Section 10: FISSION POWER

This section briefly describes the basic principles underlying the development of the fission power reactor, some examples of commercial reactors in use today and issues about the possible future of nuclear fission power.

1. Nuclear fission and the chain reaction

Energy:

Nuclear energy is used to provide power in nuclear reactors.

Fission generates a large amount of energy [~200 MeV/fission] because the BE/A for the fragments is about 0.9 MeV *higher* than that of the parent fissioning nucleus.

Neutron emission:

Several prompt neutrons are emitted during fission. These can go on to induce further fissions and if this number $k > 1$, the result is a **chain reaction** in which the number of neutrons increases exponentially. k is called the neutron multiplicity factor.

- An uncontrolled chain reaction is the basis of the nuclear bomb.
- The chain reaction can be controlled by ensuring that k is always kept close to unity. In a reactor, this is done by inserting control rods containing neutron-absorbing material [e.g. boron or cadmium].
- A fraction of emitted neutrons are delayed [see Section 8]. This delay allows enough time to control the neutron flux (provided k is close to unity) and hence, the power level in a reactor safely.

Cross section:

The probability [cross section σ] of a neutron inducing fission generally increases as the neutron energy is reduced. In the case of ^{235}U, σ is about 1000 times higher at thermal energies [~0.025 eV] than it is at a typical emission energy [~MeV].

However, this is not the case for ^{238}U, for which there is a threshold energy >1 MeV for a neutron to induce fission. Natural uranium consists almost entirely [99.27%] of ^{238}U and a reactor cannot be constructed with natural uranium unless the neutrons are slowed down (moderated) to increase the fission probability of the minority isotope ^{235}U.

Moderation:

Neutrons are slowed down in a reactor by surrounding the uranium fuel with light materials e.g. carbon or water in which neutrons lose energy in successive collisions with light nuclei in the moderator.

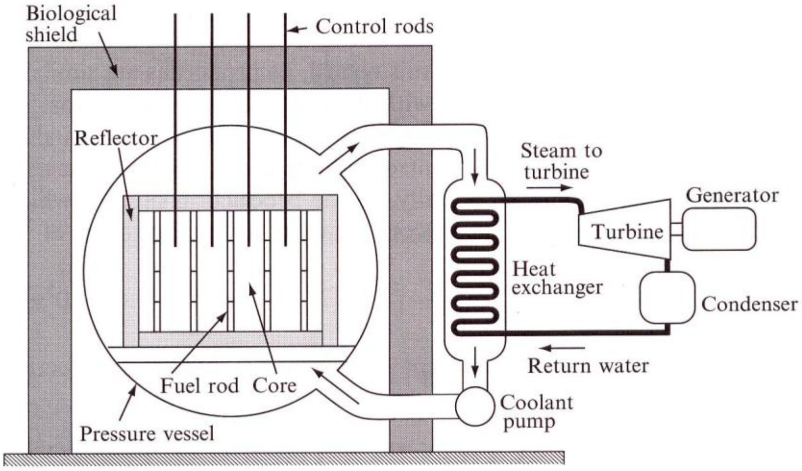
A good moderator should have a low probability for absorbing neutrons i.e. σ(scattering) >> σ(absorption). Heavy water (D_2O), in which hydrogen in replaced by deuterium, would be ideal as has a very low probability of absorbing neutrons.

2. Commercial Thermal Reactors

Early gas cooled reactors

Uranium isotopes are difficult to separate from each other and shortly after WWII neither heavy water nor enriched uranium was available. Natural water (H_2O) absorbs too many neutrons and so the early, commercial reactors were built using natural uranium moderated with carbon.

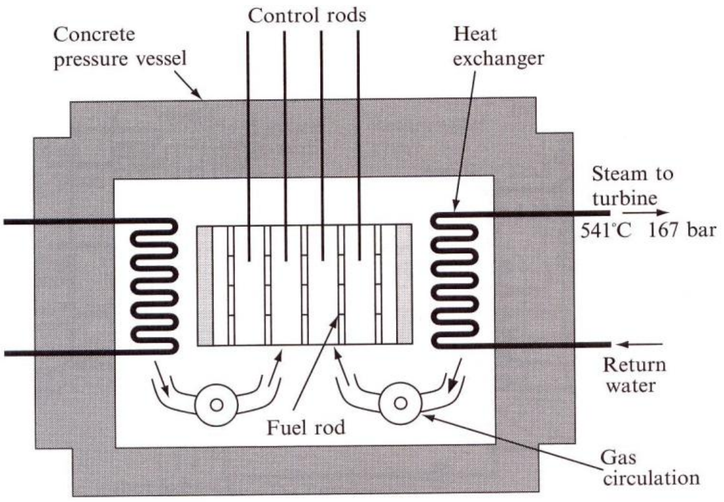
In these reactors, the uranium fuel was encased in a tube of magnesium alloy called Magnox. A matrix of rods, surrounded with graphite moderator, formed the reactor core. Cooling was done by circulating CO_2 gas through the core, which then was passed through a heat exchanger where steam was produced to drive turbines for generating electricity.



The figure shows a schematic layout of a typical Magnox power station. It has a large core, about 10m high and16m in diameter. This is supported in a spherical pre-stressed concrete pressure vessel, about 30m in diameter.

Advanced gas-cooled reactor [AGR]

The Magnox reactor had an efficiency of only about 28% and the AGR was designed to have an improved efficiency by allowing higher operating temperatures.



The figure shows a simplified layout of an AGR.

- Stainless steel or a zirconium alloy (Zircalloy) is used for fuel cladding.
- Fuel is uranium oxide or carbide – both of which have very high melting points.
- Operating temperature is up to 650 ^{o}C

AGRs are more efficient than the Magnox reactor. However, graphite-moderated reactors are large and their capital cost is high.

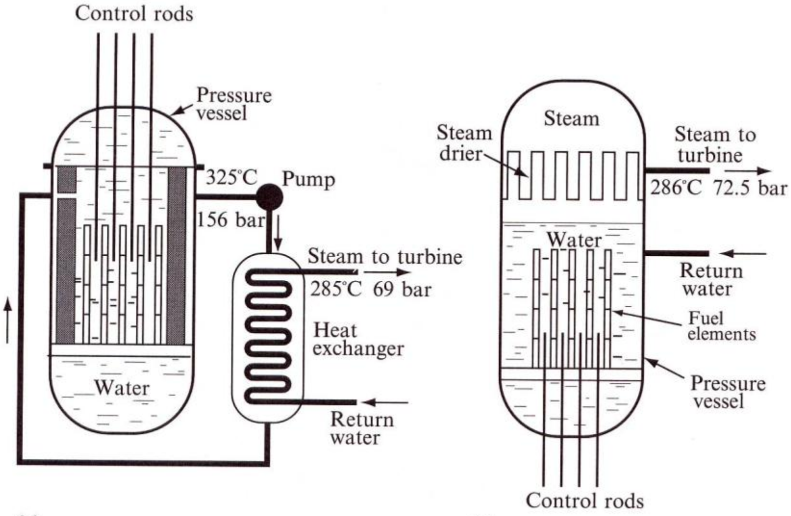
Nowadays, uranium enriched in ^{235}U is available and modern, much more compact (and cheaper) reactors can be built using natural (light) water as moderator.

Light-water reactors.

Most modern reactor designs capitalise on the good qualities of light water as a moderator. Its excellent slowing-down properties enables the size of the core to be reduced considerably to one which is ~3m in height and diameter.

Enriched fuel must be used to offset the greater tendency of water to absorb neutrons.

There are two types of water-cooled reactor: the Pressurized water reactor [PWR] and the Boiling water reactor [BWR]. They are shown schematically below as (a) and (b) respectively. In both types, water is used as both moderator and coolant.



(a) (b)

In the **PWR**, water is prevented from boiling to maintain its effectiveness as a coolant. It is subjected to very high pressure (up to 160 bar) to raise its boiling point.

The PWR was developed in the U.S.A. Many have been built throughout the world and most new reactors are of this type. An example, is the reactor at Sizewell B in the U.K. It contains 4.5% enriched uranium fuel as UO_2 pellets encased in tubes arranged in a core of diameter 3.4 m and height 3.7 m. The pressure vessel has a wall thickness of 21.5cm and the operating pressure is 156 bar.

Small PWRs are used to power naval vessels, most notably submarines. This allows them to be able to remain at sea for long periods of time without the need for refuelling.

In the **BWR,** water is allowed to boil and steam is passed directly to the turbines, thus avoiding the need for a heat exchanger. The design is such that if there is a sudden increase in boiling and steam replaces water, the effect of the moderator is reduced leading to a drop in power. This self-stabilizing behaviour allows the BWR to follow a load demand.

A typical BWR contains 140 tonnes of fuel in a core approximately 4.7 m diameter and 3.7 m high. Operating pressure is about 70 bar.

Heavy-water reactors

There are some commercial reactors that use heavy water as moderator and coolant. These are the CANDU reactors that were developed in Canada after the end of WWII. They can use natural uranium as fuel, however, heavy water is expensive and, as with a light water reactor, it is necessary to operate at high pressure.

An alternative version uses light water as coolant, which reduces the cost.

3. Future of Nuclear Fission Power

There are concerns about the proliferation of nuclear power throughout the world - the possibility of accidents, control of fissile material and dealing with radioactive waste. Yet despite these concerns, there is a continuing and growing demand for energy and the need for the source of that energy to be secure. There is also concern about the impact on the planet of the continuing use of fossil fuel.

Here we briefly consider two issues:

- Dealing with waste
- Future availability of fissile fuel

Waste

Waste is produced when a reactor is refuelled and especially when it is decommissioned.

Most of the radioactivity in the waste is from **fission fragments**, most of which have half-lives of a few 10s of years and require to be safely stored for several 100 years.

However, some radio nuclides have very long half-lives. These are mainly certain actinide nuclei: e.g. ^{239}Pu ($t_{1/2}$ = 24000y), ^{242}Pu ($t_{1/2}$ = 373000y) and ^{243}Am ($t_{1/2}$ = 7400y). These need to be stored for hundreds of thousands of years.

Currently, it is proposed to deal with this high-level waste, by extracting it from the rest, converting it into inert, glassified material, containing it in concrete and then storing it underground in a geologically stable location.

However, since the actinide products are effectively unburned fuel, an alternative possibility would be to extract these and put them back into a reactor. Further neutron captures would occur and in time, they would be transformed into fissile nuclei, undergo fission and be converted into much shorter-lived fission fragments.

Fuel supply

The only naturally-occurring fissile nucleus (^{235}U) constitutes only about 0.7% of natural uranium. This means that nuclear power cannot be considered as a source of energy for the indefinite future unless more fissile fuel is created. This is the purpose of the *breeder reactor*, which is designed to convert ^{238}U and ^{232}Th into fissile materials: ^{239}Pu and ^{233}U.

Prototypes of one type of breeder have been built. A much more speculative concept is the proposed *accelerator-driven system*. We comment on each of these below:

Breeder reactor: All reactors produce new fissile material, but consume more than they produce: i.e. the so-called breeding ratio $B < 1$. A true breeder has $B > 1$.

A critical quantity is η - the number of neutrons produced in fission per neutron absorbed in fuel. In operation, one neutron is used to maintain the reactor; others are lost to capture (C) and leakage (L).

Typically, $C + L \sim 0.2$. Thus, $B \sim \eta - 1.2$. Some η values are shown in the following table:

Nucleus	Thermal neutrons (0.025 eV)	High-energy neutrons (0.5 MeV)
^{235}U	2.06	2.35
^{239}Pu	2.16	2.90
^{233}U	2.29	2.40

These figures show that only ^{233}U could be used in a thermal breeder reactor, which is why only **fast** breeder reactors – for which η is sufficiently high - have been built so far.

The core of a fast breeder must contain highly-enriched uranium. This is surrounded with a blanket of uranium and/or thorium, where new fissile material is produced. The core is very compact. It operates at very high power density and cooling is done using liquid metal – sodium or a sodium/potassium alloy.

Breeder reactors have been built and run successfully. However, they are more expensive to build and operate than a conventional reactor and cannot produce fuel sufficiently cheaply to make them commercially viable, at present.

Although fuel breeding is not needed now, it should be noted that it takes considerable time to breed enough new fissile material to fuel another reactor. For example, using values for a reasonably high-rated reactor, with $B = 1.2$, the time to double the initial fuel load is about 45 years, which is comparable with some time estimates for the lifetime of existing oil and gas reserves.

Accelerator-driven system [ADS]: In an ADS, an accelerator injects a high intensity beam of energetic, charged particles (protons or deuterons) into a core, made of uranium and thorium, which is sub-critical, i.e. one for which the neutron multiplicity factor $k < 1$. The interaction of the beam in the core generates neutrons - about 30 neutrons per 1 GeV proton. Some of these neutrons would induce further fissions, producing energy and amplifying the initial neutron flux. Others would be used to breed new fuel. The compact core means the neutron flux is high and proposed cooling uses a liquid metal, as in the fast breeder.

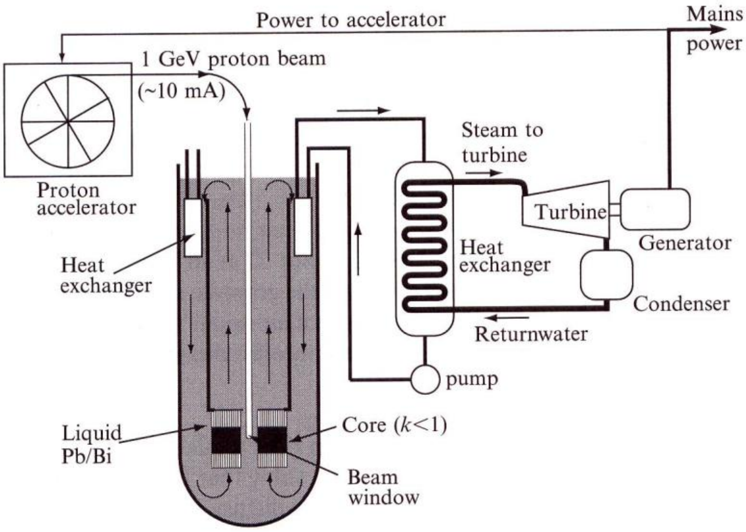
Advantages:

- A critical accident should be impossible because the system is operated with $k < 1$.
- The ADS could incinerate actinides to the extent that geological waste disposal would be greatly reduced.

A schematic layout of a proposed ADS is shown in the figure.

The accelerator delivers an intense beam of protons into a sub-critical, fast-fission unit. With $k \approx 0.95$, neutron amplification could be such that the system produces a useful excess of power above that needed to power the accelerator.

For fuel breeding, the core would contain ^{232}Th, which would be converted into ^{233}U. Thorium if more plentiful than uranium and thus constitutes a huge potential source of energy for the far distant future.



There are considerable technical challenges facing the proponents of the ADS. Some of these are:

1. Beam intensity at 1 GeV is many times the current world record.
2. Beam transport must be reliable with negligible loss.
3. The window between the evacuated beam line and the reactor core must be capable of functioning in an extremely hostile environment.
4. Efficiency for converting electrical energy into beam energy must be high (40-50%).

Achieving these targets will require major investment in R&D, but the possibility of safer nuclear power without the need for long term waste disposal may be sufficiently attractive to warrant the effort.