon technology, operating at very high levels of safety. The installed nuclear electricity capacity in the Europe (EU) for example, is 132 GWe, which provides one third of the EU's generated electricity. Most of the current designs are Light Water Reactors (LWR) of the second generation, capable of providing base-load electricity often with availability factors of over 90 %. There have been only a few new nuclear power plants connected to the grid in the last two decades, and as a result of decommissioning of old plants the total number of reactors in Europe has decreased. Nevertheless, electricity supply from nuclear has remained constant and the levelised cost has decreased owing to improved efficiency, power upgrade and improved availability factor.

More recently, there has been a renewed interest in nuclear energy, referred to as "nuclear renaissance", mainly driven by concerns over climate change, security and independence of supply and energy costs.

18.12 Generation IV Nuclear Energy Systems

Concerns over energy resource availability, climate change, air quality, and energy security suggest an important role for nuclear power in future energy supplies. While the current Generation II and III nuclear power plant designs provide a secure and low-cost electricity supply in many markets, further advances in nuclear energy system design can broaden the opportunities for the use of nuclear energy. To explore these opportunities, the U.S. Department of Energy's Office of Nuclear Energy has engaged governments, industry, and the research community worldwide in a wide ranging discussion on the development of next generation nuclear energy systems known as "Generation IV".

The goal of the Gen IV Nuclear Energy Systems is to address the fundamental research and development (R&D) issues necessary to establish the viability of next-generation nuclear energy system concepts to meet tomorrow's needs for clean and reliable electricity, and non-traditional applications of nuclear energy. Successfully addressing the fundamental Research and Development (R&D) issues will allow Gen IV concepts that excel in safety, sustainability, cost-effectiveness, and proliferation risk reduction to be considered for future commercial development and deployment by the private sector (Fig. 18.19).

Gen IV reactor concepts are being developed to use advanced fuels, fashioned from recycled reactor fuel and capable of high-burnups. The corresponding fuel cycle strategies allow for efficient utilization of domestic uranium resources while minimizing waste. Reduction of proliferation risk and improvements in physical protection are being designed into Gen IV concepts to help thwart those who would target nuclear power plants for terrorist acts or use them improperly to develop materials for nuclear weapons. Gen IV concepts will feature advances in safety and reliability to improve public confidence in nuclear energy while providing enhanced investment protection for plant owners. Competitive life-cycle costs and acceptable financial risk are being factored into Gen IV concepts with high-efficiency electricity generation systems, modular construction, and shortened development schedules before plant startup.

Gen IV is also an active participant in the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). INPRO was established in 2001 in response to a resolution by the IAEA General Conference to help to ensure that

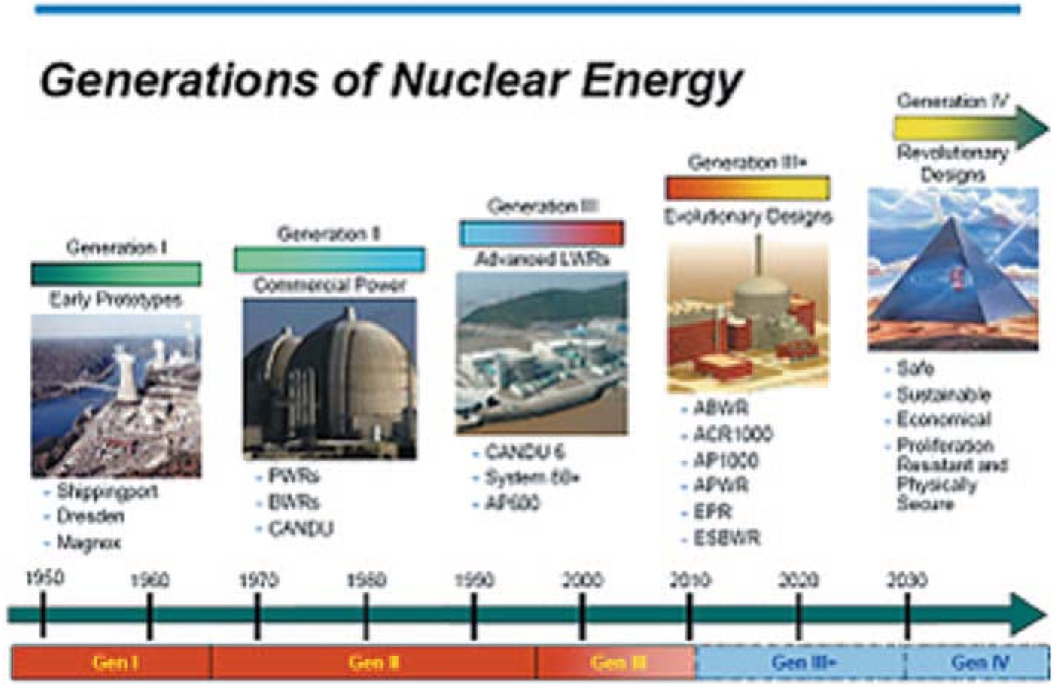


Fig. 18.19 The evolution of nuclear power

nuclear energy is available to contribute, in a sustainable manner, to meeting the energy needs of the twenty-first century and to bring together technology holders and users so that they can consider jointly the international and national actions required for achieving desired innovations in nuclear reactors and fuel cycles. INPRO provides a forum for discussion for experts and policy makers from industrialized and developing countries on all aspects of nuclear energy planning as well as on the development and deployment of innovative nuclear energy systems in the twenty-first century.

The Generation IV International Forum (GIF) was chartered in May 2001, to lead the collaborative efforts of the world's leading nuclear technology nations to develop the next generation of nuclear energy systems. The initial efforts of GIF resulted in the identification of the six most promising reactor concepts to be investigated by this international research community and are documented in the Generation IV Technology Roadmap. Thirteen members have signed the GIF Charter: Argentina, Brazil, Canada, People's Republic of China, Euratom, France, Japan, Republic of Korea, the Russian Federation, Republic of South Africa, Switzerland, the United Kingdom and the United States. This unique international effort reached a major milestone on February 28, 2005, as five of the Forum's member countries (Canada, France, Japan, United Kingdom, and United States) signed the world's first multi-lateral agreement aimed at the international development of advanced nuclear energy systems—the Framework Agreement for International Collaboration on Research and Development of Generation IV Nuclear Energy Systems. Subsequent signatories to the Framework Agreement included People's Republic of China, Euratom, Republic of Korea, Republic of South Africa, and Switzerland.

Fig. 18.20 Map of member countries

The United Kingdom is a signatory of the Framework but is currently a non-active member. Argentina and Brazil have not ratified the Framework Agreement and are therefore considered non-active. The Russian Federation is working on the necessary approvals for its accession to the Framework (Fig. 18.20).

As detailed in its Charter and subsequent GIF Policy Statements, GIF is led by the Policy Group (PG), which is responsible for the overall coordination of GIF's Research and Development (R&D) collaboration, policy formation and for interactions with other organizations. France with currently chairs the Policy Group vice chairs from the U.S. and Japan. An Experts Group and the Senior Industry Advisory Panel advises the Policy Group on (R&D) strategy, priorities, and methodology and on evaluating research plans for each Generation IV System. The Framework Agreement establishes two levels of implementing arrangements in order to conduct the joint (R&D). The first level consists of a System Arrangement for each Generation IV reactor concept directed by a System Steering Committee (SSC). Under each SSC, Project Arrangements are established with Project Management Boards to manage and implement the joint (R&D).

18.13 Technological State of the Art and Anticipated Developments

It has been demonstrated that Generation-II plants can be safely and economically operated for up to 60 years through the development of improved harmonized Plant-Life Management technologies and Plant License Extension practices (PLIM/PLEX) and that developments in fuel technologies can still lead to improvements in reactor performance. The first Generation-III reactors, which are an evolution of thermal reactors with even further improved safety characteristics and economy, are now being built. In the coming decades, nuclear electricity generation should increase or at least maintain its current level by a combination of lifetime extension and power upgrades of Generation-II reactors and new build of Generation-III reactors. Two 1.6 GWe Generation-III reactors are presently under construction in Finland and France, targeted for connection to the grid in 2012.

The Finnish reactor was a First-of-a-Kind (FOAK) and the construction has suffered delays with the Overnight Cost increasing from 2000 to 3100 €/kWe, whereas the Overnight Cost for the second reactor in France is now 2400 €/kWe. In series production, the industry expects the cost to be 2000 ± 500 €/GWe, which is in line with recent international studies. An additional capacity of 100 GWe of Generation-III reactors over the next 25 years is a reasonable estimate, which would require an investment in the range of € 200–280 billion. The capital costs represent typically 60–70 % of the levelised cost for nuclear electricity, operation and maintenance 20–25 % and fuel 10–15 %. The front-loaded cost profile means that the levelised cost is very sensitive to construction time and the financial schemes for the investment. Estimates in 2007 for UK resulted in range of 31–44 £/MWh (37–53 €/MWh).

Though uranium is relatively abundant in the Earth's crust and oceans, estimates of natural reserves are always related to the cost of mineral extraction. As the price of uranium increases on world markets, the number of economically exploitable deposits also increases. The most recent estimates [6] identified 5.5 Million t of Uranium (MtU) that could be exploited below 130 $/kg. The total amount of undiscovered resources (reasonably assured and speculative) available at an extraction cost below 130 $/kgU is estimated at 10.5 MtU. Unconventional resources, from which uranium is extracted as a by-product only, e.g. in phosphate production, lie between 7 and 22 MtU, and reserves in sea water are estimated to be 4000 MtU. Japanese studies suggest that uranium from sea water can be extracted at 300 €/kg. At a conservative estimate, 25,000 t of the uranium are required to produce the fuel to generate 1000 TWhe in an open fuel cycle. The global electricity supplied by nuclear is 2600 TWhe, which means that the conventional resources below 130 $/kgU at the current rate of consumption would last for at least 85 years with the already identified resources (5.5 MtU) and 246 years, if the undiscovered are also included (5.5 + 10.5 MtU). In addition to uranium, it is also possible to use thorium, which is three times more abundant in the Earth's crust, though would require different reactors and fuel cycles. Nonetheless, natural resources are plentiful and do not pose an immediate limiting factor for the development of nuclear energy.

However, in a scenario with a large expansion of nuclear energy, resources will become an issue much earlier, especially since new plants have at least a 60-year lifetime and utilities will need assurances when ordering new build that uranium supply can be maintained for the full period of operation. Eventually, known conventional reserves will all be earmarked for current plants or those under construction, and this could happen by the middle of this century. This underlines the need to develop the technology for a new generation, the so-called Generation-IV, of reactors and fuel cycles that are more sustainable. In particular, fast neutron breeder reactors could produce up to 100 times more energy from the same quantity of uranium than current designs and may significantly reduce the amount of ultimate radioactive waste for disposal.

Fast reactors convert non-fissile material (U-238) in the fuel into fissile material (Pu-239) during reactor operation so that the net amount of fissile material increases (breeding). After re-processing of the spent fuel, the extracted fissile materials are then re-cycled as new fuels. Reduction of the radio-toxicity and heat load of the waste is achieved by separating some long-lived radionuclides, the minor actinides, which could then be "burned" in fast reactors or alternatively in Accelerator Driven Systems (ADS), through transmutation. The fast reactor concept has been demonstrated in research programs and national prototypes in the past, but further R&D is needed to make it commercially viable and to develop the designs in compliance with true Generation-IV criteria. Major issues involve new materials that can withstand higher temperatures, higher burn-ups and neutron doses, corrosive coolants; reactor designs that eliminate severe accidents; and development of fuel cycles for waste minimization and elimination of proliferation risks. Fast Reactors are expected to be commercially available from 2040.

So far nuclear power has primarily been used to produce electricity, but it can also be used for process heat applications [7]. Currently, LWRs are already being used to a limited extent for some lower temperature applications (200 °C), such as district heating and desalination of seawater. Existing designs of High-Temperature Reactors (HTR) that can reach 800 °C can be deployed in the coming decades and Very-High Temperature Reactors (VHTR) that can reach gas coolant temperatures beyond 1000 °C are being studied as a Generation-IV concept for later deployment. Process heat applications include petroleum refinery applications (400 °C), recovery of oil from tar sands (600–700 °C), synthetic fuel from CO_2 and hydrogen (600–1000 °C), hydrogen production (600–1000 °C) and coal gasification (900–1200 °C). Small reactors that can be inherently safe and used to support specific high energy applications and often in remote areas are another very interesting application that is receiving more attention, in particular in the IAEA INPRO Initiative.

The management of radioactive waste, as either spent fuel or ultimate waste, depending on the national strategy, is a key issue for public acceptance of nuclear energy. There is scientific consensus that geological disposal is the only safe long-term solution for the management of ultimate waste. After a long period of intensive research and development coupled with in-depth political and social engagement, the world's first deep geological repositories for nuclear waste will be in operation in Sweden and Finland by 2020, with France following a few years later, demonstrating that practical solutions exist for the safe long-term management of hazardous waste from the operation of nuclear power plants. Though there will also be ultimate waste from Generation-IV fast reactor fuel cycles after reprocessing, the volumes and heat loads will be greatly reduced thereby facilitating disposal operations and optimizing use of space in available geological repositories.

18.14 Next Generation Nuclear Plant (NGNP)

The Next Generation Nuclear Plant (NGNP) demonstration project forms the basis for an entirely new generation of advanced nuclear plants capable of meeting the Nation's emerging need for greenhouse-gas-free process heat and electricity. The NGNP is based on the Very-High-Temperature gas-cooled Reactor (VHTR) technology, which was determined to be the most promising for the U.S. in the medium term. The determination is documented as part of the Generation IV implementation strategy in a report submitted to Congress in 20031 following an extensive international technical evaluation effort. The VHTR technology incorporates substantive safety and operational enhancements over existing nuclear technologies. As required by the Energy Policy Act of 2005 (EPAct), the NGNP will be a prototype nuclear power plant, built at the Idaho National Laboratory (INL). Future commercial versions of the NGNP will meet or exceed the reliability, safety, proliferation-resistance, and economy of existing commercial nuclear plants.

It is envisioned that these advanced nuclear plants would be able to supply cost-competitive process heat that can be used to power a variety of energy intensive industries, such as the generation of electricity, hydrogen, enhanced oil recovery, refineries, coal-to-liquids and coal-to-gas plants, chemical plants, and fertilizer plants.

The U.S. Nuclear Regulatory Commission (NRC) is responsible for licensing and regulating the construction and operation of the NGNP. The EPAct authorizes the U.S. Department of Energy (DOE) to build the NGNP at the Idaho National Laboratory and charges INL with responsibility for leading the project development. The project's completion depends on the collaborative efforts of DOE and its national laboratories, commercial industry participants, U.S. universities, and international government agencies as well as successful licensing by the NRC. At present, and pending further evaluation as the NGNP proceeds through Phase 1 in cost-shared collaboration with industry as required by the EPAct, DOE has not made a final determination on whether the license applicant will be DOE or one or more entities that reflect a partnership between DOE and private sector firms.

Under the provisions of Section 644 of the EPAct, the Secretary of Energy and the Chairman of the Nuclear Regulatory Commission are to jointly submit to Congress a licensing strategy for the NGNP within 3 years of the enactment of the Act on August 8, 2005. This report addresses the requirement by outlining a NGNP licensing strategy jointly developed by the NRC and DOE. The scope of the document includes all four elements of the NGNP licensing strategy described in Section 644 (b) of the EPAct:

1. A description of the ways in which current NRC light-water reactor (LWR) licensing requirements need to be adapted for the types of reactors considered for the project.
2. A description of the analytical tools that the NRC will need to develop in order to independently verify the NGNP design and its safety performance.
3. A description of other research or development activities that the NRC will need to conduct for the review of an NGNP license application.
4. A budget estimate associated with the licensing strategy.

DOE has determined that the NGNP nuclear reactor will be a very-high-temperature gas-cooled reactor (VHTR) for the production of electricity, process heat, and hydrogen. The VHTR can provide high-temperature process heat (up to 950 °C) that can be used as a substitute for the burning of fossil fuels for a wide range of commercial applications. Since the VHTR is a new and unproven reactor design, the NRC will need to adapt its licensing requirements and process, which have historically evolved around light-water reactor (LWR) designs, for licensing the NGNP nuclear reactor. Thus, Section 644 of the EPAct recognized the need for an alternative licensing strategy. This report provides the recommended NGNP licensing strategy, jointly developed by the NRC and DOE. As the technology matures, the government/industry partnership evolves, and input is provided by the general public, revisions to the strategy may be necessary and appropriate.

The report addresses the four elements of the licensing strategy set forth in Section 644(b) of the EPAct. These elements are summarized above.

18.15 Generation IV Systems

The world's population is expected to expand from 6.7 billion people today to over 9 billion people by the year 2050, all striving for a better quality of life. As the earth's population grows, so does the demand for energy and the benefits that it brings: improved standards of living, better health and longer life expectancy, improved literacy and opportunity, and many others. Simply expanding the use of energy along the same mix of today's production options, however, does not satisfactorily address concerns over climate change and depletion of fossil resources. For the earth to support its population while ensuring the sustainability of humanity's development, we must increase the use of energy supplies that are clean, safe, cost effective, and which could serve for both basic electricity production and other primary energy needs. Prominent among these supplies is nuclear energy.

There is currently 370 GWe of nuclear power capacity in operation around the world, producing 3000 TWh each year—15 % of the world's electricity—the largest share provided by any non-greenhouse gas- emitting source. This reduces significantly the environmental impact of today's electricity generation and affords a greater diversity of electricity generation that enhances energy security.

For more than a decade, Generation IV International Forum (GIF) has led international collaborative efforts to develop next-generation nuclear energy systems that can help meet the world's future energy needs. Generation-IV designs will use fuel more efficiently, reduce waste production, be economically competitive, and meet stringent standards of safety and proliferation resistance.

As, we said the Generation IV International Forum (GIF) was initiated in May 2001 and formally chartered in mid 2001. It is an international collective representing government of 13 countries where nuclear energy is significant now and also seen as vital for the future. Most are committed to joint development of the next generation of nuclear technology. Led by the USA, Argentina, Brazil, Canada,

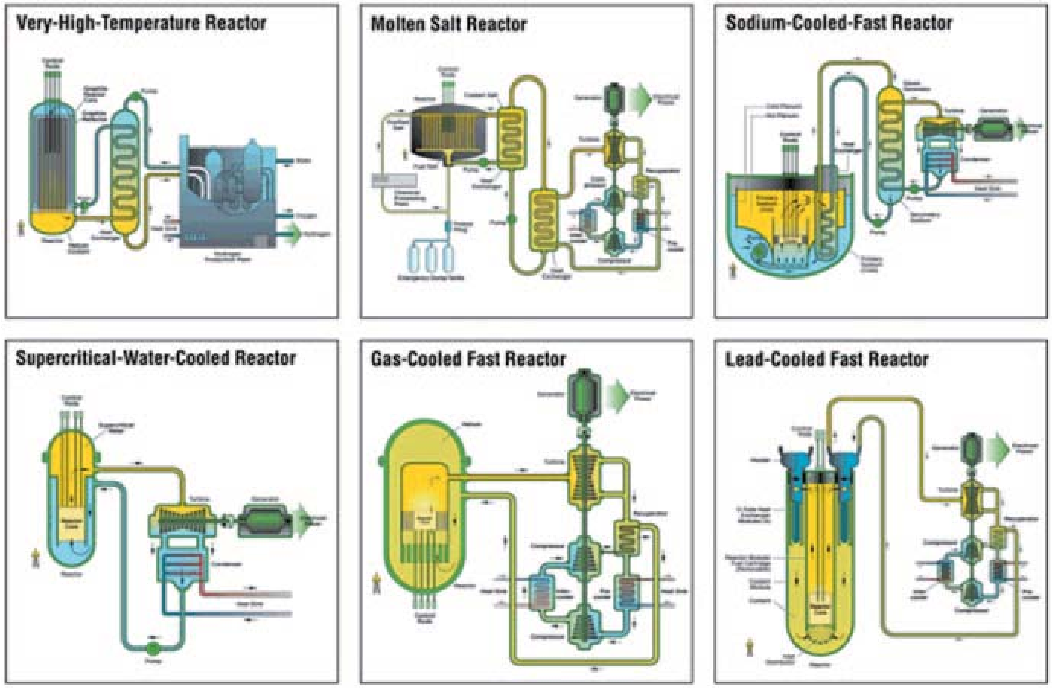


Fig. 18.21 Six reactor technologies of generation IV. (Courtesy of the Generation IV International Forum)

China, France, Japan, Russia, South Korea, South Africa, Switzerland, and the UK are charter members of the GIF, along with the EU (Euratom). Most of these are party to the Framework Agreement (FA), which formally commits them to participate in the development of one or more Generation IV systems selected by GIF for further R&D. Argentina, and Brazil did not sign the FA, and the UK withdrew from it; accordingly, within the GIF, these three are designated as "inactive Members." Russia formalized its accession to the FA in August 2009 as its tenth member, with Rosatom as implementing agent. In 2011 the 13 members decided to modify and extend the GIF charter indefinitely

With these goals in mind, some 100 experts evaluated 130 reactor concepts before GIF selected six reactor technologies for further research and development.

These include:

1. Very High Temperature Reactor (VHTR)
2. the Molten Salt Reactor (MSR),
3. the Sodium-cooled Fast Reactor (SFR),
4. the Super Critical Water-cooled Reactor (SCWR),
5. the Gas-cooled Fast Reactor (GFR), and
6. the Lead-cooled Fast Reactor (LFR)

Figure 18.21 below, is illustration of the six types of reactors that are considered as part of Generation IV power plant. More details of each of these reactors are provided in later sections.

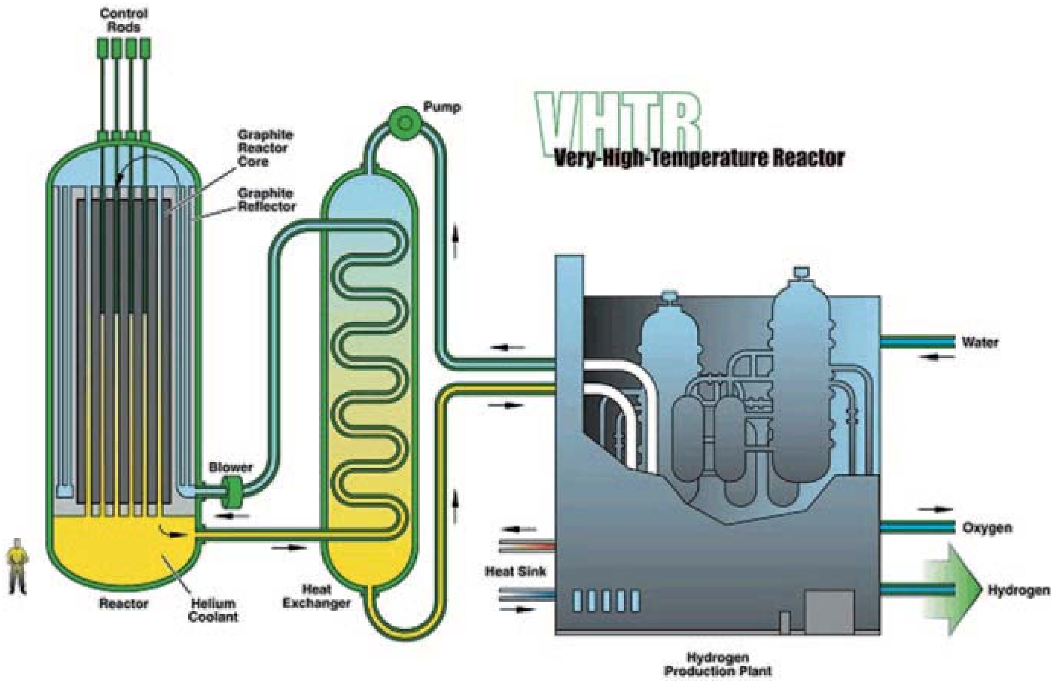


Fig. 18.22 Very high temperature reactor. (Courtesy of the Generation IV International Forum)

18.15.1 Very High Temperature Reactor (VHTR)

Among the six candidates of the Gen IV nuclear systems in the technical roadmap of Gen IV International Forum (GIF), the Very High Temperature Reactor (VHTR) is primarily dedicated to the cogeneration of electricity and hydrogen, the latter being extracted from water by using thermo-chemical, electro-chemical or hybrid processes. Its high outlet temperature makes it attractive also for the chemical, oil and iron industries. Original target of outlet temperature of 1000 °C from VHTR can support the efficient production of hydrogen by thermo-chemical processes. The technical basis for VHTR is the TRISO coated particle fuel, the graphite as the core structure, helium coolant, as well as the dedicated core layout and lower power density to removal decay heat in a natural way. The VHTR has potential for inherent safety, high thermal efficiency, process heat application capability, low operation and maintenance costs, and modular construction (Fig. 18.22).

The VHTR is a next step in the evolutionary development of high-temperature gas-cooled reactors. It is a graphite-moderated, helium-cooled reactor with thermal neutron spectrum. It can supply nuclear heat and electricity over a range of core outlet temperatures between 700 and 950 °C, or more than 1000 °C in future. The reactor core type of the VHTR can be a prismatic block core such as the Japanese HTTR, or a pebble-bed core such as the Chinese HTR-10. For electricity generation, a helium gas turbine system can be directly set in the primary coolant loop, which is called a direct cycle or at the lower end of the outlet temperature range, a steam generator can be used with a conventional rankine cycle. For nuclear heat applications such as process heat for refineries, petrochemistry, metallurgy, and hydrogen production, the heat application process is generally coupled with the reactor through an intermediate heat exchanger (IHX), the so-called indirect cycle. The VHTR can produce hydrogen from only heat and water by using thermochemical processes (such as the sulfur-iodine (S-I) process or the hybrid sulfur process), high temperature steam electrolysis (HTSE), or from heat, water, and natural gas by applying the steam reformer technology.

While, the original approach for VHTR at the start of the Generation IV program focused on very high outlet temperatures and hydrogen production, current market assessments have indicated that electricity production and industrial processes based on high temperature steam that require modest outlet temperatures (700–850 °C) have the greatest potential for application in the next decade. This also reduces technical risk associated with higher outlet temperatures. As a result, over the past decade, the focus has moved from higher outlet temperature designs such as GT-MHR and PBMR to lower outlet temperature designs such as HTR-PM in China and the NGNP in the US.

The VHTR has two typical reactor configurations, namely:

1. the pebble bed type and
2. the prismatic block type

Although the shape of the fuel element for two configurations are different, the technical basis for both configuration is same, such as the TRISO coated particle fuel in the graphite matrix, full ceramic (graphite) core structure, helium coolant, and low power density.

This will allow achieving high outlet temperature and the retention of fission production inside the coated particle under normal operation condition and accident condition. The VHTR can support alternative fuel cycles such as U-Pu, Pu, MOX, U-Th.

18.15.2 Molten Salt Reactor (MSR)

The MSR is distinguished by its core in, which the fuel is dissolved in molten fluoride salt. The technology was first studied more than 50 years ago. Modern interest is on fast reactor concepts as a long-term alternative to solid-fuelled fast neutrons reactors. The onsite fuel-reprocessing unit using pyrochemistry allows breeding plutonium or uranium-233 from thorium. R&D progresses toward resolving feasibility issues and assessing safety and performance of the design concepts. Key feasibility issues focus on a dedicated safety approach and the development of salt redox potential measurement and control tools in order to limit corrosion rate of structural materials. Further work on the batch wise online salt processing is required. Much work is needed on molten salt technology and related equipments.

Molten Salt Reactor (MSR) technology was partly developed, including two demonstration reactors, in the 1950s and 1960s in the USA (Oak Ridge National Laboratory). The demonstrations MSRs were thermal-neutron-spectrum graphite-moderated concepts. Since 2005, R&D has focused on the development of fast-spectrum MSR concepts (MSFR) combining the generic assets of fast neutron reactors (extended resource utilization, waste minimization) with those relating to molten salt fluorides as fluid fuel and coolant (low pressure and high boiling temperature, optical transparency).

In contrast to most other molten salt reactors previously studied, the MSFR does not include any solid moderator (usually graphite) in the core. This design choice is motivated by the study of parameters such as feedback coefficient, breeding ratio, graphite lifespan and 233U initial inventory. MSFR exhibit large negative temperature and void reactivity coefficients, a unique safety characteristic not found in solid-fuel fast reactors.

Compared with solid-fuel reactors, MSFR systems have lower fissile inventories, no radiation damage constraint on attainable fuel burn-up, no requirement to fabricate and handle solid fuel, and a homogeneous isotopic composition of fuel in the reactor. These and other characteristics give MSFRs potentially unique capabilities for actinide burning and extending fuel resources.

MSR developments in Russia on the Molten Salt Actinide Recycler and Transmuter (MOSART) aim to be used as efficient burners of transuranic (TRU) waste from spent UOX and MOX light water reactor (LWR) fuel without any uranium and thorium support and also with it. Other advanced reactor concepts are being studied, which use the liquid salt technology, as a primary coolant for Fluoride salt-cooled High-temperature Reactors (FHRs), and coated particle fuels similar to high temperature gas-cooled reactors (Fig. 18.23).

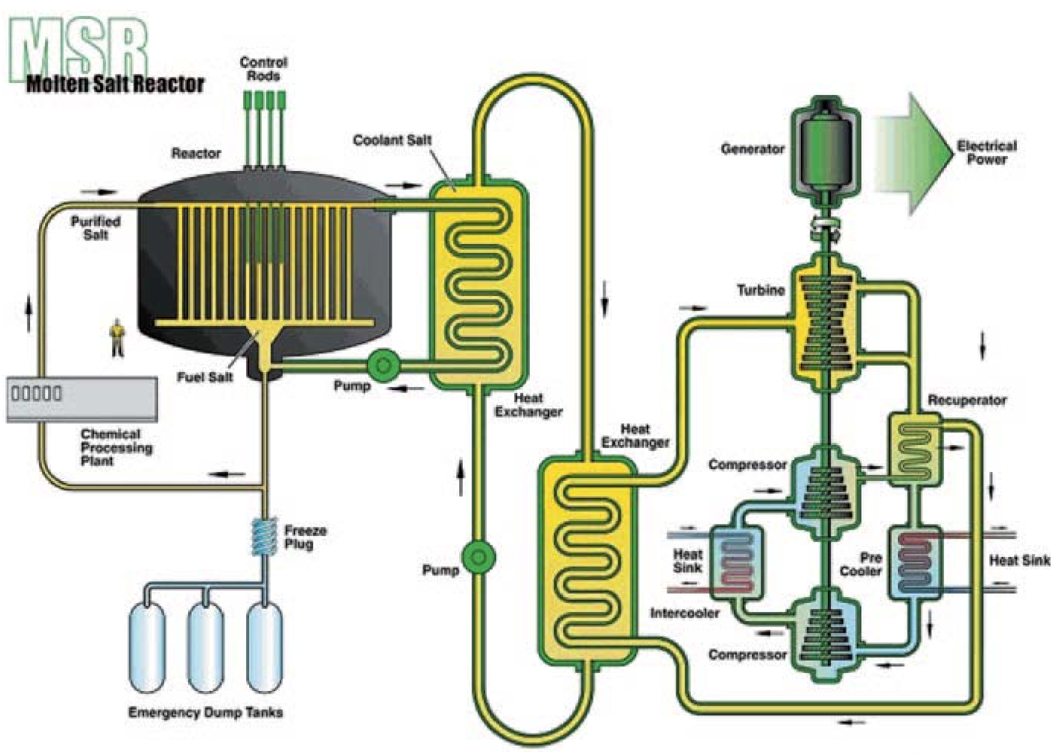


Fig. 18.23 Molten salt reactor. (Courtesy of the Generation IV International Forum)

More generally, there has been a significant renewal of interest in the use of liquid salt as a coolant for nuclear and non-nuclear applications. These salts could facilitate heat transfer for nuclear hydrogen production concepts, concentrated solar electricity generation, oil refineries, and shale oil processing facilities amongst other applications.

18.15.3 Sodium Cooled Fast Reactor (SFR)

The Sodium Cooled Fast Reactor (SFR) uses liquid sodium as the reactor coolant, allowing high power density with low coolant volume fraction and operation at low pressure. While the oxygen-free environment prevents corrosion, sodium reacts chemically with air and water and requires a sealed coolant system.

Plant size options under considerations are ranging from small, 50 to 300 MWe, modular reactors to larger plants up to 1500 MWe. The outlet temperature is 500–550 °C for the options, which allows the use of the materials developed and proven in prior fast reactor programs.

The SFR closed fuel cycle enables regeneration of fissile fuel and facilitates management of minor actinides. However, this requires that recycle fuels be developed and qualified for use. Important safety features of the Generation IV system include a long thermal response time, a reasonable margin to coolant boiling, a primary system that operates near atmospheric pressure, and an intermediate sodium system between the radioactive sodium in the primary system and the power conversion system. Water/steam, supercritical carbon-dioxide or nitrogen can be considered as working fluids for the power conversion system to achieve high performance in terms of thermal efficiency, safety and reliability. With innovations to reduce capital cost, the SFR is aimed to be economically competitive in future electricity markets. In addition, the fast neutron spectrum greatly extends the uranium resources compared to thermal reactors. The SFR is considered to be the nearest-term deployable system for actinide management (Fig. 18.24).

Much of the basic technology for the SFR has been established in former fast reactor programmes, and is being confirmed by the Phenix end-of-life tests in France, the restart of Monju in Japan and the lifetime extension of BN-600 in Russia. New programs involving SFR technology include the Chinese experimental fast reactor (CEFR) which was connected to the grid in July 2011, and India's prototype fast breeder reactor (PFBR) which is currently planned to go critical in 2013.

The SFR is an attractive energy source for nations that desire to make the best use of limited nuclear fuel resources and manage nuclear waste by closing the fuel cycle.

Fast reactors hold a unique role in the actinide management mission because they operate with high energy neutrons that are more effective at fissioning actinides. The main characteristics of the SFR for actinide management mission are:

- Consumption of transuranics in a closed fuel cycle, thus reducing the radiotoxicity and heat load, which facilitates waste disposal and geologic isolation.

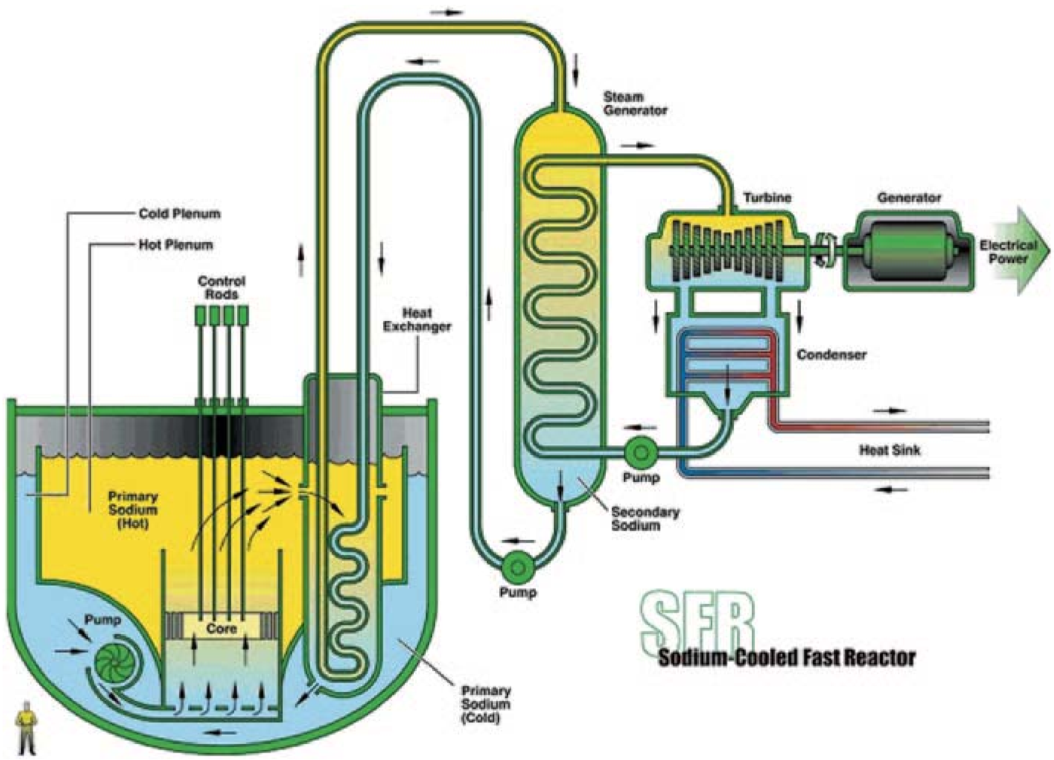


Fig. 18.24 Sodium cooled fast reactor. (Courtesy of the Generation IV International Forum)

- Enhanced utilization of uranium resources through efficient management of fissile materials and multi-recycle.

High level of safety achieved through inherent and passive means also allows accommodation of transients and bounding events with significant safety margins.

The reactor unit can be arranged in a pool layout or a compact loop layout. Three options are considered:

- A large size (600–1500 MWe) loop-type reactor with mixed uranium-plutonium oxide fuel and potentially minor actinides, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors.
- An intermediate-to-large size (300–1500 MWe) pool-type reactor with oxide or metal fuel.
- A small size (50–150 MWe) modular-type reactor with uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyro-metallurgical processing in facilities integrated with the reactor.

18.15.4 Super Critical Water Cooled Reactor (SCWR)

The Super Critical Water Cooled (SCWRs) are high temperature, high-pressure, light-water-cooled reactors that operate above the thermodynamic critical point of water (374 °C, 22.1 MPa).

The reactor core may have a thermal or a fast-neutron spectrum, depending on the core design. The concept may be based on current pressure vessel or on pressure tube reactors, and thus use light water or heavy water as moderator. Unlike current water-cooled reactors, the coolant will experience a significantly higher enthalpy rise in the core, which reduces the core mass flow for a given thermal power and increases the core outlet enthalpy to superheated conditions. For both pressure vessel and pressure-tube designs, a once through steam cycle has been envisaged, omitting any coolant recirculation inside the reactor. As in a boiling water reactor, the superheated steam will be supplied directly to the high-pressure steam turbine and the feed water from the steam cycle will be supplied back to the core. Thus, the SCWR concepts combine the design and operation experiences gained from hundreds of water-cooled reactors with those experiences from hundreds of fossil-fired power plants operated with supercritical water (SCW). In contrast to some of the other Generation IV nuclear systems, the SCWR can be developed incrementally step-by-step from current water-cooled reactors.

a. **Advantage and Challenges**
   Such SCWR designs have unique features that offer many advantages compared to state-of the-art water-cooled reactors:

   - SCWRs offer increases in thermal efficiency relative to current-generation water-cooled reactors. The efficiency of a SCWR can approach 44 % or more, compared to 34–36 % for current reactors.
   - Reactor coolant pumps are not required. The only pumps driving the coolant under normal operating conditions are the feed water pumps and the condensate extraction pumps.
   - The steam generators used in pressurized water reactors and the steam separators and dryers used in boiling water reactors can be omitted since the coolant is superheated in the core.
   - Containment, designed with pressure suppression pools and with emergency cooling and residual heat removal systems, can be significantly smaller than those of current water-cooled reactors can.
   - The higher steam enthalpy allows to decrease the size of the turbine system and thus to lower the capital costs of the conventional island.

These general features offer the potential of lower capital costs for a given electric power of the plant and of better fuel utilization, and thus a clear economic advantage compared with current light water reactors.

However, there are several technological challenges associated with the development of the SCWR, and particularly the need to validate transient heat transfer models (for describing the depressurization from supercritical to sub-critical conditions), qualification of materials (namely advanced steels for cladding), and demonstration of the passive safety systems.

b. **GIF Progress up to 2012**
   Pre-conceptual core design studies for a core outlet temperature of more than 500 °C have been performed in Japan, assuming either a thermal neutron

spectrum or a fast neutron spectrum. Both options are based on a coolant heat-up in two steps with intermediate mixing underneath the core. Additional moderator for a thermal neutron spectrum is provided by feed water inside water rods. The fast-spectrum option uses zirconium-hydride ($ZrH2$) layers to minimize hardening of the neutron spectrum in case of core voiding. A pre-conceptual design of safety systems for both options has been studied with transient analyses.

A pre-conceptual plant design with 1700 MW net electric power based on a pressure-vessel-type reactor has been studied by Yamada et al. and has been assessed with respect to efficiency, safety and cost. The study confirms the target net efficiency of 44 % and estimates a cost reduction potential of 30 % compared with current pressurized water reactors. Safety features are expected to be similar to advanced boiling water reactors.

A pre-conceptual design of a pressure-vessel-type reactor with a 500 °C core outlet temperature and 1000 MW electric power has been developed in Europe, as summarized by Schulenberg and Starflinger. The core design is based on coolant heat-up in three steps. Additional moderator for the thermal neutron spectrum is provided in water rods and in gaps between assembly boxes. The design of the nuclear island and of the balance of the plant confirms results obtained in Japan, namely an efficiency improvement up to 43.5 % and a cost reduction potential of 20–30 % compared with latest boiling water reactors. Safety features as defined by the stringent European Utility Requirements are expected to be met.

Canada is developing a pressure-tube-type SCWR concept with a 625 °C core outlet temperature at the pressure of 25 MPa. The concept is designed to generate 1200 MW electric power (a 300 MW concept is also being considered). It has a modular fuel channel configuration with separate coolant and moderator. A high-efficiency fuel channel is incorporated to house the fuel assembly. The heavy-water moderator directly contacts the pressure tube and is contained inside a low-pressure calandria vessel. In addition to providing moderation during normal operation, it is designed to remove decay heat from the high-efficiency fuel channel during long-term cooling using a passive moderator cooling system. A mixture of thorium oxide and plutonium is introduced as the reference fuel, which aligns with the GIF position paper on thorium fuel. The safety system design of the Canadian SCWR is similar to that of the ESBWR. However, the introduction of the passive moderator cooling system coupled with the high-efficiency channel could reduce significantly the core damage frequency during postulated severe accidents such as large-break loss-of-coolant or station blackout events.

Pre-conceptual designs of three variants of pressure vessel supercritical reactors with thermal, mixed and fast neutron spectrum have been developed in Russia, which joined the SCWR System Arrangement in 2011.

Outside of the GIF framework, two conceptual SCWR designs with thermal and mixed neutron spectrum cores have been established by some research institutes in China. This is done, under framework of the Chinese national R&D projects from 2007 to 2012, covering some basic research projects on materials and thermo hydraulics, the core/fuel design, the main system design (including the conventional part), safety systems design, reactor structure design and fuel assembly structure design. The related feasibility studies have also been completed, and show that the design concept has promising prospects in terms of the overall performance, integration of design, component structure feasibility and manufacturability.

Prediction of heat transfer in SCW can be based on data from fossil fired power plants as discussed by Pioro et al. Computational tools for more complex geometries like fuel assemblies are available but still need to be validated with bundle experiments. System codes for transient safety analyses have been upgraded to include SCW, including depressurization transients to subcritical conditions. Flow stability in the core has been studied numerically. As in boiling water reactors, flow stability can be ensured using suitable inlet orifices in fuel assemblies.

A number of candidate cladding materials have been tested in capsules, autoclaves and recirculating loops up to 700 °C at a pressure of 25 MPa. Stainless steels with more than 20 % chromium (Cr) are expected to have the required corrosion resistance up to a peak cladding temperature of 650 °C. More work is needed to develop alloys suitable for use at the design peak cladding temperatures of 850 °C for the Canadian SCWR concept. Further work is also needed to better identify the coolant conditions that lead to stress corrosion cracking. It has been shown that the creep resistance of existing alloys can be improved by adding small amounts of elements, such as zirconium (Zr), as reported by Kaneda et al. In the longer term, the steel experimental oxide dispersion strengthened (ODS) alloys offer an even higher potential, whereas nickel-base alloys are being considered for use in ultra supercritical fossil fired plants are less favorable for use in SCWRs due to their high neutron absorption and associated swelling and embrittlement.

Key water chemistry issues have been identified by Guzonas et al.; predicting and controlling water radiolysis and corrosion product transport (including fission products) remain the major R&D areas. In this regard, the operating experience using nuclear steam reheat at the Beloyarsk nuclear power plant in Russia is extremely valuable (Fig. 18.25).

18.15.5 Gas Cooled Fast Reactor (GFR)

The Gas Cooled Reactor (GFR) system is a high-temperature helium-cooled fast-spectrum reactor with a closed fuel cycle. It combines the advantages of fast-spectrum systems for long-term sustainability of uranium resources and waste minimization (through fuel multiple reprocessing and fission of long-lived actinides), with those of high-temperature systems (high thermal cycle efficiency and industrial use of the generated heat, for hydrogen production for example).

The GFR uses the same fuel recycling processes as the SFR and the same reactor technology as the VHTR. Therefore, its development approach is to rely, in so far as feasible, on technologies developed for the VHTR for structures, materials, components and power conversion system. Nevertheless, it calls for specific R&D

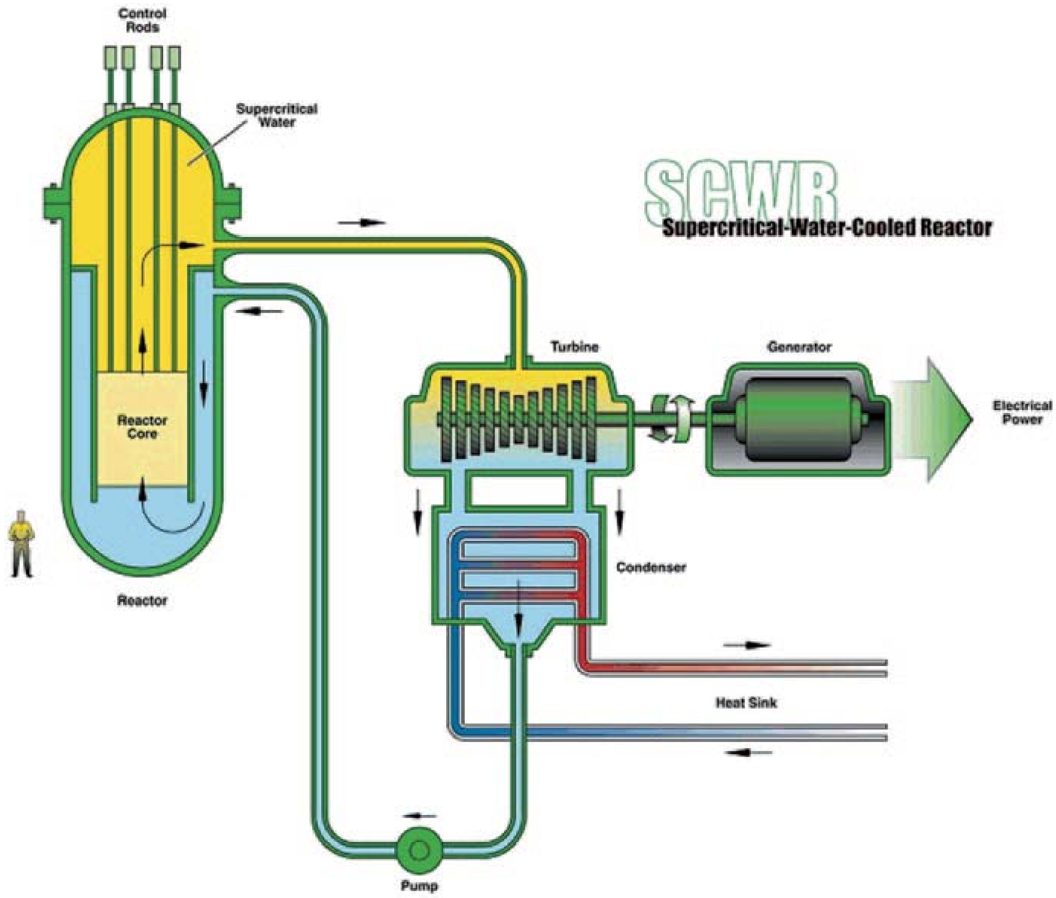


Fig. 18.25 Super critical water cooled reactor. (Courtesy of the Generation IV International Forum)

beyond the current and foreseen work on the VHTR system, mainly on core design and safety approach.

The reference design for GFR is based around a 2400 MWth reactor core contained within a steel pressure vessel. The core consists of an assembly of hexagonal fuel elements, each consisting of ceramic-clad, mixed-carbide-fuelled pins contained within a ceramic hex-tube. The favoured material at the moment for the pin clad and hex-tubes is silicon carbide fibre reinforced silicon carbide. The figure below shows the reactor core located within its fabricated steel pressure vessel surrounded by main heat exchangers and decay heat removal loops. The whole of the primary circuit is contained within a secondary pressure boundary, the guard containment (Figs. 18.26 and 18.27).

The coolant is helium and the core outlet temperature will be of the order of 850 °C. A heat exchanger transfers the heat from the primary helium coolant to a secondary gas cycle containing a helium-nitrogen mixture, which, in turn drives a closed cycle gas turbine. The waste heat from the gas turbine exhaust is used to raise steam in a steam generator, which is then used to drive a steam turbine. Such a combined cycle is common practice in natural gas-fired power plant so represents an established technology, with the only difference in the GFR case being the use of a closed cycle gas turbine.

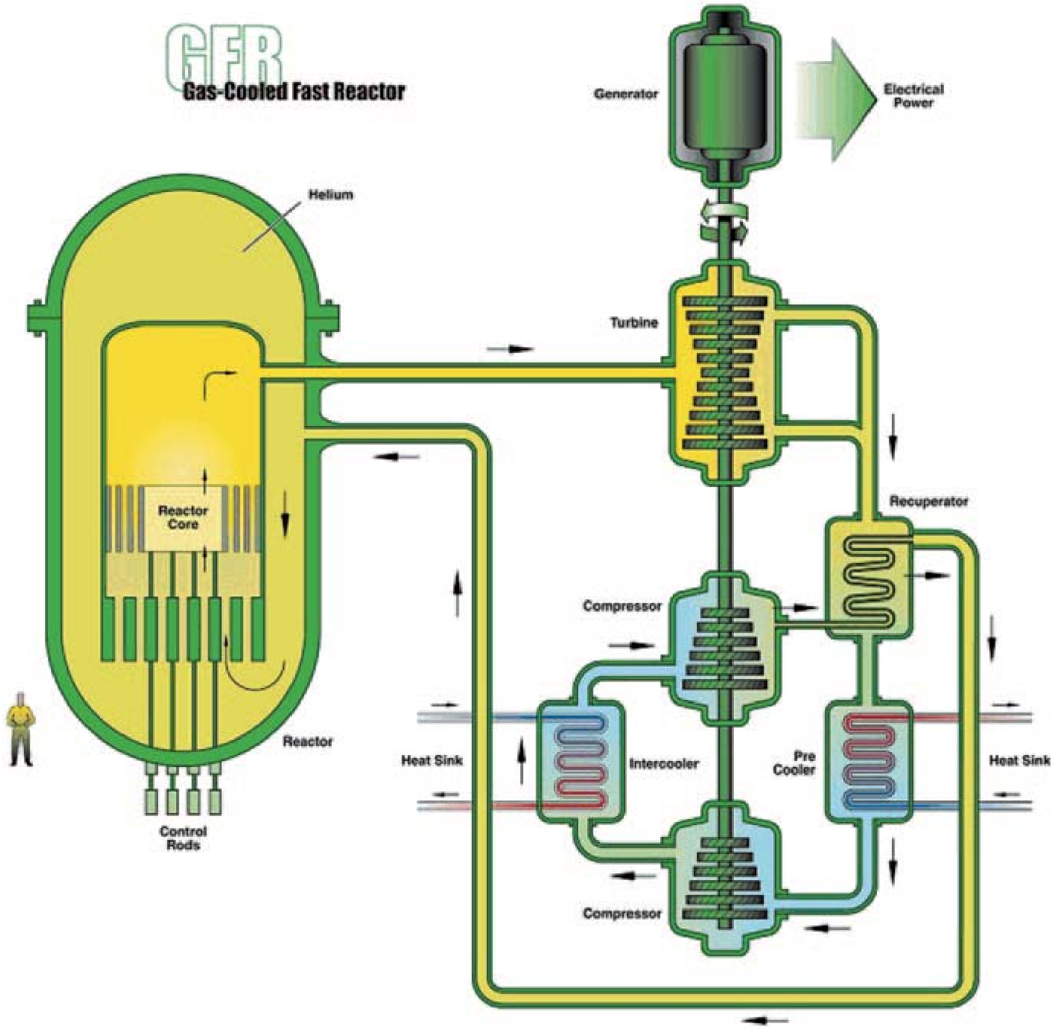


Fig. 18.26 Gas cooled fast reactor. (Courtesy of the Generation IV International Forum)

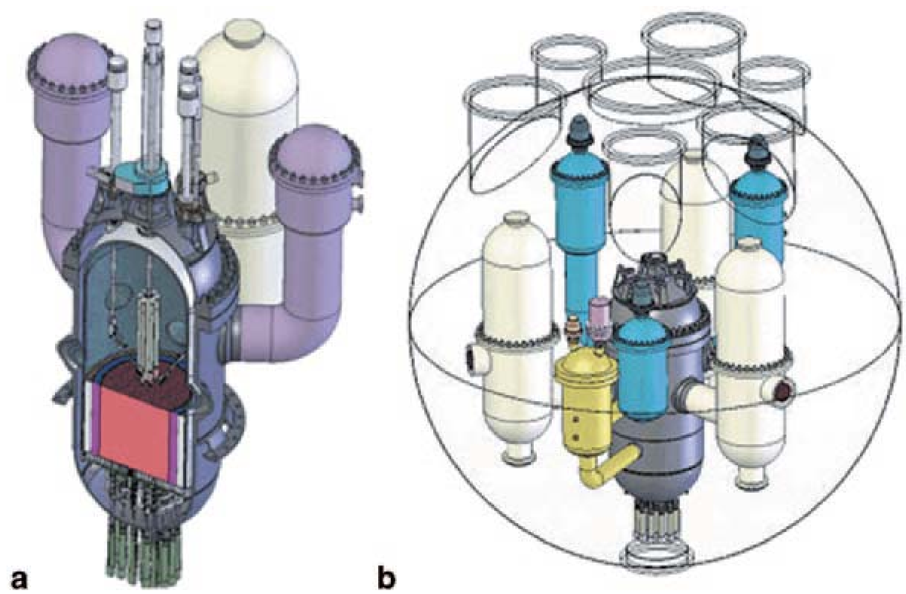


Fig. 18.27 **a** GFR-reactor, decay heat loops, main heat exchangers and fuel handling equipment. **b** GFR spherical guard vessel. (Courtesy of the Generation IV International Forum)

18.15.6 Lead Cooled Fast Reactor (LFR)

The Lead-cooled Fast Reactor (LFR) features a fast neutron spectrum, high temperature operation, and cooling by molten lead or Lead-Bismuth Eutectic (LBE), low-pressure, chemically inert liquids with very good thermodynamic properties. It would have multiple applications including production of electricity, hydrogen and process heat. System concepts represented in plans of the Generation-IV International Forum (GIF) System Research Plan (SRP) are based on Europe's ELFR lead-cooled system, Russia's BREST-OD-300 and the SSTAR system concept designed in the US.

The LFR has excellent materials management capabilities since it operates in the fast-neutron spectrum and uses a closed fuel cycle for efficient conversion of fertile uranium. It can also be used as a burner to consume actinides from spent LWR fuel and as a burner/breeder with thorium matrices. An important feature of the LFR is the enhanced safety that results from the choice of molten lead as a chemically inert and low-pressure coolant. In terms of sustainability, lead is abundant and hence available, even in case of deployment of a large number of reactors. More importantly, as with other fast systems, fuel sustainability is greatly enhanced by the conversion capabilities of the LFR fuel cycle. Because they incorporate a liquid coolant with a very high margin to boiling and benign interaction with air or water, LFR concepts offer substantial potential in terms of safety, design simplification, proliferation resistance and the resulting economic performance. An important factor is the potential for benign end state to severe accidents (Fig. 18.28).

The LFR has development needs in the areas of fuels, materials performance, and corrosion control. During the next 5 years progress is expected on materials, system design, and operating parameters. Significant test and demonstration activities are underway and planned during this period.

18.16 Next Generation of Nuclear Power Reactors for Power Production

Experts are projecting worldwide electricity consumption will increase substantially in the coming decades, especially in the development world, accompanying economic growth and social progress that has direct impact on rising electricity prices have focused fresh attention on nuclear power plants. New, safer and more economical nuclear reactors could not only satisfy many of our future energy needs but could combat global warming as well. Today's existing nuclear power plants on line in the United States provide fifth of the nation's total electrical output.

Taking into account the expected increase in energy demand worldwide and the growing awareness about global warming, climate change issues and sustainable development, nuclear energy will be needed to meet future global energy demand.

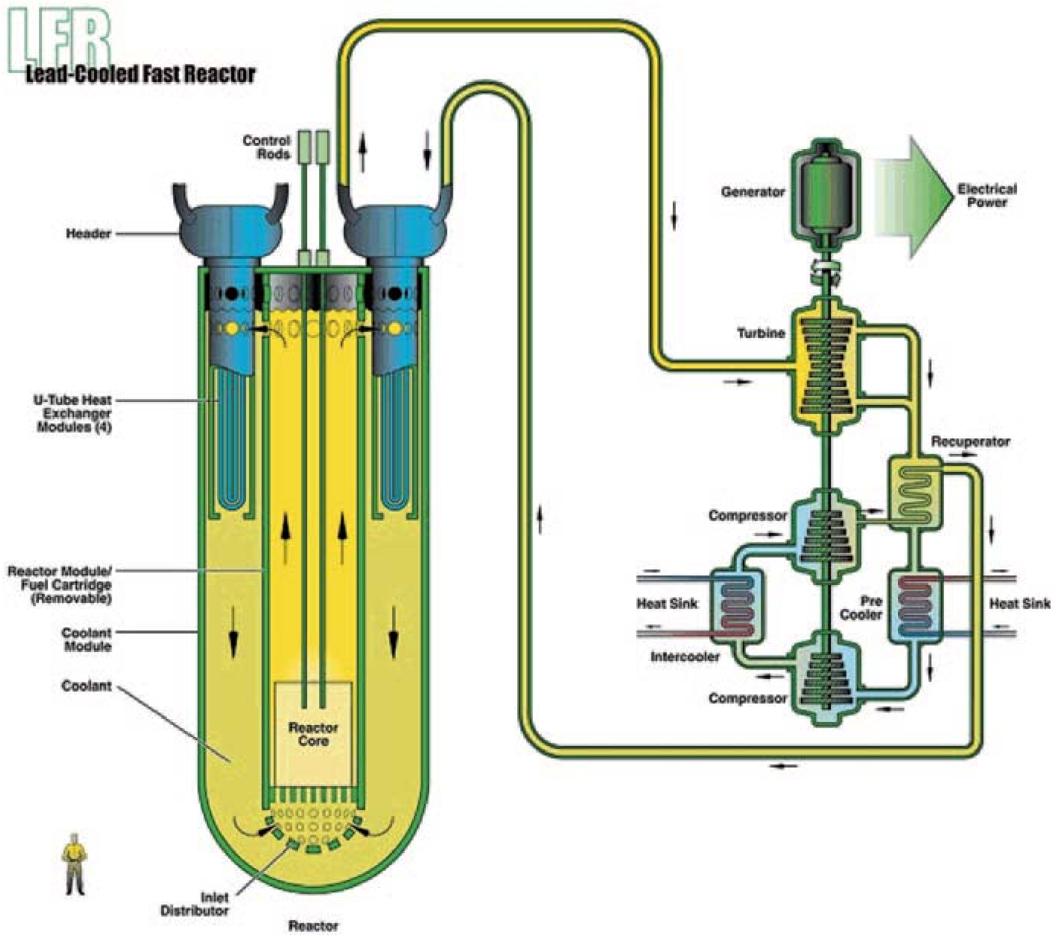


Fig. 18.28 Lead cooled fast reactor. (Courtesy of the Generation IV International Forum)

Nuclear power plant technology has evolved as distinct design generations as we mentioned in previous section and briefly summarized here again as follows:

- First Generation: prototypes, and first realizations (~1950–1970)
- Second Generation: current operating plants (~197–2030)
- Third generation: deployable improvements to current reactors (~2000 and on)
- Fourth generation: advanced and new reactor systems (2030 and beyond)

The Generation IV International Forum, or GIF, was chartered in July 2001 to lead the collaborative efforts of the world's leading nuclear technology nations to develop next generation nuclear energy systems to meet the world's future energy needs.

Eight technology goals have been defined for Generation IV systems in four broad areas:

1. Sustainability,
2. Economics,
3. Safety and Reliability, and finally,
4. Proliferation resistance and Physical protection.

A large number of countries share these ambitious goals as they aim at responding to economic, environmental and social requirements of the twenty-first century. They establish a framework and identify concrete targets for focusing GIF R&D efforts

Eight technology goals have been defined for Generation IV systems in four broad areas: sustainability, economics, safety and reliability, and proliferation resistance and physical protection.

18.17 Goals for Generation IV Nuclear Energy Systems

The next generation ("Generation IV") of nuclear energy systems is intended to meet the below goals (while being at least as effective as the "third" generation in terms of economic competitiveness, safety and reliability) in order to provide a sustainable development of nuclear energy.

In principle, the Generation IV Systems should be marketable or deployable from 2030 onwards. The systems should also offer a true potential for new applications compatible with an expanded use of nuclear energy, in particular in the fields of hydrogen or synthetic hydrocarbon production, seawater desalination and process heat production.

It has been recognized that these objectives, widely and officially shared by a large number of countries, should be at the basis of an internationally shared R&D program, which allows keeping open and consolidating the technical options, and avoiding any early or premature down selection.

Sustainability—1	Generation IV nuclear energy systems will provide sustainable energy generation that meets clean air objectives and provides long term availability of systems and effective fuel utilization for worldwide energy production.
Sustainability—2	Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden, thereby improving protection for the public health and the environment.
Economics—1	Generation IV nuclear energy systems will have a clear life cycle cost advantage over other energy sources.
Economics—2	Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.
Safety and reliability—1	Generation IV nuclear energy systems operations will excel in safety and reliability.
Safety and reliability—2	Generation IV nuclear systems will have a very low likelihood and degree of reactor core damage.
Safety and reliability—3	Generation IV nuclear energy systems will eliminate the need for offsite emergency response.
Proliferation resistance and physical protection	Generation IV nuclear energy systems will increase the assurance that they are very unattractive and the least desirable route for diversion or theft of weapons usable materials, and provide increased physical protection against acts of terrorism.

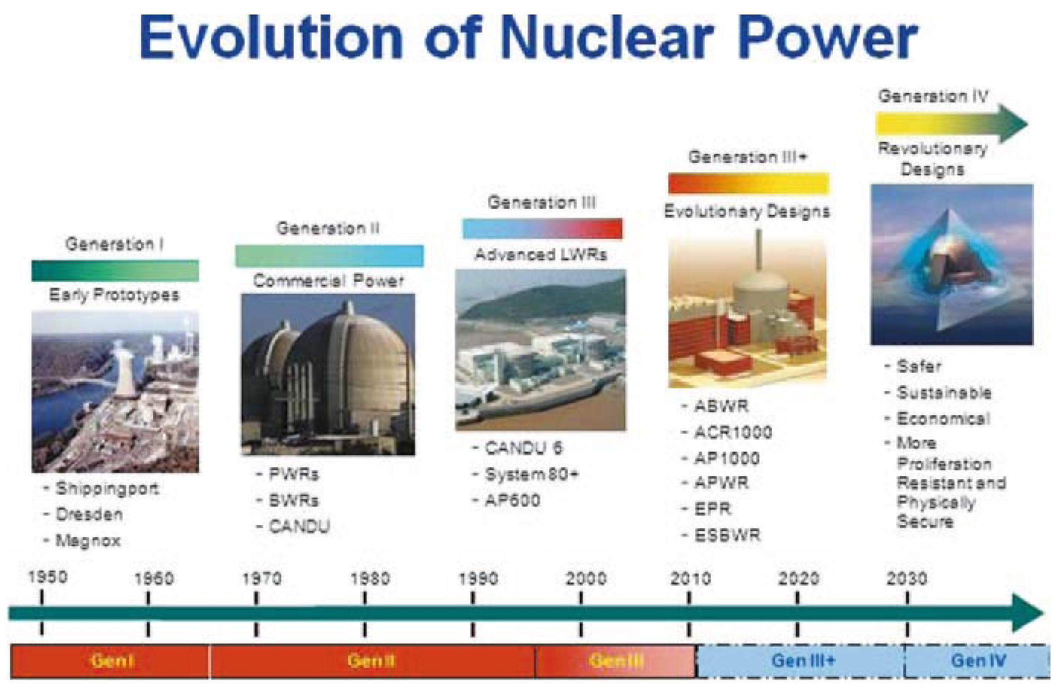


Fig. 18.29 Evolution of nuclear power plants

In fact, because the next generation nuclear energy systems will address needed areas of improvement and offer great potential, many countries share a common interest in advanced R&D that will support their development. The international research community should explore such development benefits from the identification of promising research areas and collaborative efforts that. The collaboration on R&D by many nations on the development of advanced next generation nuclear energy systems will in principle aid the progress toward the realization of such systems, by leveraging resources, providing synergistic opportunities, avoiding unnecessary duplication and enhancing collaboration (Fig. 18.29).

In 2009, the Experts Group published an outlook on Generation IV R&D, to provide a view of what GIF members hope to achieve collectively in the period 2010–2014. All Generation IV systems have features aiming at performance improvement, new applications of nuclear energy, and/or more sustainable approaches to the management of nuclear materials. High-temperature systems offer the possibility of efficient process heat applications and eventually hydrogen production. Enhanced sustainability is achieved primarily through adoption of a closed fuel cycle with reprocessing and recycling of plutonium, uranium and minor actinides using fast reactors; this approach provides significant reduction in waste generation and uranium resource requirements. The following Table summarizes the main characteristics of the six Generation IV systems.

System	Neutron spectrum	Coolant	Temp. °C	Fuel cycle	Size (MWe)
VHTR (Very high temperature gas reactor)	Thermal	Helium	900–1000	Open	250–300
SFR (Sodium-cooled fast reactor)	Fast	Sodium	550	Closed	30–150, 300–1500 1000–2000
SCWR (Supercritical water—cooled reactor)	Thermal/fast	Water	510–625	Open/Closed	300–700 1000–2000
GFR (Gas—cooled fast reactor)	Fast	Helium	850	Closed	1200
LFR (Lead—cooled fast reactor)	Fast	Lead	480–800	Closed	20–180 300–1200 600–1000
MSR (Molten salt reactor)	Epithermal	Fluoride salt	700–800	Closed	1000

18.18 Why We Need to Consider the Future Role of Nuclear Power Now

The following reasoning's are some arguments that show why we need to consider the future role in design of new nuclear power plant;

1. Nuclear power has been part of the global energy need mix for the past five decades. Currently it provides about 18 % of the electricity we use in our homes and workplaces. For example in the UK, about one third of our emissions of carbon dioxide come from electricity generation. The vast majority of those emissions come from coal and gas power plants.
2. Energy companies will need to invest in around 30–35 GW of new electricity generating capacity—as coal and nuclear plants retire—over the next two decades, with around two-thirds needed by 2020. This is equivalent to about one-third of our existing capacity. The world needs a clear and stable regulatory framework to reduce uncertainty for business to help ensure sufficient and timely investment in technologies that contribute to our energy goals.
3. Of the capacity that is likely to close over the two decades, two thirds is from carbon intensive fossil fuel generation and about 10 GW is nuclear and therefore low carbon. So companies' decisions on the type of power stations they invest in to replace this capacity will have significant implications for the level of carbon emissions. As an illustration, if our existing nuclear power stations were all replaced with fossil fuel fired power stations, our emissions would be between 8 and 16 MtC (million tons of carbon) a year higher as a result (depending on the mix of gas and coal-fired power stations). This would be equivalent to about 30–60 % of the total carbon savings we project to achieve under our central

scenario from all the measures we are bringing forward in the Energy White Paper. Our gas demand would also be higher, at a time when we are becoming more dependent on imported sources of fossil fuels.

4. Electricity demand in the United States is expected to grow significantly in the future. Over the past decade, Americans used 17 % more electricity, but domestic capacity rose only 2.3 % (National Energy Policy, May 2001). Unless the United States significantly increases its generating capacity, the country will face an energy shortage that is projected to adversely affect our economy, our standard of living, and our national security. Coupled with this challenge is the need to improve our environment.
5. New nuclear power stations have long lead times. This time is necessary to secure the relevant regulatory and development consents, which must be obtained before construction can begin, and there is also a long construction period compared to other generating technologies. Our conservative assumption is that for the first new nuclear plant the pre-construction period would last around 8 years (to secure the necessary consents) and the construction period would last around 5 years. For subsequent plants, this is assumed to be 5 and 5 years; respectively. New nuclear power stations are therefore unlikely to make a significant contribution to the need for new capacity before 2020.
6. Even with our expectations that the share of renewable will grow, it is likely that fossil fuel generation will meet some of this need. However, beyond that date there are still significant amounts of new capacity needed; for example, in 2023 one third or 3 GW of our nuclear capacity will still be operational, based on published lifetimes. Given the likely increase in fossil fuel generation before this date, it is important that much of this capacity is replaced with low carbon technologies. New nuclear power stations could make an important contribution, as outlined in this consultation document, to meeting our needs for low carbon electricity generation and energy security in this period and beyond to 2050. Because of the lead-times, without clarity now we will foreclose the opportunity for nuclear power.
7. The existing approach on new nuclear build was set out in 200311:"Nuclear power is currently an important source of carbon-free electricity. However, its current economics make it an unattractive option for new, carbon-free generating capacity and there are also important issues of nuclear waste to be resolved. These issues include our legacy waste and continued waste arising from other sources. This white paper does not contain specific proposals for building new nuclear power stations. However, we do not rule out the possibility that at some point in the future new nuclear build might be necessary if we are to meet our carbon targets. Before any decision to proceed with the building of new nuclear power stations, there will need to be the fullest public consultation and the publication of a further white paper setting out our proposals."
8. Since 2003 there have been a number of developments, which have led the Government to consider afresh the potential contribution of new nuclear power stations. Firstly, there has been significant progress in tackling the legacy waste issue:

  - we have technical solutions for waste disposal that scientific consensus and experience from abroad suggest could accommodate all types of wastes from existing and new nuclear power stations;
  - there is now an implementing body (the Nuclear Decommissioning Authority), with expertise in this area, and Government is reconstituting the Committee on Radioactive Waste Management (CoRWM) in order to provide continued independent scrutiny and advice; and
  - a framework for implementing long-term waste disposal in a geological repository will be consulted on in the coming months.

9. The Government has also made progress in considering the issue of waste management in relation to potential new nuclear power stations:

   - This consultation provides the opportunity to discuss the ethical, intergenerational and public acceptability issues associated with a decision to allow the private sector to invest in new nuclear power stations and generate new nuclear waste;
   - The Government is developing specific proposals to protect the taxpayer. Under these proposals, private sector developers would meet the full decommissioning costs and full share of waste management costs. The proposals would be implemented in the event that we conclude that energy companies should be allowed to invest in new nuclear power stations. They would need to be in place before proposals for new power stations could go ahead.

10. Secondly, the high-level economic analysis of nuclear power, prepared for the Energy Review, concluded that under likely scenarios for gas and carbon prices and taking prudent estimates of nuclear costs, nuclear power would offer general economic benefit to the UK in terms of reduced carbon emissions and security of supply benefits. Therefore, the Government believes that it has a potential contribution to make, alongside other low-carbon generating technologies.
11. Thirdly, some energy companies have expressed a strong interest in investing in new nuclear power stations. They assess that new nuclear power stations could be an economically attractive low-carbon investment, which could help diversify their generation portfolios. Their renewed interest reflects assessments that with carbon being priced to reflect its impacts and gas prices likely to be higher than previously expected, the economics of new nuclear power stations are becoming more favorable.
12. Nuclear power stations have long lead times. If they are to be an option to replace the capacity closing over the next two decades, and in particular after 2020, a decision on whether allowing energy companies the option of investing in new nuclear power stations would be in the public interest, needs to be taken now. Energy companies would need to begin their initial preparations in the near future in order to have a reasonable prospect of building new generation in this period. Not taking the public interest decision now would foreclose the option of new nuclear being one of our options for tackling climate change and achieving energy security.

18.19 The Generation IV Roadmap Project

As the Generation IV goals were being finalized, preparations were made to develop the Generation IV technology roadmap. The organization of the roadmap is shown in the Fig. 15.21 below. The Roadmap Integration Team (RIT) is the executive group. Groups of international experts were organized to undertake identification and evaluation of candidate systems, and to define R&D to support them (Fig. 18.30).

In a first step, an Evaluation Methodology Group was formed to develop a process to systematically evaluate the potential of proposed Generation IV nuclear energy systems to meet the Generation IV goals. A discussion of the Evaluation Methodology Group's evaluation methodology is included in this report. At the same time, a solicitation was issued worldwide, requesting that concept proponents submit information on nuclear energy systems that they believe could meet some or all of the Generation IV goals. Nearly 100 concepts and ideas were received from researchers in a dozen countries [8].

Technical Working Groups (TWGs) were formed—covering nuclear energy systems employing water-cooled, gas-cooled, liquid-metal-cooled, and non-classical reactor concepts—to review the proposed systems and evaluate their potential using the tools developed by the Evaluation Methodology Group. Because of the large number of system concepts submitted, the TWGs collected their concepts into sets of concepts with similar attributes. The TWGs conducted an initial screening, *termed screening for potential*, to eliminate those concepts or concept sets that did not have reasonable potential for advancing the goals, or were too distant or technically infeasible [9].

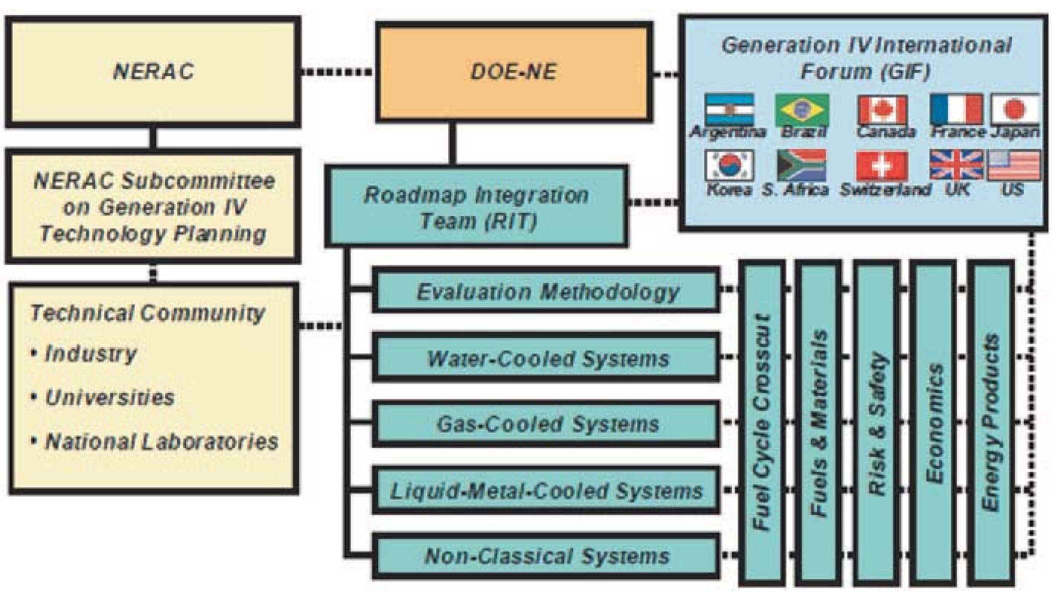


Fig. 18.30 The roadmap organization

A Fuel Cycle Crosscut Group (FCCG) was also formed at a very early stage to explore the impact of the choice of fuel cycle on major elements of sustainability—especially waste management and fuel utilization. Their members were equally drawn from the working groups, allowing them to compare their insights and findings directly. Later, other Crosscut Groups were formed covering economics, risk and safety, fuels and materials, and energy products. The Crosscut Groups reviewed the TWG reports for consistency in the technical evaluations and subject treatment, and continued to make recommendations regarding the scope and priority for crosscutting R&D in their subject areas. Finally, the TWGs and Crosscut Groups worked together to report on the R&D needs and priorities of the most promising concepts.

The international experts that contributed to this roadmap represented all ten GIF countries, the Organization for Economic Cooperation and Development Nuclear Energy Agency, the European Commission, and the International Atomic Energy Agency.

18.20 Licensing Strategy Components

A DOE and NRC working group was formed to develop the licensing strategy. This group conducted an in-depth analysis of LWR licensing process and technical requirements options, which was performed by the experienced senior staff of the two agencies. The methodology used in formulating the NGNP licensing strategy alternatives also included development of a phenomena identification and ranking table (PIRT) for a prototypical NGNP by subject matter experts in the nuclear field. The PIRT results assisted in the identification of key R&D needs. Based on the detailed analysis of these alternatives and balancing schedule considerations with licensing risk and other pertinent factors, the Secretary of Energy and the Commission concluded that the following NGNP licensing strategy provides the best opportunity for meeting the 2021 date for initial operation of a prototype NGNP, which details of such analysis can be found in NGNP report to Congress.

NGNP reactor technology will differ from that of commercial LWRs currently used for electric power generation. LWRs have a well-established framework of regulatory requirements, a technical basis for these requirements, and supporting regulatory guidance on acceptable approaches an applicant can take to show that NRC requirements are met. The NRC uses a Standard Review Plan to review licensing applications for these reactor designs. Additionally, the NRC has a well-established set of validated analytical codes and methods and a well-established infrastructure for conducting safety research needed to support its independent safety review of an LWR plant design and the technical adequacy of a licensing application.

New nuclear power plants can be licensed under either of two existing regulatory approaches. The first approach is the traditional "two-step" process described in Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the *Code of Federal Regulations* (10 CFR Part 50), which requires both a Construction Permit (CP) and a separate Operating License (OL). The second approach is the new "one-step" licensing process described in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," which incorporates a combined Construction and Operating License (COL). Both of these processes allow a deterministic or risk-informed performance-based approach to technical requirements.

Many of the regulatory requirements and supporting review guidance for LWRs are technology-neutral; that is, they are applicable to non-LWR designs as well as LWR designs. However, certain LWR requirements may not apply to the unique aspects of a VHTR design. Accordingly, in developing the NGNP licensing strategy, the NRC and DOE considered the various options available to the NRC staff for adapting current NRC LWR licensing requirements for the NGNP VHTR. These options related to legal, process, technical, research, and regulatory infrastructure matters and included an examination of historical licensing activities. These considerations led to selection of a licensing strategy that would comply best with the considerations identified in the EPAct.

The licensing strategy outlined in this report is composed of two distinct aspects. The first aspect is a recommended approach for how the NRC will adapt the current LWR technical requirements to apply to a VHTR. The second aspect is a recommended licensing process alternative that identifies which of the procedural alternatives in the NRC regulations would be best for licensing the NGNP. To arrive at these recommendations, NRC and DOE evaluated a number of options and alternatives.

18.21 Market and Industry Status and Potentials

Europe plays a leading role in the development of nuclear energy and has 35 % of the globally installed capacity. The reactors in Europe have been in operation for 27 years on average. Current plans in most EU member countries are to extend their lifetime on a case by case basis beyond 40 years, and even beyond 60 years in some cases, in combination with power upgrades. The first two Generation-III reactors, European Pressurized-water Reactor (EPR) are currently being constructed.

The global growth of the nuclear energy can be measured by the increasing number of reactors (three more in 2005 and 2006; seven in 2007 and ten in 2008), but with a strong concentration in Asia. Nevertheless a number of these reactors are of European design. There are presently four reactors under construction in Europe: the EPRs in Finland and France and two smaller reactors of Generation-II type (VVER 440) in Slovakia and with plans to build new reactors in France, Romania, Bulgaria and Lithuania. Perhaps more importantly the UK has taken concrete steps towards new build with bidding beginning in 2009 from leading utilities, and Italy has declared that it intends to start a nuclear program with a target to produce 25 % of the electricity by 2030. The estimated maximum potential installed capacities of nuclear fission power for the EU-27, (150 GWe by 2020 and 200 GWe by 2030) appear more realistic than the baseline (115 GWe in 2020 and 100 GWe in 2030).

Programs to build fast reactor and high-temperature reactor demonstrators are being implemented in Russia and several Asian countries. Although these are not Generation-IV designs, transfer of knowledge and experience from operation will contribute significantly to future Generation-IV development. In Europe, a concerted effort is proposed in the form of a European Industrial Initiative in sustainable nuclear fission as part of the Community's SET-Plan. The EII has singled out the Sodium Fast Reactor (SFR) as its primary system with the basic design selected by 2012 and construction of a prototype of 250–600 MWe that is connected to the grid and operational by 2020.

In parallel, a gas- or lead-cooled fast reactor (GFR/LFR) will also be investigated. The selection of the alternative fast reactor technology is scheduled for 2012 on the basis of a current program of pre-conceptual design research. The reactor will be a 50–100 MWth demonstrator reactor that should also be in operation by 2020. The SFR prototype and LFR/GFR demonstrator will be complemented by a fuel fabrication workshop that should serve both systems, and by a range of new or refurbished supporting experimental facilities for qualification of safety systems, components, materials and codes. A commercial deployment for a SFR reactor is expected from 2040 and for the alternative design a decade later.

High temperature reactors dedicated to cogeneration of process heat for the production of synthetic fuels or industrial energy products could be available to meet market needs by 2025, which would trigger requirements to construct "first of a kind" demonstrators in the next few years. Indeed, such programs are currently being set up in some countries (USA, Japan, South Africa and China). The key aspect is the demonstration of the coupling with the conventional industrial plant. Supercritical water reactors and molten salt reactors, as well as accelerator driven sub-critical systems dedicated to transmutation of nuclear waste, are currently being assessed in terms of feasibility and performance, though possible industrial applications have yet to be clearly identified.

18.22 Barriers

The high capital cost of nuclear energy in combination with uncertain long-term conditions constitutes a financial risk for utilities and investors. The lack of widespread support in the EU Member States may undermine the strength of EU industry for the development of new technologies. Harmonized regulations, codes and standards at the EU-level would strengthen the competitiveness of Europe's nuclear sector and promote deployment of Generation-III technology in the near term. The industry, infrastructures and services that support nuclear power has shrunk significantly during the last decades. This situation in Europe is not unique but it may pose a bottleneck for the deployment of reactors in the relatively near future. One example is large forgings needed for pressure vessel heads. World capacity is limited and even at the present new build construction rate there is a waiting list for delivery of these components.

Public acceptance remains an important issue, but even though opinion is not very favorable in a number of Member States, there are signs that the mood is changing. Nevertheless, concerted efforts are still required, based on objective and open dialogue amongst all stakeholders. International cooperation currently exists at the level of research, and this is being facilitated in the area of Generation-IV systems by the Generation-IV International Forum (GIF). However, EU industry is facing stiff competition, especially in Asia where strong corporate support for R&D is putting industry in a better position to gain leadership in the near future. Another significant potential barrier for nuclear fission is the shortage of qualified engineers and scientists as a result of the lack of interest in nuclear careers during the 1990s and the reduced availability of specialist courses at universities. Preservation of nuclear knowledge remains a major issue, especially since most of the current generations of nuclear experts are nearing retirement.

18.23 Needs

The high initial capital investments and sensitive nature of the technology involved means that renewed deployment of currently available nuclear technology can only take place in a stable (or, at least, predictable) regulatory, economic and political environment. In June 2009, the EU established a common binding framework on nuclear safety with the adoption of the Council Directive establishing a Community framework for the safety of nuclear installations [8, 9]. This is the first binding EU legislation in this field.

In order to retain its leading position and to overcome bottlenecks in the supply chain, Europe also needs to re-invigorate the industrial supply chains supporting the nuclear sector. Apart from this overriding requirement for a clear European strategy on nuclear energy, a new research and innovation system is needed that can assure additional funding, especially for the development of Generation-IV technology. In this context the Sustainable Nuclear Energy Technology Platform plays a key role. The timescales involved, and the fact that key political and strategic decisions are yet to be taken regarding this technology, mean that a significant part of this additional funding must be public. The launch of the European Sustainable Nuclear Industrial Initiative under the Community's SET-Plan, bringing together key industrial and R&D organizations would be a very significant step towards the construction and operation of the necessary demonstrators and prototypes.

High temperature reactors based on existing technology can also be deployed in the near future with the aim of demonstrating the co-generation of process heat and the coupling with industrial processes. This would need to be built and funded through a European or International consortium, which should also include the process heat end-user industries. In the meantime, an enhanced research effort is needed to ensure Europe's leadership in sustainable nuclear energy technologies that include continuous innovation in LWRs, qualification and development of materials, closed fuel cycle with U-Pu multi-recycling and (very) high temperature reactors and related fuel technology.

Breakthroughs are especially sought in the fields of materials to enhance safety, nuclear fuels and fuel cycle processes. Additionally, there is a need for harmonization of European standards and a strategic planning of national and European research infrastructures for use by the European research community. The implementation of geological disposal of high-level waste is also being pursued as part of national waste management programs, though some countries are not as advanced as others. The new Implementing Geological Disposal Technology Platform, launched in November 2009, is coordinating the remaining necessary applied research in Europe leading up to the start of operation of the first geological repositories for high-level and long-lived waste around 2020, and will facilitate progress in and technology transfer with other national programs.

More effort is needed to inform and interact with the public and other stakeholders, and the education and training of a new generation of nuclear scientists and engineers and transfer of knowledge from the generation that designed and built reactors in the seventies and eighties needs urgent attention. The European Nuclear Energy Forum (ENEF) provides a unique platform for a broad open discussion on the role nuclear power plays today and could play in the low carbon economy of the future. ENEF analyses and discusses the opportunities (competitiveness, financing, grid, etc) and risks (safety, waste) and need for education and training associated with the use of nuclear power and proposes effective ways to foster communication with and participation of the public.

18.24 Synergies with Other Sectors

Nuclear energy provides a very stable base-load electricity supply and can therefore work in synergy with renewable energies that are more intermittent. Nuclear energy should also contribute significantly to a low-carbon transport sector as high temperature applications can provide synthetic fuel and hydrogen, while generated electricity could provide a large share of the energy for electrical cars. Interactions are anticipated with activities in “Hydrogen Energy and Fuel Cells” through the potential of nuclear hydrogen production and with “grids” from the characteristics of nuclear electricity generation. With respect to basic materials research, there should be synergies with other applications, such as “Biofuels” and “Clean Coal”, where materials are subjected to extreme environments. In addition, the opportunities for important common research with the fusion program, especially in the area of materials, need to be fully exploited. The European Energy Research Alliance under the SET-Plan is also expected to provide opportunities for synergies and collaborative work in the area of nuclear materials. In general, cross cutting research would benefit from more clearly defined channels of interaction, responsibilities and increased flexibility regarding funding and programming.

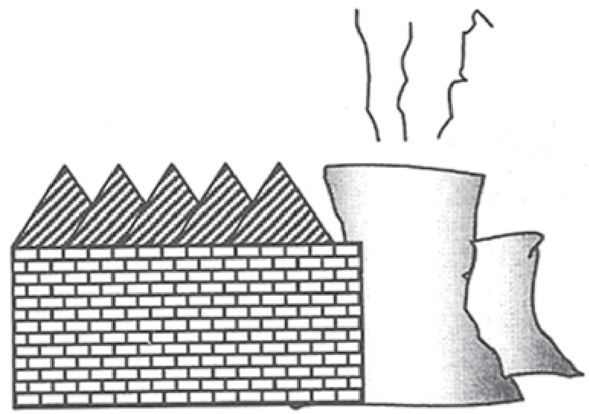


Fig. 18.31 Sketch for Problem 18.1

Problems

Problem 18.1: A 1000 MW power plant is powered by nuclear fuel. Determined the amount of nuclear fuel consumed per year. See Fig. 18.31 below.

Problem 18.2: A 1000 MW power plant is powered by burning coal. Calculate the amount of coal consumed per year.

Problem 18.3: A power plant that burns coal produces 1.1 kg of carbon dioxide (CO_2) per kWh. Determine the amount of CO_2 production that is due to the refrigerators in a city. Assume that the city uses electricity produced by a coal power plant.

Problem 18.4: A person trades in his Ford Taurus for a Ford Explorer. Calculate the amount of CO_2 emitted by the Explorer within 5 years. Assume the Explorer is assumed to use 940 gallons a year compared to 715 gallons for Taurus.

Problem 18.5: A power plant that burns natural gas produces 0.59 kg of carbon dioxide (CO_2) per kWh. Calculate the amount of CO_2 production that is due to the refrigerators in the city. Assume the city uses electricity produced by a natural gas power plant. Give the fact that 0.59 kg of CO_2 is produced per kWh of electricity generated. Noting that there are 200,000 households in the city and each household consumes 700 kWh of electricity for refrigeration.

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