

DIFFERENT TYPES OF NUCLEAR POWER PLANT

INTRODUCTION

A nuclear reactor produces and controls the release of energy from splitting the atoms of certain elements. In a nuclear power reactor, the energy released is used as heat to make steam to generate electricity. (In a research reactor the main purpose is to utilise the actual neutrons produced in the core. In most naval reactors, steam drives a turbine directly for propulsion.)

The principles for using nuclear power to produce electricity are the same for most types of reactor. The energy released from continuous fission of the atoms of the fuel is harnessed as heat in either a gas or water, and is used to produce steam. The steam is used to drive the turbines which produce electricity (as in most fossil fuel plants).

The world's first nuclear reactors operated naturally in a uranium deposit about two billion years ago. These were in rich uranium orebodies and moderated by percolating rainwater. Those at Oklo in west Africa, each less than 100 kW thermal, together consumed about six tonnes of that uranium.

Today, reactors derived from designs originally developed for propelling submarines and large naval ships generate about 85% of the world's nuclear electricity. The main design is the pressurised water reactor (PWR) which has water at over 300°C under pressure in its primary cooling/heat transfer circuit, and generates steam in a secondary circuit. The less numerous boiling water reactor (BWR) makes steam in the primary circuit above the reactor core, at similar temperatures and pressure. Both types use water as both coolant and moderator, to slow neutrons. Since water normally boils at 100°C, they have robust steel pressure vessels or tubes to enable the higher operating temperature. (Another type uses

heavy water, with deuterium atoms, as moderator. Hence the term ‘light water’ is used to differentiate.)

Components of a nuclear reactor

There are several components common to most types of reactors:

Fuel. Uranium is the basic fuel. Usually pellets of uranium oxide (UO_2) are arranged in tubes to form fuel rods. The rods are arranged into fuel assemblies in the reactor core.

Moderator. Material in the core which slows down the neutrons released from fission so that they cause more fission. It is usually water, but may be heavy water or graphite.

Control rods. These are made with neutron-absorbing material such as cadmium, hafnium or boron, and are inserted or withdrawn from the core to control the rate of reaction, or to halt it. In some PWR reactors, special control rods are used to enable the core to sustain a low level of power efficiently. (Secondary control systems involve other neutron absorbers, usually boron in the coolant – its concentration can be adjusted over time as the fuel burns up.)

Coolant. A fluid circulating through the core so as to transfer the heat from it. In light water reactors the water moderator functions also as primary coolant. Except in BWRs, there is secondary coolant circuit where the water becomes steam. (See also later section on primary coolant characteristics).

Nuclear power plants in commercial operation

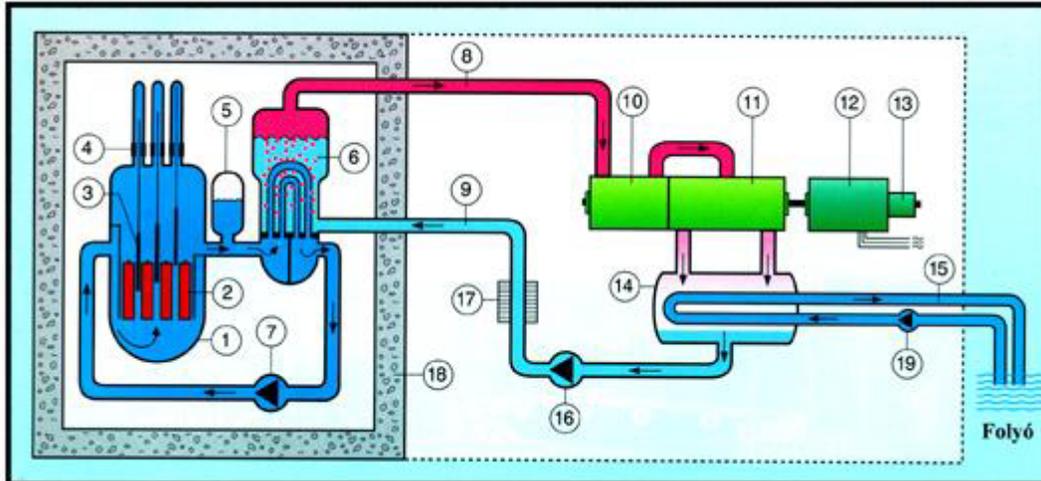
Reactor Type	Reactor Type Descriptive Name	Number of Reactors	Total Net Electrical Capacity [MW]
PWR	Pressurized Light-Water-Moderated and Cooled Reactor	274	253510
BWR	Boiling Light-Water-Cooled and Moderated Reactor	81	75773
PHWR	Pressurized Heavy-Water-Moderated and Cooled Reactor	48	23900
GCR	Gas-Cooled, Graphite-Moderated Reactor	15	8040
LWGR	Light-Water-Cooled, Graphite-Moderated Reactor	15	10219
FBR	Fast Breeder Reactor	2	580
Total		435	372022

Main Types of nuclear power plant:

1. Pressurized Water Reactor (PWR)

This is the most common type, with 274 in use for power generation and several hundred more employed for naval propulsion. The design of PWRs originated as a submarine power plant.

The pressurized water reactor belongs to the light water type: the moderator and coolant are both light water (H₂O). It can be seen in the figure that the cooling water circulates in two loops, which are fully separated from one another. In Russia these are known as VVER types - water-moderated and -cooled.



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|-------------------------|--------------------------|-------------------------|
| 1 Reactor vessel | 8 Fresh steam | 14 Condenser |
| 2 Fuel elements | 9 Feedwater | 15 Cooling water |
| 3 Control rods | 10 High pressure turbine | 16 Feedwater pump |
| 4 Control rod drive | 11 Low pressure turbine | 17 Feedwater pre-heater |
| 5 Pressurizer | 12 Generator | 18 Concrete shield |
| 6 Steam generator | 13 Exciter | 19 Cooling water pump |
| 7 Main circulating pump | | |

A PWR has fuel assemblies of 200-300 rods each, arranged vertically in the core, and a large reactor would have about 150-250 fuel assemblies with 80-100 tonnes of uranium.

The primary circuit water (dark blue) is continuously kept at a very high pressure and therefore it does not boil even at the high operating temperature. (Hence the name of the type.) Constant pressure is ensured with the aid of the pressurizer (expansion tank). (If pressure falls in the primary circuit, water in the pressurizers is heated up by electric heaters, thus raising the pressure. If pressure increases, colder cooling water is injected to the pressurizer. Since the upper part is steam, pressure will drop.) The primary circuit water transfers its heat to the secondary circuit water in the small tubes of the steam generator; it cools down and returns to the reactor vessel at a lower temperature.

Since the secondary circuit pressure is much lower than that of the primary circuit, the secondary circuit water in the steam generator starts to boil (red). The steam goes from here to the turbine, which has high and low pressure stages. When steam leaves the turbine, it becomes liquid again in the condenser, from where it is pumped back to the steam generator after pre-heating.

Normally, primary and secondary circuit waters cannot mix. In this way it can be achieved that any potentially radioactive material that gets into the primary water should stay in the primary loop and cannot get into the turbine and condenser. This is a barrier to prevent radioactive contamination from getting out.

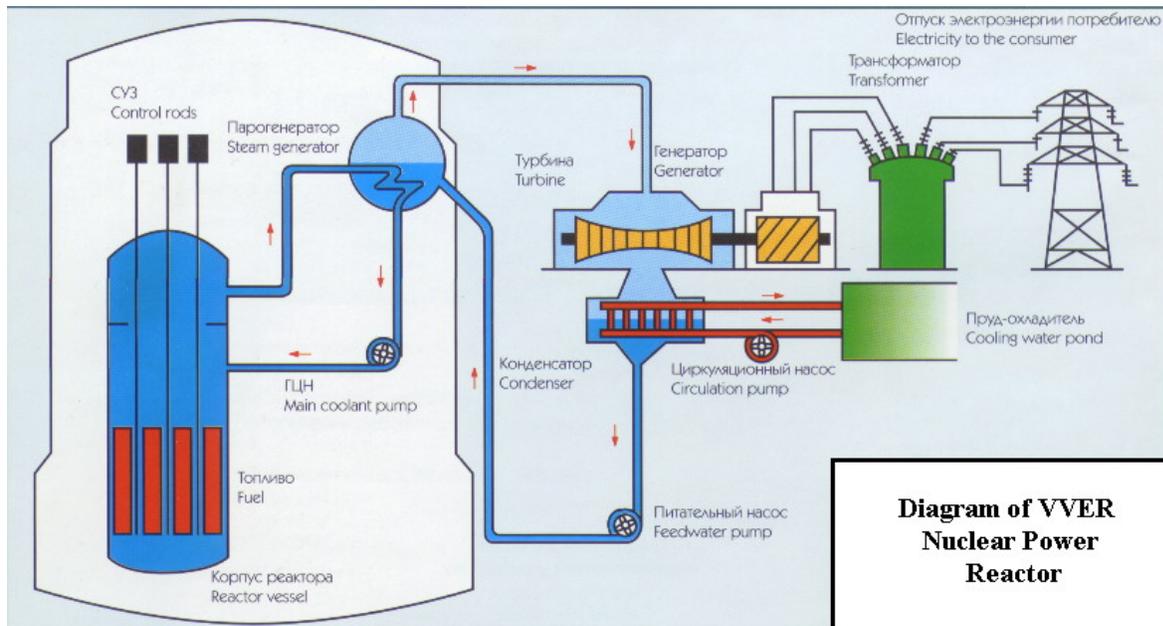
In pressurized water reactors the fuel is usually low (3 to 4 per cent) enriched uranium oxide, sometimes uranium and plutonium oxide mixture (MOX). In today's PWRs the primary pressure is usually 120 to 160 bars, while the outlet temperature of the coolant is 300-320 °C. The PWR is the most widespread reactor type in the world: they account for about 64 % of the total power of the presently operating nuclear power plants.

❖ **VVER**

The **VVER**, or **WWER** is a series of pressurised water reactor designs originally developed in the Soviet Union, and now Russia, by OKB Gidropress. Power output ranges from 300 MWe to 1700 MWe with the latest Russian development of the design. VVER power stations are used by Armenia, Bulgaria, China, Czech Republic, Finland, Hungary, India, Iran, Slovakia, Ukraine, and the Russian Federation.

The Russian abbreviation VVER stands for 'water-water energy reactor' (i.e. water-cooled water-moderated energy reactor). This describes the pressurised water reactor (PWR) design. The main distinguishing features of the VVER compared to other PWRs are:

- Horizontal steam generators
- Hexagonal fuel assemblies
- No bottom penetrations in the pressure vessel
- High-capacity pressurisers providing a large reactor coolant inventory



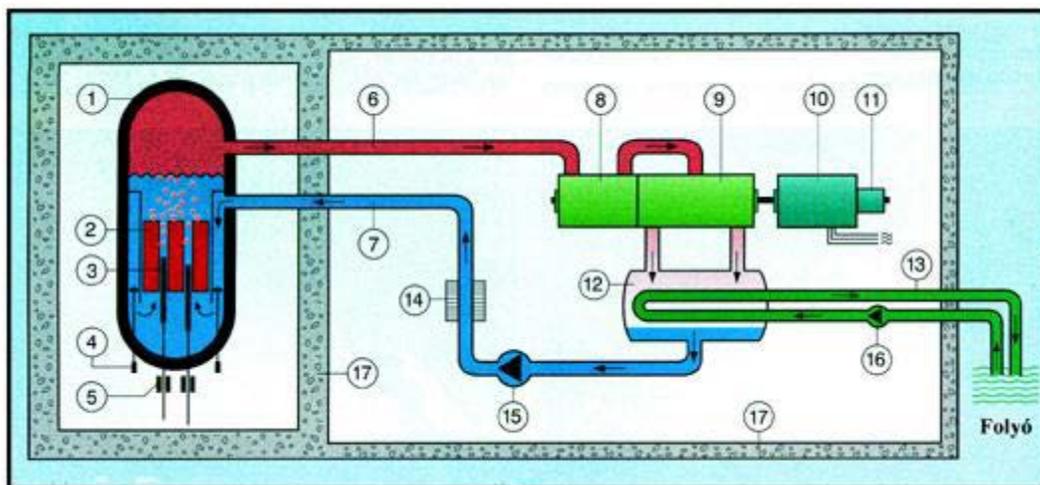
Reactor fuel rods are fully immersed in water kept at 15 MPa of pressure so that it does not boil at normal (220 to over 300 °C) operating temperatures. Water in the reactor serves both as a coolant and a moderator which is an important safety feature. Should coolant circulation fail the neutron moderation effect of the water diminishes, reducing reaction intensity and compensating for loss of cooling, a condition known as negative void coefficient. Later versions of the reactors are encased in massive steel pressure shells. Fuel is low enriched (ca. 2.4–4.4% ^{235}U) uranium dioxide (UO_2) or equivalent pressed into pellets and assembled into fuel rods.

Reactivity is controlled by control rods that can be inserted into the reactor from above. These rods are made from a neutron absorbing material and depending on

depth of insertion hinder the chain reaction. If there is an emergency, a reactor shutdown can be performed by full insertion of the control rods into the core.

2.Boiling Water Reactor (BWR)

In a boiling water reactor, light water (H₂O) plays the role of moderator and coolant, as well. Part of the water boils away in the reactor pressure vessel, thus a mixture of water and steam leaves the reactor core. The thus generated steam directly goes to the turbine, therefore steam and moisture must be separated (water drops in steam can damage the turbine blades). Steam leaving the turbine is condensed in the condenser and then fed back to the reactor after preheating. Water that has not evaporated in the reactor vessel accumulates at the bottom of the vessel and mixes with the pumped back feedwater.



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|---------------------------|-------------------------|-----------------------|
| 1 Reactor pressure vessel | 7 Feedwater | 13 Cooling water |
| 2 Fuel rods | 8 High pressure turbine | 14 Preheater |
| 3 Control rod | 9 Low pressure turbine | 15 Feedwater pump |
| 4 Circulating pump | 10 Generator | 16 Cooling water pump |
| 5 Control rod drive | 11 Exciter | 17 Concrete shield |
| 6 Fresh steam | 12 Condenser | |

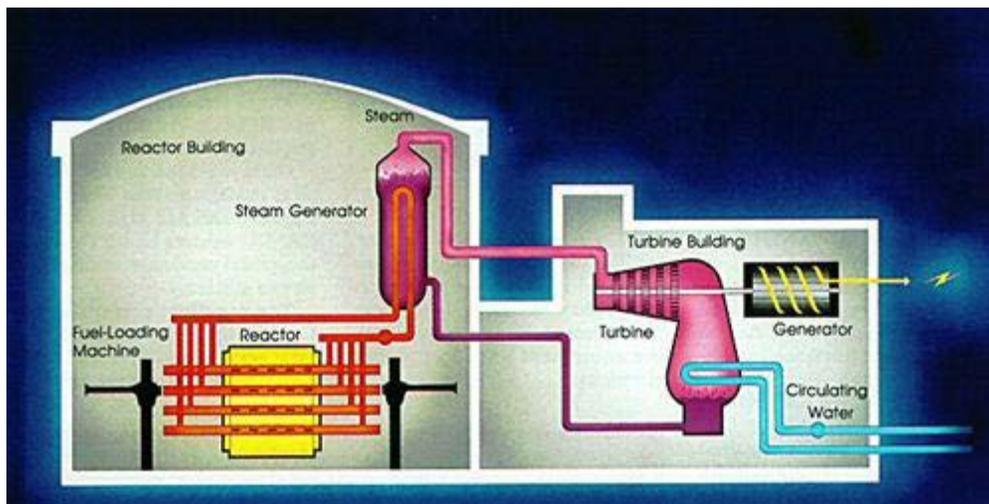
Since boiling in the reactor is allowed, the pressure is lower than that of the PWRs: it is about 60 to 70 bars. The fuel is usually uranium dioxide. Enrichment of the fresh fuel is normally somewhat lower than that in a PWR. The advantage of this

type is that - since this type has the simplest construction - the building costs are comparatively low.

20% of the total power of presently operating nuclear power plants is provided by BWRs.

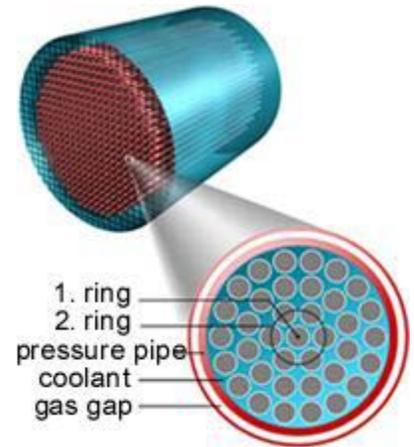
3.Heavy Water Reactor (HWR)

In heavy water reactors both the moderator and coolant are heavy water (D₂O). 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 % to 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.



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; its 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.

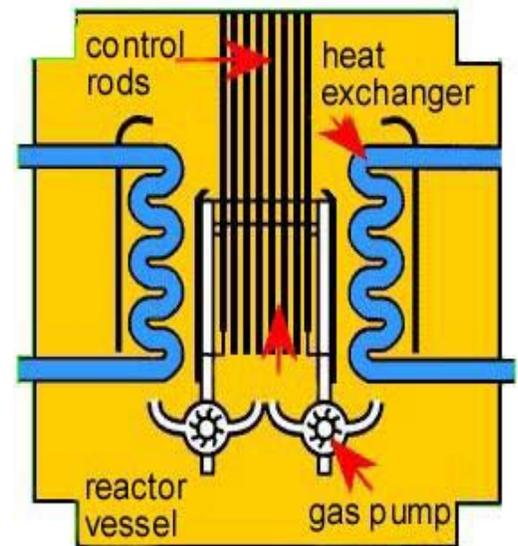


The other type of heavy water reactor is the pressurized heavy water reactor (PHWR). In this type the moderator and coolant are the same and the reactor pressure vessel has to stand the high pressure.

Heavy water reactor is the third common type, with 48 number in use for power generation and 7% of the total power of presently operating nuclear power plants is provided by HWRs; however 10 % 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.

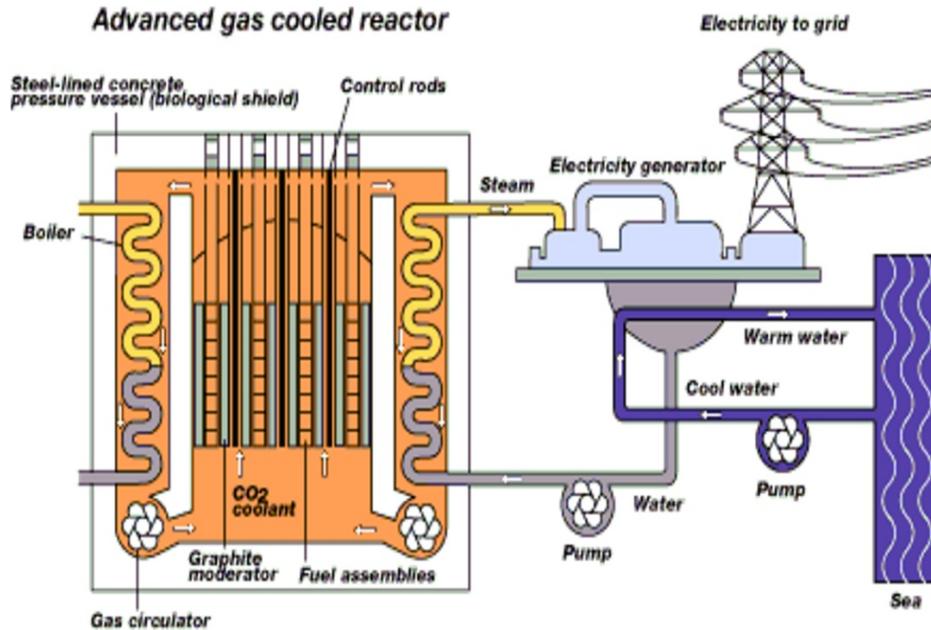
4. Gas Cooled Reactors (GCR)

The gas-graphite reactors operate using graphite as moderator and some gas (mostly CO₂, 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.



Advanced Gas-cooled Reactors (AGR) are the second generation of British gas-cooled reactors, using graphite moderator and carbon dioxide as primary coolant. The fuel is uranium oxide pellets, enriched to 2.5-3.5%, in stainless steel tubes. The carbon dioxide circulates through the core, reaching 650°C and then past steam generator tubes outside it, but still inside the concrete and steel pressure vessel (hence 'integral' design). Control rods penetrate the moderator and a secondary shutdown system involves injecting nitrogen to the coolant.

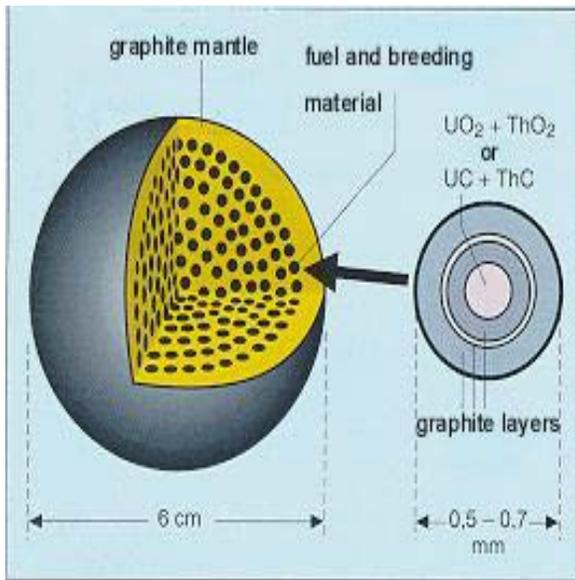
The AGR was developed from the Magnox reactor, also graphite moderated and CO₂ cooled, and one of these is still operating in UK to late 2014. They use natural uranium fuel in metal form. Secondary coolant is water.



The newest gas cooled reactor type is the HTGR (High Temperature Gas cooled Reactor), 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 %.

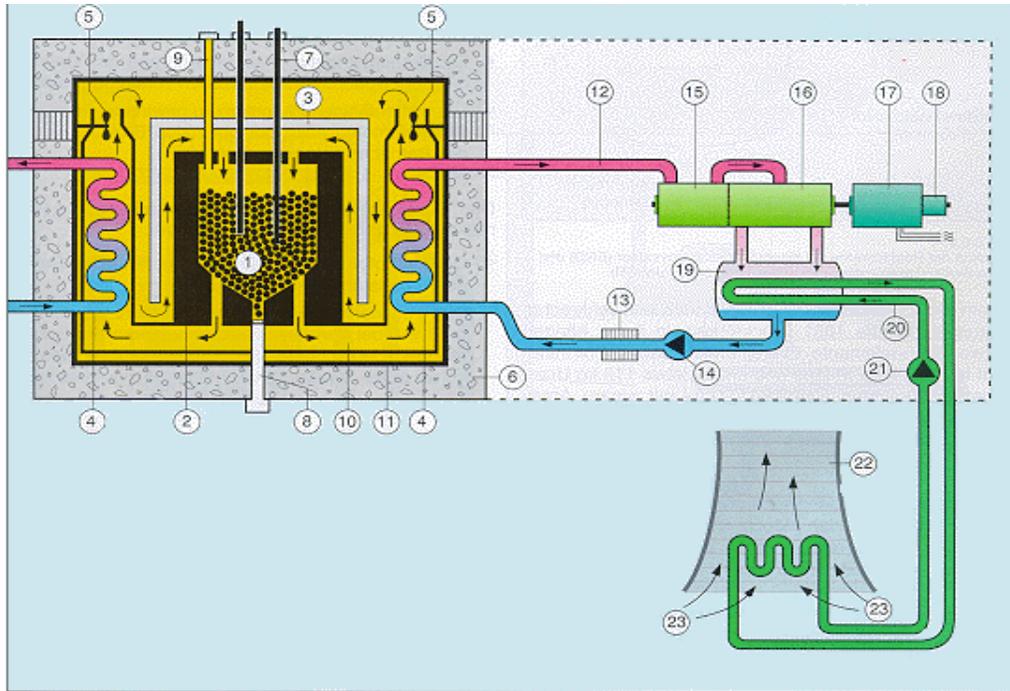
5. Thorium high temperature reactor (THTR)

The high temperature thorium fuelled reactor is a special type of the gas cooled reactor. So far only one of this type has operated, between 1985 and 1989 in Germany. The thermal power of the reactor was 760 MW while the electrical power was 307 MW, with an efficiency of 40.5%. (This is very high taking into account the light water moderated reactors' 32 to 33 per cent efficiency.) The other advantage of this type is outstanding safety.



The fuel elements of THTR-300 were balls of 6 cm diameter, in each of which 35,000 smaller balls (diameter between 0.5 and 0.7 mm) could be found. Each small ball contained 1 g of U-235 and 10 g of Th-232, as breeder material. There are 360,000 such balls in the reactor. The moderator was graphite, layered on the U-Th balls, and a further 280,000 balls made from pure graphite. Upon neutron capture, U-233 is produced from Th-232, which is fissionable for slow neutrons. Correspondingly, during its operation the reactor itself produced part of its fuel.

The heat produced in the reactor was conveyed using helium, which entered the reactor at the top at 250 °C and exited at the bottom at 750 °C (the name is related to this high temperature, which is responsible for the high efficiency). The helium gave its heat to a water-steam loop in six heat-exchangers (only two are visible in the figure). In order to control and shut down the reactor, 51 control rods could be inserted between the balls from the top.



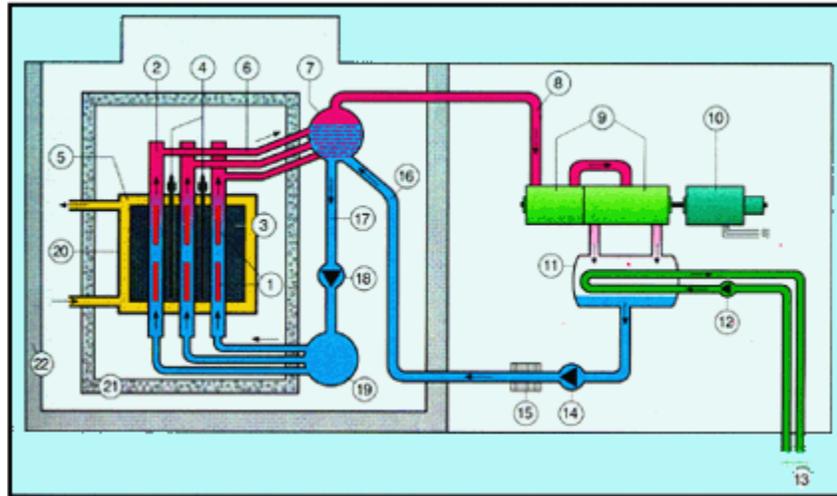
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| 1 Reactor core | 9 Inlet of balls | 17 Generator |
| 2 Graphite neutron reflector | 10 He coolant gas | 18 Exciter |
| 3 Iron shield | 11 Gas tight steel shell | 19 Condenser |
| 4 Steam generator | 12 Fresh steam | 20 Cooling water |
| 5 Gas circulator | 13 Preheater | 21 Cooling water pump |
| 6 Prestressed concrete container | 14 Feedwater pump | 22 Dry cooling tower |
| 7 Control rods | 15 High pressure turbine | 23 Air flow |
| 8 Outlet of balls | 16 Low pressure turbine | |

The refuelling machine built into the reactor vessel made it possible to replace the spent fuel balls with new ones during operation. In THTR-300 approximately 620 balls were replaced by fresh ones every day; the balls spent 3 years in the reactor and during this period they passed through the core six times.

6.RBMK

The RBMK is a unique reactor type: its moderator is graphite (in this respect it resembles the AGRs), the coolant is boiling light water (as in the case of BWRs), moreover it has pressure tubes (like the CANDUs). The world's first nuclear power

plant reactor was an RBMK (Soviet Union Obninsk, 1954). To this type belongs the Chernobyl reactor unit that suffered an accident on 26 April 1986.



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| 1 Uranium fuel | 9 Turbine | 16 Feedwater |
| 2 Pressure tube | 10 Generator | 17 Water counterflow |
| 3 Graphite moderator | 11 Condenser | 18 Circulating pump |
| 4 Control rod | 12 Cooling water pump | 19 Water dispenser tank |
| 5 Protective gas | 13 Heat transport | 20 Steel casing |
| 6 Water/steam | 14 Feedwater pump | 21 Concrete shield |
| 7 Moisture separator | 15 Preheater | 22 Reactor building |
| 8 Steam to turbines | | |

The reactor core consists of graphite blocks, between them vertically stand the pressure tubes. These embody the fuel assemblies and the in between flowing coolant. A mixture of water and steam leaves the core (hence the reactor is a boiling water type), which goes to the moisture separator. The separated steam goes to the turbine and then, after condensing and preheating, back to the reactor.

RBMK reactors only operate in a few successor states of the former Soviet Union. The type's share in total NPP capacity of the world is 1-2 %. These reactors have a lot of technical and economical advantages, but they have big security risk as well.

The type has two major advantages. One is the enormous power that can be achieved: since the pressurized coolant flows in tubes and thus a pressure vessel is not needed, moreover practically any number of pressure tubes can be built into a reactor, there is no theoretical limit of the power that an RBMK can produce. (The power of the Chernobyl reactors was 1,000 MW electrical, but a 2,000 MW type was also designed.) The other advantage of the type over other light water reactors is that there is no need for outages for refuelling; the fuel assemblies can be replaced during operation (as with CANDUs).

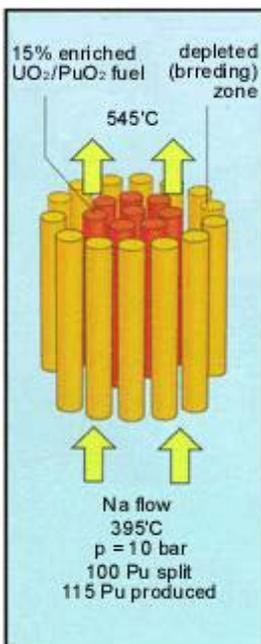
We have to emphasize one of the disadvantages of this type: the core is very large and therefore control is very difficult. In Chernobyl, for example, there were 200 control rods (as a comparison, in Paks there are only 37). There is another factor, however, that played an even more significant role in the Chernobyl accident. The so-called void reactivity coefficient can be positive in certain cases, which means that under extreme circumstances coolant boiling away may result in a rise in reactivity. This is a positive feedback and is caused by the fact that the coolant, which is light water absorbs more neutrons than the moderator, which is graphite. When water (which can be considered accordingly as a neutron absorber) boils away, its density decreases and thus the number of neutrons will increase. (In the Paks VVER reactor, as well as in other PWRs, the void coefficient is always negative, since the moderator is the same as the coolant. In a potential case of boiling, the number of hydrogen nuclei, which slow down neutrons, decreases and therefore less neutrons will be able to induce fission again. Eventually, the result of the process is that the chain reaction stops.) With many of the RBMK reactors it has been achieved that the void coefficient has become practically zero and in this way their safety has been enhanced.

Although in the 1950s graphite moderated and light water cooled reactors were used in the USA for plutonium production, this type could not spread in power

plants there because the Americans realized its disadvantages. Plutonium was never produced at Chernobyl, because the Pu made by an RBKM is useless for military purposes.

7. Fast breeder reactors

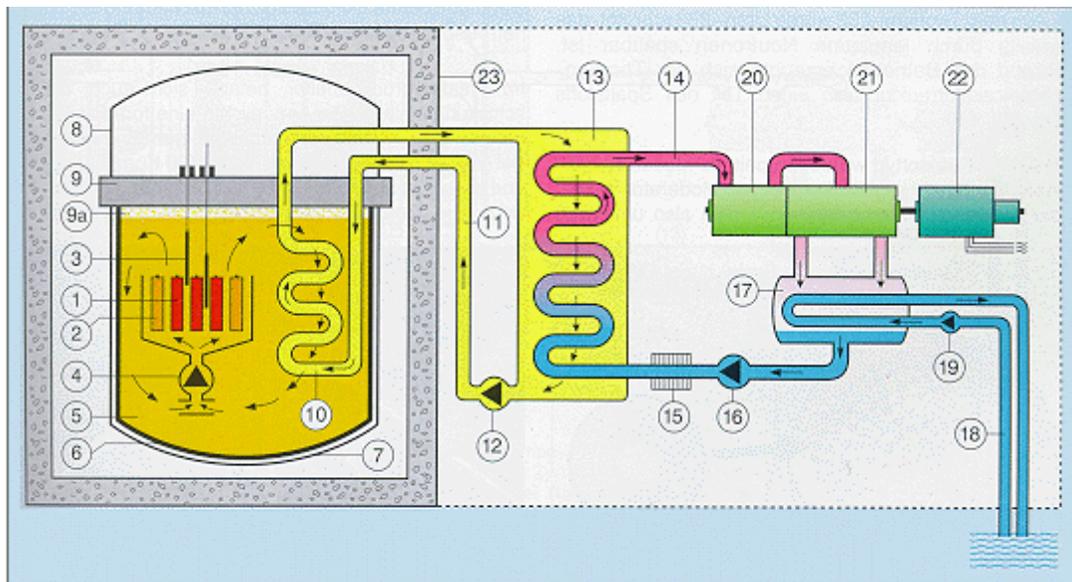
In PWRs and BWRs, a vast majority of the fission reactions occur in U-235, which makes up for only 0.7 % of natural uranium and during fuel fabrication for these reactors it is enriched to a few percent. Accordingly, in the already mentioned reactor types (sometimes referred to as thermal reactors) U-238 is hardly applied as fissionable material. However, upon capturing a neutron, the nucleus of U-238 can transform into Pu-239 (via radioactive decay), which is a fissile material. For Pu-239, fission can also be induced using fast neutrons. The fast breeder reactors use both processes. The largest nuclear power plant with a fast breeder reactor is the Superphenix in France, which started operation in 1986. Its thermal power is 3000 MW, while the electrical power is 1180 MW (this corresponds to an efficiency of 39%). Fast breeder reactors have a share of less than 1% of the total power of the world's NPPs.



The core of a fast breeder reactor consists of two parts. The fuel rods, which contain a mixture of uranium dioxide and plutonium dioxide, are found in the inner part. Here fission reactions dominate, while in the outer part the predominant process is conversion of U-238 to Pu-239. This part contains depleted uranium (i.e. uranium in which the U-235 content is even lower than the natural 0.7%). In such a reactor one can achieve a situation where more fissile plutonium nuclei are produced in a unit time than the number of fissile nuclei which undergo fission (hence the name "breeder"). On the other hand, neutrons are not thermalized, since fast neutrons are needed for the above described processes.

In the French Phénix breeder reactor it was determined that for 100 fission reactions there are 115 newly produced fissile nuclei. Correspondingly, more fissile material is produced than used, and this can later be used in other thermal (such as light water moderated) or breeder reactors.

Obviously, in a fast reactor there must not be any moderator, which implies that water is not at all suitable as coolant. Instead some liquid metal, usually sodium is applied. In the Superphénix, sodium enters the core at 395°C and exits at 545°C. Since the boiling point of sodium is very high even at comparatively low pressures (at 10 bars about 900°C), there is no need to maintain a high pressure in the primary circuit and thus the construction and manufacturing of the reactor vessel is easier.



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|---------------------------|-------------------------|--------------------------|
| 1 Fuel (fissile material) | 9 Cover | 17 Condenser |
| 2 Fuel (breeder material) | 10 Na/Na heat exchanger | 18 Cooling water |
| 3 Control rods | 11 Secondary Na | 19 Cooling water pump |
| 4 Primary Na pump | 12 Secondary Na pump | 20 High pressure turbine |
| 5 Primary Na coolant | 13 Steam generator | 21 Low pressure turbine |
| 6 Reactor vessel | 14 Fresh steam | 22 Generator |
| 7 Protective vessel | 15 Feedwater pre-heater | 23 Reactor building |
| 8 Reactor cover | 16 Feedwater pump | |

The heat of primary sodium is transferred to the secondary sodium in an intermediate heat exchanger, while the third heat exchanger is the steam generator. Application of three loops is necessitated by safety considerations (liquid sodium is very dangerous: the primary sodium is highly radioactive because of neutrons activation, which results in Na-24; the second sodium loop prevents radioactive sodium from accidental contact with water).

Floating nuclear power plants:

Apart from over 200 nuclear reactors powering various kinds of ships, Rosatom in Russia has set up a subsidiary to supply floating nuclear power plants ranging in size from 70 to 600 MWe. These will be mounted in pairs on a large barge, which will be permanently moored where it is needed to supply power and possibly some desalination to a shore settlement or industrial complex. The first has two 40 MWe reactors based on those in icebreakers and will operate at a remote site in Siberia. Electricity cost is expected to be much lower than from present alternatives.

The Russian KLT-40S is a reactor well proven in icebreakers and now proposed for wider use in desalination and, on barges, for remote area power supply. Here a 150 MWt unit produces 35 MWe (gross) as well as up to 35 MW of heat for desalination or district heating. These are designed to run 3-4 years between refuelling and it is envisaged that they will be operated in pairs to allow for outages, with on-board refuelling capability and used fuel storage. At the end of a 12-year operating cycle the whole plant is taken to a central facility for 2-year overhaul and removal of used fuel, before being returned to service. Two units will be mounted on a 21,000 tonne barge. A larger Russian factory-built and barge-mounted reactor is the VBER-150, of 350 MW thermal, 110 MWe. The larger VBER-300 PWR is a 325 MWe unit, originally envisaged in pairs as a floating

nuclear power plant, displacing 49,000 tonnes. As a cogeneration plant it is rated at 200 MWe and 1900 GJ/hr.

Advanced reactors:

Several generations of reactors are commonly distinguished. Generation I reactors were developed in 1950-60s and only one is still running today. They mostly used natural uranium fuel and used graphite as moderator. Generation II reactors are typified by the present US fleet and most in operation elsewhere. They typically use enriched uranium fuel and are mostly cooled and moderated by water. Generation III are the Advanced Reactors evolved from these, the first few of which are in operation in Japan and others are under construction and ready to be ordered. They are developments of the second generation with enhanced safety. There is no clear distinction Gen II to Gen III.

Generation IV designs are still on the drawing board and will not be operational before 2020 at the earliest, probably later. They will tend to have closed fuel cycles and burn the long-lived actinides now forming part of spent fuel, so that fission products are the only high-level waste. Of seven designs under development, 4 or 5 will be fast neutron reactors. Four will use fluoride or liquid metal coolants, hence operate at low pressure. Two will be gas-cooled. Most will run at much higher temperatures than today's water-cooled reactors.

More than a dozen (Generation III) advanced reactor designs are in various stages of development. Some are evolutionary from the PWR, BWR and CANDU designs above, some are more radical departures. The former include the Advanced Boiling Water Reactor, a few of which are now operating with others under construction. The best-known radical new design has the fuel as large 'pebbles' and uses helium as coolant, at very high temperature, possibly to drive a turbine directly.

Considering the closed fuel cycle, Generation 1-3 reactors recycle plutonium (and possibly uranium), while Generation IV are expected to have full actinide recycle.

Some Design Descriptions:

[ABWR](#): The U.S. Advanced Boiling Water Reactor design uses a single-cycle, forced circulation, reactor with a rated power of 1,300 megawatts electric (MWe). The design incorporates features of the BWR designs in Europe, Japan, and the United States, and uses improved electronics, computer, turbine, and fuel technology.

The design is expected to increase plant availability, operating capacity, safety, and reliability. Improvements include the use of internal recirculation pumps, control rod drives that can be controlled by a screw mechanism rather than a step process, microprocessor-based digital control and logic systems, and digital safety systems. It also includes safety enhancements such as protection against over pressurizing the containment, passive core debris flooding capability, an independent water makeup system, three emergency diesels, and a combustion turbine as an alternate power source.

[ACR700](#): The ACR-700 ® is an evolutionary, Generation III+, 750 MWe class pressurized tube reactor, designed to meet industry and public expectations for safe, reliable, environmentally friendly, low-cost nuclear generation. The ACR-700 is designed for a 2016 in-service date, and is currently undergoing a pre-licensing review in Canada.

[ACR1000](#): The ACR-1000 ® is an evolutionary, Generation III+, 1200 MWe class pressurized tube reactor, designed to meet industry and public expectations for safe, reliable, environmentally friendly, low-cost nuclear generation. The ACR-

1000 is designed for a 2016 in-service date, and is currently undergoing a pre-licensing review in Canada.

[AP600](#): This is a 600 MWe advanced pressurized water reactor that incorporates passive safety systems and simplified system designs. The passive systems use natural driving forces without active pumps, diesels, and other support systems after actuation. Use of redundant, non-safety-related, active equipment and systems minimizes unnecessary use of safety-related systems.

[AP1000](#): This is a larger version of the previously approved AP600 design. It is a 1,000 MWe advanced pressurized water reactor that incorporates passive safety systems and simplified system designs. It is similar to the AP600 design but uses a longer reactor vessel to accommodate longer fuel, and also includes larger steam generators and a larger pressurizer.

[EPR](#) : The EPR is a large pressurized water reactor of evolutionary design, with design output of approximately 1,600 MWe. Design features include four 100% capacity trains of engineered safety features, a double-walled containment, and a "core catcher" for containment and cooling of core materials for severe accidents resulting in reactor vessel failure. The design does not rely on passive safety features. The first EPR is currently being constructed at the Olkiluoto site in Finland. Framatome also hopes to build EPR's at the Flammanville site in France, and has submitted a bid for EPR construction in China.

[ESBWR](#): The Economic and Simplified Boiling Water Reactor (ESBWR) is a 1,390 MWe, natural circulation boiling water reactor that incorporates passive safety features. This design is based on its predecessor, the 670 MWe Simplified BWR (SBWR) and also utilises features of the certified Advanced Boiling Water

Reactor (ABWR). Natural circulation was enhanced in the ESBWR by using a taller vessel, a shorter core, and by reducing the flow restrictions. The ESBWR design utilizes the isolation condenser system for high-pressure water level control and decay heat removal during isolated conditions. After the automatic depressurization system operates, low-pressure water level control is provided by the gravity-driven cooling system. Containment cooling is provided by the passive containment cooling system.

[IRIS](#): The International Reactor Innovative and Secure is a pressurized light water cooled, medium-power 335MWe reactor that has been under development for several years by an international consortium. IRIS is a pressurized water reactor that utilizes an integral reactor coolant system layout. The IRIS reactor vessel houses not only the nuclear fuel and control rods, but also all the major reactor coolant systems components including pumps, steam generators, pressurizer and neutron reflector. The IRIS integral vessel is larger than a traditional PWR pressure vessel, but the size of the IRIS containment is a fraction of the size of corresponding loop reactors.

[PBMR](#): The Pebble Bed Modular Reactor is a modular HTGR that uses helium as its coolant. PBMR design consists of eight reactor modules, 165 MWe per module, with capacity to store 10 years of spent fuel in the plant (there is additional storage capability in onsite concrete silos). The PBMR core is based on the German high-temperature gas-cooled reactor technology and uses spherical graphite elements containing ceramic-coated fuel particles.

[System 80+](#): This standard plant design uses a 1,300 MWe pressurized water reactor. It is based upon evolutionary improvements to the standard CE System 80 nuclear steam supply system and a balance-of-plant design developed by Duke

Power Co. The System 80+ design has safety systems that provide emergency core cooling, feedwater and decay heat removal. The new design also has a safety depressurization system for the reactor, a combustion turbine as an alternate AC power source, and an in-containment refuelling water storage tank to enhance the safety and reliability of the reactor system.