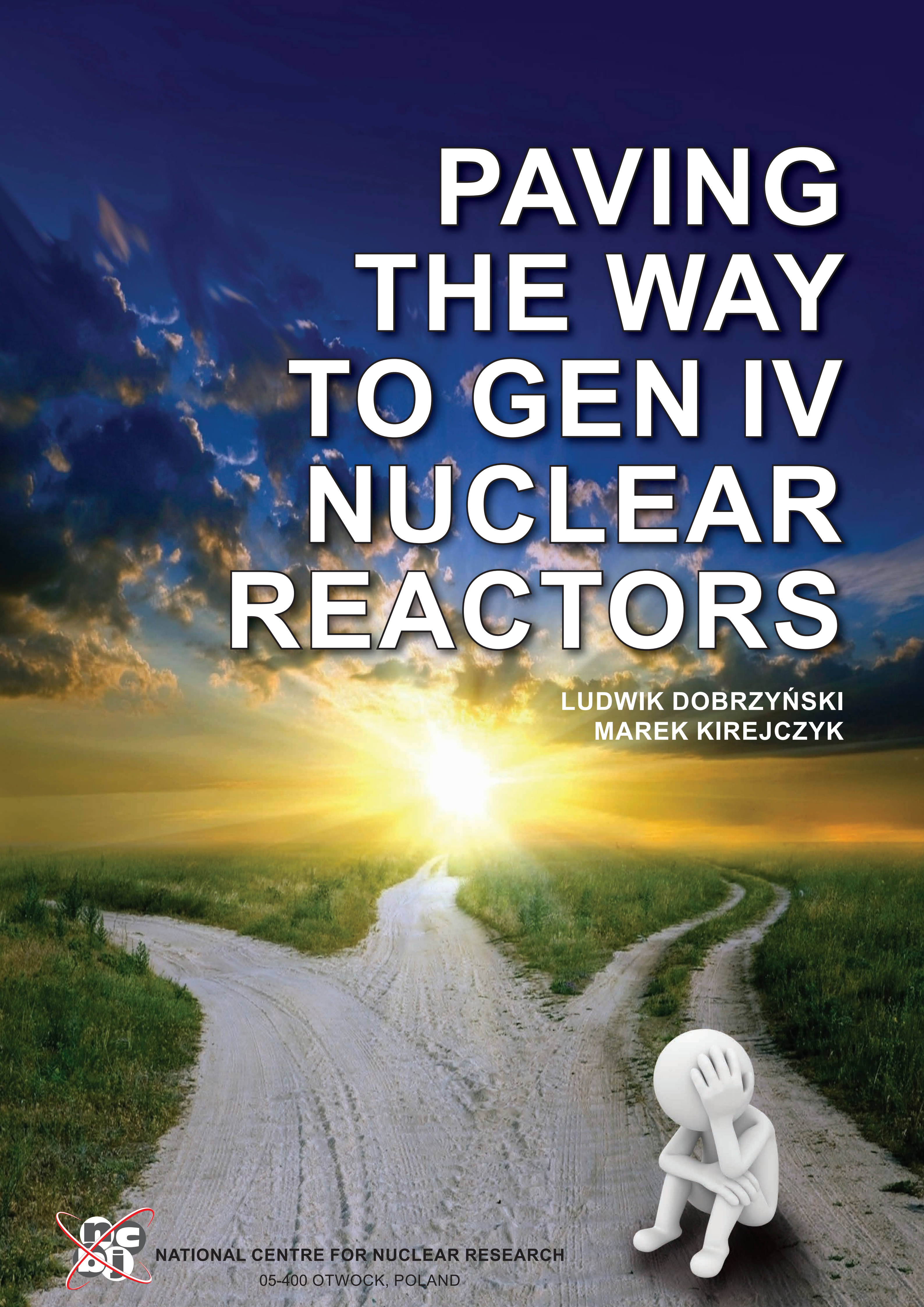

PAVING THE WAY TO GEN IV NUCLEAR REACTORS

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TABLE OF CONTENTS

1. Foreword

The problem of the future of nuclear power is still debated. Serious accidents in Chernobyl (1986) and Fukushima (2011) stopped development of the technology in many OECD countries even if stress-tests carried out soon after the Fukushima-Daiichi accident proved generally high safety of nuclear reactors. Unfortunately, the decisions about nuclear power are often dictated by political pressure rather than analysis of strengths and weaknesses of the nuclear power technology and possible improvements, if necessary.

On the other hand a number of new reactors in nuclear power plants (NPPs) have been put into operation in numerous other countries, most notably China, Korea, Russia and India. In Europe, new projects in France and Finland will hopefully be finished soon. There are also plans to build or modernize NPPs in Lithuania, UK, Hungary, Romania and Poland. So there is no doubt that in recent years we have been witnessing both setbacks and rapid development of new NPPs. The development of safety systems in "conventional" reactors has led to new builds of Generation III and III+. Developed in parallel Generation IV reactors will be able not only to deliver electric energy to power grids, but also high temperature steam useful in various chemical processes (like hydrogen production or desalination of seawater). Besides, they will use nuclear fuel more efficiently and reduce the volume of nuclear waste since they may be fueled by fuel earlier burnt in conventional reactors. It is important to note that the plan of "burning" nuclear waste has its direct impact on reducing the risk of nuclear weapons proliferation. To achieve these goals a strong international collaboration is needed. Generation IV International Forum, a platform for such cooperation, was established in 2000 under auspices of Euratom.

This brochure attempts to describe basics of operation of NPPs, to outline development of nuclear power technologies from the now obsolete Gen-I reactors to future Gen-IV ones, and to express our belief in the need to promote nuclear power even if economic terms are currently unfavorable. Authors of this text have drawn heavily on two brochures prepared earlier in NCBJ mainly for secondary school teachers: "Nuclear Power. The first encounter" by L. Dobrzyński and K. Żuchowicz (2013), and "Nuclear Power. The second encounter" by L. Dobrzyński, K. Samul and K. Różycki (2014). We are grateful for authors' permission to use fragments of their texts. We are also indepted to our VINCO colleagues for their remarks and helpful discussion. Special thanks must go to Kajetan Różycki for corrections and inspiring comments.

2. On electric power needs and possible solutions

It is a well-known fact that our civilization is powered by huge amounts of energy and constant growth is demanding ever larger amounts of energy. Most probably traditional energy sources will become exhausted, or their price will skyrocket, in some not-so-distant future. Current estimates are that the time left is between 50 and 150 years – but even if they are too pessimistic, sooner or later mankind will be in need of energy from some alternate sources. Certainly saving the energy is something we have to do, however, it will not stop the growing demand for energy. All available socio-economic data show that GNP is positively correlated with both energy production output and amount of consumed electric energy. The data indicate also that life expectancy increases with energy consumption.

So far, majority of energy consumed in the world is produced by combustion of either biomass (mainly timber) or fossil fuel (coal, oil, natural gas). The so-called "green" sustainable sources (windmills, solar panels, hydro etc.) do not seem to be an option due to their intermittent availability (except hydropower which, however, is not at all available in many countries, in others is used close to the limits; some countries don't even consider it "green"). This makes the production of baseload electricity necessary. Nuclear power could satisfy mankind's energy hunger for a very long time (up to a million years). Compared to other industrial power sources nuclear power is environmentally friendly and affordable to electricity consumers. Experience accumulated over about 60 year-long history of NPP operation and technology development is vast and modern NPPs are extremely safe facilities. Demand for energy – a driving force of investments in nuclear power – has been recently growing, especially in Asia. Therefore it is not surprising that majority of new reactors currently under construction are located in Korea, China, India and Vietnam.

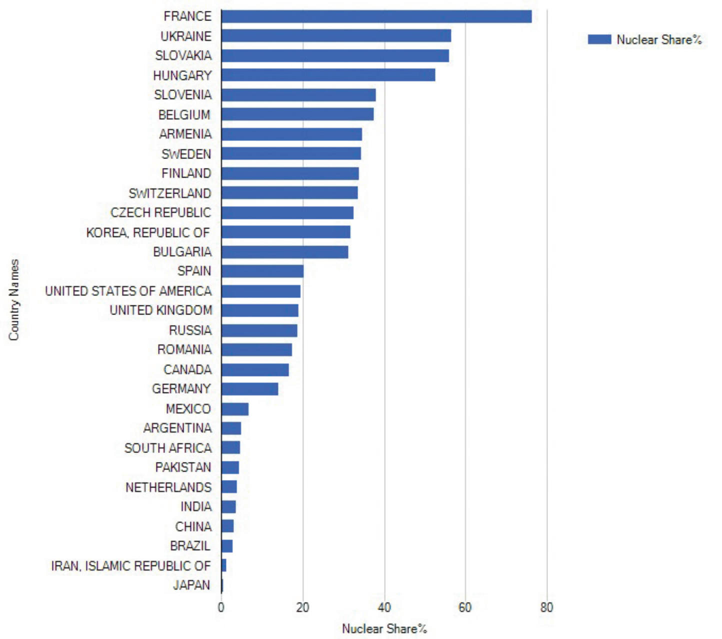
Mining/drilling and bulk transport (indispensable to supply classical power industry with fossil fuels) are quite risky operations. Let's just mention serious accidents in coal mines that occurred in 2010 in China and Ukraine, or pollution of Mexican Gulf waters with crude oil flowing out of the damaged "Deep Horizon" BP rig for three months in 2010. Even "green" energy has some environmental problems: photovoltaic panels are quite polluting in their front-end production, while windmills disturb the landscape and pose some threats to birds and bats. Both sources supply energy intermittently, which means that some big/expensive energy storage systems are necessary and the facility peak power must be 4-5 times larger than its required average power. Facilities of both types occupy rather large areas of land. There is one more factor: typical lifetime of a solar panel/wind turbine/conventional coal-burning plant/nuclear reactor is about 20/25/40/60 years, respectively.

One may argue that people will sooner or later invent some efficient and easily accessible sources of cheap energy, so why to bother to invest now? Such discoveries are difficult to imagine right now, but who knows the future… However, in no case we can expect that these hypothetical sources become any commercially viable alternative in a period shorter than about 50 years after their discovery. That period is comparable to the time in which currently identified fossil fuel resources will start to run low. So, we have to make some important decisions in that matter, and to make them soon. Nuclear energy is a very efficient source of power, already practically available. True, it requires huge financial investments. However, there is no other more promising energy source for the near future. Even if investment outlays necessary to develop a NPP are very high, cost of electricity produced in a NPP (to be borne by consumers) turns out to be relatively low, in fact it is among the lowest costs in the whole power industry. Estimates of gross costs of electricity produced in various type power plants are shown in Table 1. NPP-related estimates include costs of necessary safeguards, systems to protect fissile materials against uncontrolled spreading, radioactive waste management, and total decommissioning of the plant down to the so-called "green grass" level after its lifetime is over.

Table 1. Comparison of costs of electric energy and drawbacks of various type power plants (after the "Electricity produced in nuclear/coal-fired/gas-fired power plants and from renewable sources: cost comparison" report by Agencja Rynku Energii S.A., December 2009). Notes: (i) 1 PLN = 0.20-0.25 EUR; (ii) The estimates strongly depend on assumptions concerning cost of capital/CO_2 emissions, technology developments, and (in some cases) fuel prices. As such, they should be treated only as some indicators of general trends.

Plant type	Cost of 1 kWh (PLN)	Draw-backs
Hard coal-fired w/ system to remove SO_x/NO_x from flue gases	0.36	Air pollution
Hard coal-fired w/ system to remove SO_x/NO_x from flue gases and a system to remove and store CO_2	0.36	Large quantity of ash
Brown coal-fired w/ system to remove SO_x/NO_x from flue gases	0.36	Air pollution
Brown coal-fired w/ system to remove SO_x/NO_x from flue gases and a system to remove and store CO_2	0.34	Large quantity of ash
Nuclear power plant (NPP) equipped with Gen-III PWR reactors	0.29	Radioactive waste
Natural-gas-fired	0.37	Uncertain fuel cost
Fired by gas from an integrated hard coal gasification facility	0.40	Air pollution
Fired by gas from an integrated hard coal gasification facility, equipped with a system to remove and store CO_2	0.34	
Fired by gas from an integrated brown coal gasification facility	0.40	Air pollution
Fired by gas from an integrated brown coal gasification facility, equipped with a system to remove and store CO_2	0.32	
Land-based wind generators	0.43	Costly back up system necessary
Sea-based wind generators	0.44	Costly back up system necessary

Currently about 11% of electric energy produced in the world is supplied by NPPs[1]. Data on nuclear power in individual countries is shown in Fig.1[2] .



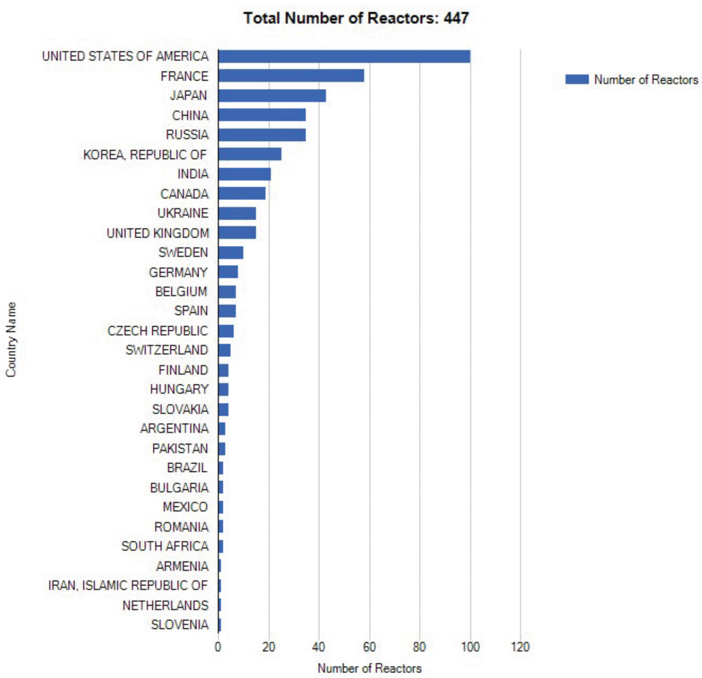


Fig.1 Nuclear power share in supplies and number of reactors in various countries

Nuclear power saves environment from pollution since neither flue gases nor carbon dioxide are produced in NPPs. Therefore it does not contribute to the so-called global warming effect. Each 22 tons of uranium "burned" in nuclear reactors prevent emission of about one million (sic!) tons of carbon dioxide that would accompany combustion of coal if an equivalent amount of electricity was produced in classical power plants.

Emerging market countries do not disregard opportunities brought about by nuclear power. Programmes to develop nuclear power industry are most impressive in countries in which shortages of power and smog caused by coal burning are most acute e.g. in China and India.

Some countries like Italy, Germany, Japan and Switzerland have revised their approach to nuclear power after the Fukushima accident in March 2011. Italians shut their NPPs down in 1990 and voted (2011 referendum) not to build new ones. Germans intend to decommission their NPPs before 2022, and Swiss – before 2034. Japan government has decided to keep the nuclear option open and to resume operation of Japanese NPPs shut down after the Fukushima accident; however, this process is very slow as of 2016. On the other hand governments of as many as 60 countries asked in 2011 International Atomic Energy Agency in Vienna to consult their programmes to develop new NPPs. While USA have not initiated construction of new NPPs since the Three Mile Island accident in 1979, the US Nuclear Regulatory Commission issued in 2011 several new licenses for reactor construction. 9 new reactors are to be connected to the power grid in 2016-2017 in China, Republic of Korea, Pakistan, Russia, Slovakia and United Arab Emirates.

China is now the leader in nuclear power development. There were 14/26/28 operational/under construction/ planned power reactors in China in March 2012, while the respective figures for January 2016 were 35/20/35. Chinese government plans to cover about 10% of the country's demand for electricity in 2025 by NPP-produced supplies.

According to World Nuclear Association, 442 power reactors in the world were in December 2015 operational, 63 were under construction and 158 were planned (approval granted and/or funds committed by the developer). As one can see, nuclear power industry is currently being rather revived in spite of negative decisions made in the above mentioned countries and in spite of rather low prices of oil, natural gas and coal. Since some currently operated reactors will have to be decommissioned within the coming decades, total output of nuclear power industry will not grow as much as the number of new reactors might suggest. Also, nuclear share in electricity supplies has been globally declining for the last 5 years, in part because quite a large number of dirty but rather straightforward to develop coal-burning power plants have recently been built in China. But even if the nuclear power industry is not in its best days, the news about its death are definitely premature. Even Japanese have not completely abandoned their involvement in nuclear power industry after the Fukushima accident.

Why the industry seems to be revived just five years after the accident? The answer is quite simple: it's just a matter of costs. Nuclear power is just competitively cheap and does not emit either carbon dioxide or ashes into the environment. One should also note that nuclear power is really compact. Fuel requirements (per year) of a 1,000 MWe power plant and an area of land necessary to provide the power are shown in Table 2 for several technologies.

[1]Source: Nuclear Energy Institute, http://www.nei.org/Knowledge-Center/Nuclear-Statistics/World-Statistics
[2]Source: PRIS, http://pris.iaea.org/PRIS, July 2016

Table 2. Fuel (per year of operation) / land area needs of a 1,000 MWe power plant[3]

Energy source	"Fuel" needs	For comparison
Biomass	2,000 km^2 of energy crop	3 times surface of Lake Geneva
Wind	2,700 windmills each of 1.5 MW (25% capacity factor)	That would occupy 486 km^2
Photovoltaics	23 km^2 of solar panels on the equator[4]	2,555 soccer fields
Biogas	20 million pigs	
Natural gas	1.2 km^3	47 Cheops pyramids
Oil	1.4 million tons	10 million barrels/ 100 supertankers
Coal	2.5 million tons	26,260 train cars (2 trains/d)
Nuclear fission[5]	20 tons of UO_2	Air pollution
Nuclear fusion	100 kg D and 150 kg T	2,850 m^3 of sea water and 10 tons of Lithium ore

There are still hopes that thermonuclear (fusion) technology will become feasible within a few coming decades. However, even if that technology is very efficient in the fuel heating value sense, its practical utilization (commercial viability) is a matter of a rather distant future.

3. World resources of nuclear fuel and fuel independence

These days nuclear fuel is mainly produced from uranium ores mined from deposits in Earth crest. How long the reserves might supply the world nuclear power industry? Contrary to all appearances the answer is not straightforward at all. Firstly, one has to assume some method of uranium utilization; the current technology is the most obvious choice. Secondly, available reserves depend on some acceptable price of uranium ore; as the reserves will be running low, the price will undoubtedly be rising. At current (mid 2016) uranium prices around 60 USD/kg of U_3O_8 ("yellow cake"), uranium reserves will get depleted in less than 100 years at the current consumption rate. It might seem a pretty gloomy picture, but it's not so bad. As uranium prices rise, several factors come into play:

(i) Prices rise dramatically increases amount of economically minable ores.

(ii) World will be switching to other currently not cost-effective types of nuclear reactors, first of all to the so-called breeders and/or thorium-based "fast" reactors (discussed later in this text). As global reserves of Th are much richer than global reserves of U, future nuclear power technologies that would allow to fully utilize both resources might be capable to satisfy global demand for energy even for a million years.

(iii) Even current technologies allow to extract uranium from seawater at a cost of about 500 USD/kg. Uranium in seawater is in dynamic equilibrium with uranium in under-water crust, so it is replenished if extracted and that way is practically inexhaustible[6]. The 500/60 price ratio seems drastic, but cost of nuclear fuel contributes quite little to price of electricity produced in NPPs. On the contrary, cost of the fuel is the most significant economic factor in fossil fuel fired power plants.

The fuel independence issue is also often raised. Again, nuclear power is favored here, even if no high quality uranium ores are deposited in Central Europe. Uranium is offered for sale by vendors from many countries. Taking into account only uranium extractable at a cost below 130 USD/kg, deposits identified on territories of the 4 largest potentates (Australia/Kazakhstan/Canada/Russia) amount to 1,700/650/485/480 thousand tons, respectively. Global reserves amount to about 5.4 million tons. About 35 million tons are qualified as minable at higher prices. Potentially the richest deposits of uranium in the world were found in Sweden at the end of July 2010. Since import is possible from numerous countries, there is no hazard of becoming dependent on any single supplier.

4. Conventional nuclear power reactors

In principle, nuclear power reactors substitute coal, oil or gas burners in conventional power plants. Fig.2 shows layout of a typical NPP based on the workhorse among various types of nuclear reactors i.e. on Pressurized Water Reactor (PWR). Heat produced in uranium fission reactions is taken away by water circulating the primary cooling loop. Inside reactor core, the water is heated up to a temperature of about 330°C. To prevent boiling, relatively high pressure (about 150 times atmospheric pressure) is maintained inside the loop. The pressure is stabilized by means of a pressure stabilizer with a gas cushion. In heat exchanger (steam generator) the primary loop water heats water circulating the secondary loop (maintained at a much lower pressure) up to a temperature sufficiently high to convert it into high pressure steam. The steam drives a turbine coupled with electricity generator. Used (depressurized) steam is condensed in a condenser and the resulting water is pumped again into the heat exchanger. The steam condensation process is assisted by cold water drawn from a reservoir situated outside the plant, such as a lake, a river, or a sea.

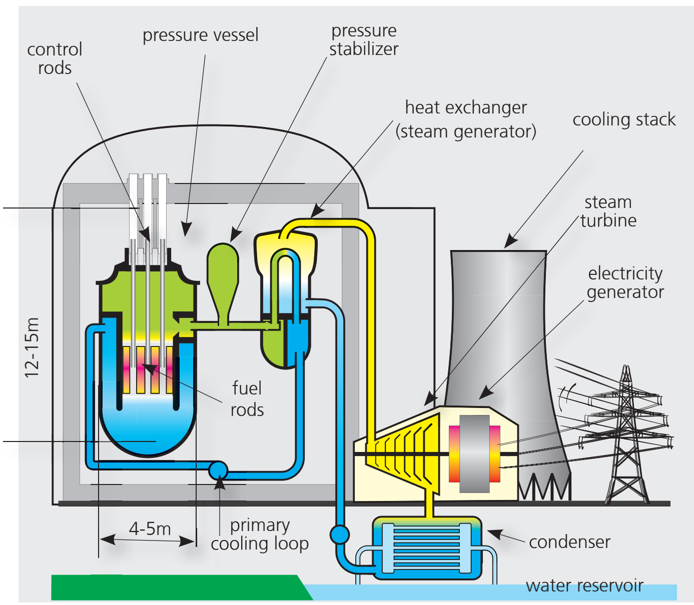


Fig. 2 Layout of a pressurized water reactor NPP

[3]Main source: M.-T. Westra, S. Kuyvenhoven: "Energy, Powering Your World", FOM - Institute for Plasma Physics Rijnhuizen 2002, 2005, available e.g. from http://fire.pppl.gov/energy_overview_EFDA.pdf
[4]In central Europe necessary area would be roughly 8 times larger
[5]For Generation III reactors
[6]http://www.forbes.com/sites/jamesconca/2016/07/01/uranium-seawater-extraction-makes-nuclear-power-completely-renewable/; http://pubs.acs.org/toc/iecred/55/15#UraniuminSeawater

PWR is the most popular reactor type currently used in NPPs. Boiling Water Reactor (BWR) is the second most popular type, see Fig.3. Fission reactions in both reactors depend on relatively slow neutrons, so fast neutrons produced by the reactions must be slowed down by collisions with nuclei of some moderating medium. Water, heavy water, and graphite are efficient neutron moderators. In both PWRs and BWRs water plays double role: a coolant and a moderator.

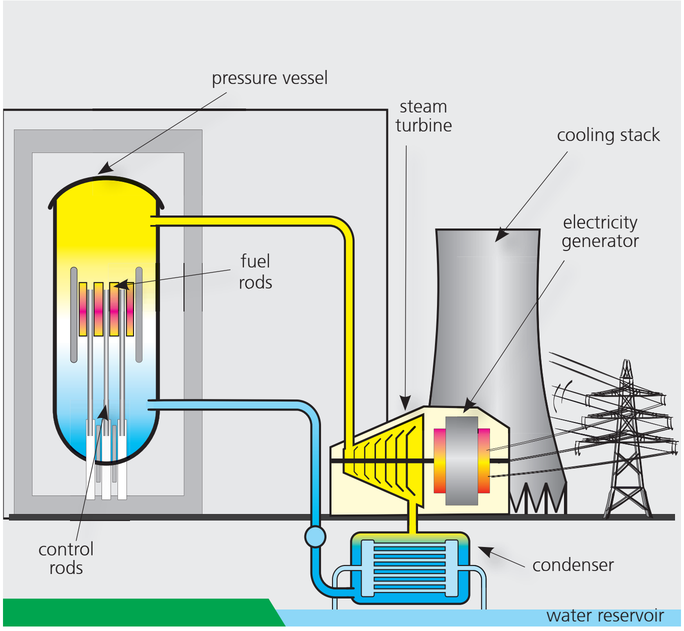


Fig. 3 Layout of a Boiling Water Reactor NPP

BWR reactor directly produces steam necessary to drive the turbine. The steam collects at the top of the reactor pressure vessel. Since the chain reaction runs predominantly at the bottom part of the BWR core where there is enough water to efficiently slow down (moderate) fast neutrons, control rods must be inserted from below also (they would not survive long if operated in hot & wet steam environment and they might collide with steam drier). Before entering the turbine, the hot and wet steam (about 76 times atmospheric pressure, 285°C) is dried. BWRs are simpler to build than PWRs (single cooling loop rather than two loops), but require some shielding of their turbines because turbine working medium is contaminated with short-lived activation products from the reactor core (mainly ^{16}N that decays in seconds, so the turbine chamber may be entered already about 2 minutes after turbine shut-down).

Other reactor types in use include gas-cooled reactors, liquid-metal-cooled reactors, or reactors employing heavy water as the moderator (e.g. Canadian CANDU). However, light-water moderated/cooled reactors clearly dominate the nuclear power market: about 2/3 of electric power delivered in 2011 by NPPs all over the world (370 GWe) came from PWR reactors, and about 20% from BWR ones.

Regardless of the type, the single most important issue that must be scrutinized in detail by reactor designers is reactor safety. From the very beginning, nuclear industry was imposing highly restrictive measures concerning safety, therefore modern reactors are impressively safe. A very small number of serious accidents with nuclear reactors that occurred during all 60 years of their operation all over the world confirms that latter statement. Nevertheless, the issue is still on the table, and a number of even more modern solutions are still introduced or proposed. All introduced improvements have collectively resulted in a safety level not found in other industry branches. Probability of core meltdown in a typical Gen-II reactor (dominant today) is on the order of 10^{-5}/year. Given roughly 400 reactors in operation today that latter figure translates into 1 expected meltdown in 25 years of operation – quite rare, but still not acceptably rare accidents. However, in some modern reactors (e.g. AP1000 made by Westinghouse) meltdown probability is 20 times smaller, on the order of $5x10^{-7}$/year i.e. 1 expected meltdown in 500 years of operation.

The other problem which needs a better and a more socially acceptable solution *is what to do with used nuclear fuel (also called "spent fuel")*? Although used fuel is for layman just another term for highly dangerous waste, it is not so for professionals who know that nuclei of used nuclear fuel are still binding a lot of energy. Why not to liberate that energy through recycling of fissile materials and to use it for the good of people? The used fuel should not therefore be treated as a waste but rather as a future energy source.

Reactor operation, including the issues mentioned above, will be discussed in more detail in the next chapter.

5. Basics on operation and safety of nuclear reactors

5.1. Reactor core

Nuclear reactor may be described as a device built to run a controlled chain reaction in some fissile material (usually uranium ^{235}U). Fission is a process in which free neutron hits some fissionable nucleus causing it to split into (typically) two fragments plus a number of new-generation neutrons (2.5 on the average in case of ^{235}U); the split is accompanied by liberation of some pretty high amount of energy. Each of the new-generation neutrons can hit another fissionable nucleus causing a new fission, and so on – that explains the chain reaction term. Chain reactions occurring in atomic bombs are uncontrolled: fission rates grow exponentially and a huge amount of energy is liberated in a fraction of a second that way giving rise to a nuclear explosion. Fission rates and other parameters of a running chain reaction depend on probability that a liberated neutron hits ^{235}U, and that the hit nucleus will split. That in turn depends on the uranium load geometry (size and shape) and on energy of the hitting neutrons (i.e. on moderator type and geometry). For example, no chain reaction can sustain if a relatively small amount of metallic uranium is shaped as an ideal sphere since majority of neutrons escape from the sphere. Such state is referred to as *sub-critical*. The larger the sphere, the lower the fraction of escaping neutrons. For a sufficiently large radius the chain reaction becomes self-supporting or critical. Such mass of a fissile material is referred to as critical mass. To be critical without any moderator, a ^{235}U sphere must contain about 50 kg of uranium (sphere diameter of about 17 cm). The larger mass, the faster the chain reaction runs (the *super-critical* mode).

To sum up: chain reactions may run in three distinct modes:

- sub-critical: majority of liberated neutrons fail to split other nuclei, the number of neutrons is dropping with time, the reaction is being extinguished (or it does not start)

- critical: exactly one neutron out of all fission-liberated ones splits another nuclei, the number of neutrons is more or less constant with time, the reaction is running steadily; and
- super-critical: more than one fission-liberated neutron splits other nuclei, the number of neutrons is increasing with time eventually leading to explosion. Such a state may be created when two sub-critical masses of fissionable uranium are suddenly joined into a larger super-critical mass. If separated, each of the masses would be back in the sub-critical state.

Each nuclear reactor must provide some measures to prevent uncontrolled chain reaction. Usually this is achieved in two ways. First, uranium fuel is hermetically sealed in fuel elements, the quantity in each element being much smaller than the critical mass. Each fuel element contains much more ^{238}U isotope than ^{235}U. The former isotope absorbs neutrons preventing any chain reaction inside the element. Second, fuel elements inside reactor are separated not only by distance, but also by control rods made of a material strongly absorbing neutrons, e.g. boron carbide or cadmium (boron and cadmium nuclei absorb neutrons very strongly). Unless control rods are fully lifted up, they maintain the neutron population at some desirably low level. Grid of equidistant fuel elements and control rods is the heart of any nuclear reactor referred to as reactor core.

Each nuclear reactor is normally operated in the *critical state*. In such conditions the number of neutrons is more or less constant in time, and the chain reaction is running steadily at some adjusted power level (is stationary). The critical state may be relatively easily maintained thanks to the so-called delayed neutrons produced in decay of some fission products. About 0.65% of neutrons from fissions of ^{235}U nuclei are delayed by more than 5 s. The delay may reach even 1 minute, while the average value is several seconds. Presence of *delayed neutrons* makes the mechanical control rod management a doable task. If there were no such neutrons, control rod would have to react to fluctuations of instantaneous number of neutrons with the time constant on the order of 1/1000 second. There are no equally fast mechanical systems.

Typical nuclear reactor is built in such a way that it is sub-critical without delayed neutrons (more neutrons absorbed than produced in fissions). ^{235}U nuclei are most readily split by thermal neutrons of a kinetic energy comparable to energy of thermal vibrations at room temperature (a fraction of one electron volt). However, *fission neutrons* have energies on the order of one million electron volts. Probability that such highly energetic neutrons will initiate next fissions is small. To use as much neutrons as possible we have to deprive fission neutrons of most of their energy, which explains the need for neutron moderators. However, there are reactors where fast neutrons are used. They will be discussed in more detail in the chapter on Gen-IV reactors.

5.2. Reactor safety systems

Hazards related to operation of nuclear facilities have been analyzed with utmost care since the time first such facilities appeared. Steps taken to protect personnel and population against consequences of possible failures come from the requirement that risks of running NPPs must not be higher than risks associated with other electric power generation technologies. Sixty years of practice have also enabled us to acquire vast experience in all matters related to radioactive waste management. General safety principles that must be observed during development and operation of any nuclear facility include:

- Design of each individual facility must guarantee its reliable, continuous and easy operation, in which an overriding rule is "safety first". All NPP employees are taught that safety is more important than any electricity production schedule.
- Design must follow the *defense in depth* principle: multitude of protective levels, multiple barriers preventing release of radioactive materials. Probability of each failure (or combinations of failures) that could give rise to any serious consequences must be reasonably minimized. Since such failures cannot be ruled out, one must take measures to reduce consequences of the failure.
- Technical solutions that are not verified, either practically in previously operated facilities, or experimentally, must not be used.
- Design and operational instructions of each individual facility must take into account the possibility of human error at every stage of operation of the facility.
- Design must keep exposition of staff to ionizing radiation and risk of releasing radioactive materials into the environment as low as reasonably possible.

There are multiple safety barriers built into nuclear reactors. Reactor designers strictly adhere to the safety system *redundancy principle* – safety systems are multiple, with each system or subsystem based (if possible) on different physical law/principle (such as gravitation, convection, pressure difference etc.), or different power supply, so that no single failure could make them all simultaneously inoperative.

Usually control and safety systems require some external power (e.g. coolant pumps must be supplied with electric power). However, some systems do not need this, they work thanks to the laws of nature: gravity can pull safety rod down if released, fluid can flow from a higher to a lower pressure etc. Safety systems based on physical phenomena and not requiring any external power supply are known as *passive safety systems*. In most of the currently developed reactors of Generation III or III+ safety systems are not only redundant, but in part also passive, and are therefore extremely reliable: calculated probability of reactor core meltdown is smaller than once per one hundred thousand years of operation. No other industry meets so stringent safety requirements. Poor reactor safety is not an issue in the state-of-the art constructions.

In that context the question of the catastrophic Chernobyl accident that occurred in 1986 may naturally be asked. Not going into intricate details, it must be pointed out that the RBMK-type reactors were designed with military applications in mind (although the one deployed in Chernobyl was not used for such purposes, as far as we know). Their construction would not be approved as safe to operate (even at that time) in any other country than Soviet Union. The Chernobyl accident resulted also from numerous mistakes made by operators of the reactor. One should note that Ukrainian and Lithuanian reactors of that type have been afterwards decommissioned, but 15 reactors operating on similar principles, with some of their safety features corrected, are still operated in Russia.

Rate of work-related accidents may be used as a measure of day-to-day overall safety of nuclear power industry. In USA the rate of serious accidents (ones that force some limitation of access or force the worker to change occupation altogether) is much lower in nuclear power industry than in classical power industry, not to mention construction industry notoriously hazardous for workers.

Vulnerability of NPPs to terrorist attacks (e.g. consequences of an airplane strike into reactor building) is another often raised issue. However, present day safety standards applied to reactor building walls sufficiently protect such facilities. It was experimentally verified: a decommissioned real airplane was allowed to strike a mock-up of modern reactor building. Damage to the building was insignificant, while the airplane got totally disintegrated. Alike, no other terrorist attack can seriously put any reactor in jeopardy since nuclear facilities are designed and constructed with exceptional care regarding physical security.

5.3. Passive elements of reactor safety systems

Currently developed reactors are equipped with both passive and active safety systems and are therefore extremely reliable. As we have already said, reactor safety is based on multitude of barriers. For instance let us examine the structural elements which prevent the release of fission products outside the reactor room. The four major barriers are:

- fuel element matrix (which directly entraps uranium fission products)
- fuel element cladding
- walls of primary cooling circuit (reactor vessel, pressurizer, cooling loop piping, heat exchanger etc.)
- reactor safety containment.

Those barriers are obviously passive. In the rest of this chapter we will discuss other passive systems - the ones that are in place in case of a failure. Passive systems are driven by simple physical forces (such as gravitation or convection) even in absence of external power and without operator intervention.

The first action during each reactor start-up is to pull emergency rods up and to drive them outside the reactor core. The rods are hanged under electromagnets. In case of power loss or breaking the safety circuit attractive forces of electromagnets disappear, the rods gravitationally fall down into their positions among fuel rods and automatically extinguish the chain reaction. Gravitation is passive element of the safety system.

Perhaps the most serious hazard in any nuclear reactor is loss of cooling. Heat is produced even if no chain reaction is running, as even partly used nuclear fuel contains a lot of radioactive material. Therefore in the absence of cooling the reactor core may melt down. Reactors must be ready for such failures. Typical solution is to pump emergency cooling water from a system of multiple emergency reservoirs. Normally the pumps need electric power and it may fail, therefore there must be additional solution in place. For example hydro-accumulators may be located in the vicinity and above the reactor core (Fig.4) and be connected with the reactor vessel by a short tubing equipped with a check valve. During normal plant operation compressed nitrogen pumped to the reactor vessel maintains pressure p_0 inside the vessel higher than pressure p_1 exerted on the check valve by mass of water in the hydro-accumulator, so the valve is closed. However, as soon as the p_0 pressure drops, the valve opens enabling the

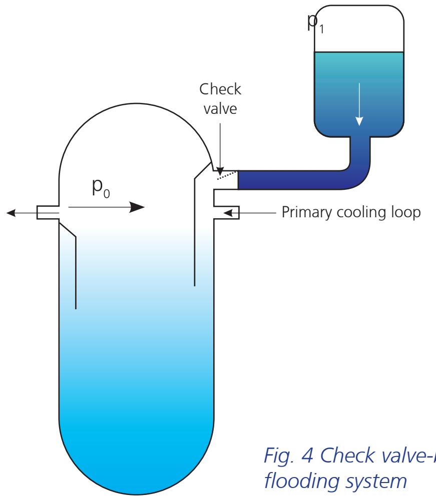


Fig. 4 Check valve-based passive core flooding system

water to flood the core until p_1 drops below the check valve threshold. Safety depends here on static pressure difference, core flooding is triggered without any operator intervention and may proceed without any external power source.

Of course no hydro-accumulator is inexhaustible. Nevertheless, should the primary loop be broken, such hydro-accumulator can provide some time needed to start up other (active) systems capable to take over the core cooling function before core melts down. In emergencies electric power should be supplied to pumps by some Diesel generators.

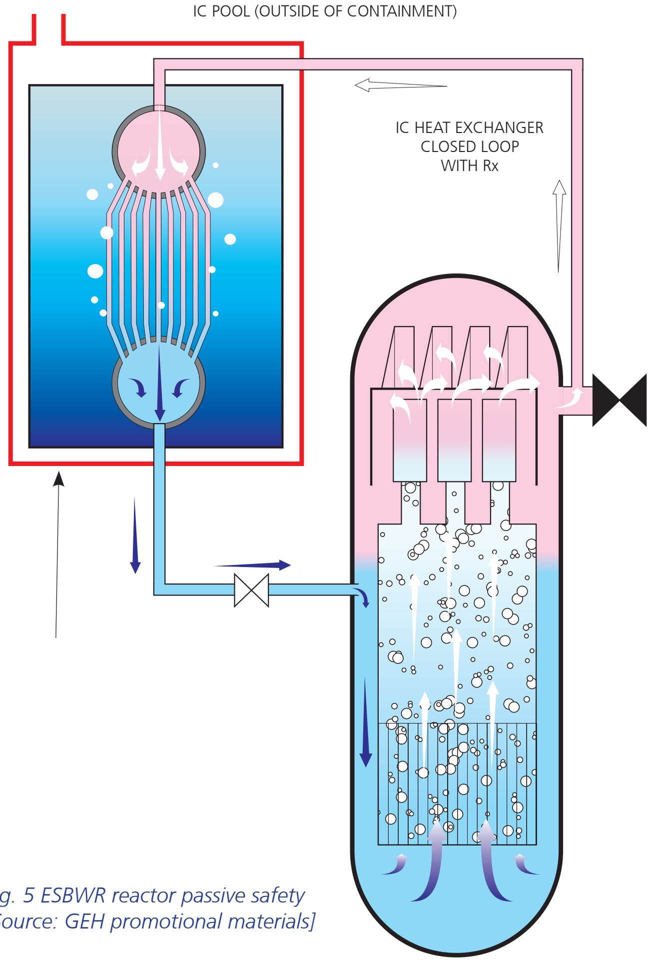


Fig. 5 ESBWR reactor passive safety [Source: GEH promotional materials]

Another example of passive safety element is shown in Fig.5. Circulation of water (hence cooling) is guaranteed even in absence of power in pumps by different density of hot water inside reactor vessel and colder water inside external tank with heat exchanger (depicted IC POOL in the Figure) i.e. by convection. In emergencies valve on hot water pipeline to the pumps (depicted ▶◀) closes and heat generated inside the reactor core is carried away by water driven by convection forces to a heat exchanger situated above the core.

If cooling is lost, overheated water vapour may react with zirconium cladding to produce zirconium oxide and hydrogen. Hydrogen mixed with air explodes. This happened in Chernobyl and Fukushima. In the latter case some of the generators were flooded and some Diesel fuel tanks were flushed to the ocean by the tsunami wave. The reactors did lose their cooling, which resulted in hydrogen explosion. All European reactors built in XXI century are provided with passive hydrogen *recombiners* that keep hydrogen cocentration below flammability limits, thus preventing hydrogen explosion.

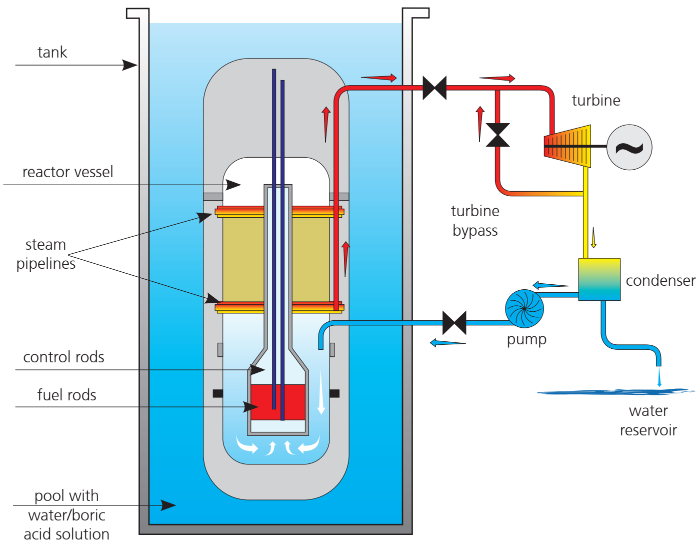


Fig. 6 The PIUS concept (after Wikipedia Commons)

PIUS (Process Inherent Ultimate Safety) concept is depicted in Fig. 6. The reactor can be immersed in an external pool filled up with solution of boric acid in water. The solution does not mix with the cooling water unless the core becomes overheated in emergencies, when the solution automatically floods the core. Water cools the core down, while boron atoms (which strongly absorb neutrons) stop the chain reaction. PIUS was developed as an extremely safe solution which could be placed in populated areas. Unfortunately, no reactor was ever built according to that concept.

Finally let us mention a simple solution designed to eliminate overpressure in emergencies: a cooling tower (to the right in Fig.7). It's role is similar to the role of safety containment. Such a cooling tower was designed for the never built Żarnowiec nuclear power plant planned in Poland. In emergencies steam pressure may suddenly soar; such overpressure would be however quickly eliminated because overheated steam would pass through a series of special water tanks stacked into a tower. Passing through cold water steam would condense, hence its pressure would drop.

Fig. 7 Model of the never built Żarnowiec NPP. The plant was to be the first NPP ever built in Poland, however the project was abandoned in 1990. Bubble condenser tower visible to the right was to protect the plant against sudden increase of steam pressure in emergencies

6. Reactor generations

Constructions of nuclear reactors are by convention classified into a few "generations", usually as follows.

First commercial reactors built in 1950s and 60s made up the first generation (Gen-I in short). Examples include CO_2 cooled Magnox reactors built in the UK, and the first PWR and BWR reactors built in the US. This early generation was composed of a real multitude of types and models, out of which majority turned out unsatisfactory and were eventually abandoned (reactors with organic moderators, graphite-sodium reactors to name a few). On the other hand, the Calder Hall plant operated in UK between 1956 and 2001 is an example of a very successful Gen-I construction. A single reactor/power generation unit of those times could deliver 50-200 MWe[7].

Second generation reactors appeared in late 1960s. The many types of the Gen-I gave way to just a few constructions present in Gen-II: PWR (and VVER Soviet counterpart)[8], BWR[9], PHWR[10] a.k.a. CANDU[11], RBMK[12], and AGR[13]. Gen-II reactors are still being built in some countries, in particular in China. The power of a single reactor/power generation unit can reach 1,300 MWe, however typical range is 900 - 1,100 MWe.

The accident in the Three Mile Island plant (1979) was an event that ended the era of Gen-II. The lessons learnt on that occasion motivated nuclear agencies in many countries to toughen up the regulations. The new requirements demanded Gen-III reactors to have much lower probability of serious accidents, and reactor buildings to be specially designed to cope with such emergencies. It is not an easy task to meet such demands, and raised requirements coincided with general slowdown in nuclear business worldwide.

[7]MWe = megawatt of electrical power, as opposed to MWt or MWth = megawatt of thermal power
[8]PWR = Pressurized Water Reactor, WWER or VVER = Water-Water Power Reactor (Vodo-Vodyanoy Enegetichesky Reactor)
[9]BWR = Boiling Water Reactor
[10]PHWR = Pressurized Heavy Water Reactor
[11]CANDU = Canadian Deuterium Uranium; the only commercial application of PHWR type
[12]RBMK = Large Power Channel Reactor (Reaktor Bolshoy Moshchnosti Kanalniy)
[13]AGR = Advanced Gas Reactor

As a result the number of vendors capable (and willing) to deliver Gen-III reactors has dropped down to just a few in the world, while reactor/power plant costs have soared. The remaining reactor types are Advanced PWR, BWR and PHWR. Some manufacturers claim their reactors belong to an upgraded 3+ (III+) generation, but the criteria accepted in the US and in Europe for reactors to be classified as III+ are different and the whole thing seems to be a marketing catch. Generation III reactor-based nuclear power plants are currently under construction in several countries in the world. Besides, a few ABWR[14] boiling water reactors classified as generation III have been operated in Japan for almost twenty years.

Future technologies are rated as Gen-IV. Reactors of that generation will be designed using radically different technologies and radically different approaches to safety issues. With exception of the Russian BN-800 breeder no reactor of that generation is so far (2016) operational. The list of expected improvements is quite long:

- radically decreased amount of produced nuclear waste
- at least partially closed fuel cycle (waste recycling)
- power generation efficiency 40-50% (currently about 35%)
- no fission material produced within the reactor core should have any military application
- increased safety level.

Specific designs belonging to Generation IV are discussed in chapter 9.

Fig. 8 shows time evolution of reactor generations. In each subsequent generation the safety is better than in the previous one. Technical solutions that have not proved their merits in practice are eliminated.

Majority of reactors operated these days belong to the Gen-II, while reactors under construction belong to Gen-II, Gen-III and Gen-III+.

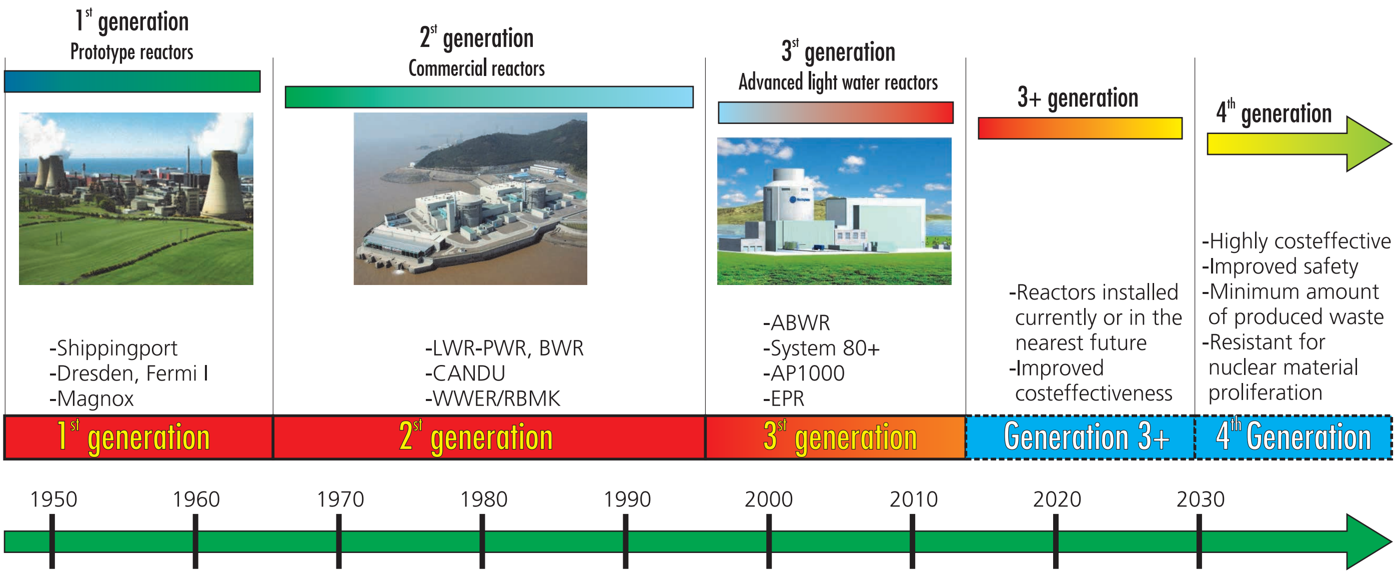


Fig. 8 Time evolution of reactor generations

7. The spent fuel problem

7.1. Spent fuel and nuclear waste

Contrary to coal/liquid fuel/gas, nuclear fuel never gets completely burned because: (i) every reactor needs some minimum concentration of fissile nuclei in its core, and (ii) the fuel gets "poisoned" with time since reactors produce during their operation also a couple of isotopes that strongly absorb neutrons (^{135}Xe is a typical example). Spent fuel elements have been so far treated as nuclear (or radioactive) waste dangerously radioactive for thousands of years, see Fig.9. That waste, an issue specific for nuclear power, is one of the major arguments against the strategy to promote that technology.

Spent fuel – composed mainly (94…95%) of uranium – comprises various radioactive products of the fission reaction. Uranium isotopes emit relatively weakly penetrating alpha particles, but many of the fission products emit much more penetrating beta and gamma rays. Although not so long-lived as uranium (^{238}U: $T_{1/2}$=4.5 billion years, ^{235}U: $T_{1/2}$=0.7 billion years), some fission products also live long enough to be a real nuisance for nuclear power developers. Long-lived isotopes found in spent fuel are listed in Table 3.

Table. 3 Major long-lived components of spent nuclear fuel and some of their properties

Some long-lived isotopes found in spent fuel		
Isotope	$T_{1/2}$ (years)	Radiotoxicity (Sv/kg)*
^{99}Tc	$2.1\cdot10^5$	$4.9\cdot10^2$
^{129}I	$1.6\cdot10^7$	$0.7\cdot10^3$
^{135}Cs	$2.3\cdot10^6$	$0.8\cdot10^2$
^{237}Np	$1.1\cdot10^6$	$0.3\cdot10^4$
^{238}Pu	88	$1.4\cdot10^8$
^{239}Pu	$2.4\cdot10^4$	$0.6\cdot10^6$
^{240}Pu	$6.6\cdot10^3$	$2.1\cdot10^6$
^{242}Pu	$3.7\cdot10^5$	$0.4\cdot10^5$
^{241}Am	432	$0.3\cdot10^8$
^{243}Am	$7.4\cdot10^3$	$1.5\cdot10^6$
^{244}Cm	18	$0.5\cdot10^9$

*Sv (Sievert) is an unit of effective dose of ionizing radiation

[14] ABWR = Advanced Boiling Water Reactor

Besides, neutrons absorbed by nuclear fuel may also produce trans-uranium (Z>92) radioisotopes, in particular fissile ^{239}Pu (Z=94). Since plutonium is (i) deadly poisonous; and (ii) may be used in military applications (although those applications are not at all as straightforward as someone might expect), a particular care must be exercised during handling/storage of plutonium-containing materials, including spent fuel. Also structural materials in vicinity of a reactor core get activated during operation of the reactor (^{60}Co is a typical result), and after several years of reactor operation they finally become radioactive waste as well.

Because of that, management of nuclear waste is an essential social, political and economic issue that must be solved if nuclear power technology is to be deemed fully safe and accepted by public. Fortunately, a typical 1,000 MWe NPP produces annually just 3 m^3 (about 27 tons) of high-activity waste; that activity drops 1,000 times after just 10 years. Some industrial waste including plastics, Eternit (fibre-cement roofing material containing asbestos), some chemicals, scrap metals etc. may survive much longer in the environment.

About 1% of plutonium (mostly the ^{239}Pu fissile isotope) in spent fuel elements seems not much. However, plutonium holds huge amounts of energy: 1 g of plutonium is equivalent to 1 ton of crude oil or 100 g of natural uranium. Recycled uranium contains also about 1% of fissile ^{235}U, i.e. more than natural uranium.

Decaying radioactive nuclei produce heat. Therefore spent fuel elements freshly removed from the reactor core are first stored in storage water pools to cool them down. With time their activity drops. If no recycling is planned, spent fuel is stored inside of a water pool localized at NPP premises for 20-50 years. For the next 30-50 years they are stored in a "dry" bunker in a gaseous atmosphere. Finally the waste may be buried inside a special underground *radioactive waste repository* (cemetery, storage yard), which might be arranged for in a former salt mine, loam deposits, or granite rocks.

An alternate scenario is spent fuel *recycling*. After a few years of cooling down in water (in the reactor storage pool) spent fuel may be shipped to special processing plants to be chemically processed to separate fissionable elements (uranium and plutonium that may be used to manufacture fresh fuel elements) and some economically valuable materials (e.g. rare earth metals or some radioisotopes). About 3% of the starting mass ("true waste") remains, generally in the form of a liquid. The residues are glazed (vitrified), packed into large metal casks, and shipped to a radioactive waste repository. Vitrified fission products form some oxides of a structure typical for glass. Such glass is very resistant to washing away and sufficiently durable not to change its properties during the entire time needed for their activity to decay. Unfortunately the glazing procedure is not commonly applied since it requires a very advanced technology and is expensive. Spent fuel processing plants in France, UK and Belgium are currently producing about 1,000 tons of glazed nuclear waste per year (2,500 canisters of 400 kg each). A 1,000 MWe nuclear reactor produces 5 tons (12 canisters) of such glass per year. Such quantities are relatively easy to transport and store behind necessary shields.

Another approach worked out in Australia is to trap fission products into the Synroc ceramic, which is a synthetic rock containing a mixture of titanium dioxide (TiO_2), Bahollandite ($BaAl_2Ti_6O_{16}$) and perovskite ($CaTiO_3$). Fission products may be built into the rock crystallite structure. Trapping into such rocks is very efficient, waste content may attain 30% of the rock mass.

Influence of spent fuel processing on radiotoxicity of highly active waste produced in nuclear power reactors is charted in Fig.9.

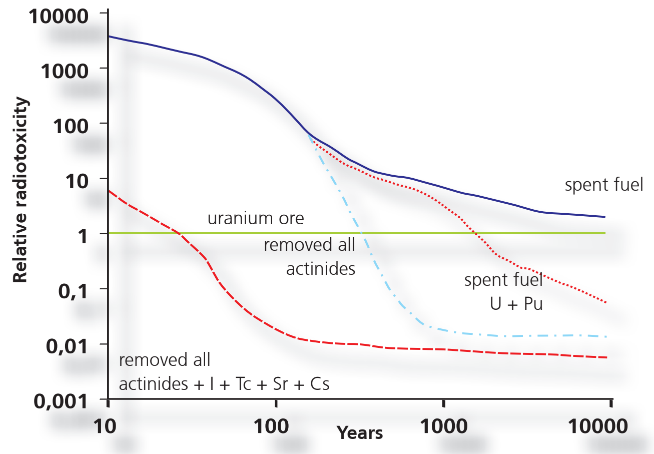


Fig. 9 Influence of spent fuel processing on radiotoxicity of highly active waste produced in nuclear power reactors (after G.J. van Tuyle et al., Nucl. Tech. 101 (1993) 1)

As one can see, spent fuel waste from which all actinides and basic fission fragments were removed would need to be stored only for a couple decades, i.e. would present no storage problem. On the other hand something needs to be done with long-lived radioactive isotopes extracted from the spent fuel elements. The technology to deal with them, that is currently under heavy research is called *transmutation*. Transmutation, see Fig.10, is conversion of one isotope into another isotope as a result of absorbing a neutron. New isotopes have usually much shorter half-lives. For this reason the so-called P&T (Partitioning and Transmutation) technology capable to extract plutonium, minor actinides (in particular Am and Cm), rare earth elements, other long-lived fission and to transmute them into other short-lived or stable isotopes gets much attention regardless of efforts devoted to arranging for geologic storage yards for nuclear waste. However, it is a technology of the future. It may be difficult to believe that the glazed portion of nuclear waste remaining after producing electric energy sufficient to satisfy lifetime needs of a statistical man fits in a handful.

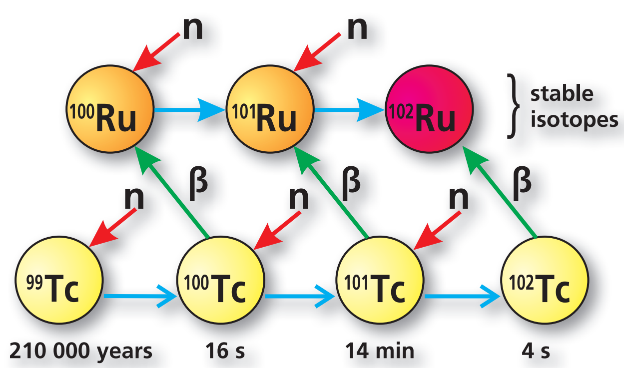


Fig. 10 Sample transmutations of the long-lived technetium isotope

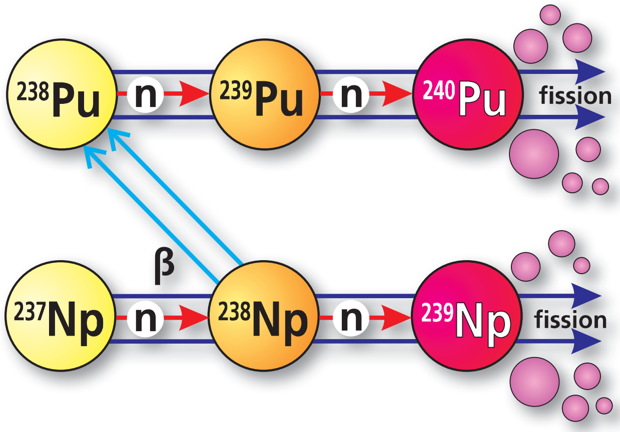


Fig. 11 Sample "burning" of neptunium and plutonium. After W. Gudowski, Royal Inst. Techn. Stockholm, Sweden

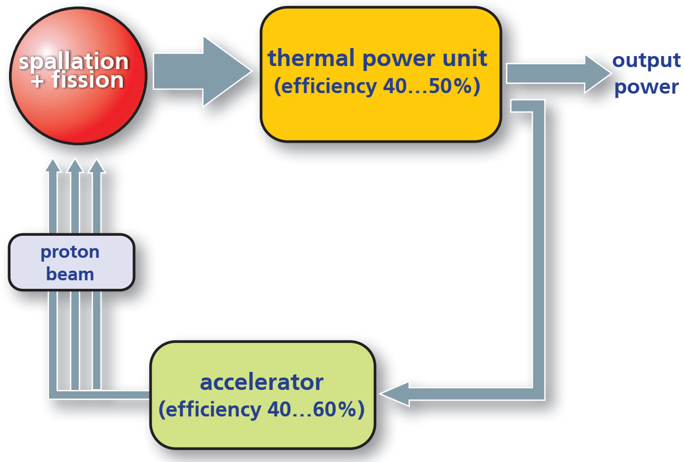


Fig. 12 The "energy amplifier" concept

While nuclear waste is indeed a problem to be dealt with, let us recall some numbers regarding other technologies. A typical 1,000 MW conventional power plant produces each year about 7 million tons of carbon dioxide, 200,000 tons of sulphur dioxide (both these gases are pollutants and/or greenhouse gases), and 200,000 tons of ash containing quite large amounts of toxic metals and radioactive elements. Let's look at France (where nuclear power is particularly well advanced) – less than 1 kg of nuclear waste is produced per capita per year, to be compared with 14 tons of industrial waste including 140 kg of hazardous materials. Long-lived isotopes account for only 20-30 g in all that nuclear waste, including 10 g of high-activity waste.

7.2. Will we be ever able to "burn" radioactive waste?

Let us define first what is the meaning of the "burning radioactive waste" and "transmuting radioactive waste" terms. To "transmute" a long-lived radioisotope into a short-lived or stable one means to convert it without employing any fission process. We are talking about "burning" if the isotope undergoes fission. Sample processes depicted in Figs. 10 and 11 must be initiated by fast neutrons (energies on the order of MeV). Both mechanisms are currently very intensely researched. Fast neutron reactors and accelerator-driven reactors give some hopes that we will be able to convert both already accumulated and future-produced nuclear waste into short-lived isotopes that are significantly easier to manage.

Energy amplifier (do not take that term too seriously; there is no energy amplification, just some external source provides additional neutrons to increase reactor transmutation capabilities) is the idea proposed at the turn of centuries by Professor Carlo Rubbia from CERN (Switzerland). Additional neutrons are produced in the so-called *spallation* reaction – protons accelerated to high energies (~1 GeV) strike heavy nuclei (e.g. lead), which as a result just crumble into many tiny pieces, including a large number of free neutrons. This high intensity flux of fast neutrons may be used in a subcritical reactor to convert ^{232}Th into fissile ^{233}U (the *thorium-uranium* cycle) and/or to "burn" nuclear waste (cause actinides to burn and/or transmute light isotopes). Ratio of fission-produced energy to the energy necessary to run the accelerator may be estimated to be 4 - 8, which explains the "energy amplifier" term coined for marketing purposes. The whole process is illustrated in Fig.12.

Sub-criticality of the reactor guarantees safety of that hybrid system: it is enough to turn the accelerator down to stop the chain reaction running within the reactor core. Besides, the thorium-uranium cycle does not produce any military grade plutonium at all, and quantities of transuranium isotopes produced in the system are two or three orders of magnitude lower than quantities produced in the currently operated "ordinary" nuclear power reactors. Research on accelerator-driven systems (ADS) is currently conducted in Europe, Japan, Korea, Belgium, Russia and US. Such systems may be demonstrated in a not-far distant future, especially in the Belgian MYRRHA project. The layout of the proposed system is shown in Fig.13.

It should be remembered, that while ADS design is discussed in this chapter as a part of thorium-uranium cycle, it is driven mostly by the need to deal with long-living actinides produced during operation of conventional nuclear reactors.

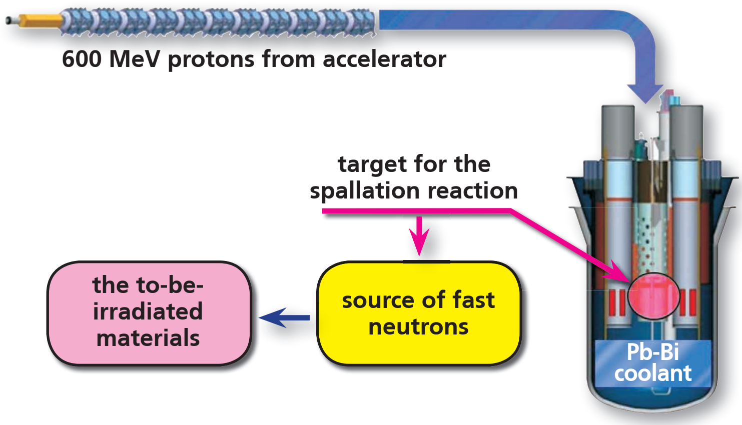


Fig. 13 Layout of the MYRRHA system. Reactor output 50-100 MWth in subcritical state, around 100 MWth in critical state

Two other innovative approaches include Inert Matrix Fuel and High Temperature Gas Reactors. New fuel type is tested since plutonium isotopes by-produced during operation of classical uranium-fueled reactors pose a serious problem.

Inert Matrix Fuel consists of grains of fissile ^{239}Pu dispersed within a chemically simple inert matrix (e.g. silicon carbide or magnesium oxide). Pay attention that no ^{238}U isotope is present. That latter isotope does not undergo fission in flux of thermal neutrons which is the main source of ^{239}Pu. New fuel type gives hope to gradually get rid of an excessive stock of dangerously poisonous, military-grade plutonium accumulated all over the world. Currently operated NPPs and spent fuel processing plants produce about 100 tons of plutonium each year, of which quantity only a small portion is used to manufacture the so-called MOX fuel (Mixed Oxides, a mixture of uranium and plutonium oxides). Stored plutonium must be very reliably protected due to its potential military applications and extremely high toxicity. Inert matrix fuel could also give a chance to "burn" minor actinides (Am, Np, Cm).

8. The uranium fuel cycle

Uranium fuel cycle consists of several distinctive stages.

Stage 1 is producing uranium oxide U_3O_8 out of mined uranium ore. The diggings are crushed and ground to a fine dust. The dust is chemically processed to separate uranium oxide from the rock. A 1,000 MWe NPP needs about 200 tons of U_3O_8 each year.

Stage 2 is enriching natural uranium, i.e. increasing content of the ^{235}U isotope. The oxide is chemically converted into the UF_6 gas. The gas is centrifuged in multistage cascades of high-speed centrifuges so that the heavier ^{238}U isotope is gradually separated from the lighter ^{235}U isotope. The enriched fraction is used to produce nuclear fuel, the depleted fraction (after conversion to uranium – a very dense metal) may be used as a very effective shield against gamma radiation. The majority of NPPs need uranium enriched to 4-6% of ^{235}U with CANDU being the only type that may be fueled by natural uranium. However, the price of the final product (electric energy) produced by CANDU reactors is not lower at all, since savings made on skipping the enrichment operation are outweighed by expensive heavy water needed in those reactors as moderator.

Stage 3 is "burning" the fissionable ^{235}U isotope contained within the fuel elements in the core of a reactor. We say the fuel inside reactor core is getting "spent", or "burned", although "used" is more appropriate name.

Stage 4 is storing the spent fuel nearby the reactor to cool it down. Finally it is either shipped to a processing plant to recycle fissionable materials (^{235}U and ^{239}Pu produced within the reactor) or else is prepared to be stored for a long period then shipped to a nuclear waste repository.

The "closed fuel cycle" term is presently understood as recycling of fissionable isotopes. The MOX (mixed-oxide fuel) fresh fuel produced after chemical separation in a process referred to as PUREX consists of some suitably processed plutonium mixed with enriched uranium. Stock reserves of accumulated military-class plutonium may be gradually used up to produce the MOX fuel. Five nuclear fuel processing plants are currently operated in Europe (including whole Russia)[15], about 30 reactors may be fueled by the MOX fuel. Out of about 7,000 tons of spent fuel produced each year by all operated light water reactors only about 15% is recycled. After recycling the volume of high-activity nuclear waste is decreased 5 times, while radiotoxicity of the waste is decreased 10 times.

Other products (fission fragments, minor actinides) are the remaining 4% of the spent fuel. The actinides are long-lived radioisotopes. They may be pressed into pellets waiting for future reactors that will transmute them into some short-lived/stable isotopes and/or "burn" them down. Should such transmutation/"burning" operation turn out to be feasible in a single step employing an ADS accelerator-driven sub-critical reactor, a closed fuel cycle of the future would become a reality.

High-activity solid nuclear waste remaining after fuel recycling is in most cases glazed, loaded into stainless steel containers and shipped to an underground nuclear repository (Fig.14). If spent fuel is not recycled, we are talking about open fuel cycle. Such spent fuel is normally cooled down in order to decrease its activity and radiotoxicity at least 100 times prior to shipment to an underground nuclear repository. A possible block diagram of closed a cycle is shown in Fig.15.

One may note, that similar (in principle at least) to *uranium-plutonium cycle* is *thorium-uranium cycle* (with ^{232}Th in the role of ^{238}U and ^{233}U in the role of ^{239}Pu). Since thorium is several times more abundant in Earth's crust than all isotopes of uranium combined mastering this cycle would open huge additional fuel reserves – but at this point there seems to be even smaller economic incentive for thorium reactors then there is for closing of uranium-plutonium cycle.

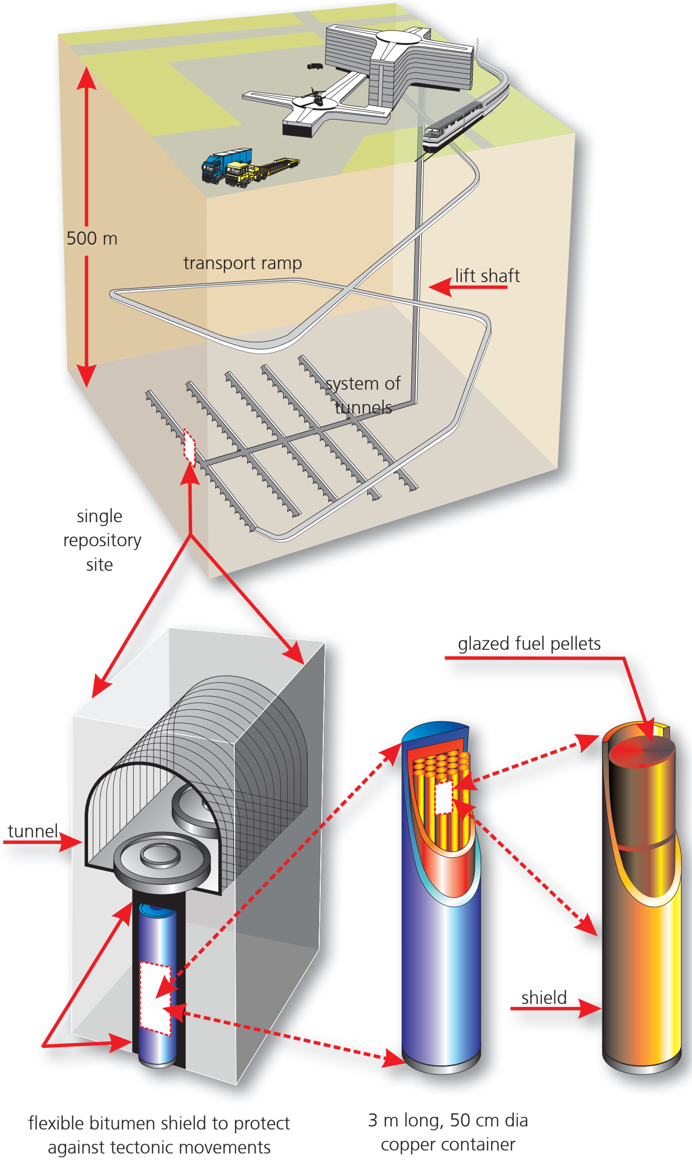


Fig. 14 Layout of a typical nuclear waste underground repository
The copper container may also hold glazed recycled nuclear waste. Typical diameters in the Onkalo, Finland repository: vent shaft 5.7 m, tunnel 3.5 m, entrance ramp tunnel 5.5 m. Depth of the lowest level in Onkalo is 520 m. Combined length of tunnels at the 420 m level is 5.5 km, tunnel slope 1:10.

[15]Source: http://www.euronuclear.org/info/encyclopedia/r/reprocessing-plants-ww.htm

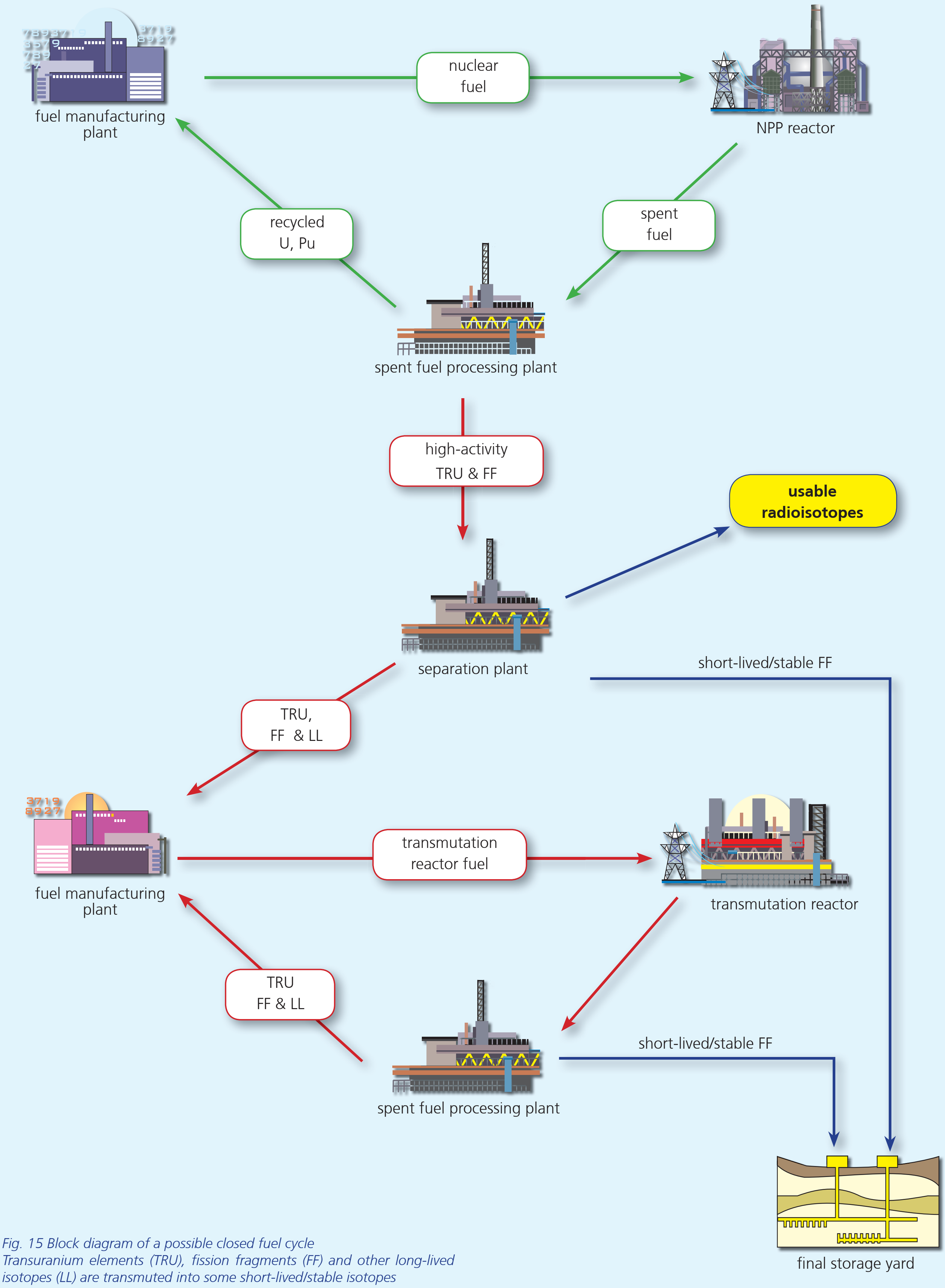


Fig. 15 Block diagram of a possible closed fuel cycle
Transuranium elements (TRU), fission fragments (FF) and other long-lived isotopes (LL) are transmuted into some short-lived/stable isotopes

9. 4th generation reactors

9.1. Introduction: fast reactors and breeders

Neutrons emitted in uranium fission reactions are fast: their velocities are about 10% of the speed of light, their energies are on the order of millions electron-volts (MeV). Such neutrons are easily absorbed by ^{238}U and they do not split ^{235}U nuclei easily. Therefore fast neutrons in conventional reactors must be slowed down by a moderator.

However, nuclear reactor may run as well without a moderator, if sufficiently enriched fuel (either ^{235}U, ^{233}U or ^{239}Pu) is provided. Four major advantages of fast-neutron reactors (in short: fast reactors) over typical thermal (slow) ones include:

- possibility to close the fuel cycle and to produce much more energy from uranium than that obtainable in conventional reactors. This is achieved through conversion of otherwise unusable ^{238}U into ^{239}Pu (which is a good fuel) by fast neutrons. In ordinary reactor (light water), this process occurs as well, although on smaller scale
- possibility to run on thorium fuel (fissile ^{233}U is being created from ^{232}Th)
- possibility to "burn down" spent fuel used in conventional reactors and thus decrease the amount of radioactive waste to be stored in underground repositories
- possibility to work at higher temperatures and in consequence to raise efficiency of the turbines.

In spite of these advantages, high investment outlays necessary to develop reactors of Generation IV are a problem. Besides, even if only 0.7% of natural uranium is fissile and is "burnt" in conventional reactors, demand for more efficient technologies of "burning" is not high taking into account that (i) uranium prices are currently rather low; (ii) the more efficient technologies are rather costly. Therefore, fast reactors are currently perceived in majority of countries as future facilities to burn down radioactive waste from conventional reactors if and when such waste accumulate. The issue of better utilization of uranium resources may be placed on the agenda in the future if (and when) uranium prices rise.

The cooling medium is the key technical problem with any fast reactor. Coolant must not slow fast neutrons down, therefore water is excluded. Various molten metals that may be pumped just like any liquid are tried as alternatives. Essential parameters of the most common such alternatives are given in Table 4.

Table 4 Liquid metals used (or planned) as fast reactor coolants[16]

	Sodium	Lead	Lead-bismuth eutectics
Melting point [°C]	98	327	125
Boiling point [°C]	883	1,745	1,670
Density at 450°C [kg/m³]	845	10,520	10,150
Specific heat at 450°C [kJ/kg/K]	1.23	0.127	0.128
Melting point [°C]	98	327	125
Boiling point [°C]	883	1,745	1,670
Density at 450°C [kg/m³]	845	10,520	10,150
Specific heat at 450°C [kJ/kg/K]	1.23	0.127	0.128
Melting point [°C]	98	327	125
Boiling point [°C]	883	1,745	1,670
Density at 450°C [kg/m³]	845	10,520	10,150

Reactors that produce nuclear fuel (some were discussed before, c.f. plutonium use) are referred to as breeders. As a matter of fact, the reactor that provided first nuclear electricity (EBR in Idaho, USA, 1951) was a breeder. Let us remind that conventional power reactors are ^{235}U-based, while ^{238}U nuclei just absorb or scatter neutrons. However, fast neutrons of energy on the order of 1 MeV may induce fissions of ^{238}U nuclei. Besides, low-energy neutrons may be absorbed by ^{238}U nuclei, in effect producing fissile ^{239}Pu plutonium. For that reason ^{238}U isotope is referred to as a *fertile material*[17]. ^{232}Th thorium is another fertile isotope and one day it might replace uranium as the major nuclear fuel.

Breeders are built to optimize nuclear reactions in which some fissile isotopes are produced. The possibility of producing nuclear fuel as a by-product sounds great, but in reality very few currently used power reactors are breeders. Economic terms are not favourable because:

(i) rate at which new fuel is produced is rather low;
(ii) plutonium is of low usability in the to-day world dominated by uranium-based reactors and at currently low prices of uranium.

Nevertheless, it is worth knowing that modern PWR/BWR reactors utilize only about 1% of energy contained in uranium or thorium, while breeders can utilize almost all that energy. There are some estimates that breeders might supply mankind with electricity for more than 1 million years (at the current energy consumption level and provided that also uranium contained in sea water might be used up). Besides, breeders would be able to effectively "burn down" (i.e. convert to other isotopes) actinides present in spent fuel/nuclear waste, thus reducing the time necessary for specific activity of the spent-fuel to decrease to that of uranium ore (as was discussed in section 7).

For technical reasons breeders are preferably fast neutron reactors.

Three major functions of the breeders envisioned for the future include:

- To produce trans-uranium elements (i.e. elements heavier than uranium[18]) from uranium, or ^{233}U from thorium, all usable as nuclear fuel. Breeders may limit demand for uranium even 100 times in relation to current demand of light water reactor fuel, not to mention the possibility to use the vast thorium deposits.
- To play a role of an isotope converter making possible to balance production and consumption of various trans-uranium elements.
- To convert minor actinides[19] and other long-lived isotopes present in nuclear waste into much shorter-lived isotopes (transmutation).

Potential advantages of fast neutron reactors are accompanied by some disadvantages:

(i) Much larger (than in thermal reactors) power density inside the core.

[16] http://www-pub.iaea.org/MTCD/Publications/PDF/P1567_web.pdf
[17] Of course the ^{238}U isotope remains fertile also in conventional light water reactors. However in such reactors ^{239}Pu is produced at a significantly lower rate than the rate at which ^{235}U is used up.
[18] Every reactor-produced trans-uranium element is simultaneously an actinide.
[19] All actinides except uranium and plutonium (neptunium, americium, curium, berkelium, californium, einsteinium, fermium).

(ii) Very short lifetime of free neutrons (ones living from one fission act to another fission act).
(iii) Smaller fraction of delayed neutrons (0.35% in comparison to about 0.6% in thermal reactors). The core must not be operated at the maximum reactivity in nominal conditions taking into account that coolant voiding may increase the reactivity.
(iv) Liquid metals as coolants are much more difficult to use than water.

The disadvantages generally mean that the core must be smaller, temperature in it may change more rapidly, and control circuitry must be able to make decisions to shut the reactor down in a time shorter than 1 s. It is indeed a challenge but not any fundamental technical problem.

9.2. Sodium-cooled fast reactors (SFR)

9.2.1. Introduction

Fast reactors must be cooled down with a medium which neither slows neutrons down nor absorbs them. Mercury – the only metal which is liquid at room temperatures – was the first choice both in USA and in the former Soviet Union. It is a heavy metal that does not slow neutrons down and does not freeze during shutdowns. However, its disadvantages: toxicity for humans, high vapour pressure, and low boiling point (reactor would have to be operated at a relatively low temperature) limited its application to just a few prototypes.

Sodium is an alternative[20]. As a light metal it is supposed to somewhat slow neutrons down, but it has got no mercury disadvantages. It melts at 98^{0}C so the reactor must be heated during idle time to avoid solidification of sodium inside tubing. Single stable isotope of sodium gets relatively easily activated and the resulting isotope (^{24}Na) has life-time of ~15 h. But the greatest problem is sodium strong reactivity with air and water – for this reason every tube and every tank with liquid sodium must have double walls, and the space between the walls must be filled up with some inert gas. Leak detectors (often complicated) are necessary. Special attention must be given to steam generator – a place where sodium coolant must be close to water to vaporize it into steam needed to power the turbine (therefore options using gas power conversion system instead of conventional steam generators are being considered, like in the framework of the French ASTRID demonstrator). To limit radiological risks in case of any leak, two sodium loops are necessary: sodium that carries away heat from the reactor core (and contains ^{24}Na radioactive isotope) transfers it to the secondary loop sodium, and only this latter, not radioactive, medium is allowed into steam generator. All in all, investment costs are high. Two possible solutions are schematically shown in Figure 16. In the loop design, sodium circulates outside the reactor vessel, although inside biological shield. In the pool design, primary heat exchanger and pumps are immersed in the reactor pool. Costs of expensive tubing are reduced in the latter approach, but the pool must be larger.

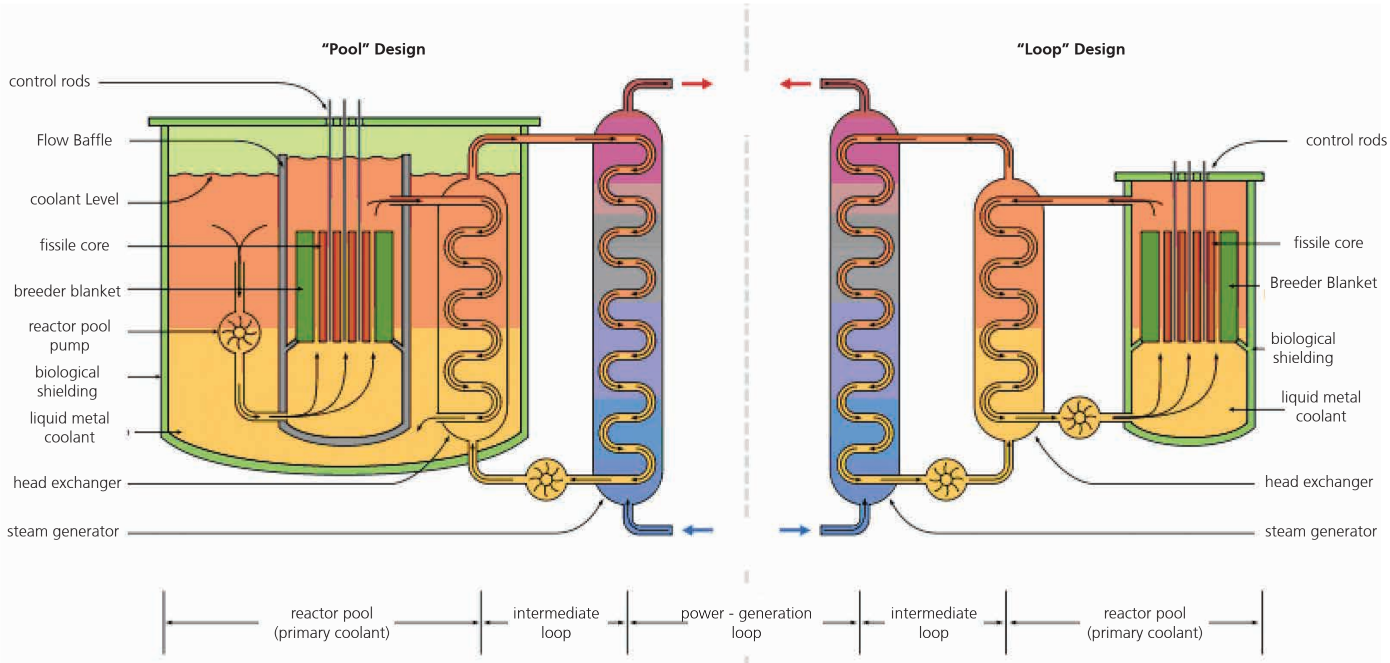


Fig. 16 Two variants of sodium reactor: pool design (left) and loop design (right) (source: Wikipedia Commons)

In spite of all these problems, sodium reactor technology is the most mature among all available technologies of the Gen IV reactors. Several such facilities have already been built and are now tested, work on subsequent facilities of that type is in progress. Approaches followed in various countries (USA, France, China, India, Russia, Japan and UK) are briefly presented in subsequent sections.

[20] Some prototypes in USSR and USA used sodium-potassium eutectic, which has lower melting temperature, but the same chemical disadvantages.

9.2.2. PRISM (USA)

Research on fast reactors has a long tradition in USA. Clementine (the first mercury-cooled reactor in the world), Fermi-1 (a pilot facility operated between 1969-1972), Clinch River Breeder Reactor Plant (a full-scale project never finished because of an unexpected rise of costs and some political issues), Integral Fast Reactor (also never finished project) are just a few examples. Drawing on that rich experience GE-Hitachi is currently promoting their PRISM reactor designed as a part of a plant built to re-process nuclear fuel spent in conventional reactors (Fig.17). Its primary task would be to "burn down" actinides present in the reprocessed fuel, 311 MWe power would be a by-product. The remaining waste would contain much less much shorter-lived isotopes. The PRISM technology might be interesting for the UK, where stock of plutonium acquired during Cold War times is now a problem[21].

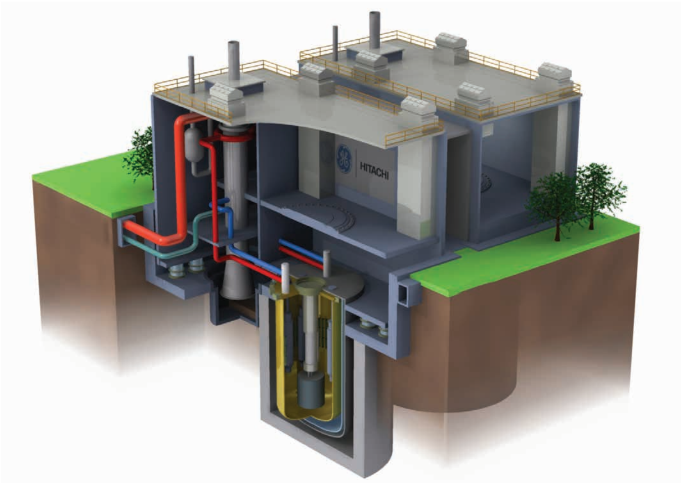


Fig. 17 PRISM (Power Reactor Innovative Small Module) layout, GE Hitachi http://en.wikipedia.org/wiki/S-PRISM

9.2.3. ASTRID (France)

Fig. 18 Astrid layout.

Research on sodium-cooled reactors has also a long tradition in France. Rapsodie, the first French reactor of that type (22, and later 40 MWt) was put in operation already in 1968. Much larger Phenix (250 MWe) was operated between 1973 and 2009. Still larger Superphenix (1,200 MWe), was operated only between 1986 and 1997. Both projects were troubled by sodium leaks with Superphenix having additional problems: technical (roof over the turbine room fell under the load of snow), social (it was producing large amount of plutonium), and legal (licence for operation was withdrawn in 1991, it took three years to get a new licence). Decrease of uranium prices in 1980's and 90s' was the final blow. As a result the reactor was longer idle than worked. The decision to decommission it, taken in 1997, was also a consequence of the Green Party's participation in the a French coalition government of that time.

Recently 650 million have been allocated for design work on ASTRID[22] (Fig.18), a new 600 MWe fast sodium-cooled reactor. If a decision to build the reactor is made, it should be put in operation around 2030.

9.2.4. CEFR (China)

20 MWe China Experimental Fast Reactor[23] (CEFR, Fig.19) was connected to the Chinese power grid in 2011. That experimental facility is to verify solutions to be applied in 600 MWe CFR-600 prototype reactor (scheduled for 2023)[24], which in turn is to be followed by 1,000 MWe CFR-1000 commercial reactor (scheduled for 2030)[25].

Fig. 19 CEFR visualisation.

9.2.5. FBR (India)

40 MWt small Fast Breeder Testing facility based on the French Rapsodie project has been operated in India (Kalpakkam) since the end of '80. The acquired know-how is currently used to build a much larger (500 MWe) Prototype Fast Breeder Reactor (PFBR). Commercial objects of that type are planned for the future.

9.2.6. BN (Russia)

Fast reactors were studied in the former Soviet Union equally intensely as in USA. BR-2, the first mercury-cooled fast reactor, was put in operation already in 1955. It was a very small facility of thermal power just 0.1 MW. Subsequent larger facilities (BR-5, BR-10, BOR-60) were cooled using a sodium-potassium eutectic or pure sodium.

125 MWe BN-350 put in operation in 1972 close to Aktau (Shevchenko) in Kazakhstan (on the banks of the Caspian sea) was the first sodium-cooled power reactor in the USSR. It was among the most successful sodium-cooled constructions ever. It was producing electricity for almost 20 years, while the produced steam was used to run seawater desalination plant (100,000 m^3 fresh water per day)[26]. However, it was decommissioned when the Soviet Union disintegrated because of large operational costs.

BN-600 (Byeloyarsk-3) was the next step. It was put in operation in 1980 and has been reliably operated till now. Change in location of the sodium-sodium heat exchangers

[21]https://www.theguardian.com/environment/2012/jul/30/fast-breeder-reactors-nuclear-waste-nightmare
[22]http://www.princeton.edu/sgs/publications/sgs/archive/17-1-Schneider-FBR-France.pdf
[23]http://www.iaea.org/NuclearPower/Downloadable/Meetings/2013/2013-09-11-09-13-TM-NPTD/7.yang.pdf
[24]https://aris.iaea.org/sites/..%5CPDF%5CCFR-600.pdf
[25]http://www.iaea.org/NuclearPower/Downloadable/Meetings/2013/2013-03-04-03-07-CF-NPTD/5.zhang.pdf
[26]http://www.iaea.org/inis/collection/NCLCollectionStore/_Public/28/008/28008858.pdf

was the major technological advancement from BN-350 to BN-600: external in respect to the reactor vessel in BN-350 exchangers were in BN-600 moved into the vessel. As a result the vessel had to be much larger, but costly sodium tubing has been greatly reduced.

Construction of the next step – 864 MWe BN-800 reactor (Fig.20) – started in 1983 at Byeloyarsk NPP. Works were halted when the Soviet Union disintegrated, and resumed in 2006. The BN-800 Byeloyarsk-4 reactor was connected to the grid in December 2015. A larger BN-1200 reactor is planned for around 2020.

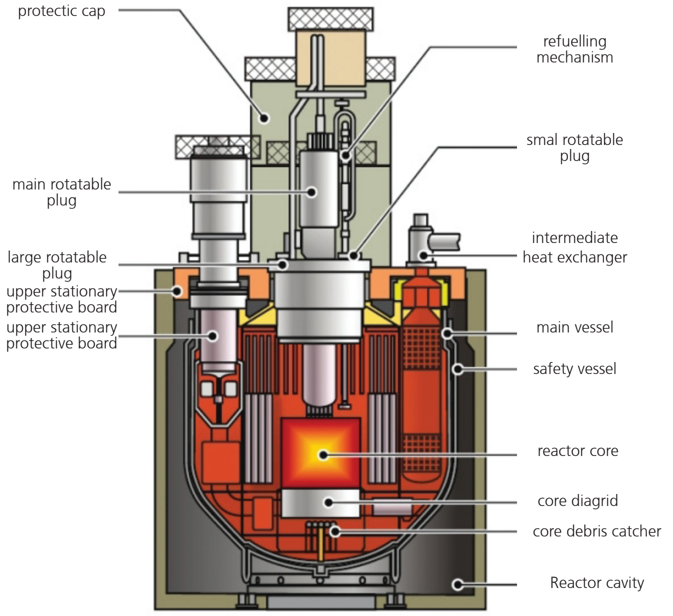


Fig. 20 BN-800 layout[27]

9.2.7. Japan

Japan was running in the past a rather ambitious programme to develop sodium-cooled breeders. 50 MWt JOYO reactor was put in operation in 1978; its power was later increased to 140 MW[28]. A much larger (280 MWe) MONJU reactor was connected to the power grid in 1995. After failures in 2007 and 1995 (respectively)[29] currently both reactors are shut down. In view of the current political situation in Japan one should not expect the Japanese programme to develop fast reactors will soon be continued, although both reactors are maintained so their restart in some future is possible.

9.2.8. UK

The program to ensure long-term security of electricity supplies from fast reactor NPPs in UK (should eventual shortages of uranium limit the deployment of thermal reactor NPPs) was initiated early in 1950s. Prototype Fast Reactor (PFR) built and operated at the UK Atomic Energy Authority's site at Dounreay in Scotland was the peak point of that programme. The facility was built to validate and provide operational experience with a large pool-type fast reactor and as a test bed for fuel, components, materials and instrumentation needed for an eventual commercial fast reactor NPP.

9.3. Lead- and lead-bismuth-cooled fast reactors (LFR)

Lead-cooled reactors (Fig.21) are an interesting alternative to sodium-cooled reactors, since lead is not flawed with the largest sodium disadvantage, namely high reactivity with water. Very high boiling point (1745°C) is another lead advantage: a sodium-cooled reactor might theoretically boil out its coolant in some emergency, while it is practically impossible in case of a lead-cooled reactor.

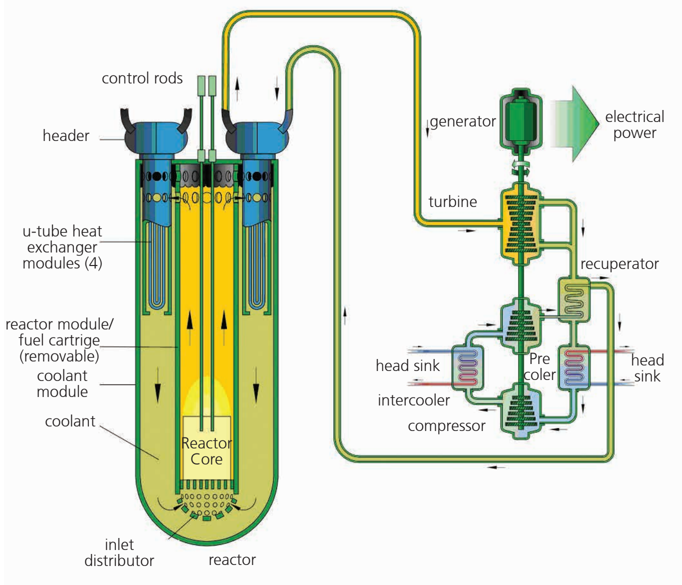


Fig. 21 Lead-or lead-bismuth-eutectic-cooled reactor layout (source: Wikipedia Commons).

However, lead has also some very serious drawbacks: (i) is very dense (difficult to pump); (ii) it erodes pump rotors; (iii) its relatively high melting point 327°C needs more heating in idle periods to prevent solidification of the coolant inside tubes/tanks. In that latter respect lead-bismuth-eutectic (44.5% Pb + 55.5% Bi alloy) may be an interesting alternative: its melting point is only 125°C.

Former Soviet Union has been the sole country which practically tried the lead-bismuth-eutectic technology for their submarines. In such applications a possibility to obtain larger power density i.e. smaller reactor sizes compared to conventional PWR reactors[30] is a major advantage. Currently Russians are trying to use the acquired know-how to work out 300 MWe BREST reactor to be located close to Tomsk in Siberia[31].

Less advanced studies are conducted also in Europe. Mentioned before MYRRHA (Fig.13) lead-bismuth cooled research reactor (discussed as critical or accelerator-driven system) planned in Mol (Belgium) is also to produce radioisotopes and transmute long-lived isotopes present in spent fuel. ALFRED lead-cooled power reactor is planned in Romania (with Italian companies engaged in that project). However, so far neither of those projects[32] has acquired funding sufficient to start construction phase.

Lead cooling technology (and even more lead-bismuth cooling technology) requires a very careful control of coolant contamination level. In particular presence of even residues of oxygen gives rise to corrosion products which accumulate very easily and may quickly clog the coolant channels. It has happened on board of one of Soviet submarines.

Generally lead is a better coolant in power reactors operated most of the time (hence not requiring long periods

[27]http://www.okbm.nnov.ru/npp#fast
[28]http://www.jaea.go.jp/jnc/jncweb/02r-d/fast.html
[29]http://www.iaea.org/NuclearPower/Downloadable/Meetings/2012/2012-06-20-06-22-TWG-NPTD/10_Japan.pdf
[30]http://www.gidropress.podolsk.ru/en/projects/nuclear-submarines.php
[31]http://world-nuclear-news.org/NN_Fast_moves_for_nuclear_development_in_Siberia_0410121.html
[32]http://www.nineoclock.ro/new-type-of-nuclear-reactor-to-be-commissioned-in-mioveni/

of external heating) since it is not as demanding in terms of contamination level. The more demanding lead-bismuth eutectic is better in research or military reactors since less energy is required to keep it in the molten state during long stand-by periods.

9.4. Gas-cooled fast reactors (GFR)

Gas instead of liquid metal may also be used as fast reactor coolant (Fig.22). Inert helium is the best choice: it is transparent to neutrons, resistant to neutron activation, optically transparent (the reactor can be relatively easily controlled and coolant temperature measured by means of some optical instruments). GFR has a potential to combine advantages of a high-temperature reactor and a breeder, with the breeding rate close to 1.

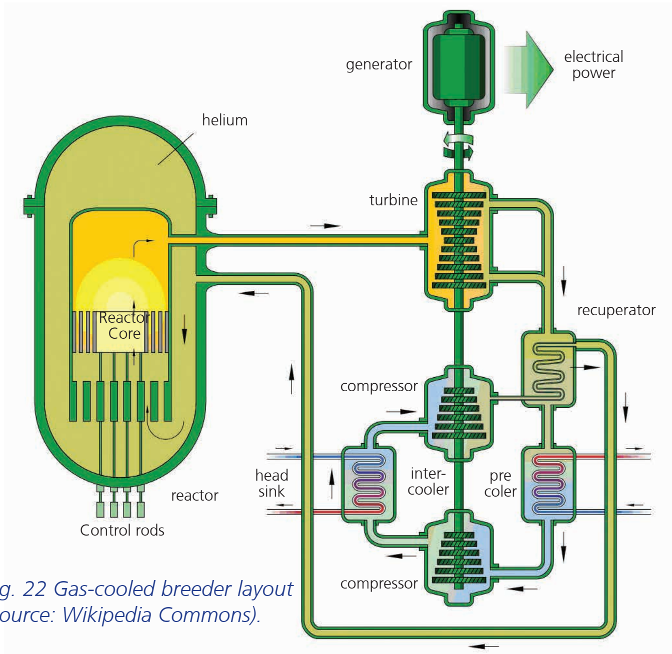


Fig. 22 Gas-cooled breeder layout (source: Wikipedia Commons).

This solution is of particular interest to parties from the Visegrad Group of Nations (V4) involved in the EU VINCO project (ŮJV Řež a.s. and Centrum Výzkumu Řež s.r.o. from Czech Republic, VUJE a.s. from Slovakia, Academy of Sciences Centre for Energy Research from Hungary, and National Centre for Nuclear Research from Poland), supported by the French CEA. Advantages of a helium coolant when compared to liquid metal include:

- no corrosion issues
- no coolant activation issues
- reactor core may be easily visually inspected by means of some cameras
- a better neutron balance (helium does not absorb neutrons), therefore waste may be "burned down" with a better efficiency
- high temperature (about 850°C) of helium at the core outlet is possible, which allows to achieve higher thermal-to-electric conversion coefficient (about 45%)
- possible achievement of a zero-breeding gain core, i.e. production of fissile material inside the core is equal to its consumption
- possibility to build a reactor with just one cooling loop (direct cycle) – with generator being powered by gas turbine powered directly by helium coming from the core instead of steam turbine requiring second loop and steam generator.

Major disadvantages include:

- an overpressure must be kept at all times inside the reactor vessel to preserve cooling
- relatively low gas heat capacity in comparison to liquids which requires much larger volumes and flow velocity of coolant
- shielding requirements (since radiation is not absorbed by the coolant).

The disadvantages significantly complicate the task to cool the reactor down in emergencies.

Two currently considered gas-cooled designs are EM² (Energy Multiplier Module by US General Atomics) and ALLEGRO (originally by French CEA, now by Vysehrad countries' V4G4 Center of Excellence).

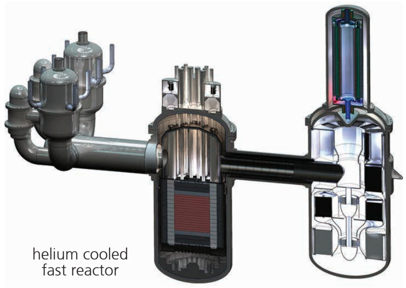


Fig. 23 EM² reactor layout (source: General Atomics promotional materials).

EM² reactor would be fuelled with uranium nitride and helium-cooled. Hot helium would directly drive a gas turbine.

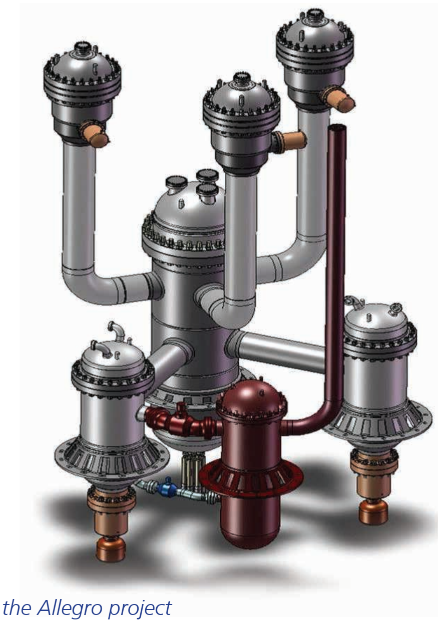


Fig. 24 Idea of the Allegro project

ALLEGRO (Fig. 24) helium-cooled experimental fast reactor currently under development by the V4G4 Centre of Excellence in cooperation with the French CEA is one of the six GFR concepts selected by the Generation IV International Forum. The main purpose of the facility is to develop:

- innovative refractory GFR fuel
- components and systems related to helium-cooling technologies, and
- GFR-specific safety solutions.

Preliminary design studied started in 2009. The project is exploring the rated power range (30–75 MWt) and the power density range (50–100 MW/m³) compatible with safety limits.

At the same time, feasibility of a LEU UOX (Lowly-Enriched Uranium: Uranium Oxide fuel) start-up core as an alternative to the standard MOX core is considered. This start-up core, to be used within the first period of reactor operation, will include some experimental channels dedicated to research on some new high-temperature resistant (refractory) fuel. The currently studied approach is uranium & plutonium carbide housed in some SiC tubes.

Both projects are currently in their early stages of development and still need plenty of time and effort before eventual implementation. In particular the to-be-solved barriers include the technology of helium-driven turbines (some successfully concluded tests cannot be regarded as a proof of a mature technology) and the technology of the new fuel.

9.5. High temperature graphite reactors (HTGR)

Helium is also the coolant of choice in the High Temperature Graphite Reactor approach, but HTGR is not a fast reactor: it uses graphite moderator to slow neutrons down. It is a sole Gen IV reactor discussed here to use slow neutrons. The key innovation is the fuel: instead of regular rods/pellets, very fine spheres (of diameter of a fraction of one millimetre) each covered with several ceramic layers are planned (Fig.25). Years of experiments resulted in a combination of materials resistant to high temperatures, tight for fission products, and radiation resistant. The technology has been dubbed TRISO.

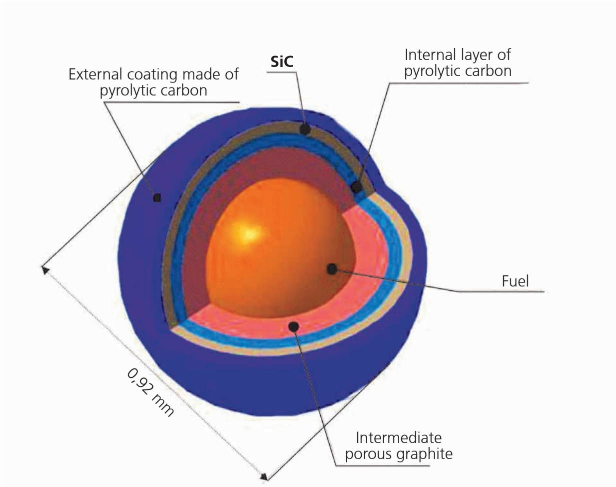


Fig. 25 TRISO fuel sphere

Introduction of such fuel flips over the entire nuclear reactor safety philosophy. The entire set of barriers whose task is to prevent release of fission products outside the conventional reactors are here replaced with ceramic coatings covering each individual fuel sphere. The coatings form a kind of "safety containment". Such fuel is resistant to very high temperatures, but requires a very stringent quality control during production to preserve tightness. Moderator is part of the fuel, not an outside block.

New fuel concept together with gas-cooled core dramatically reduces complexity, improves safety and makes the reactor more economical. Gas coolant circulates around "pebbles" without need for special in-core coolant piping. Coolant contains no hydrogen thus vastly reducing the explosion risk. Fuel spheres are removed from the core at the constant rate, each one is examined and either returned into the core or replaced by the fresh one. Reactor can use ^{235}U, ^{239}Pu or MOX fuel, though not at the same time.

The most important are safety features of this reactor type. If the temperature raises, neutron are absorbed in the non-fissile isotopes contained in the fuel and the reactor passively switches off. It cannot explode, but its inflammability is disputed. It has been tested that if all control rods are removed, and cooling is switched off the reactor goes into safe mode of small thermal power, that can be radiated through the reactor vessel at safe temperature.

HTGR reactor concept was born in UK, West Germany and USA, where a few such facilities were operated in the past. A number of technical problems typical for each new technology were encountered during operation, however most seem to be solvable with modern computational, technical and measurement techniques.

Works on HTGR reactors both in Germany and in USA were practically stopped in early 1990s, when low oil prices made investing in new reactor types an economically unjustified venture. Additionally, after the Chernobyl accident, political attitude in Germany became very unfavourable. However, China bought documentation from Germany and a small (10 MW) prototype facility has been operated in Beijing for a few years. Construction of a larger facility started in 2012. The HTR-PM (Power Module) facility (Fig.26) will consist of two 250 MWt reactors, each with its steam generator. Helium will be heated up to 750°C, 550°C steam produced by both reactors will drive a single 210 MWe turbine (42% electricity production efficiency). Six-reactor blocks are planned for a more distant future.

The HTGR technology has been developed for years also by:

- US company General Atomics (MHTGR of mid 80s employing heat exchanger and steam turbine, followed by potentially very attractive Gas Turbine – Modular Helium

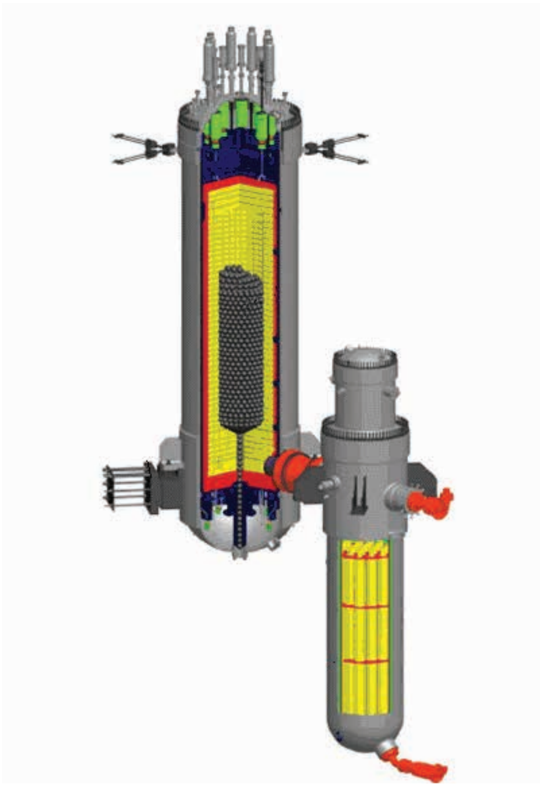


Fig. 26 HTR-PM unit layout (reactor to the left, steam generator to the right). Each fuel sphere (of diameter of about 6 cm) is composed of a few thousand TRISO grains pressed into graphite. Reactor core contains a few hundred thousand such spheres. Source: INET.

Reactor, in which hot helium directly drives a helium turbine, see Fig.27, of early 90s, both never implemented);

- French company Areva (ANTARES project, based on MHTGR documents purchased from GA)
- a consortium of companies from South Korea
- Japanese Atomic Energy Agency (the HTR prototype 30 MW reactor).

HTGR can use wide variety of fissile material – TRISO pebbles can use enriched uranium, Mixed OXides (of uranium and plutonium) and also thorium with ^{233}U.

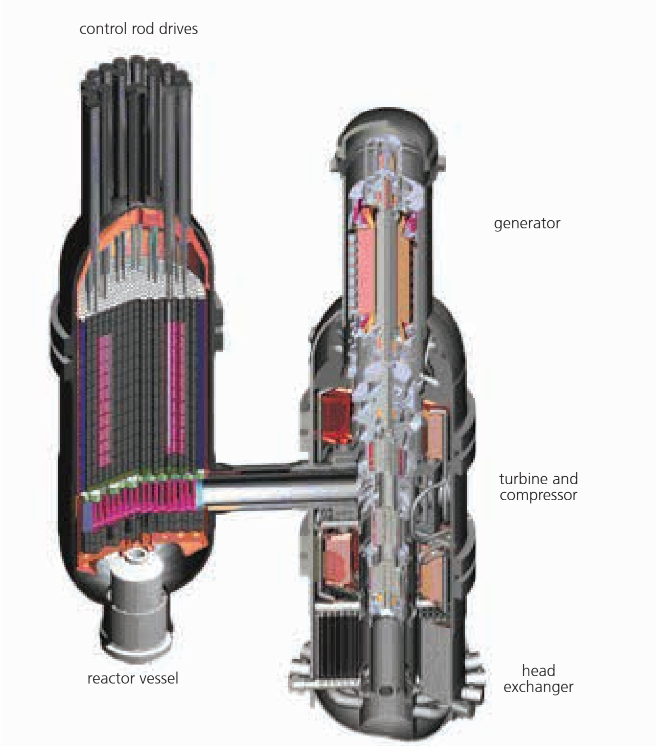


Fig. 27 GT-MHR reactor cross-section.

Safety is an essential advantage of the HTR technology. Reactor vessel is filled up only with some refractory materials (graphite, ceramics), a large graphite mass provides a large thermal inertia so potential incidents run relatively slowly. However, the key feature is capability to release decay heat from the shut-down reactor through vessel walls into the surrounding atmosphere. Neither complicated cooling systems (of necessarily limited reliability) nor safety containment are required.

Disadvantages of the technology include:

- Large volume reactor vessel (in proportion to the generated power)
- Problems with lifetime of graphite, which would be very difficult to replace. This can be a limiting factor for plant lifetime. The problem has been observed for many years in UK, where graphite reactors have been in operation since early 50's
- Generation of graphite waste, which is troublesome (although definitely manageable)
- Requirement for fuel enriched to higher levels than fuel for light water reactors.

High helium temperatures make possible electricity production efficiency above 40%. Chinese experts estimate that investment outlays (per 1 MWe) in their HTR-PM technology are comparable to outlays necessary in the PWR conventional technology. High coolant temperatures open up possibilities for applications other than electricity production, supply of industrial heat for large chemical or desalination plants in the first place. At this point chemical plants use large amounts of coal and natural gas just to produce hot (and sometime superheated) steam, this steam is transported through the web of pipes.

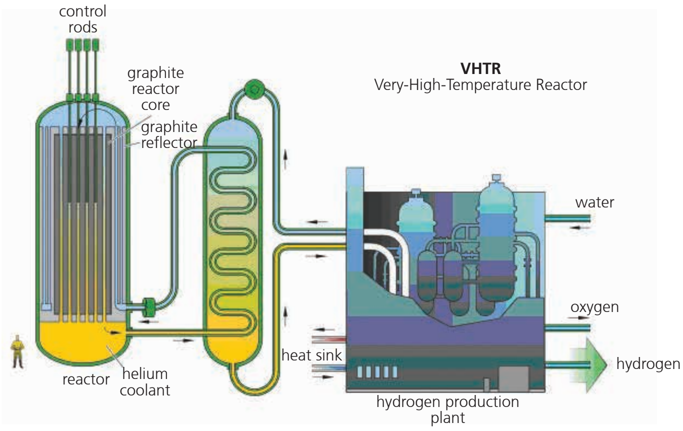


Fig. 28 Layout of a very high temperature reactor that might power up a hydrogen producing plant (source: Wikipedia Commons).

Currently used high-temperature nuclear reactors may produce steam of temperatures up to 550°C (~300°C in case of PWRs and BWR) and can replace fossil-fuel-fired boilers as sources of the hot steam. High temperature helium- or molten metal-cooled reactors might replace natural gas as sources of the heat necessary for the superheated steam production. Of course chemical plants' installations would have to be suitably adopted. The highest helium temperature so far experimentally obtained at the output of AVR reactors in Germany and HTTR reactors in Japan was about 950°C. It was a value close to the limits of modern technology set by strength of materials of which reactor vessel and heat exchangers are made. Therefore the possibility that HTGR or Very High Temperature Reactors VHTR (Fig.28) reactors will replace natural gas burnt in chemical plants is a distant future perspective.

10. Can we safely live with nuclear reactors around us?

Operation of every nuclear facility – as is the case for any industrial facility – is accompanied by some risks of small probability, but perhaps large consequences. They include the risk of reactor failure, risk of liberating radioactive substances to the atmosphere, risk of environment pollution resulting from incorrect nuclear waste management, or risk of spreading fissionable/radioactive materials. It is difficult to assess exposition of individuals or consequences of such accidents since the involved risks do not belong to the category of voluntarily accepted risks such as the risk of participating in a traffic accident one accepts at the moment of getting on a car to travel a highway. The involved risks may be estimated by number of death casualties per unit of produced energy. This ratio estimated for coal mining or oil/gas drilling industry plus conventional power generation industry plus consequences of air pollution resulting from combustion of fossil fuels is about 40 times higher than the ratio estimated for uranium mining industry plus nuclear power generation industry including waste management and decommissioning of totally depreciated plants.

Death casualties following large industrial catastrophes of the 20th century were counted in thousands. Overtopping of the Vaiont Dam (Italy, 1963) claimed 2,000 casualties; poisonous chemicals leaking from a Bhopal (India) pesticide plant in 1984 instantly killed over 30,000 people, the number of aggregated casualties reached 200,000; failure of the Banqiao Dam (China, 1975) was a cause of death of 171,000 people. For comparison, the worst disaster in the entire history of nuclear power, namely the fire of the Chernobyl reactor (1986), directly claimed lives of 31 rescuers, including 28 who died within a few days because they were exposed to lethal doses of radiation. Another 19 members of the 106-men rescue team died before 2010, several children died of thyroid cancer. About 6,700 new thyroid cancer cases were noted, however none of them turned out to be mortal.

Natural disasters may sometimes claim much more human lives. Tsunami on the Indian Ocean in 2005 claimed lives of about 300,000 people. Tsunami that destroyed the Fukushima NPP in March 2011 claimed lives of about 20,000 people. However, much less is talked about those casualties than about the destroyed reactors and the increased radiation level in the area around the plant even if nobody was injured nor lost their life because of the nuclear power accident itself or the aftermath radiation. Some estimate that the entire US nuclear power programme has increased the radiation risk by an amount comparable to consequences of a hypothetical rise of car speed limit from 80 to 81 km/h.

Nuclear power is not risk free – but no human activity is risk free. The relative dangers of every new technology should be carefully weighed out and honestly compared with other technologies. We believe that nuclear power is probably the best solution capable to satisfy mankind's hunger for power in a honest comparison, in which all costs and dangers have been duly taken into account.

One may ask the following question: if the above is true, why NPPs have not taken already the power industry? Apart the negative attitude against nuclear power currently prevailing in numerous societies, there is a matter of cost, or – more precisely – the amount of capital outlays required to develop a NPP. Electricity from nuclear power is cheap under condition that initial investment is to be returned in about 50 years. The price of nuclear fuel and workforce salaries is negligible compared to construction costs. On one hand it is a good thing – one does not care whether uranium costs 30 or 600 USD/kg (which opens up perspectives that abundant new resources may become available). But on the other hand someone has to put some tens billion USD upfront.

Also the process to develop new nuclear technologies is very costly. The need to provide a working demonstrator before commercialization of any new solution means that someone has to pay for it. Commercial sector avoids investments in possibly risky scenarios, and taxpayers are scared. It is no surprise that in such a climate so many R&D programmes have been stopped, officially due to lack of funding. However, if this is not going to change, we will probably have to depend on suppliers of coal, oil and natural gas for many years to come.

11. Afterword

This brochure offers a review of problems encountered by nuclear power technology on its evolutionary way from Gen-I reactors towards Gen-IV reactors. The road is neither easy nor cheap. Nevertheless there is no good alternative to nuclear energy. The history shows us how useful, environment-friendly, and safe nuclear power can be. It can provide us with cheap energy for many thousands of years ahead. It would be very unfortunate if the recently amplified anti-nuclear fears and preconceptions outweighed the benefits offered by that technology. And one should remember that the time left for humanity for putting new nuclear power plants into operation is not long, may be 100 years or less.

12. GLOSSARY

ABWR	Advanced Boiling Water Reactor. Reactor worked out in 80'/90', currently offered for sale by General Electric, Hitachi, and Toshiba. A few such facilities are operated in Japan, other are currently under construction on Taiwan.
AGR	Advanced Gas Cooled Reactor. British reactor of 2nd generation evolved from the 1st generation Magnox reactors.
AP1000	Advanced Passive 1000. PWR-type reactor of power 1,000 MWe, currently offered for sale by Westinghouse. A few such facilities are currently under construction in USA and China.
BN 350/600/ 800/1200	Russian family of sodium cooled fast reactors of power 350/600/800/1,200 MWe, currently shut-down/ operated in Byeloyarsk/operated in Byeloyarsk/under design, respectively.
BWR	Boiling Water Reactor. One of two major types of conventional power reactors.
CANDU	CANadian Deuterium Uranium. Canadian family of PHWR-type reactors exported to India, Pakistan, Romania, South Korea, Argentina, China.
EM2	Energy Multiplier Module. Helium cooled fast reactor project promoted by the General Atomics Company (San Diego, Ca, USA).
ESBWR	Economic Simplified Boiling Water Reactor. BWR-type reactor of a new generation, offered for sale by General Electric/Hitachi consortium
GT-MHR	Gas Turbine Modular Helium Reactor. HTR-type reactor/helium turbine combination project worked out in 90' by General Atomics.
HTR	High Temperature Reactor. Helium cooled reactor with graphite moderator.
HTGR	High Temperature Graphite Reactor. US equivalent for HTR, used to distinguish such reactors from other technologies also capable to produce high temperature heat.
HTR-PM	HTR-Power Module. Chinese use that name for two HTR-type reactors currently under development in China.
IAEA	International Atomic Energy Agency. UN agency promoting peaceful applications of nuclear energy and preventing proliferation of nuclear weapons.
INES	International Nuclear Event Scale.
Magnox	Magnesium, non-oxidizing. Magnesium alloy used for cladding of fuel applied in 1st generation British CO_2 cooled reactors. Commonly used name for all those reactors.
MW	Megawatt. Unit of power.
MWe	Megawatt of electric power. Unit of electric power.
MWt or MWth	Megawatt of thermal power. Unit of thermal power.
MWh	Megawatt hour. Unit of energy.
PHWR	Pressurized Heavy Water Reactor. Reactor type similar to PWR, but heavy water rather than ordinary light water is used as the moderator and coolant. The type popular in Canada (CANDU) and India (licenced by Canadians).
PWR	Pressurized Water Reactor. One of two major types of conventional power reactors.
RBMK	In Russian: Large Power Channel Reactor. Soviet reactor type with moderator graphite, cooled by pressurized boiling water. Never offered for export since it is capable to produce military-grade high purity plutonium. Chernobyl power plant employed just such reactors. Currently RBMK reactors located in Lithuania and Ukraine are shut down, a few RBMK reactors are operated exclusively in Russia.
TMI	Three Mile Island. Power plant in Pennsylvania (USA). One of the two PWR-type Babcock&Wilcox reactors installed in that plant failed in 1979. It was one of the few famous accidents in history of civil nuclear power.
TSO	Technical Support Organisation. A body with scientific/technical potential in the field of nuclear power technologies necessary to deliver expert services, to conduct R&D works, to verify not yet checked technical solutions etc. In some countries TSOs are parts of Nuclear Regulatory Agencies, in others – independent organizations that may be hired by Nuclear Regulatory Agencies or nuclear industry.
WANO	World Association of Nuclear Operators
WWER or VVER	Soviet family of PWR-type reactors exported to former eastern bloc countries, India and Iran. Power of the most popular variants is 440 and 1,000 MWe.

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