INTERNATIONAL JOURNAL OF
ENERGY AND ENVIRONMENT

Volume 4, Issue 5, 2013 pp.743-776
Journal homepage: www.IJEE.IEEFoundation.org

An overview of future sustainable nuclear power reactors

Andreas Poullikkas*

Electricity Authority of Cyprus, P.O. Box 24506, 1399 Nicosia, Cyprus.

Abstract
In this paper an overview of the current and future nuclear power reactor technologies is carried out. In particular, the nuclear technology is described and the classification of the current and future nuclear reactors according to their generation is provided. The analysis has shown that generation II reactors currently in operation all around the world lack significantly in safety precautions and are prone to loss of coolant accident (LOCA). In contrast, generation III reactors, which are an evolution of generation II reactors, incorporate passive or inherent safety features that require no active controls or operational intervention to avoid accidents in the event of malfunction, and may rely on gravity, natural convection or resistance to high temperatures. Today, partly due to the high capital cost of large power reactors generating electricity and partly due to the consideration of public perception, there is a shift towards the development of smaller units. These may be built independently or as modules in a larger complex, with capacity added incrementally as required. Small reactors most importantly benefit from reduced capital costs, simpler units and the ability to produce power away from main grid systems. These factors combined with the ability of a nuclear power plant to use process heat for co-generation, make the small reactors an attractive option. Generally, modern small reactors for power generation are expected to have greater simplicity of design, economy of mass production and reduced installation costs. Many are also designed for a high level of passive or inherent safety in the event of malfunction. Generation III+ designs are generally extensions of the generation III concept, which include advanced passive safety features. These designs can maintain the safe state without the use of any active control components. Generation IV reactors, which are future designs that are currently under research and development, will tend to have closed fuel cycles and burn the long-lived actinides now forming part of spent fuel, so that fission products are the only high-level waste. Relative to current nuclear power plant technology, the claimed benefits for generation IV reactors include nuclear waste that lasts a few centuries instead of millennia, 100-300 times more energy yield from the same amount of nuclear fuel, the ability to consume existing nuclear waste in the production of electricity and improved operating safety. Generation V+ reactors are designs which are theoretically possible, but which are not being actively considered or researched at present. Though such reactors could be built with current or near term technology, they trigger little interest for reasons of economics, practicality or safety.

Keywords: Nuclear power; Nuclear reactor; Power generation.

* Parts of this work were undertaken while the author was a Visiting Professor in the Department of Mechanical Engineering, College of Engineering, American University of Sharjah, PO Box 26666, Sharjah, United Arab Emirates.

1. Introduction
The nuclear technology is widely used by the developing and industrial countries and still is an option for the expansion of generation systems. The nuclear industry has evolved greatly over the last 50 years. It has accumulated several hundreds of years of experience on various types of reactors. It has been constantly researching ways to improve safety, efficiency and waste disposal problems. The latest technologies on the field are quite promising in terms of safety and waste disposal problems while achieving high efficiency and low overall costs. The nuclear power generation remains one of the cleanest energy forms in the world in comparison with the fossil fuel technologies [1].
Although, the nuclear power industry has improved the safety and performance of reactors and has proposed new safer but generally untested generation III, IV and V+ reactor designs, there is no guarantee that the reactors will be designed, built and operated correctly. Mistakes do occur and the designers of reactors at Fukushima in Japan did not anticipate that a tsunami generated by an earthquake would disable the backup systems that were supposed to stabilize the reactor after the earthquake. Catastrophic scenarios involving terrorist attacks are also conceivable [2].
In this work, an overview of current and future sustainable nuclear energy is carried out. In particular, the nuclear technology is described and the classification of the current and future nuclear reactors according to their generation is discussed in detail.
In section 2, the nuclear technology is described and in section 3, the generation II nuclear reactors are presented. In section 4, the different types of generation III nuclear reactors are discussed and in section 5 the generation IV nuclear reactors are presented. The future generation V+ nuclear reactors are described in section 6. The conclusions are summarized in section 7.

2. The nuclear technology
Just as conventional power stations generate electricity by harnessing the thermal energy released from burning fossil fuels, the nuclear reactor produces and controls the release of energy from splitting the atoms of certain elements as illustrated in the fission chemical reaction, known as nuclear chain reaction. When a large fissile atomic nucleus such as Uranium-235 (^{235}U) or Plutonium-239 (^{239}Pu) absorbs a neutron, it may undergo nuclear fission. The heavy nucleus splits into two or more lighter nuclei, releasing kinetic energy, gamma radiation and free neutrons, which are collectively known as fission products. A portion of these neutrons may later be absorbed by other fissile atoms and trigger further fission events, which release more neutrons, and so on. This nuclear chain reaction can be controlled by using neutron poisons and neutron moderators to change the portion of neutrons that will go on to cause more fission. Nuclear reactors generally have automatic and manual systems to shut the fission reaction down if unsafe conditions are detected [2].
The energy released is used as heat to make steam to generate electricity. The principles for using nuclear power to produce electricity are the same for most types of reactors. The energy released from the continuous fission of atoms of the fuel is harnessed as heat in either a gas or water, and is used to produce steam. The steam is used to drive the turbines, which in turn drive generators and produce electricity, as in most fossil fuel plants [1]. There are several components [3, 4] common to most types of reactors. Fuel, usually pellets of uranium dioxide (UO_2) arranged in tubes to form fuel rods. The rods are arranged into fuel assemblies in the reactor core. The moderator, slows down the neutrons released from the fission reaction so that they cause more fission. It is usually water, but may be D_2O or graphite. Control rods, are made of neutron-absorbing material such as cadmium, hafnium or boron, and are inserted or withdrawn from the core, which allow controlling the rate of fission reaction, or to halt it. Secondary shutdown systems involve adding other neutron absorbers, usually as a fluid, to the system.

3. Generation II reactors
Generation II reactors are the reactors that are currently in operation all around the world. They typically use enriched uranium fuel and are mostly cooled and moderated by water. The main types and characteristics of generation II reactors are tabulated in Table 1[1].

3.1 Pressurized water reactor
This is the most common reactor type, with over 230 reactors in use around the world for power generation and a further several hundred in naval propulsion. It uses ordinary water as both coolant and moderator [3, 4]. In pressurized water reactor (PWR) the nuclear fuel in the reactor pressure vessel is engaged in a chain reaction, which produces heat. The water of the primary coolant loop is then heated

through the fuel cladding. The hot water is pumped into a steam generator in, which the secondary coolant is heated up without mixing the two fluids. This is desirable, since the primary coolant is necessarily radioactive. The steam formed in the steam generator is then used for power generation. The primary coolant is used in a PWR flows through the reactor core at a pressure of around 155bar and temperature of roughly 315°C [5].

Table 1. Generation II reactors designs

Reactor type	Light water reactor (LWR)		Heavy water reactor (HWR)	Graphite moderated reactor (GMR)		Fast breeder reactor (FBR)
	Boiling water reactor (BWR)	Pressurized water reactor (PWR)		Gas cooled (GCR)	Water cooled	
Purpose	Electricity	Electricity; nuclear powered ships (USA)	Electricity; plutonium production	Electricity; plutonium production	Electricity; plutonium production	Electricity; plutonium production
Coolant type	Water	Water	Heavy water (D_2O)	Gas (CO_2 or helium)	Water	Molten, liquid sodium
Moderator type	Water	Water	Heavy water	Graphite	Graphite	Not required
Fuel-chemical composition	Uranium dioxide (UO_2)	Uranium dioxide (UO_2)	Uranium dioxide (UO_2) or metal	Uranium dicarbide (UC_2) or uranium metal	Uranium dioxide (UO_2) (RBMK) or metal (N-reactor)	Plutonium dioxide (PuO_2) and uranium dioxide (UO_2) in various arrangements
Fuel-enrichment level	Low-enriched	Low-enriched	Natural uranium (not enriched)	Slightly-enriched natural uranium	Slightly-enriched	Various mixtures of ^{239}Pu and ^{235}U

In PWRs the coolant water is used as a moderator by letting the neutrons undergo multiple collisions with light hydrogen atoms in the water, losing speed in the process. This moderating of neutrons will happen more often when the water is denser (more collisions will occur). The use of water as a moderator is an important safety feature of PWRs, as any increase in temperature causes the water to expand and become less dense, thereby, reducing the extent to which neutrons are slowed down and hence reducing the reactivity in the reactor. Therefore, if reactor activity increases beyond normal, the reduced moderation of neutrons will cause the chain reaction to slow down, producing less heat. This property, which is known as the negative temperature coefficient of reactivity, makes PWR reactors very stable.

The uranium used in ^{235}U fuel is usually enriched. After enrichment the UO_2 powder is fired in a high-temperature, sintering furnace to create hard, ceramic pellets of enriched uranium metal. The cylindrical pellets are then put into tubes of a corrosion-resistant zirconium metal alloy (zircaloy) which are backfilled with helium to aid heat conduction and detect leakages. The finished fuel rods are grouped in fuel assemblies, called fuel bundles that are then used to build the core of the reactor. As a safety measure PWR designs do not contain enough fissile uranium to sustain a prompt critical chain reaction (i.e., sub-stained only by prompt neutrons). Avoiding prompt criticality is important as a prompt critical chain reaction could very rapidly produce enough energy to damage or even melt the reactor. A typical PWR has fuel assemblies of 200 to 300 rods each, and a large reactor would have about 150-250 such assemblies with 80-100t of uranium in all. Refueling for most commercial PWRs is on an 18-24 month cycle. Approximately one third of the core is replaced each refueling.

Boron and control rods are used to maintain primary system temperature at the desired point. In order to decrease power, the operator throttles shut turbine inlet valves. This would result in less steam being drawn from the steam generators. This results in the primary loop increasing in temperature. The higher temperature causes the reactor to fission less and decrease in power. The operator could then add boric acid and/or insert control rods to decrease temperature to the desired point.

Reactivity adjustments to maintain 100% power as the fuel is burned up in most commercial PWR's is normally controlled by varying the concentration of boric acid dissolved in the primary reactor coolant. The boron readily absorbs neutrons and increasing or decreasing its concentration in the reactor coolant will therefore affect the neutron activity correspondingly. An entire control system involving high pressure pumps, usually called the charging and letdown system, is required to remove water from the high pressure primary loop and re-inject the water back in with differing concentrations of boric acid. The reactor control rods, inserted through the top directly into the fuel bundles, are normally only used for power changes [5].

One disadvantage of PWR is that the coolant water must be highly pressurized to remain liquid at high temperatures. This requires high strength piping and a heavy reactor pressure vessel and hence increases construction costs [6]. The higher pressure can increase the consequences of a loss of coolant accident (LOCA), following shutdown of the primary nuclear reaction, the fission products continue to generate decay heat at initially roughly 7% of full power level, which requires 1 to 3 years of water pumped cooling. If cooling fails during this post-shutdown period, the reactor can still overheat and meltdown. Upon LOCA the decay heat can raise the rods above 2200^{o}C [7], where upon the hot zircaloy used for casing the nuclear fuel rods spontaneously explodes in contact with the cooling water or steam, which leads to the separation of water into its constituent elements (hydrogen and oxygen). In this event there is a high danger of hydrogen explosions, threatening structural damage and the exposure of highly radioactive stored fuel rods in the vicinity outside the plant in pools.

3.2 Boiling water reactor

The boiling water reactor (BWR) is characterized by two-phase fluid flow (water and steam) in the upper part of the reactor core. Light water (i.e., common distilled water) is the working fluid used to conduct heat away from the nuclear fuel. The water around the fuel elements also thermalizes neutrons, i.e., reduces their kinetic energy, which is necessary to improve the probability of fission of fissile fuel. Fissile fuel material, such as the ^{235}U and ^{239}Pu isotopes, has large capture cross sections for thermal neutrons [4].

In BWRs the steam going to the turbine that powers the electrical generator is produced in the reactor core rather than in steam generators or heat exchangers. This design has many similarities to the PWR, except that there is only a single circuit in which the water is at lower pressure at about 75bar so that it boils in the core at about 285°C. The reactor is designed to operate with 12-15% of the water in the top part of the core as steam, and hence with less moderating effect and thus efficiency [3].

A BWR can be designed with no recirculation pumps and rely entirely on the thermal head to recirculation the water inside of reactor pressure vessel however, the forced recirculation head from the recirculation pumps is very useful in controlling power. The thermal power level is easily varied by simply increasing or decreasing the forced recirculation flow through the recirculation pumps. Reactor power is controlled via two methods, (a) by inserting or withdrawing control rods and (b) by changing the water flow through the reactor core. Since the water around the core of a reactor is always contaminated with traces of radio nuclides, it means that the turbine must be shielded and radiological protection provided during maintenance. Most of the radioactivity in the water is very short-lived so the turbine hall can be entered soon after the reactor is shut down [8].

The BWR reactor core continues to produce heat from radioactive decay after the fission reactions have stopped, making nuclear meltdown possible in the event that all safety systems have failed and the core does not receive coolant. A BWR has a negative void coefficient, that is, the thermal output decreases as the proportion of steam to liquid water increases inside the reactor. A sudden increase in BWR steam pressure (caused, for example, by a blockage of steam flow from the reactor) will result in a sudden decrease in the proportion of steam to liquid water inside the reactor. The increased ratio of water to steam will lead to increased neutron moderation, which in turn will cause an increase in the power output of the reactor. Because of this effect in BWRs, operating components and safety systems are designed to ensure that no credible, postulated failure can cause a pressure and power increase that exceeds the safety systems' capability to quickly shutdown the reactor before damage to the fuel or to components containing the reactor coolant can occur. In the event of an emergency that disables all of the safety systems, each reactor is surrounded by a containment building designed to seal off the reactor from the environment [4].

A modern BWR fuel assembly comprises 74 to 100 fuel rods, and there are up to approximately 800 assemblies in a reactor core, holding up to approximately 140t of uranium. The secondary control system involves restricting water flow through the core so that steam in the top part means moderation is reduced [8].

3.3 Pressurized heavy water reactor

The pressurized heavy water reactor (PHWR) or CANDU reactor design has been developed since the 1950s in Canada. The acronym CANDU stands for Canada deuterium uranium. All current power reactors in Canada are of the CANDU type. It uses natural uranium (0.7% ^{235}U) oxide as fuel, hence needs a more efficient moderator, such as, D_2O [3, 9].

The coolant is kept under high pressure to raise its boiling point and avoid significant steam formation in the core. The hot D_2O generated in this primary cooling loop is passed into a heat exchanger heating light water in the less-pressurized secondary cooling loop. The generated steam drives a conventional turbine with a generator for power generation [9].

The moderator is in a large tank called a calandria, penetrated by several hundred horizontal pressure tubes which form channels for the fuel, cooled by a flow of D_2O under high pressure in the primary cooling circuit, reaching 290°C. Traditional designs using light water as a moderator will absorb too many neutrons to allow a chain reaction to occur in natural uranium due to the low density of active nuclei. D_2O absorbs fewer neutrons than light water, allowing a high neutron economy that can sustain a chain reaction even in unenriched fuel. Also, the low temperature of the moderator (below the boiling point of water) reduces changes in the neutrons' speeds from collisions with the moving particles of the moderator (neutron scattering). The neutrons therefore are easier to keep near the optimum speed to cause fissioning, therefore, they have good spectral purity. At the same time, they are still somewhat scattered, giving an efficient range of neutron energies [3, 4].

The large thermal mass of the moderator provides a significant heat sink that acts as an additional safety feature. If a fuel assembly were to overheat and deform within its fuel channel, the resulting change of geometry permits high heat transfer to the cool moderator, thus preventing the breach of the fuel channel, and the possibility of a meltdown. Furthermore, because of the use of natural uranium as fuel, this reactor cannot sustain a chain reaction if its original fuel channel geometry is altered in any significant manner.

The central functionality behind the CANDU design is D_2O moderation and on-line refueling, which permits a range of fuel types to be used, including natural uranium, enriched uranium, thorium, and used fuel from light water reactors (LWRs). Significant fuel cost savings can be realized if the uranium does not have to be enriched, but simply formed into ceramic natural UO_2 fuel. This saves not only on the construction of an enrichment plant, but also on the costs of processing the fuel. However, some of this potential savings is offset by the initial, one time cost of the D_2O. The D_2O required must be more than 99.75% pure and tones of this are required to fill the calandria and the heat transfer system [9].

CANDU reactors do have some drawbacks. D_2O generally costs hundreds of dollars per kilogram, though this is a trade-off against reduced fuel costs. It is also notable that the reduced energy content of natural uranium as compared to enriched uranium necessitates more frequent replacement of fuel, which is normally accomplished by use of an on-power refueling system. The increased rate of fuel movement through the reactor also results in higher volumes of spent fuel than in reactors employing enriched uranium. However, as the unenriched fuel was less reactive, the heat generated is less, allowing the spent fuel to be stored much more compactly [10].

3.4 Graphite moderated reactors

Gas cooled reactors (GCR) and advanced gas cooled reactors (AGR) use carbon dioxide (CO_2) as the coolant to carry the heat to the turbine, and graphite as the moderator. Like D_2O, a graphite moderator allows natural uranium, usually in GCR or slightly enriched uranium, usually in AGR, to be used as fuel [3, 4].

3.4.1 Advanced gas cooled reactor

The advanced gas cooled reactor (AGR) reactor is a British design generation II GCR, using graphite moderator and CO_2 as coolant. The mean temperature of the hot coolant leaving the reactor core was designed to be 648°C. In order to obtain these high temperatures, yet ensure useful graphite core life (graphite oxidises readily in at high temperature) a re-entrant flow of coolant at the lower boiler outlet temperature of 278°C is utilised to cool the graphite, ensuring that the graphite core temperatures do not vary too much. The superheater outlet temperature and pressure are designed to be 170bar and 543°C. The fuel is UO_2 pellets, enriched to 2.5-3.5%, in stainless steel tubes. The original design concept of the AGR was to use a beryllium based cladding. When this proved unsuitable, the enrichment level of the fuel was raised to allow for the higher neutron capture losses of stainless steel cladding. This significantly increased the cost of the power produced by an AGR. The CO_2 circulates through the core, reaching 650°C and then past steam generator tubes outside it, but still inside the concrete and steel reactor pressure vessel. Control rods penetrate the moderator and a secondary shutdown system involves injecting nitrogen to the coolant.

The AGR was designed to have a high thermal efficiency of about 41%, which is better than modern PWRs which have a typical thermal efficiency of 34%. This is due to the higher coolant outlet temperature of about 640°C practical with gas cooling, compared to about 325°C for PWRs. However the reactor core has to be larger for the same power output, and the fuel burn-up ratio at discharge is lower so the fuel is used less efficiently, countering the thermal efficiency advantage. AGRs are designed to be refueled without being shut down first. This on-load refueling is an important part of the economic case for choosing the AGR over other reactor types [11].

3.4.2 Water cooled light water graphite moderated reactor

The light water graphite moderated reactor (RBMK) is a Soviet design, developed from plutonium production reactors. It employs long vertical pressure tubes running through graphite moderator, and is cooled by water, which is allowed to boil in the core at 290°C, much as in a BWR. Fuel is low-enriched UO_2 made up into fuel assemblies 3.5m long. With moderation largely due to the fixed graphite, excess boiling simply reduces the cooling and neutron absorption without inhibiting the fission reaction and a positive feedback problem can arise [3, 4].

It is estimated that about 5.5% of the core thermal power is in the form of graphite heat. About 80-85% of this heat is removed by the fuel rod coolant channels, via the graphite rings. The rest of the heat is removed by the control rod channel coolant. The gas circulating in the reactor plays the role of assisting the heat transfer to the coolant channels. There are 1661 fuel channels and 211 control rod channels in the reactor core. The fuel assembly is suspended in the fuel channel on a bracket, with a seal plug. The seal plug has a simple design, to facilitate its removal and installation by the remotely controlled refueling machine. The fuel channels may, instead of fuel, contain fixed neutron absorbers or be empty and just filled with the cooling water. The small clearance between the pressure channel and the graphite block makes the graphite core susceptible to damage. If the pressure channel deforms, e.g., by too high internal pressure, the deformation or rupture can cause significant pressure loads to the graphite blocks and lead to their damage, and possibly propagate to neighboring channels.

The fuel pellets are made of UO_2 powder sintered with a suitable binder into barrels. The material may contain added europium oxide as a burnable nuclear poison to lower the reactivity differences between a new and partially spent fuel assembly. To reduce thermal expansion issues and interaction with the cladding, the pellets have hemispherical indentations. The enrichment level is 2% (0.4% for the end pellets of the assemblies). Maximum allowable temperature of the fuel pellet is 2100°C. The rods are filled with helium at 5bar and hermetically sealed. Retaining rings help to seat the pellets in the center of the tube and facilitate heat transfer from the pellet to the tube. The pellets are axially held in place by a spring. Each rod contains 3.5kg of fuel pellets. The fuel rods are 3.64m long, with 3.4m of that being the active length. The maximum allowed temperature of a fuel rod is 600°C. The fuel assemblies consist of two sets of 18 fuel rods. The rods are arranged along the central carrier rod and held in place with 10 stainless steel spacers separated by 360mm distance. The two sub-assemblies are joined with a cylinder at the center of the assembly and during the operation of the reactor, this dead space without fuel lowers the neutron flux in the central plane of the reactor [12].

3.5 Fast breeder reactors

As of 2006, all large-scale fast breeder reactor (FBR) power stations have been liquid metal fast breeder reactors (LMFBR) cooled by liquid sodium. These have been of one of two designs (a) Loop type, in which the primary coolant is circulated through primary heat exchangers outside the reactor tank, but inside the biological shield due to radioactive sodium-24 (^{24}Na) in the primary coolant and (b) Pool type, in which the primary heat exchangers and pumps are immersed in the reactor tank.

All current FBR designs use liquid metal as the primary coolant, to transfer heat from the core to steam used to power the electricity generating turbines. FBRs have been built cooled by liquid metals other

than sodium (some early FBRs used mercury), other experimental reactors have used a sodium-potassium alloy. Both have the advantage that they are liquids at room temperature, which is convenient for experimental rigs but less important for pilot or full scale power stations. Lead and lead-bismuth alloy have also been used. FBRs usually use a mixed oxide fuel core of up to 20% plutonium dioxide (PuO_2) and at least 80% UO_2. Another fuel option is metal alloys, typically a blend of uranium, plutonium, and zirconium (used because it is transparent to neutrons). Enriched uranium can also be used on its own.
In many designs, the core is surrounded in a blanket of tubes containing non-fissile uranium-238 (^{238}U) which, by capturing fast neutrons from the reaction in the core, is converted to fissile ^{239}Pu (as is some of the uranium in the core), which is then reprocessed and used as nuclear fuel. Other FBR designs rely on the geometry of the fuel itself (which also contains ^{238}U), arranged to attain sufficient fast neutron capture. The ^{239}Pu (or the fissile ^{235}U) fission cross-section is much smaller in a fast spectrum than in a thermal spectrum, as is the ratio between the ^{239}Pu / ^{235}U fission cross-section and the ^{238}U absorption cross-section. This increases the concentration of the ^{239}Pu / ^{235}U needed to sustain a chain reaction, as well as the ratio of breeding to fission. On the other hand, a fast reactor needs no moderator to slow down the neutrons at all, taking advantage of the fast neutrons producing a greater number of neutrons per fission than slow neutrons. For this reason ordinary liquid water, being a moderator as well as a neutron absorber is an undesirable primary coolant for fast reactors. Because large amounts of water in the core are required to cool the reactor, the yield of neutrons and therefore breeding of ^{239}Pu are strongly affected. Theoretical work has been done on reduced moderation water reactors, which may have a sufficiently fast spectrum to provide a breeding ratio slightly over 1. This would likely result in an unacceptable power derating and high costs in an liquid water cooled reactor, but the supercritical water coolant of the supercritical water reactor (SCWR) has sufficient heat capacity to allow adequate cooling with less water, making a fast-spectrum water cooled reactor a practical possibility. In addition, a D_2O moderated thermal breeder reactor, using thorium to produce uranium-233 (^{233}U), is also possible [13].

3.6 Aqueous homogeneous reactor
Aqueous homogeneous reactor (AHR) is a type of nuclear reactor in which soluble nuclear salts, which are usually uranium sulfate or uranium nitrate, are dissolved in water. The fuel is mixed with the coolant and the moderator, thus the name homogeneous. The water can be either D_2O or light water, both which need to be very pure. A D_2O AHR can achieve criticality (turn-on) with natural uranium dissolved as uranium sulfate. Thus, no enriched uranium is needed for this reactor. The D_2O versions have the lowest specific fuel requirements (least amount of nuclear fuel is required to start them). Even in light water versions less than 0.454kg of ^{239}Pu or ^{233}U is needed for operation. Neutron economy in the D_2O versions is the highest of all reactor designs.
Their self-controlling features and ability to handle very large increases in reactivity make them unique among reactors, and possibly safest. AHRs were sometimes called water boilers, although they are not boiling water reactors. They seem to be boiling their water, but in fact this bubbling is from the production of hydrogen and oxygen as the radiation, and especially the fission particles, dissociate the water into its constituent gases. Corrosion problems associated with sulfate base solutions limited their application as breeders of ^{233}U fuels from thorium. Current designs use nitric acid base solutions (e.g., uranyl nitrate) eliminating most of these problems in stainless steels [14].

4. Generation III reactors
Generation III reactors have emerged through the '90's, with evolutionary designs, they are the evolution of generation II, as illustrated in Figure 1, with significant advances in terms of safety and economics resulting in near-term deployment in several countries. Some are evolutionary from the generation II PWR, BWR and CANDU designs, and some designs are more radical. The former include the advanced boiling water reactor (ABWR), two of which are now operating with others under construction. The best-known radical new design is the pebble bed modular reactor (PBMR), which uses helium as coolant at very high temperature to drive a turbine directly. Generation III reactors are undergoing deployment and will be doing so up to the arrival of generation IV reactors after 2030. Table 2 tabulates the various generation III reactors designs found in the literature and Table 3 provides the associated capital cost estimates based on various projects around the world. Generation III reactors have (a) a standardized design for each type to expedite licensing, reduce capital cost and reduce construction time, (b) a simpler and more rugged design, making them easier to operate and less vulnerable to operational upsets, (c) higher availability and longer operating life, typically 60 years, (d) reduced possibility of core melt

accidents, (e) minimal effect on the environment, (f) higher burn-up to reduce fuel use and the amount of waste and (g) burnable absorbers to extend fuel life. The greatest departure from generation II designs are the passive or inherent safety features that require no active controls or operational intervention to avoid accidents in the event of malfunction, and may rely on gravity, natural convection or resistance to high temperatures [15].

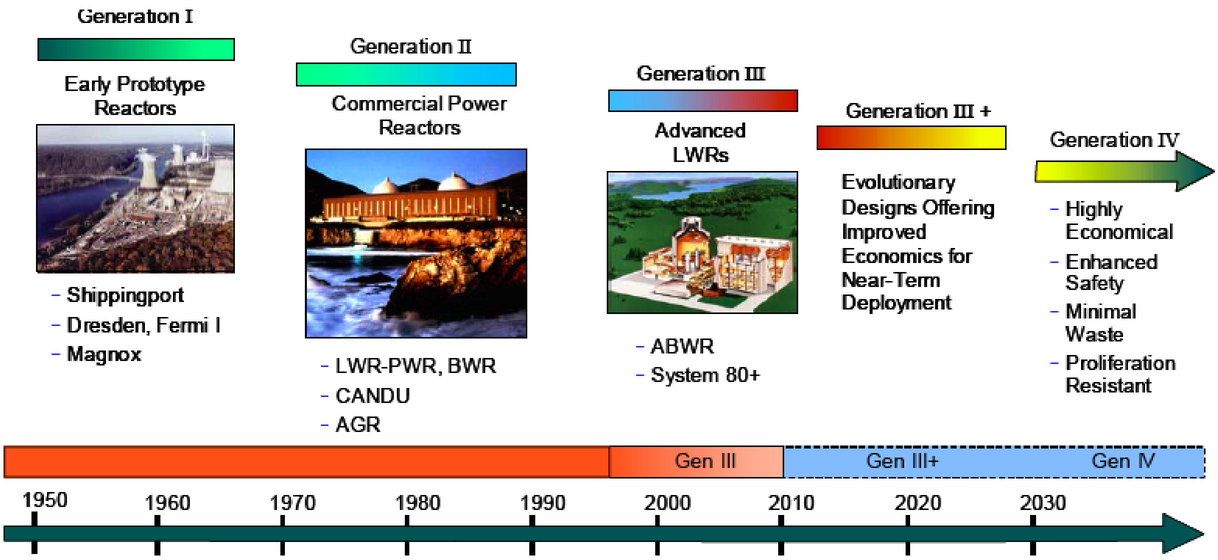

Figure 1. Nuclear reactors evolution

Table 2. Generation III reactors designs

No.	Reactor	Capacity (MWe)	Power cycle
	ALWR Advanced light water reactors		
1	EPR European pressurized water reactor	1600-1750	Rankine
2	ABWR Hitachi	600-1700	Rankine
3	ESBWR Economic simplified boiling water reactor	1390-1550	Rankine
4	APWR Advanced pressurized water reactor	1500	Rankine
5	BWR 90+	1500	Rankine
6	VVER-448	1500	Rankine
7	APR - 1400 Advanced pressurized water reactor 1400 (System 80+)	1400	Rankine
8	ABWR Advanced boiling water reactor	1300	Rankine
9	SWR-1000 Siedewasser boiling water reactor	1000-1290	Rankine
10	AP1000 Advanced passive 1000	1100	Rankine
11	VVER-91	1000	Rankine
12	V-392	950	Rankine
13	VVER-640	640	Rankine
14	VPBER-600	600	Rankine
15	AP600 Advanced passive 600	600	Rankine
16	IRIS International reactor innovative and secure	335	Rankine
17	MSBWR Modular simplified boiling water reactor (under development)	50 & 200	Rankine
18	IRIS-50 International reactor innovative and secure (under development - GIII+)	>50	Rankine
19	KLT-40	30-35	Rankine
20	TRIGA power system (pressurized water reactor)	16,4	Rankine
21	VBER-150	110	Rankine
22	VBER-300	295	Rankine
23	VK-300 (under development - boiling water reactor)	250	Rankine
24	ABV (under development - pressurized water reactor)	10-12	Rankine

Table 2. (Continued)

No.	Reactor	Capacity (MWe)	Power cycle
25	CAREM (under development)	27	Rankine
26	SMART system - integrated modular advanced reactor	110	Rankine
27	MRX (under development)	30	Rankine
28	NP-300	100-300	Rankine
29	NHR-200	N/A	Rankine
	PHWR's Pressurized heavy water reactors		
30	CANDU-9 Canadian deuterium uranium	925-1300	Rankine
31	ACR-1000 Advanced CANDU reactor 1000 hybrid PHWR/PWR	1100-1200	Rankine
32	CANDU-X	350-1150	Supercritical Rankine
33	AHWR Advanced heavy water reactor	300	Rankine
34	ACR-700 Advanced CANDU reactor 700 hybrid PHWR/PWR	750	Rankine
	HT GCR's High temperature gas cooled reactors		
35	GTHTR Gas turbine high temperature reactor	300	Brayton
36	GT- MHR Gas turbine modular helium reactor	285	Brayton
37	HTR-PM High temperature pebble bed gas cooled reactor	195	Rankine
38	PBMR Pebble bed modular reactor	165	Brayton
39	HTTR High temperature test reactor	N/A	Rankine/ Brayton
	Fast neutron reactors (Liquid metal cooled fast reactors)		
40	Super PRISM	2280	N/A
41	BN-800	880	N/A
42	BN-600	600	N/A
43	FBR	500	N/A
44	BREST	300	Rankine
45	BN-350	350	N/A
46	STAR Secure transportable autonomous reactor	N/A	Brayton
47	PRISM Liquid metal cooled	150	N/A
48	SVBR Lead-bismuth fast reactor	75-100	Rankine
49	SSTAR Small sealed transportable autonomous reactors	10-100	Brayton
50	LSPR Lead-bismuth cooled reactor	53	Rankine
51	ENHS Encapsulated nuclear heat source	50	Rankine
52	4S Super safe, small & simple, nuclear battery	10 & 50	Rankine
53	Rapid-L (under development)	0.2	Rankine
	MSRs Molten salt reactors		
54	AHTR Advanced high temperature reactor	1000	Brayton
55	FUJI MSR	100	Brayton

Generally, modern small nuclear reactors for power generation are expected to have greater simplicity of design, economy of mass production and reduced siding costs. Many are also designed for a high level of passive or inherent safety in the event of malfunction. Some are conceived for areas away from transmission grids and with small loads, others are designed to operate in clusters in competition with large units. Generation III+ designs are generally extensions of the generation III concept which include advanced passive safety features. These designs can maintain the safe state without the use of any active control components [16].

Table 3. Capital cost estimates of generation III nuclear reactors

No	Reactor	Capacity (MWe)	Cost (US$/kW)*	Ref
1	EPR (Olkilmoto 3)	2x860	3341	[18]
2	EPR (Flamanville 3)	1600	3203	[18]
3	ABWR (Hitachi/Toschiba GE KK-6)	1315	2974	[19]
4	ABWR (Hitachi/Toschiba GE KK-7)	1315	2686	[19]
5	ESBWR (GE)	1560	1160-1250	[20]
6	APWR (Mitshubishi)	2x1700	1529	[21]
7	BWR 90+ (Westinghouse)	1650	1400	[22]
8	VVER-1500/V448	1500	1200	[23]
9	APR-1400 (South Korea)	1450	1400	[24]
10	ABWR (GE)	1326	1390	[25]
11	SWR-1000	1000-1290	1800	[26]
12	AP-1000 (Westinghouse Electric)	1100	1000	[25]
13	AP-1000 (Westinghouse)	1100	1200	[24]
14	VVER-91 (China)	2x1060	1245-1831	[27]
15	VVER-1000/V392 (Koodankulam)	2x1000	1500	[28]
16	VVER-640	645	1980	[29]
17	AP-600 (Westinghouse Electric)	600	1420	[25]
18	AP-600	600	166	[24]
19	IRIS	335	1000-1200	[24]
20	MSBWR	50	1950	[30]
21	IRIS-50	50	1950	[30]
22	KLT-40 (Severodrinsk)	2x40	4213	[31]
23	VBER-300	300	331	[32]
24	VK-300 (MED)	2x200	1140	[33]
25	ABV	2x38	3158	[32]
26	CAREM	300	1000	[34]
27	SMART	2x100	1615	[35]
28	SMART	1000	1800	[36]
29	SMART (South Korea)	65	6458	[37]
30	NP-300 (Technicatome) (MED)	300	442	[38]
31	NHR-200 (China)	200	552	[36]
32	CANDU-9 (Darlington)	4x881	3973	[39]
33	ACR-1000	1200	1000	[40]
34	AHWR	300	1176-1411	[41]
35	ACR-700	681	1000	[25]
36	GT-HTR (JAERI)	300	1300-1700	[16]
37	GT-MHR	288	972	[25]
38	GT-MHR (GA+Afrikantov)	285	1000	[16]
39	HTR-PM (Huaneng)	200	1500	[16]
40	PBMR (Escon)	165	108	[16]
41	Super PRISM	2280	1300	[42]
42	BN-800	800	1875	[43]
43	BN-600	560	10714	[44]
44	FBR	1250	4800	[45]
45	SVBR	75/100	661.5	[34]
46	4S (Toshiba+Criebi)	10 & 50	2500	[16]
47	AHTR	1000	1000	[16]

*Exchange rates used: 1 Japanese yen = 0.009593 US$, 1 Euro = 1.5531 US$, 1 tenge= 0.0082843 US$, 1 korean Won= 0.0009763 US$, 1 crore= 10000000 Indian Rupee= 235127.93 US$.

4.1 Mitsubishi advanced pressurized water reactor

The Mitsubishi advanced pressurized water reactor (APWR) is a generation III nuclear reactor developed by Mitsubishi Heavy Industries based on PWR technology. It features several design enhancements

including a neutron reflector, improved efficiency and improved safety systems, including a combination of passive and active systems. The core is surrounded by a steel neutron reflector which increases reactivity. In addition, the APWR uses more advanced steam generators compared to the PWR, which creates drier steam allowing for higher efficiency and more delicate turbines. This leads to an almost 10% efficiency increase compared to the PWR.
Several safety improvements are also notable. The safety systems have enhanced redundancy, utilizing 4 trains each capable of supplying 50% of the needed makeup water instead of 2 trains capable of 100%. Also, more reliance is placed on the accumulators which have been redesigned and increased in size. The improvements in this passive system have led to the elimination of the safety injection system, which is an active system [17].

4.2 Advanced light water reactors
The advanced light water reactors (ALWR) incorporate all of the improved features of generation III but have no real difference in terms of their operation between their generation II counterparts. The major improvements are notably in the field of safety and improved economics [15, 46].

4.2.1 European pressurized reactor
The main design objectives of the European pressurized reactor (EPR) design are increased safety while providing enhanced economic competitiveness through evolutionary improvements to previous PWR designs scaled up to an electrical power output of 1600MWe. The reactor can use 5% enriched UO_2 or uranium plutonium mixed oxide fuel. The EPR design has several active and passive protection measures against accidents [18], such as, four independent emergency cooling systems, each capable of cooling down the reactor after shutdown (i.e., 300% redundancy), leak tight container around the reactor, extra container and cooling area if a molten core manages to escape the reactor. Ex-vessel cooling - corium catcher and two-layer concrete wall with total thickness 2.6m, designed to withstand impact by airplanes and internal over pressure.

4.2.2 Economic simplified boiling water reactor
The economic simplified boiling water reactor (ESBWR) is a passively safe generation III+ reactor which builds on the success of the ABWR. Both are designs by General Electric Hitachi Nuclear Energy (GEH), and are based on their BWR design. The ESBWR uses natural circulation with no recirculation pumps or their associated piping. The passive safety systems in an ESBWR operate without using any pumps at all, thereby further increasing design safety integrity and reliability, while simultaneously reducing overall reactor cost. The technology also uses natural circulation for coolant recirculation within reactor pressure vessel, therefore, there are no recirculation pumps and none of the associated piping, power supplies, heat exchangers and instrumentation and controls.
ESBWR's passive safety systems include a combination of systems that allow for the efficient transfer of decay heat from the reactor to pools of water outside of containment. These systems utilize natural circulation based on simple laws of physics to transfer the decay heat outside of containment while maintaining water inventory inside the reactor keeping the nuclear fuel submerged in water and adequately cooled.
The core is shorter than conventional BWR plants because of the smaller core flow, which is caused by the natural circulation. There are 1132 bundles and the electric power can reach 1550MWe. Below the vessel, there is a piping structure which allows for cooling of the core during a very severe accident. These pipes divide the molten core and cool it with water flowing through the piping. The probability of radioactivity release to the atmosphere is several orders of magnitude lower than conventional nuclear power plants, and the building cost is 60-70% of other LWRs [47]. The energy production cost is lower than other plants due to lower initial capital cost and lower operational and maintenance cost.

4.2.3 Advanced boiling water reactor
The advanced boiling water reactor (ABWR) is a generation III reactor based on the boiling water reactor. The ABWR was designed by GEH and Toshiba. The ABWR generates electrical power by using steam, which is boiled from water using heat generated by fission reactions within nuclear fuel, to power a turbine connected to a generator. The standard ABWR plant design has a net output of about 1350MWe.

The addition of reactor internal pumps mounted on the bottom of reactor pressure vessel achieves improved performance while eliminating large recirculation pumps in containment and associated large-diameter and complex piping interfaces with reactor pressure vessel. Only the reactor internal pumps motor is located outside of reactor pressure vessel in the ABWR.
A fully digital reactor protection system with redundant digital backups as well as redundant manual backups, ensures a high level of reliability and simplification for safety condition detection and response. This system initiates rapid hydraulic insertion of control rods for shutdown when needed. Fully digital reactor controls, with redundant digital backup and redundant manual backups, allow the control room to easily and rapidly control plant operations and processes. Separate redundant safety and non-safety related digital multiplexing buses allow for reliability and diversity of instrumentation and control. In particular, the reactor is automated for startup (i.e., initiate the nuclear chain reaction and ascent to power) and for standard shutdown using automatic systems only. Of course, human operators remain essential to reactor control and supervision, but much of the busy-work of bringing the reactor to power and descending from power can be automated at operator discretion.
The internal pumps reduce the required pumping power for the same flow to about half that required with the jet pump system with external recirculation loops. Thus, in addition to the safety and cost improvements due to eliminating the piping, the overall plant thermal efficiency is increased. Eliminating the external recirculation piping also reduces occupational radiation exposure to personnel during maintenance [48].

4.2.4 Advanced passive 1000 reactor
The advanced passive 1000 (AP1000) reactor is a two-loop PWR which produces a net of 1117MWe. The safety systems apply passive protection, which is designed to yield such high degree of safety that there is no need for the usual diesel generators, which provide the equipment with power in the case of a loss of electrical supply. In the event of an accident they require little intervention, which reduces the chance of human error and other failures. Safety enhancement is also achieved by using modern, reliable devices. The probability of failures is further decreased by applying the concept of diversity (several different types of systems are used and thus the effect of potential intrinsic failures can be avoided).
The design is less expensive to build partly due to the fact that it uses existing technology. The expense is also reduced by rationalizing technology, which means decreasing not only the number of pipes, wires, and valves necessary, but reducing a number of other components, and therefore reducing cost. Standardization and type-related licensing will also help reduce the time and cost of construction. The safety systems in the AP1000 are passive, relying on things like gravity and natural recirculation rather than active systems such as pumps [49].

4.2.5 KLT-40C reactor
The KLT-40C reactor takes advantage of the experience gained through the operation of the KLT-40 reactors used to provide power for icebreaker propulsion. The KLT-40C utilises a reactor pressure vessel and a loop nuclear steam plant configuration similar to a conventional PWR, incorporating forced reactor coolant circulation at power. Four separate, helical coil, once-through steam generators and four canned reactor coolant pumps are used. No boron is added to the reactor coolant during normal reactor operation. Reactor pressure vessel and steam generators are adjacent to each other and at approximately the same elevation, with concentric piping connections between reactor pressure vessel and the steam generators. The capabilities of removing decay heat from both the primary and secondary system by natural convection has been demonstrated experimentally. The steam generators and the reactor pressure vessel are housed within cavities located in a metal-water shied tank. The air gap between the components and the walls of the metal-water shield tank minimise heat loss to the shielding water during reactor operation. Reactor pressure vessel cavity of the water-shield tank can be flooded under severe accident conditions to prevent pressure vessel melt through. The KLT-40C is provided with a steel containment structure. A secondary structure protects the containment from external events.
The major unique aspect of the KLT-40C nuclear power plant design is the incorporation of two KLT-40C reactors into a comprehensive barge, referred to as the floating power unit. This consists of two principal parts, living quarters and the process section. The living quarters provide all necessary living accommodations for the operating staff. The process section houses the two KLT-40C reactors, the control rooms, all other systems required for the normal operation of the power plants, and spent fuel and radioactive waste storage facilities. Facilities to use the steam produced by the reactors for either

electricity production or process heat application are housed in separate on-shore facilities. Other on-shore facilities include the switchyard, administration building, and the accident management centre [50].

4.2.6 CAREM reactor

The CAREM nuclear power plant has an integrated reactor. The entire high energy primary system-core, steam generators, primary coolant and steam dome is contained inside a single reactor pressure vessel. The flow rate in the reactor primary systems is maintained by natural circulation. The driving force obtained by the differences in the density along the circuit are balanced by friction and form losses, producing a flow rate in the core that allows for sufficient thermal margin to critical phenomena. The coolant acts also as a moderator.

Self-pressurization of the primary system in the steam dome is the result of the liquid vapour equilibrium, at which the core outlet bulk temperature corresponds to saturation temperature at primary pressure. Heaters and sprinklers that are typical of conventional PWR's are eliminated. Twelve identical mini-helical vertical steam generators, of the once-through type are used to transfer heat from the primary to the secondary circuit, producing dry steam at 47bar, with 30°C of superheating. The location of the steam generator above the core induces natural circulation in the primary system.

The secondary system circulates upwards within the tubes, while the primary system does so in counter-current flow (downward circulation). An external shell surrounding the outer coil layer, with an adequate seal guarantees that the entire stream of the primary system flows through the steam generators. As another safety feature, steam generators are designed to withstand the pressure from the primary system up to the steam outlet and water inlet valves in case of loss of secondary pressure. The CAREM plant has a standard steam cycle with a simple design. In accordance with the behavior of once-through boilers, steam is superheated under all plant conditions and no super-heater is needed. Likewise, no blow-down is needed in the steam generators, which reduces waste generation. A single turbine is used, and the exhaust steam at low pressure is condensed in a water cooled surface condenser. The condensate is then pumped and delivered to the full stream polishing system in order to maintain ultra-pure water conditions.

High purity water exiting the polishing system is sent to the low-pressure pre-heater using turbine extraction as a heating medium. The warm water is delivered to the water accumulator in order to perform degassing operations with additional heat using extraction steam. Water is then pumped to the high-pressure pre-heaters (two in tandem using extraction steam) and sent to the steam generators as feed-water, closing the circuit. The CAREM secondary circuit is not a safety-graded system, i.e., the nuclear safety of the plant does not rely on the functioning of the steam circuit.

4.2.7 SMART reactor

The SMART reactor is an integral type power reactor with a rated thermal power of 330MW. It is different from the loop-type reactors due to the arrangement of its primary components. All major primary components, such as core, steam generators, pressurizer, and control element drive mechanisms, and main coolant pumps, are installed in a single reactor pressure vessel. The integrated arrangement of these components enables the elimination of large pipe connections between the components of the primary reactor coolant systems, and thus fundamentally eliminates the possibility of large break LOCAs. This integral arrangement, in turn, becomes a contributing factor to the safety enhancement of the SMART. These innovative and advanced features are adopted in the SMART design to enhance its safety, reliability, performance, and operability. Most of these technologies and design features implemented in the SMART are those that have been well proven through the operation of commercial power reactors, and new features will be proven through various tests.

Twelve identical steam generator cassettes are located on the annulus formed by reactor pressure vessel and the core support barrel. Each steam generator cassette is of once through design with helically coiled tubes wound around the inner shell. The primary reactor coolant flows downward in the shell side of the steam generators tubes, while the secondary feed-water flows upward in the tube side. The secondary feed-water exits the steam generator in a superheated steam condition. For performance and safety, each steam generator cassette consists of six independent modules, and six modules from three adjacent steam generators are then grouped into one nozzle. Three nozzles eventually compose one section. This concept of steam generators grouping minimizes the asymmetric impact of a steam generator section isolation of the reactor system.

An in-vessel self-pressurizing concept is adopted for the pressurizer of the SMART. The pressurizer is located in the upper space of the reactor assembly and is filled with water and nitrogen gas. The concept of the self-pressurizing design eliminates the active mechanisms such as spray and heater. By keeping the average primary coolant temperature constant with respect to power change, the large pressure variation due to power change during normal operation can be reduced. To achieve self-pressurizing, a pressurizer cooler for maintaining a low pressurizer temperature and a wet thermal insulator for reducing heat transfer from the primary coolant are installed.

Main coolant pump is a canned motor pump that does not require any pump seal. This characteristic eliminates a small break LOCA associated with a pump seal failure in the case of a station black out. The SMART has four main coolant pumps installed vertically on reactor pressure vessel annular cover. Each pump is an integral unit consisting of a canned asynchronous 3-phase motor and an axial flow single-stage pump. A common shaft rotating on three radial and one axial thrust bearings connects the motor and pump.

Besides the inherent safety characteristics of the SMART, further safety enhancement is accomplished with highly reliable engineered safety systems. These systems are designed to function passively. The shutdown of the reactor can be achieved by one of two independent systems. The primary shutdown system is the control rods with Ag-In-Cd absorbing material. In the case of the failure of the primary shutdown system, the emergency boron injection system is provided as an active backup. One of the two trains is sufficient to bring the reactor to sub-critical condition [51].

4.2.8 MRX reactor

The MRX reactor is an integral style PWR with a thermal output of about 100MW, designed initially for ship propulsion. Currently, other applications such as desalination and district heating are envisioned. The MRX reactor size can be increased to about 300MWe without significant changes to the design concept. An innovative feature of MRX is a compact steel containment vessel, which surrounds reactor pressure vessel in relatively close proximity. The inter-space between reactor pressure vessel and the containment vessel is water filled, with a nitrogen blanket in the top portion. As reactor pressure vessel and other components that operate at elevated temperature in the inter-space between the containment vessel and reactor pressure vessel are insulated to reduce heat loss, the insulation is protected by a waterproof membrane.

Normal operating pressures in reactor pressure vessel and containment vessel are 120bar and 40bar respectively. The two canned reactor coolant pump motors are each housed in horizontal canisters that project from reactor pressure vessel above the core elevation, which serve to keep the motors isolated from the containment vessel water. Hatches are provided in the containment vessel opposite the pumps to facilitate inspection and maintenance. Although not shown in the submission, it is anticipated that a shield building will be provided to protect the steel containment from external events.

The design of the MRX appears to offer several technical challenges. Among these is designing the steam and feed-water lines to accommodate thermal expansion and seismic loads within the limited space available, and the establishment and maintenance of the waterproof cladding over the insulation that is applied to reactor pressure vessel, steam lines, feed water lines and other components that operate at elevated temperatures. There appears to be a requirement for several pressure relief systems (for example, for reactor pressure vessel, the containment vessel, and the canisters that house the reactor coolant pump motors). In addition, vent lines are needed to equalise the pressures between the insulation-filled cavities and the containment vessel. MRX incorporates pressurizer heaters conceptually similar to those of conventional PWRs. Since the MRX employs a fuel cycle, a fuel design and fuel management systems that are substantially the same as those of modern PWRs of conventional design, the general characteristics regarding proliferation and safeguards application will be similar [50].

4.2.9 NHR-200 reactor

The NHR-200 reactor is a vessel type LWR with an integrated arrangement, natural circulation, self-pressurized performance and dual vessel structure. The core is located at the bottom of reactor pressure vessel. Primary heat exchangers are arranged on the periphery in the upper part of reactor pressure vessel. The system pressure is maintained by inert gas and steam. A containment vessel fits tightly around reactor pressure vessel, so that the core will not become uncovered under any postulated leakage at the reactor coolant pressure boundary. There is a long riser on the core outlet to increase the natural circulation capacity. The primary coolant absorbs the heat from the reactor core, then passes through the riser and enters the primary heat exchangers, where the heat carried is then transferred to the intermediate

circuit. An integrated arrangement is adopted to decrease the possibility of LOCA. All main parts of the primary circuit are contained in reactor pressure vessel.
Reactor pressure vessel is the pressure boundary of the reactor cooling. All in-vessel penetrations (only with a small diameter) are located on the upper part of reactor pressure vessel. The reactor core of the NHR-200 consists of 120 assemblies (fuel ducts) and 32 control rods. The reactor core stands on the lattice-support structure, which is fixed on reactor pressure vessel. The fuel bundle is arranged in a 12x12 matrix. The cruciform type control rods are placed in the gaps between the square ducts. There are 3 kinds of enrichments in the initial loading, 1.8%, 2.4% and 3% of UO_2. Gadolinium oxide (Gd_2O_3) is used as a burnable poison to control the reactivity along with the boron carbide (B_4C) control rods. This result is in a negative temperature coefficient of reactivity over the complete core life. A low core power density enhances thermal reliability during normal and accidental operating conditions. This simplifies greatly the refueling equipment and eliminates the necessary space in the reactor building. A new type of hydraulic driving facility is used for driving the control rod in the NHR-200. In the drive system the reactor coolant is the actual medium. The water is pumped into the step-cylinders of which the movable parts contain the neutron absorber. A pulsed flow is generated by controlling magnetic valves in the control unit, and it moves the movable part of the step-cylinder step by step. The drive system is very simple both in structure and its design on the fail-safe principle, i.e., all control rods will drop into the reactor core by gravity under loss of electric power, depressurization, and postulated breaks in its piping systems and pump shut down events.
Six sets of primary heat exchangers are located on the periphery of reactor pressure vessel upper part. The triangular pitch, U-tube-shaped and vertically placed bundles are adopted for easy onsite repair. The coolant enters the upper plenum of the exchangers, and then is divided into two streams to flow downward in the tubes. The flow distribution baffles are installed to make optimum heat transfer efficiency. The operating temperature of primary heat exchangers is 210^{o}C and the operating pressure is 30bar. The initial core is divided into 4 fuel regions and contains 120 fuel assemblies. Thirty assemblies are refueled at one time. The spent fuel is then moved into spent fuel racks around the active core and stored there. This design greatly simplifies the refueling equipment and eliminates the necessary space in the reactor building. A second pressure vessel made of steel is fitted tightly around reactor pressure vessel as a guard vessel so that the core will not be uncovered under any postulated coolant leakage within reactor pressure vessel. To keep the desalination system free from radioactive contamination, an intermediate circuit is needed in the NHR-200 and its operating pressure is kept higher than that in the primary circuit.
Safety of the NHR-200 is provided through two key mechanisms, by the development of the plant self-protection features and by the creation of a multi-barrier system of functional and physical protection (defense-in-depth). A number of advanced features have been incorporated into the NHR design to achieve its safety goal [51].

4.2.10 B&W mPower
The B&W mPower is a proposed 125MWe modular ALWR. The reactor's power output is approximately 125MWe. The reactor's design includes an underground containment facility that would store all of the spent fuel the reactor would use during its expected 60 year operating lifetime. The reactor has a core that can be completely removed in a single evolution, and completely replaced in a second separate evolution, making the core nearly plug and play, unlike the reactors of today, which require fuel handling and movement of individual fuel rods during a refueling outage. The entire used core, once removed, can be placed in storage in the conveniently located spent fuel pool next to the integral reactor vessel (IRV) in the containment, which is designed to hold an entire 60 years worth of used fuel, and is accessible by the convenient containment gantry crane located above the IRV within the containment.
Like a BWR, the mPower's primary coolant and moderator is highly purified water, further, hookups for a reactor water cleanup system are specified in the design to ensure that primary system water remains at the highest level of purity. Similar to the ABWR, the mPower has numerous integral coolant recirculation pumps included inside the IRV. However, like the PWR, the mPower uses PWR fuel and a PWR style entry from top of core control rod scheme, and like a PWR, it retains nearly all of its coolant in liquid phase.
Though the mPower has certain similarities to LWR and ALWR, it is designed with numerous major advances. The pressurizer and steam generator, along with all primary coolant loop piping and appurtenances, is omitted in favor of a wholly integral design for the primary loop inside a single IRV.

Pressure is controlled by the drawing and maintenance of a steam bubble at the top of the IRV and the integral steam generator is a highly advanced once-through steam generator. Control rod drives are designed to not penetrate the IRV, as in the light water reactors of today, but instead be wholly enclosed within the IRV. Shims are omitted in favor of burnable neutron absorbers within the fuel. The mPower is designed so as to produce steam with 28°C of superheat, allowing the steam turbo-generator to run in the superheated regime and avoid the issue of having to deal with low-quality, efficiency-reducing moist steam of the saturated regime.

The mPower is designed so as to make LOCA impossible due to the IRV which contains the entire primary coolant loop within reactor pressure vessel. If secondary cooling is lost, creating an effective loss of standard heat removal, there are water supplies located above and within the containment that can be used to cool the IRV with GDCS. Further advanced means of heat removal can be used in the event that these systems are exhausted, such as by flooding the containment and establishing natural circulation [52].

4.3 Pressurized heavy water reactor

The pressurized heavy water reactor (PHWR) is mainly developed by Canada and India. It is the evolution of the generation II design. In Canada, Atomic Energy of Canada Limited (AECL) has had two designs under development, which are based on its reliable CANDU-6 reactors, the most recent of which are operating in China. Moreover, the CANDU-9 has been developed as to have a flexible fuel needs, ranging from natural uranium through recovered uranium, mixed oxide fuel, to direct spent PWR fuel, to thorium. It may be able to burn military plutonium or actinides separated from reprocessed PWR/BWR waste.

The advanced CANDU 1000 reactor (ACR-1000), a generation III reactor, is more an innovative concept. While retaining the low-pressure D_2O moderator, it incorporates some features of the pressurized water reactor. Adopting light water cooling and a more compact core reduces capital cost, and because the reactor is run at higher temperature and coolant pressure, it has higher thermal efficiency. The advanced CANDU 700 reactor (ACR-700) is physically much smaller, simpler and more efficient as well as 40% cheaper than the CANDU-6. It will run on low-enriched uranium (about 1.5-2.0% ^{235}U) with high burn-up, extending the fuel life by about three times and reducing high-level waste volumes accordingly. Safety is enhanced by negative void coefficient for the first time in CANDU, and it utilises other passive safety features. Units will be assembled from prefabricated modules, eventually cutting construction time to three years. The CANDU X is a variant of the ACR-1000, but with supercritical light water coolant (e.g. 250bar and 625°C) to provide 40% thermal efficiency. India is developing the advanced heavy water reactor (AHWR) in its plan to utilise thorium to fuel its overall nuclear power program. It is designed to be self-sustaining in relation to ^{233}U bred from thorium-232 (^{232}Th) and have a low plutonium inventory and consumption, with slightly negative void coefficient of reactivity [15].

4.3.1 Advanced heavy water reactor

The advanced heavy water reactor (AHWR) is the latest Indian design for a next generation nuclear reactor that will burn thorium in its fuel core. It is slated to form the third stage in India's 3 stage fuel cycle plan. Thorium is an element that is 3 times more abundant globally than uranium. The proposed design of the AHWR is that of a D_2O moderated nuclear power reactor that will be the next generation of the PHWR type. The AHWR is a vertical pressure tube type reactor cooled by boiling light water under natural circulation. A unique feature of this design is a large tank of water on top of the primary containment vessel (PCV), called the GDWP. This reservoir is designed to perform several passive safety functions.

The reactor design incorporates advanced technologies, together with several proven positive features of Indian PHWRs. These features include pressure tube type design, low pressure moderator, on-power refueling, diverse fast acting shut-down systems and availability of a large low temperature heat sink around the reactor core. The AHWR incorporates several passive safety features.

The ECCS injection and containment cooling can act without invoking any active systems or operator action. The reactor physics design is tuned to maximise the use of thorium based fuel, by achieving a slightly negative void coefficient. Fulfilling these requirements has been possible through the use of PuO_2 - ThO_2 mixed oxide fuel and ThO_2 - $^{233}UO_2$ mixed oxide fuel in different pins of the same fuel cluster, and the use of a heterogeneous moderator consisting of amorphous carbon (in the fuel bundles) and D_2O in 80%-20% volume ratio. The core configuration lends itself to considerable flexibility and several feasible solutions, including those not requiring the use of amorphous carbon based reflectors, are possible without any changes in reactor structure [53].

4.3.2 Advanced CANDU reactor 1000

The advanced CANDU reactor 1000 (ACR-1000) is a generation III+ design and is an evolutionary development of existing CANDU reactors designed by AECL. It is a LWR that incorporates features of both PHWR and APWR technologies. It uses a similar design concept to the steam generating heavy water reactor (SGHWR).

The design uses lightly enriched uranium fuel, light water coolant, and a separate D_2O moderator. The reactivity regulating and safety devices are located within the low pressure moderator. The ACR-1000 also incorporates characteristics of the CANDU design, including on-power refueling with the CANFLEX fuel system, a long prompt neutron lifetime, small reactivity holdup, two fast, totally independent, dedicated safety shutdown systems and an emergency core cooling system.

The use of SEU fuel allows the reduction of coolant void reactivity coefficient to a small, negative value. The compact reactor core design reduces core size by half for the same power output over the older design. The current size for the ACR-1000 is approximately 1200Mwe [54].

4.3.3 CANDU X reactor

The CANDU X program at AECL is presently considering reactor designs cooled by supercritical water. There are 5 concepts that have been developed, (generically called modular- SCWR) and these are the focus of current research. Each of these concepts will be developed to the point that a high level cost model can be established, and future work concentrated on the designs that demonstrate the highest economic and safety benefits. The CANDU X is the only reactor design in this group. CANDU X is a D_2O moderated pressure tube style reactor. In this reactor style, the reactor coolant boundary within the core consists of a large number of pressure tubes, surrounded by D_2O moderator and no reactor pressure vessel is used.

The HWR design could evolve in terms of coolant temperature and enthalpy, from conventional pressures and temperatures to supercritical pressures and temperatures. Two stages of development of a supercritical cooled HWR reactor have been studied by AECL with coolant core-mean temperatures near 400°C and 500°C, labeled Mark 1 and Mark 2 respectively. They are based on heavy- or light water coolant at a nominal pressure of 250bar. Mark 1 transfers heat from a D_2O primary system to a light-water secondary system at 190bar and is expected to operate with conventional or near-conventional zircaloy-clad fuel. Mark 2 requires advanced fuel and operates with heavy water to light water, or light water to light water, in an indirect cycle, or with light water in a direct cycle.

Further innovations adopted for reference CANDU X models under consideration include the use of supercritical light water as reactor coolant, and the use of a direct cycle CANDU plants The net output of CANDU X is in the range of 350MWe to 1150MWe (depending on the number of fuel channels used in a specific design). The temperature capability of the CANDU X design is greater than that of current PHWR designs, but well below that of high-temperature gas cooled reactors (HT-CGR) and liquid metal cooled reactors (PRISM). The reactor uses passive safety in four ways (a) passive high temperature channel (no failure in accidents), (b) elimination of consequence of channel flow blockage, (c) natural circulation heat removal wherever possible and (d) passive containment heat removal. The issues of proliferation and safeguards for a CANDU X employing a once-through fuel cycle are not significantly different than those for current CANDU plants [50].

4.4 High-temperature gas cooled reactor

The high-temperature gas cooled reactors (HT-CGR) reactor has re-emerged to become the most promising reactor design. They have derived from several innovative reactors built in the 1960s and 1970s. This design runs on a closed Brayton cycle to drive turbines and produce electricity. It uses helium as a coolant. New HT-CGRs are being developed. They will be capable of delivering high-temperature (up to 950°C) helium either for industrial application via heat exchanger or directly to drive gas turbines for electricity with almost 50% thermal efficiency possible (with efficiency increasing 1.5% with each 50°C increment). The technology developed in the last decade makes HT-CGR more practical than in the past, though the direct cycle means that there must be high integrity of fuel and reactor components.

Fuel for these reactors is in the form of TRISO particles less than a millimetre in diameter. Each has a kernel of uranium oxycarbide, with the uranium enriched up to 14% ^{235}U. This is surrounded by layers of carbon and silicon carbide (SiC), providing containment for fission products, which are stable up to 1600°C or more. There are two ways in which these particles are arranged, (a) in blocks, as hexagonal prisms of graphite, or (b) in billiard ball-sized pebbles of graphite encased in SiC, each with about 15,000 fuel particles and 9g uranium. Both have a high level of inherent safety, including strong negative temperature coefficient whereby fission slows as temperature rises. There is one inconvenience in this reactor design, as it creates a greater amount of spent fuel than from the same capacity LWR [15].

4.4.1 Gas turbine modular helium reactor

The gas turbine modular helium reactor (GT-MHR) is an advanced high temperature gas cooled reactor which is jointly being developed by a consortium including Minatom of Russia, General Atomics, Framatome and Fuji Electric with the goal of burning weapons grade plutonium. It can, however, operate on uranium fuel and be competitive as a stand-alone electricity producer.

The nuclear reactor has a 600MWth core with micro particle fuel included into prismatic fuel elements. In the modular design, the safety of the concept is simplified by the use of natural phenomena such as thermal radiation, which in any event maintains the fuel temperature below the temperature that leads to SiC cladding damage. This ensures that the nuclear material is confined within the fuel all the time. With helium as a coolant, that core is coupled directly to a gas turbine in a Brayton cycle. Helium at 850°C is expanded in a turbine that drives two compressors and an alternator yielding a net electricity production of 285MWe for an efficiency of 47.5%. A special feature of the Brayton cycle, optimised for these operating conditions is the release of heat at the cold source via the pre-cooler and intercooler at more than 100°C. Normally, this heat is released only through a cooling tower or to the river, but with proper adaptation it can be converted to useful heat to be used, for example to heat the feed-water of a multi-effect desalination unit [51].

4.4.2 Pebble bed modular reactor

In South Africa, the national utility, is performing a technical and economic evaluation of a helium cooled pebble bed modular reactor (PBMR) directly coupled to a gas turbine power conversion system. The PBMR nuclear power plant design integrates a helium cooled, graphite-moderated, high temperature reactor of the general type developed in Germany (AVR-15 and THTR-300) with a high-efficiency Brayton cycle gas turbine operating in a direct cycle. The reactor core is housed in a vertical cylindrical reactor pressure vessel located in a below-grade silo, whereas the power conversion equipment is housed in a second vertical cylindrical pressure vessel, which connects to reactor pressure vessel via a short horizontal pressure vessel. This arrangement results in a very compact plant configuration, and the elimination of most of the balance of plant that is associated with water cooled reactor types [50].

A PBMR power plant combines a gas cooled core and a novel packaging of the fuel that dramatically reduces complexity while improving safety. The uranium, thorium or plutonium nuclear fuels are in the form of a ceramic, usually oxides or carbides, contained within spherical pebbles a little smaller than the size of a tennis ball and made of pyrolytic graphite, which acts as the primary neutron moderator. The pebble design is relatively simple, with each sphere consisting of the nuclear fuel, fission product barrier and moderator, which in a traditional water reactor would all be different parts. Simply piling enough pebbles together in a critical geometry will allow for criticality.

The pebbles are held in a vessel and an inert gas, such as helium, nitrogen or CO_2, which circulates through the spaces between the fuel pebbles to carry heat away from the reactor. If helium is used, because it is lighter than air, air can displace the helium if the reactor wall is breached. PBMR need fire-prevention features to keep the graphite of the pebbles from burning in the presence of air although the flammability of the pebbles is disputed. Ideally, the heated gas is run directly through a turbine. However, if the gas from the primary coolant can be made radioactive by the neutrons in the reactor, or a fuel defect could still contaminate the power production equipment, it may be brought instead to a heat exchanger where it heats another gas or produces steam. The exhaust of the turbine is quite warm and may be used to warm buildings or chemical plants, or even run another heat engine.

Much of the cost of a conventional, water cooled nuclear power plant is due to cooling system complexity. These are part of the safety of the overall design, and thus require extensive safety systems and redundant backups. A water cooled reactor is generally dwarfed by the cooling systems attached to it. In contrast, a PBMR is gas cooled, sometimes at low pressures. The spaces between the pebbles form the piping in the core. Since there is no piping in the core and the coolant contains no hydrogen, embrittlement is not a failure concern. The preferred gas, helium, does not easily absorb neutrons or impurities. Therefore, compared to water, it is both more efficient and less likely to become radioactive.

A large advantage of the PBMR over a conventional LWR is in operating at higher temperatures. The reactor can directly heat fluids for low pressure gas turbines. The high temperatures allow a turbine to extract more mechanical energy from the same amount of thermal energy, therefore, the power system uses less fuel per kWh. A significant technical advantage is that some designs are throttled by temperature, not by control rods. The reactor can be simpler because it does not need to operate well at the varying neutron profiles caused by partially withdrawn control rods. For maintenance, many designs include control rods, called absorbers that are inserted through tubes in a neutron reflector around the reactor core. A reactor can change power quickly just by changing the coolant flow rate and can also change power more efficiently by changing the coolant density or heat capacity [55].

PBMR overall thermal process disposes almost all of its waste heat via two large helium gases to buffered de-mineralized water heat exchangers, the pre-cooler and the inter-cooler. The cooling water to these heat exchangers is in turn cooled by a common de-mineralized water to seawater heat exchanger, with a common de-mineralized water circulating pump supplying the pre-Cooler and inter-Cooler in parallel (with approximately equal flows to each). Under full load conditions the helium and water temperatures flows to and from these heat exchangers as shown on the diagram. The PBMR has a power output of about 226MWth/100MWe. The core outlet helium temperature is about 900°C [51].

4.5 Fast neutron reactors

This reactor design is a technological step beyond the conventional power reactors, which offers a more efficient use of uranium resources. The fast reactor has no moderator and uses plutonium as its basic fuel, since it fissions sufficiently with fast neutrons to keep going. At the same time the number of neutrons produced per fission is 25% more than from uranium, this means that there are enough (after losses) not only to maintain the chain reaction but also to convert the ^{238}U in the fertile blanket around the core into fissile plutonium. In other words the fast reactor burns and can breed plutonium. Natural uranium contains about 0.7% ^{235}U and 99.3% ^{238}U. In any reactor the ^{238}U component is turned into several isotopes of plutonium during its operation. Two of these, ^{239}Pu and plutonium-241 (^{241}Pu), then undergo fission in the same way as ^{235}U to produce heat. In a FBR this process can be optimised so that it breeds fuel, though reprocessing of the blanket material is required to recover it. Hence FBR's can utilise uranium at least 60 times more efficiently than a normal reactor. They are however expensive to build and operate, including the reprocessing, and could only be justified economically if uranium prices were to rise to pre-1980 values. Although, there's has been an increase in nuclear fuel price this last years, it still remains to be seen if this will push further development for this reactor design [56].

Fast neutron reactors have a high power density and are normally cooled by liquid metal such as sodium, lead, or lead-bismuth, with high conductivity and boiling point and no moderating effect. Although the design needs to ensure that there is no chemical interaction (e.g. sodium-water), in some respects a liquid metal coolant is more benign overall than very high pressure water, which requires robust engineering on account of the pressure. Also fast reactors have a strong negative temperature coefficient, an inherent safety feature. Several countries have research and development programs for improved FBR. These use the ^{238}U in reactor fuel as well as the fissile ^{235}U isotope used in most reactors.

Today there has been progress on the technical front, but the economics of FBR's still depends on the value of the plutonium fuel that is bred, relative to the cost of fresh uranium. Also there is international concern over the disposal of ex-military plutonium, and there are proposals to use fast reactors for this purpose. In both respects the technology is important to long-term considerations of world energy sustainability [57].

4.5.1 BREST-300 reactor

Based on more than 30-year experience in operation of lead-bismuth cooled submarine reactors in Russia, a number of lead cooled reactor projects of large, medium and small power capacity were developed. A BREST-300 lead cooled fast reactor is one of the most developed medium-power capacity projects. A BREST-300 lead cooled fast reactor with uranium plutonium nitride fuel was developed in a number of Russian design and research institutions to obtain high parameters both for nuclear safety and economic efficiency. It is supposed that the BREST-300 reactor can be used as a heat source for generation of steam with high parameters, as a consumer of plutonium obtained from reprocessing of

spent fuel from thermal and fast reactors or weapons-grade plutonium released in disarmament programs, as well as for burning of actinides and transmutation of long-lived fission products.
Thus, this project can be a basis for solving problems formulated above. According to calculations the reactor power of 300MWe, is minimum for providing core breeding ratio close to 1. The reactor is a pool-type two-circuit steam-generating power unit and includes core, eight once-through steam generators of spiral-tube bundle type, four axial pumps, reloading system, control and safety system, turbine and emergency heat removal system. BREST 300 operates with a breeding ratio of near 1. This precludes the use of uranium blanket surrounding the core or the installation of targets in the core for weapons material production, when the reactor is operated in the design mode. BREST operates on a closed fuel cycle, with plutonium and the other higher actinides being returned to the reactor where they are burned [50].

4.5.2 Super safe, small and simple reactor
Super safe, small and simple reactor (4S) is a small sodium cooled fast reactor (SFR), which concentrates intensive efforts on meeting energy requirements in a region where technical and engineering infrastructures are limited. To meet this objective, 4S is designed on the principles of simple operation, simplified maintenance including refueling, increased safety, and improved economic and proliferation resistant features. 4S is designed to have a long life core with a small diameter surrounded by an annular reflector to control the reactivity depletion due to burning and enhance the core safety. Its lifetime is set at ten years, to eliminate the need for refueling. A co-production 4S plant can continuously produce fresh water for more than 10 years without nuclear refueling. It has also high resistance to nuclear proliferation since there is no need to access to nuclear fuel.
It contains the reactor, secondary systems, steam generator, a coast down control system, a power switchboard and the refueling pits. The plan of the building measures 26m x 16m, requiring only a small ground base. Primary coolant flows out of the core, raises the hot pool and descends in the intermediate heat exchanger through which the heat is transferred to the secondary sodium. It is pressurized by the primary electromagnetic pump at the bottom of the intermediate heat exchanger and flows down along the inner hole of the in-vessel shielding structure. Then it turns at the bottom of reactor pressure vessel and returns to the core. 4S employs a burn-up control system with an annular reflector in place of the control rods and control rod driving mechanisms. This eliminates the need of frequent maintenance services. No replacement of the reflector is required for the entire plant life. The core geometry with a reflector control system has been chosen to meet requirements for negative void reactivity and no refueling for ten years.
The reflector is driven hydraulically at start-up and shutdown. At start-up the reflector is driven upward at a rate of 1mm/sec by the hydraulic pump. The reflector is fixed by the hydraulic and moves up for the burn-up control at a constant speed of 1mm/day by a motor, which is designed so that the reflector is positioned by integration of generated power frequency. For shutdown of the reactor, a scram valve is opened to let the reflector descend at a rate of 10cm/s down to one meter for the sub-critical cold shutdown state. In order to enhance its applicability in developing countries, 4S has a long core life with a single batch fuelling. The reference design has a ten-year life core. A longer core life can be achieved by introducing at the core centre a burnable poison assembly, which contains a mixture of gadolinium, which is poison and zirconium hydride, which is a moderator to soften the spectrum. This reduces the reactivity depletion of the core and extends its life to 30 years. The longer life core enhances proliferation resistance with no need of refueling or processing of plutonium. The driving speed of the reflector is programmed to compensate the balance of reactivity [51].

4.5.3 Integral fast rector
The integral fast reactor (IFR) is a design for a nuclear reactor using fast neutrons and no neutron moderator. IFR is distinguished by a nuclear fuel cycle that uses reprocessing via electro-refining at the reactor site. IFR is cooled by liquid sodium and fueled by an alloy of uranium and plutonium. The fuel is contained in steel cladding with liquid sodium filling in the space between the fuel and the cladding.
In traditional LWRs the core must be maintained at a high pressure to keep the water liquid at high temperatures. In contrast the core could operate at close to ambient pressure, dramatically reducing the danger of a LOCA. The entire reactor core, heat exchangers and primary cooling pumps are immersed in a pool of liquid sodium, making a loss of primary coolant extremely unlikely. The coolant loops are designed to allow for cooling through natural convection, meaning that in the case of a power loss or

unexpected reactor shutdown, the heat from the reactor core would be sufficient to keep the coolant circulating even if the primary cooling pumps were to fail. The IFR also has passive safety advantages as compared with conventional LWRs.

The fuel and cladding are designed such that when they expand due to increased temperatures, more neutrons would be able to escape the core, thus reducing the rate of the fission chain reaction. In other words, an increase in the core temperature will act as a feedback mechanism that decreases the core power. Most LWRs also have negative reactivity coefficients, however, in an IFR, this effect is strong enough to stop the reactor from reaching core damage without external action from operators or safety systems. This was demonstrated in a series of safety tests on the prototype.

Liquid sodium presents safety problems because it ignites spontaneously on contact with air and can cause explosions on contact with water. To reduce the risk of explosions following a leak of water from the steam turbines, the IFR design (as with other sodium cooled fast reactors) includes an intermediate liquid-metal coolant loop between the reactor and the steam turbines. The purpose of this loop is to ensure that any explosion following accidental mixing of sodium and turbine water would be limited to the secondary heat exchanger and not pose a risk to the reactor itself [58].

4.6 Molten salt reactors

A molten salt reactor (MSR) is a type of nuclear fission reactor in which the primary coolant or even the fuel itself is a molten salt mixture. MSRs run at higher temperatures than water cooled reactors for higher thermodynamic efficiency, while staying at low vapor pressure. The ability to operate at near atmospheric pressures reduces the mechanical stress endured by the system, thus simplifying aspects of reactor design and improving safety. MSRs have the potential of being able to be constructed and operated at less cost compared to coal power plants.

The nuclear fuel may be in solid form, or dissolved in the coolant itself. In many designs the nuclear fuel is dissolved in the molten fluoride salt coolant as uranium tetrafluoride (UF_4). The fluid becomes critical in a graphite core which serves as the moderator. Solid fuel designs rely on ceramic fuel dispersed in a graphite matrix, with the molten salt providing low pressure, high temperature cooling. The salts are much more efficient at removing heat from the core, reducing the need for pumping, piping, and reducing the size of the core as these components are reduced in size [59].

4.6.1 FUJI molten salt reactor

The FUJI, a loop fluid fuel type reactor, is the only reactor design in this group. FUJI uses graphite moderator and molten-salt coolant and the fuel is dissolved in the coolant. This design takes advantage of the molten-salt reactor technology developed at Oak Ridge National Laboratories (ORNL) in the United States, and the operating experience of the MSR experiment that was operated by ORNL for 32 months in the late 1960s. The FUJI reactor, with an electrical output of 100MWe, falls within the small reactor size classification of the International Atomic Energy Agency (IAEA).

A unique feature of the MSR is that the fuel, as uranium fluoride (fissile) and thorium fluoride (fertile), is dissolved in the molten-salt coolant. MSR have very strong negative temperature reactivity coefficients due to the negative reactivity temperature coefficient provided by the graphite moderator and due to the reduction in reactivity in the core associated with the reducing density of the molten-salt coolant/fluid fuel in the core that accompanies increasing temperature. The design assures that no materials with moderating capability are located within the vicinity of the reactor vessel. This assures that the molten-salt/fluid fuel cannot achieve criticality outside the core in the event of a LOCA. The FUJI strives to enhance safety by taking advantage of the inherent and passive safety features facilitated by the technology, including the high temperature capability of the molten-salt coolant/fluid fuel and graphite moderator.

The MSR was previously promoted by ORNL as a proliferation-resistant reactor. This resistance is based on the fact that, while uranium isotopes are easily removed from the molten-salt coolant by bubbling fluorine gas through the molten-salt, the fluorine does not remove plutonium and other higher actinides. Removal of the plutonium is technically very difficult. No spent fuel is removed from FUJI during the operating life of the reactor, as plutonium and the other higher actinides produced are consumed in the reactor. Hence, there is no spent fuel available at the site; in addition, unauthorised removal of the molten-salt/fluid fuel from the reactor systems is technically difficult. No fission products (non-gaseous) are removed from the molten-salt coolant/fluid fuel over the operating life of the reactor. The on-power refueling of FUJI necessitates the transport and on-site storage of new fuel (enriched uranium, plutonium, and/or thorium, depending on the fuel cycle employed), probably in the form of a salt. The new fuel salt can include both fissile and fertile material, and be relatively diluted in terms of fissile material content in order to minimise its attractiveness to clandestine operators. There is clearly a trade-off between the frequency at which fuel is transported to the reactor, and the amount of fuel stored at the site [50].

4.7 Advanced reactors
There are also some advanced reactors which are in various stages of development, such as the subcritical reactor, the hydrogen moderated self regulating nuclear power module (HPM) and the clean and environmentally safe advanced reactor (CAESAR).

4.7.1 Subcritical reactor
A subcritical reactor is a nuclear fission reactor that produces fission without achieving criticality. Instead of a sustaining chain reaction, a subcritical reactor uses additional neutrons from an outside source. The neutron source can be a nuclear fusion machine or a particle accelerator producing neutrons by spallation. Such a device with a reactor coupled to an accelerator is called an accelerator driven system. A subcritical reactor can be used to destroy heavy isotopes contained in the used fuel from a conventional nuclear reactor, while at the same time producing electricity. The long-lived transuranic elements in nuclear waste can in principle be fissioned, releasing energy in the process and leaving behind the fission products which are shorter-lived. This would shorten considerably the time for disposal of radioactive waste. However, some isotopes have threshold fission cross sections and have a small effective fraction of delayed neutrons and therefore require a fast reactor for being fissioned and for safety reasons preferably a subcritical reactor if they constitute a significant fraction of the fuel.
Besides nuclear waste incineration, there's interest in this type reactor because they are seen as safer than normal fission reactors. In most critical reactors, the nuclear chain reaction can potentially increase exponentially until the heat destroys the reactor, causing an expensive and potentially dangerous accident. With a subcritical reactor, the reaction will stop automatically unless continually fed neutrons from an outside source. Most current accelerator driven system designs propose a high-intensity proton accelerator with 1GeV energy, directed towards a spallation target made of thorium that is cooled by liquid lead-bismuth in the core of the reactor. In that way, for each proton interacting in the target, an average 20 neutrons are created to irradiate the surrounding fuel. Thus, the neutron balance can be regulated so that the reactor would be below criticality if the additional neutrons by the accelerator were not provided. The main advantage is inherent safety, even if the nuclear fuel under consideration lack uranium's self-regulating properties, like delayed neutrons and Doppler coefficient that make standard nuclear reactors safe. Whenever the neutron source is turned off, the reaction ceases.
There are technical difficulties to overcome before accelerator driven system can become economical and eventually be integrated into future nuclear waste management. The accelerator must provide a high intensity and be highly reliable. There are concerns about the window separating the protons from the spallation target, which is expected to be exposed to stress under extreme conditions. The chemical separation of the transuranic elements and the fuel manufacturing, as well as the structure materials, are important issues. Finally, the lack of nuclear data at high neutron energies limits the efficiency of the design [60].

4.7.2 Hydrogen moderated self regulating nuclear power module
The hydrogen moderated self regulating nuclear power module (HPM) is a new type of nuclear power reactor using hydride as a neutron moderator. The design is inherently safe as the fuel and the neutron moderator are uranium hydride (UH_3), which is with temperature reduced to uranium and hydrogen. The gaseous hydrogen exits the core, being absorbed by hydrogen absorbing material such as depleted uranium, thus making it less critical. This means that with rising temperature the neutron moderation drops and the nuclear fission reaction in the core is dampened, leading to a lower core temperature. This means as more energy is taken out of the core the moderation rises and the fission process is stoked to produce more heat.
The reactor uses UH_3, low-enriched to 5% ^{235}U (the remainder is ^{238}U) as the nuclear fuel, rather than the usual metallic uranium or UO_2 that composes the fuel rods of contemporary LWR. In fact, within the application, the contemporary rod based design with fuel rods and control rods is completely omitted

from the proposed reactor design in favor of a tub design with passive heat pipes conducting heat to the heat exchanger running through the tub of granulated UH_3. The likely coolant to be used is potassium.

The reactor design in question begins producing power when hydrogen gas at a sufficient temperature and pressure is admitted to the core (made up of granulated uranium metal) and reacts with the uranium metal to form UH_3. UH_3 is both a nuclear fuel and a neutron moderator; apparently it, like other neutron moderators, will slow neutrons sufficiently to allow for fission reactions to take place. The ^{235}U atoms within the hydride also serve as the nuclear fuel. Once the nuclear reaction has started, it will continue until it reaches a certain temperature, approximately 800°C, where, due to the chemical properties of UH_3, it chemically decomposes and turns into hydrogen gas and uranium metal. The loss of neutron moderation due to the chemical decomposition of the UH_3 will consequently slow and eventually halt the reaction. When temperature returns to an acceptable level, the hydrogen will again combine with the uranium metal, forming UH_3, restoring moderation and the nuclear reaction will start again.

This makes the reactor a self-regulating, dynamic system, as with a rise in temperature, nuclear reactivity will substantially decrease, and with a fall in temperature, nuclear reactivity will substantially increase. Thus, this reactor design is self-regulating, meltdown is impossible, and the design is inherently safe. From a safety point of view, the design leverages the technology used in the TRIGA reactor, which uses uranium zirconium hydride (UZrH) fuel and is the only reactor licensed by the U.S. Nuclear Regulatory Commission for unattended operation.

According to the reactor design specification, the UH_3 core is surrounded by hydrogen-absorbing storage trays, made of depleted uranium or thorium. The storage trays can either desorb or absorb the hydrogen gas from the core. During normal operation (with the operating temperature being approximately 550°C), the storage trays are kept at a temperature high enough to expel the hydrogen gas to the core. The storage trays are heated or cooled by means of heat pipes and an external thermal source. Thus, in a steady state, the UH_3 core is slaved to the temperature of the storage trays. Other heat pipes, protruding the UH_3 core, deliver the nuclear generated heat from the core to a heat exchanger, which in turn can be connected to a steam turbine-generator set, for the production of electricity.

The only hazards are those of all nuclear materials, namely those of radiation, but this is significantly mitigated by the fact that the reactor design is intended to be buried underground and only dug up for refueling every five years, at which point, assuming proper safeguards are used, exposure to radioactivity is a comparatively trivial concern. Spent fuel is also a concern, but this is mitigated due to certain technologies and advantages that make the design in question's used fuel more suitable for nuclear recycling. In particular, the patent application for the design indicates that using a thorium fuel cycle instead of a uranium fuel cycle with this type of reactor will allow far greater recycling potential than presently is found in standard used fuel. Furthermore, the UH_3 has the capability of a high fuel burn-up, of up to 50%, in contrast to a LWR which usually achieves a burn-up of 5%. Reprocessing of spent fuel is simplified and more economical for the hydride reactor design, because the so-called process of zone refining can be used for separation.

Apparently, the proposed reactor design will be capable of supplying 27MWe of electric power or 70MWth, weight 18-20t, measure approximately 1.5m in diameter, be mass-produced on an assembly line, and be capable of unattended, unrefueled operation for up to seven to ten years at a time. Costs are projected to be competitive with other established sources of energy, like coal, conventional nuclear and natural gas.

As the concept of a UH3 reactor is novel, further experimental work will be needed with regard to gas flow dynamics, materials selection and performance (especially with regard to hydrogen embrittlement and hydride pyrophoricity), radiation damage and fission fragment build-up. A further challenge will be posed by the remote temperature control of the storage trays as well as cooling these trays when it may be necessary, so they absorb hydrogen from the core [60].

4.7.3 Clean and environmentally safe advanced reactor

The CAESAR is a nuclear reactor concept that uses steam as a moderator. The CAESAR reactor design exploits the fact that the fission products and daughter isotopes produced via nuclear reactions also decay to produce additional delayed neutrons. When steam is used as the moderator, the average neutron speed/energy is increased from that of a liquid water moderated reactor and the delayed neutrons keep going until they hit another nucleus. The resulting extremely high neutron economy will make it possible to maintain a self-sustaining reaction in fuel rods of pure ^{238}U, once the reactor has been started by enriched fuel.

Skeptics, however point out that it is generally believed that a controlled, sustained chain reaction is not possible with ^{238}U. It can undergo fission when impacted by an energetic neutron with over 1MeV of kinetic energy. But the number of high-energy neutrons produced by ^{238}U fission are not sufficient to induce enough successive fissions in ^{238}U to create a critical system, one in which the number of neutrons created by fission is equal to the number absorbed. Instead, bombarding ^{238}U with neutrons below the 1MeV fission threshold causes it to absorb them without fissioning, thus becoming uranium-239 (^{239}U) and decay by beta emission to ^{239}Pu, which is itself fissile [61, 62].

5. Generation IV reactors

Generation IV reactors are future designs that are currently under research and development supported by the ten country consortium that makes up the generation IV international forum. Generation IV reactors will not be deployable before 2030 at the earliest. The technical characteristics of the various generation IV nuclear reactors are tabulated in Table 4.

Table 4. Generation IV reactors designs REVISE

No.	Reactor type	Capacity (MWe)	Power cycle	Efficiency (%)
1	VHTR (very high temperature reactor)	hydrogen production	Brayton	>50 (at 1000°C)
2	SCWR (supercritical water cooled reactor)	1700	Supercritical water Rankine	44
3	MSR (molten salt reactor)	1000	Brayton	44-50
4	GFR (gas cooled fast reactor)	288	Brayton	48
5	SFR (sodium cooled fast reactor)	150-500; 500-1500	N/A	N/A
6	LFR (lead cooled fast reactor)	50-150; 300-400; 1200	Supercritical water Rankine, or supercritical CO_2 Brayton cycle	N/A

Will tend to have closed fuel cycles and burn the long-lived actinides now forming part of spent fuel, so that fission products are the only high-level waste. The goals for generation IV nuclear reactors [63] are:

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

Relative to current nuclear power plant technology, the claimed benefits for generation IV reactors include nuclear waste that lasts a few centuries instead of millennia, 100-300 times more energy yield from the same amount of nuclear fuel, the ability to consume existing nuclear waste in the production of electricity and improved operating safety. One disadvantage of any new reactor technology is that safety risks may be greater initially as reactor operators have little experience with the new design [2].

5.1 Very high temperature reactor

The very high temperature reactor (VHTR) system uses a thermal neutron spectrum and a once-through uranium cycle. The VHTR system is primarily aimed at relatively faster deployment of a system for high temperature process heat applications, such as coal gasification and thermo-chemical hydrogen production, with superior efficiency. The reference reactor concept has a 600MWth helium cooled core based on either the prismatic block fuel of the GT-MHR, or the pebble fuel of the PBMR [63]. The prismatic block core configuration is a design where hexagonal graphite blocks are stacked to fit in a circular reactor pressure vessel. Pebble bed designs usually have a core where the pebbles are in an annulus, and there is a graphite center spire. The fuel is usually referenced to be UO_2 [64].

Helium is used as coolant in most VHTRs to date, and the peak temperature and power depend on the reactor design. Helium is an inert gas, so it will generally not chemically react with any material. Additionally, exposing helium to neutron radiation does not make it radioactive, unlike most other possible coolants. The molten salt cooled variant, the LS-VHTR, uses a molten salt for cooling in a prismatic core. It is essentially a standard VHTR design that uses molten salt as a coolant instead of helium. The molten salt would pass through holes drilled in the graphite blocks. The LS-VHTR has many attractive features, including the ability to work at very high temperatures (the boiling point of most molten salts being considered are >1,400°C), low pressure cooling that can be used to more easily match hydrogen production facility conditions since most thermo-chemical cycles require temperatures in excess of 750°C, better electric conversion efficiency than a helium cooled VHTR operating at similar conditions, passive safety systems, and better retention of fission products in case an accident occurred [64].

In a VHTR the primary circuit is connected to a steam reformer/steam generator to deliver process heat, as illustrated in Figure 2. The VHTR system [63] has coolant outlet temperatures above 1000°C. In the prismatic designs, control rods would be inserted in holes cut in the graphite blocks that make up the core. The VHTR will be controlled like current PBMR designs if it utilizes a pebble bed core, the control rods will be inserted in the surrounding graphite reflector. Control can also be attained by adding pebbles containing neutron absorbers [64]. It is intended to be a high-efficiency system that can supply process heat to a broad spectrum of high temperature and energy-intensive, non-electric processes. The system may incorporate electricity generation equipment to meet cogeneration needs. The system also has the flexibility to adopt uranium/plutonium fuel cycles and offer enhanced waste minimization. The VHTR requires significant advances in fuel performance and high temperature materials, but could benefit from many of the developments proposed for earlier prismatic or pebble bed gas cooled reactors.

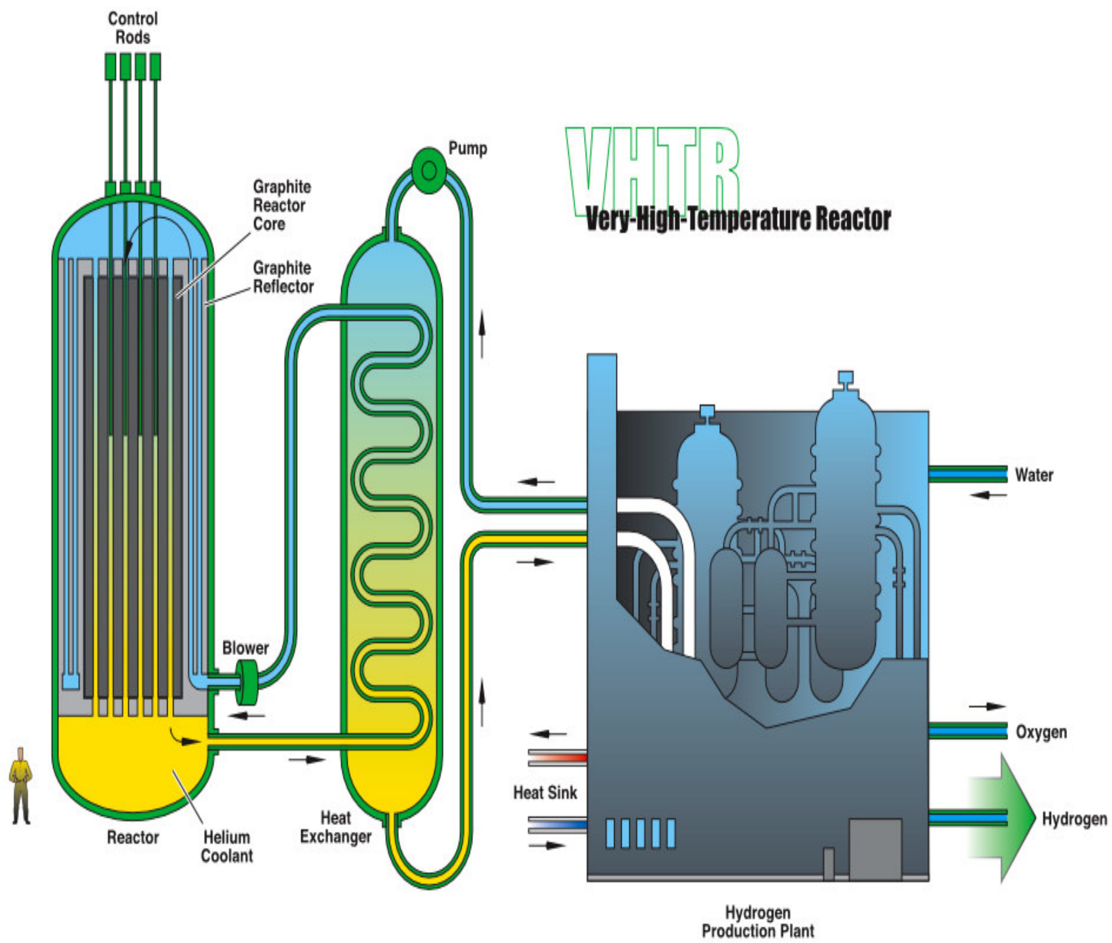

Figure 2. A typical VHTR reactor

The VHTR system is highly ranked in economics because of its high hydrogen production efficiency, and in safety and reliability because of the inherent safety features of the fuel and reactor. It is rated good in proliferation resistance and physical protection, and neutral in sustainability because of its open fuel cycle. It is primarily envisioned for missions in hydrogen production and other process-heat applications, although it could produce electricity as well. The VHTR system is the nearest-term hydrogen production system, estimated to be deployable by 2020 [63].

5.2 Supercritical water cooled reactor

The supercritical water cooled reactor (SCWR) is a generation IV reactor concept that uses supercritical water as the working fluid. SCWRs are basically LWRs operating at higher pressure and temperatures with a direct, once-through cycle. As most commonly envisioned, it would operate on a direct cycle, much like a BWR, but since it uses supercritical water as the working fluid, would have only one phase present, like the PWR. It could operate at much higher temperatures and pressure than both current PWRs and BWRs. SCWR are promising advanced nuclear systems because of their high thermal efficiency, i.e., about 45% vs. about 33% efficiency for current LWR and considerable plant simplification.

A key issue in natural circulation is constituted by the stability of the flow mainly when two phase conditions are concerned and when the feedback with neutron kinetics is possible. The main mission of the SCWR is generation of low-cost electricity. It is built upon two proven technologies, LWRs, which are the most commonly deployed power generating reactors in the world, and supercritical fossil fuel fired boilers, a large number of which are also in use around the world [65]. The SCWR system features two fuel cycle options. The first is an open cycle with a thermal neutron spectrum reactor and the second is a closed cycle with a fast-neutron spectrum reactor and full actinide recycle. Both options use a high-temperature, high-pressure, water cooled reactor that operates above the thermodynamic critical point of water at 221bar and 374°C, to achieve a thermal efficiency approaching 44%. The fuel cycle for the thermal option is a once-through uranium cycle. The fast-spectrum option uses central fuel cycle facilities based on advanced aqueous processing for actinide recycle. The fast-spectrum option depends upon the materials' R&D success to support a fast-spectrum reactor. In either option, the reference plant has a 1700MWe power level, an operating pressure of 250bar, and a reactor outlet temperature of 550°C. Passive safety features similar to those of the simplified BWR are incorporated. Owing to the low density of supercritical water, additional moderator is added to the core in the thermal option. Note that the balance-of-plant is considerably simplified because the coolant does not change phase in the reactor [63], as illustrated in Figure 3.

The SCWR uses water as a neutron moderator. Moderation comes primarily from the high density sub-critical water. This high-density water is either introduced from cooling tubes inserted into the core or as a reflector or moderated-part of the core. The neutron spectrum will be only partly moderated, perhaps to the point that the SCWR will technically become a fast neutron reactor. The advantage of using fast neutron spectrum is having higher power rating than using thermal neutron spectrum, because of high power density.

The fuel is traditional LWR fuel. However, it is likely the SCWR will use cladding fuel element like the BWR to reduce the chance of hotspots causing local variations in core properties. Because of the SCWR operating at exceeding condition than the current experience with LWRs and LMFBR, thus specific criteria for material especially for cladding must be developed for safe operation to maintain fuel rod integrity during abnormal transient, rated power operation as well as releasing the fission product caused by oxidation corrosion of the cladding. There are four failure mode considered for the fuel rod integrity at abnormal transient, (a) mechanical failure, (b) buckling collapse, (c) over pressure damage and (d) creep failure. Hydrogen injection is for reducing oxidation corrosion [65].

The coolant will be supercritical water. Operation above the critical pressure eliminates coolant boiling, so the coolant remains single-phase throughout the system. That means more of the heat produced via fission can be converted into electricity in reactors cooled with supercritical water. In addition, the elements that handle water's phase change from liquid to gas in conventional light water reactors can be cut from the design. Thus, the need for recirculation and jet pumps, pressurizers, steam generators, and steam separators and dryers in current LWRs is eliminated reducing construction costs. CWRs would likely have control rods inserted through the top, as is done in PWRs [65].

The SCWR system is highly ranked in economics because of the high thermal efficiency and plant simplification. If the fast-spectrum option can be developed, the SCWR system will also be highly ranked in sustainability. The SCWR is rated well in safety, and in proliferation resistance and physical protection. The SCWR system is primarily envisioned for missions in electricity production, with an option for actinide management [63].

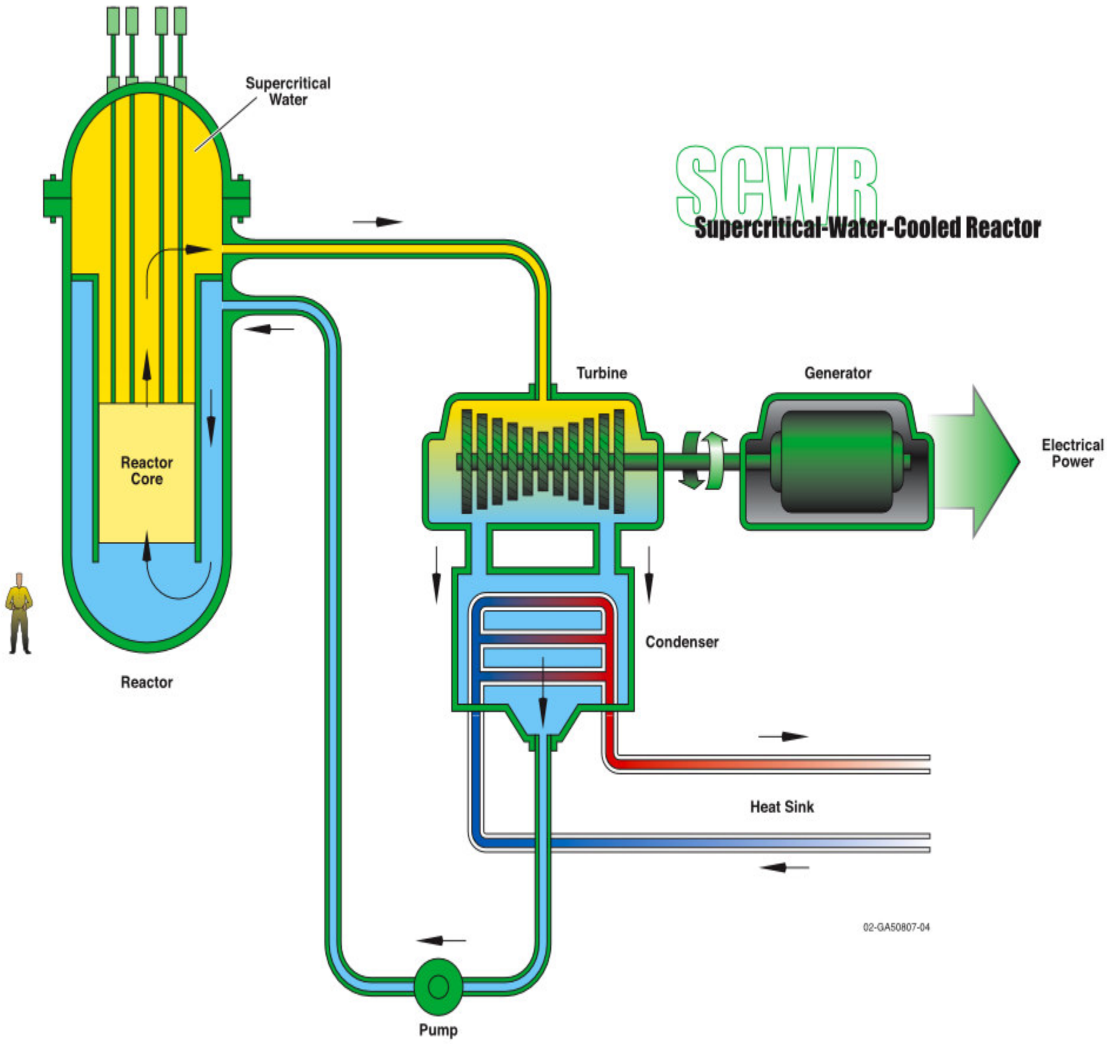

Figure 3. A typical SCWR reactor

5.3 Molten salt reactor

The molten salt reactor (MSR) features an epithermal to thermal neutron spectrum and a closed fuel cycle tailored to the efficient utilization of plutonium and minor actinides. A full actinide recycle fuel cycle is envisioned. In the MSR system, the fuel is a circulating liquid mixture of sodium, zirconium and uranium fluorides. The molten salt fuel flows through graphite core channels, producing a thermal spectrum. The heat generated in the molten salt is transferred to a secondary coolant system through an intermediate heat exchanger, and then through another heat exchanger to the power conversion system. Actinides and most fission products form fluorides in the liquid coolant [63], as illustrated in Figure 4. The homogenous liquid fuel allows addition of actinide feeds with variable composition by varying the rate of feed addition. There is no need for fuel fabrication. The reference plant has a power level of 1000MWe. The system operates at low pressure (<5bar) and has a coolant outlet temperature above 700°C, affording improved thermal efficiency. The MSR system is top-ranked in sustainability because of its closed fuel cycle and excellent performance in waste burn down. It is rated well in safety, and in proliferation resistance and physical protection, and it is rated neutral in economics because of its large number of subsystems. It is primarily envisioned for missions in electricity production and waste burn down [63].

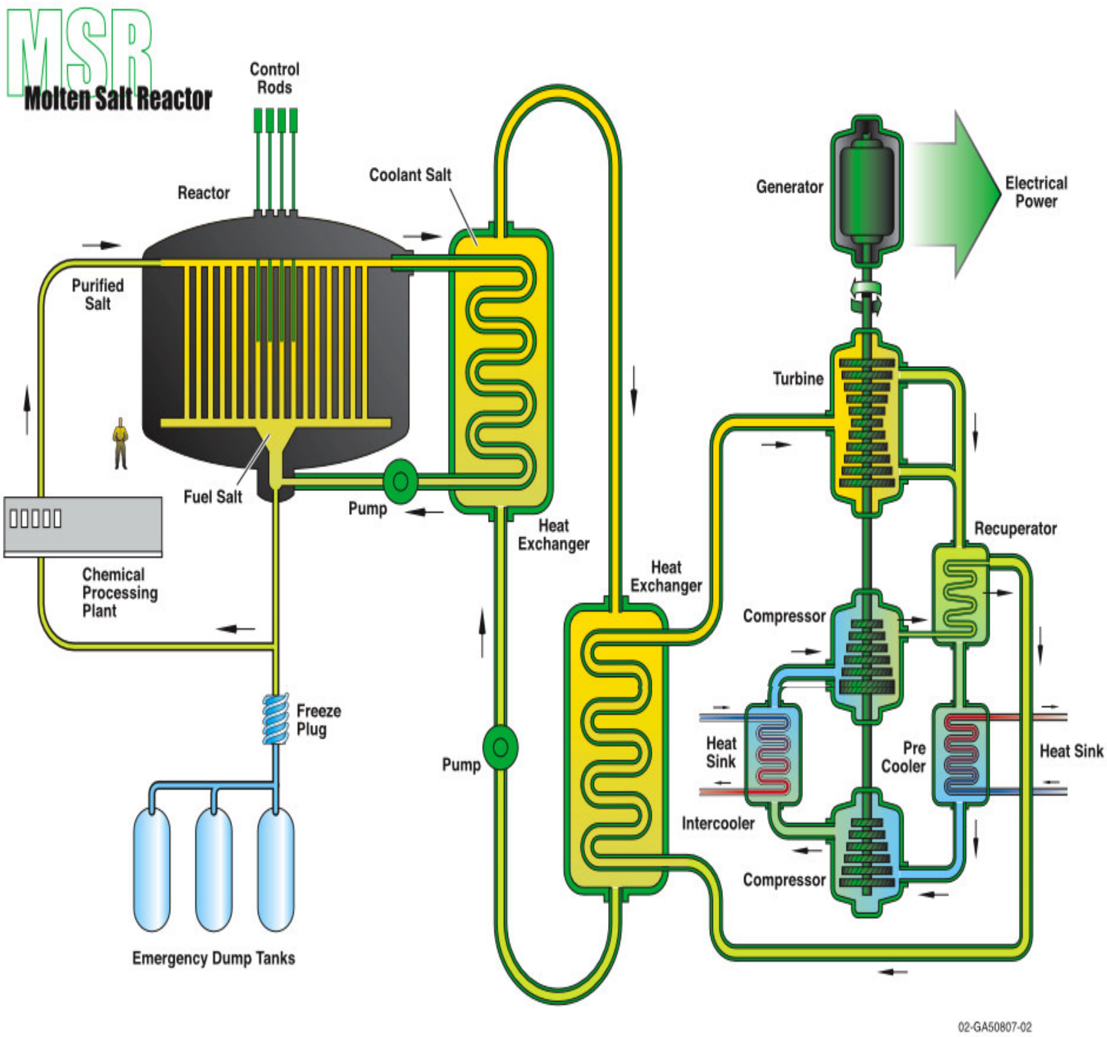

Figure 4. A typical MSR reactor

5.4 Gas cooled fast reactor

The gas cooled fast reactor (GFR) system features a fast-neutron spectrum and closed fuel cycle for efficient conversion of fertile uranium and management of actinides. A full actinide recycle fuel cycle with on-site fuel cycle facilities is envisioned. The fuel cycle facilities can minimize transportation of nuclear materials and will be based on either advanced aqueous, pyrometallurgical, or other dry processing options. The reference reactor is a 600MWth/288MWe, helium cooled system operating with an outlet temperature of 850°C using a direct Brayton cycle gas turbine for high thermal efficiency, as illustrated in Figure 5.

Several fuel forms are being considered for their potential to operate at very high temperatures and to ensure an excellent retention of fission products. The GFR system is top-ranked in sustainability because of its closed fuel cycle and excellent performance in actinide management. It is rated well in safety, economics, and in proliferation resistance and physical protection. It is primarily envisioned for missions in electricity production and actinide management, although it may be able to also support hydrogen production [63].

In a GFR reactor design, the unit operates on fast neutrons, no moderator is needed to slow neutrons down. This means that, apart from nuclear fuel such as uranium, other fuels can be used. The most common is thorium, which absorbs a fast neutron and decays into ^{233}U. This means as GFR designs have breeding properties, they can use fuel that is unsuitable in normal reactor designs and breed fuel. Because of these properties, once the initial loading of fuel has been applied into the reactor, the unit can go years without needing fuel. If these reactors are used for breeding, it is economical to remove the fuel and separate the generated fuel for future use. The gas used can be many different types, including CO_2 or helium. It must be composed of elements with low neutron capture cross sections to prevent positive void coefficient and induced radioactivity. The use of gas also removes the possibility of phase transition that induced explosions, such as when the water in a water cooled reactor (PWR or BWR) flashes to steam upon overheating or depressurization. The use of gas also allows for higher operating temperatures than are possible with other coolants, increasing thermal efficiency, and allowing other non-mechanical applications of the energy, such as the production of hydrogen fuel [66].

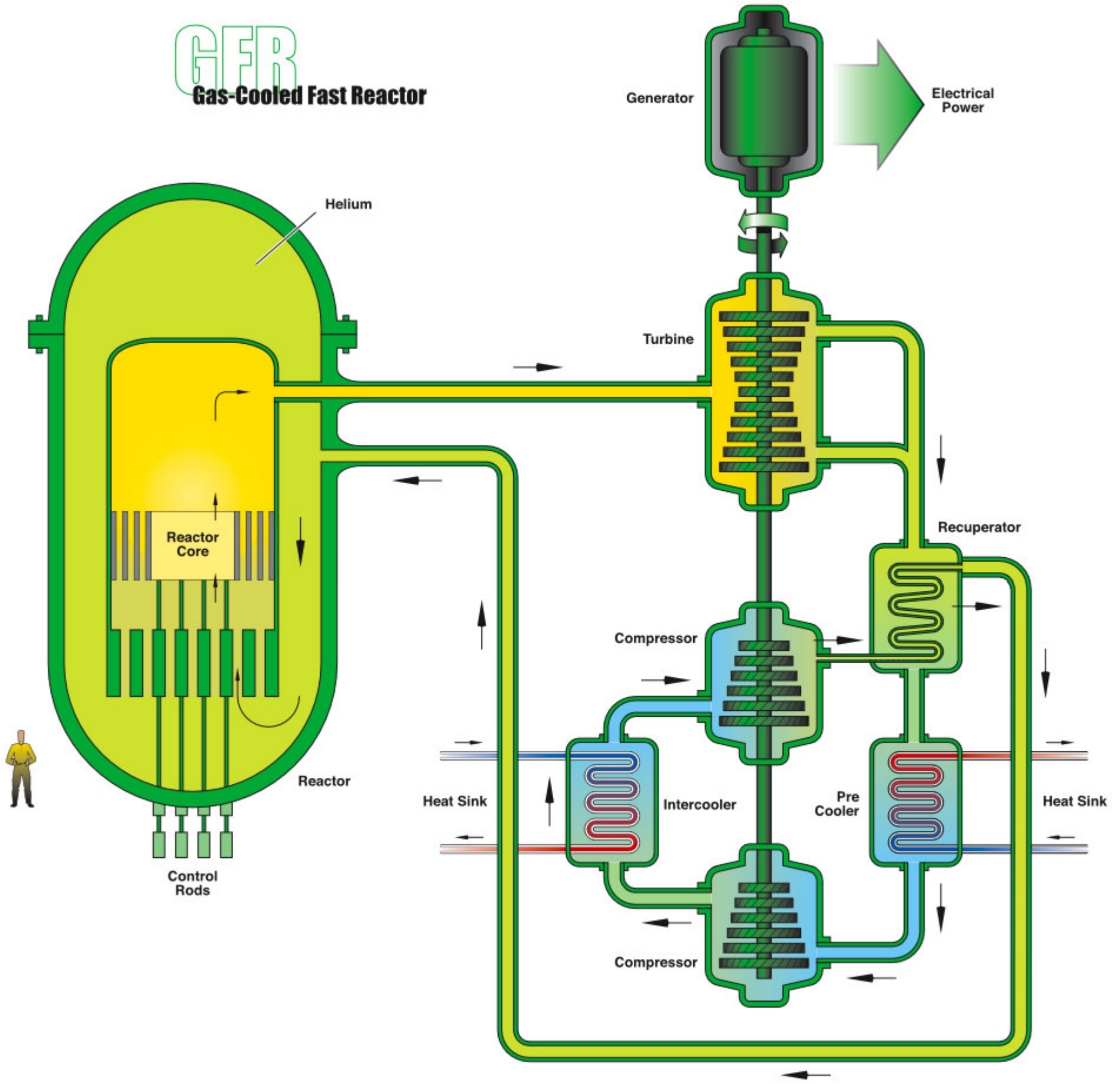

Figure 5. A typical GFR reactor

5.5 Sodium cooled fast reactor

The sodium cooled fast reactor (SFR) features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium and management of actinides. A full actinide recycle fuel cycle is envisioned with two major options. One is an intermediate size (150 to 500MWe) sodium cooled reactor with a uranium-plutonium-minor-actinide-zircaloy fuel, supported by a fuel cycle based on pyrometallurgical processing in collocated facilities. The second is a medium to large (500 to 1500MWe) SFR with mixed oxide fuel fuel, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors. The outlet temperature is approximately 550°C for both [63], as illustrated in Figure 6.

The SFR is designed for management of high-level wastes and, in particular, management of plutonium and other actinides. Important safety features of the system include a long thermal response time, a large margin to coolant boiling, a primary system that operates near atmospheric pressure, and intermediate sodium system between the radioactive sodium in the primary system and the water and steam in the power plant. With innovations to reduce capital cost, the SFR can serve markets for electricity generation. The SFR's fast spectrum also makes it possible to use available fissile and fertile materials (including depleted uranium) considerably more efficiently than thermal spectrum reactors with once-through fuel cycles.

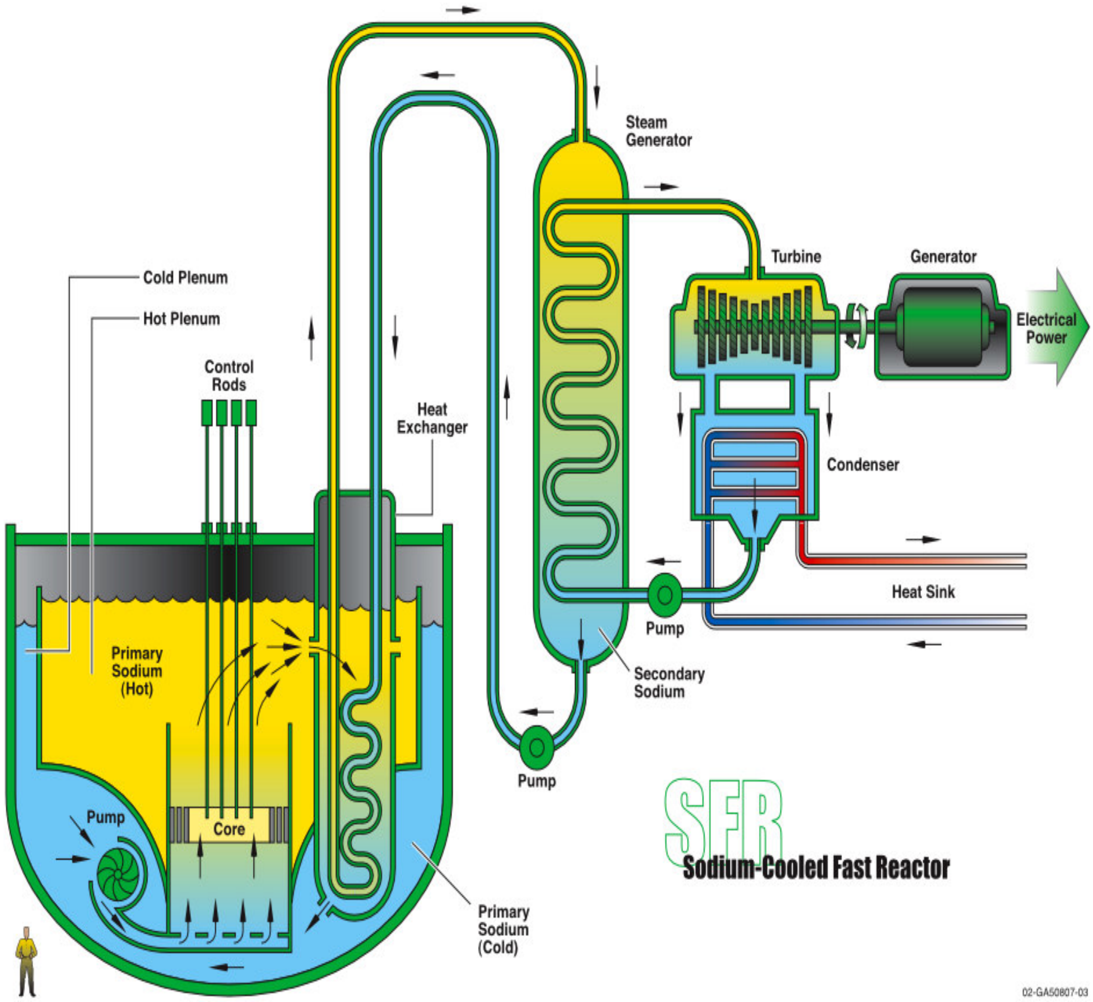

Figure 6. A typical SFR reactor

An advantage of liquid metal coolants is high heat capacity which provides thermal inertia against overheating. Water is difficult to use as a coolant for a fast reactor because water acts as a neutron moderator that slows the fast neutrons into thermal neutrons. While it may be possible to use supercritical water as a coolant in a fast reactor, this would require a very high pressure. In contrast, sodium atoms are much heavier than both the oxygen and hydrogen atoms found in water, and therefore the neutrons lose less energy in collisions with sodium atoms. Sodium also need not be pressurized since its boiling point is much higher than the reactor's operating temperature, and sodium does not corrode steel reactor parts. A disadvantage of sodium is its chemical reactivity, which requires special precautions to prevent and suppress fires. If sodium comes into contact with water it explodes, and it burns when in contact with air. In addition, neutrons cause it to become radioactive, however, activated sodium has a half-life of only 15h [67].

The SFR system is top-ranked in sustainability because of its closed fuel cycle and excellent potential for actinide management, including resource extension. It is rated good in safety, economics, and proliferation resistance and physical protection. It is primarily envisioned for missions in electricity production and actinide management. The SFR system is the nearest term actinide management system. Based on the experience with oxide fuel, this option is estimated to be deployable by 2015 [63].

5.6 Lead cooled fast reactor

The lead cooled fast reactor (LFR) system features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium and management of actinides. A full actinide recycle fuel cycle with central or regional fuel cycle facilities is envisioned. The system uses a lead or lead/bismuth eutectic liquid-metal cooled reactor. Options include a range of plant ratings, including a battery of 50-150MWe that features a very long refueling interval, a modular system rated at 300-400MWe, and a large monolithic plant option at 1200MWe. The term battery refers to the long-life, factory-fabricated core, not to any provision for electrochemical energy conversion.

The fuel is metal or nitride-based, containing fertile uranium and transhumances. The most advanced of these is the lead-bismuth battery, which employs a small size core with a very long (10-30 year) core life. The reactor module is designed to be factory fabricated and then transported to the plant site. The reactor is cooled by natural convection and sized between 120-400MWth, with a reactor outlet coolant temperature of 550°C, possibly ranging up to 800°C, depending upon the success of the materials research and development, as illustrated in Figure 7. The system is specifically designed for distributed generation of electricity and other energy products, including hydrogen and potable water.

The LFR system is top-ranked in sustainability because a closed fuel cycle is used, and in proliferation resistance and physical protection because it employs a long-life core. It is rated as good in safety and economics. The safety is enhanced by the choice of a relatively inert coolant. It is primarily envisioned for missions in electricity and hydrogen production and actinide management with good proliferation resistance [63].

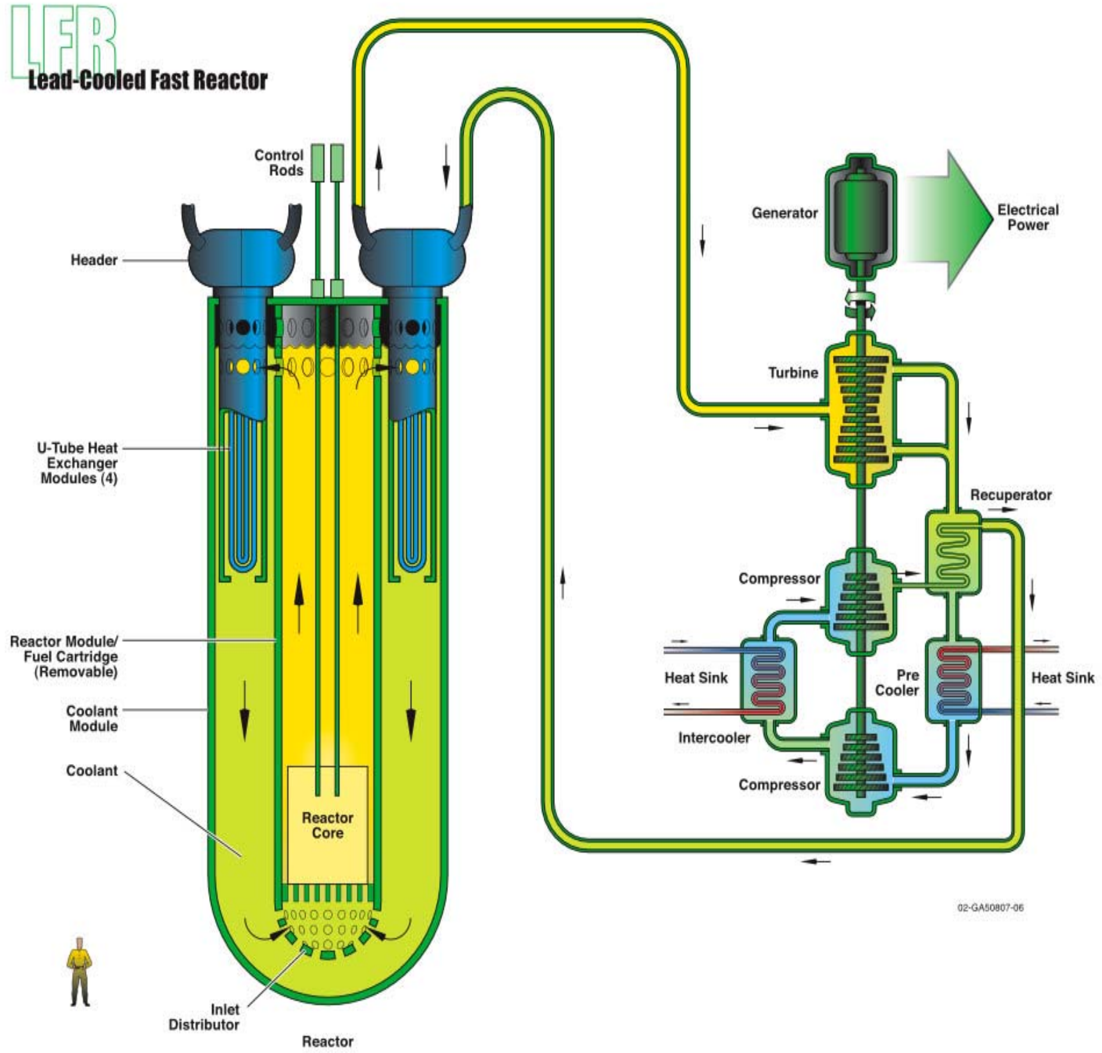

Figure 7. A typical LFR reactor

6. Conclusion

In this paper an overview of the current and future nuclear power reactor technologies was carried out. In particular, the nuclear technology was described and the classification of the current and future nuclear reactors according to their generation was provided. The analysis has shown that generation II reactors currently in operation all around the world lack significantly in safety precautions and are prone to loss of coolant accident (LOCA). In contrast, generation III reactors, which are an evolution of generation II reactors, incorporate passive or inherent safety features that require no active controls or operational intervention to avoid accidents in the event of malfunction, and may rely on gravity, natural convection or resistance to high temperatures. Today, partly due to the high capital cost of large power reactors generating electricity and partly due to the consideration of public perception, there is a shift towards the development of smaller units. These may be built independently or as modules in a larger complex, with capacity added incrementally as required. Small reactors most importantly benefit from reduced capital costs, simpler units and the ability to produce power away from main grid systems. These factors combined with the ability of a nuclear power plant to use process heat for co-generation, make the small reactors an attractive option. Generally, modern small reactors for power generation are expected to have greater simplicity of design, economy of mass production and reduced installation costs. Many are also designed for a high level of passive or inherent safety in the event of malfunction. Generation III+ designs are generally extensions of the generation III concept, which include advanced passive safety features. These designs can maintain the safe state without the use of any active control components. Generation IV reactors, which are future designs that are currently under research and development, will tend to have closed fuel cycles and burn the long-lived actinides now forming part of spent fuel, so that fission products are the only high-level waste. Relative to current nuclear power plant technology, the claimed benefits for generation IV reactors include nuclear waste that lasts a few centuries instead of millennia, 100-300 times more energy yield from the same amount of nuclear fuel, the ability to consume existing nuclear waste in the production of electricity and improved operating safety. Generation V+ reactors are designs which are theoretically possible, but which are not being actively considered or researched at present. Though such reactors could be built with current or near term technology, they trigger little interest for reasons of economics, practicality or safety.

References

[1] Lamarsh J., Baratta A., 1983, “Introduction to nuclear engineering”, Prentice Hall, 3rd Edition, ISBN 0-201-82498-1.

[2] Kessides I.N., 2010, “Nuclear power: Understanding the economic risks and uncertainties”, Energy Policy, 38, 3849-3864.

[3] Uranium Information Centre, 2005, “Nuclear power reactors”, Briefing Paper 64.

[4] Wilson P. D., 1996, “The Nuclear fuel cycle: from ore to waste”, Oxford Science Publications.

[5] Monti, S., Toti, A., 2012, “An overview of the activities carried out by the IAEA Technical Working Group on Fast Reactors”, International Congress on Advances in Nuclear Power Plants 2012, ICAPP 2012, 1, 787-792.

[6] Poullikkas A., 2003, “Effects of two-phase liquid-gas flow on the performance of nuclear reactor cooling pumps”, Progress in Nuclear Energy, 42, 3-10.

[7] Poullikkas A., 2000, “Two phase flow performance of nuclear reactor cooling pumps”, Progress in Nuclear Energy, 36, 123-130.

[8] Zhou Y., Rengifo C., Chen P., Hinze J., 2001, “Is China ready for its nuclear expansion?”, Energy Policy, 39, 771-781.

[9] Kennedy D., “New nuclear power generation in the UK: Cost benefit analysis”, Energy Policy, 35, 3701-3716.

[10] Mitenkov F.M., Antonovsky G.M., Panov Y.K, Runov B.I., “New generation medium power nuclear station with VPBER-600 passive safety reactor plant”, Nuclear Engineering and Design, 173, 99-108.

[11] Zhou Y, 20010, “Why is China going nuclear?”, Energy Policy, 38, 3755-3762.

[12] Fei, T., Richard, J.G., Kersting, A.R., Don, S.M., Oi, C., Driscoll, M.J., Shwageraus, E., 2012, “A survey of alternative once-through fast reactor core designs”, International Congress on Advances in Nuclear Power Plants 2012, ICAPP 2012, 3, 1562-1571.

[13] Michanek G., Söderholm P., 2009, “Licensing of nuclear power plants: The case of Sweden in an international comparison”, Energy Policy, 37, 4086-4097.

[14] Jewell J., 2011, “Ready for nuclear energy?: An assessment of capacities and motivations for launching new national nuclear power programs”, Energy Policy, 1041-1055.

[15] Advanced nuclear power reactors, 2005, UIC nuclear issues briefing, Paper 16.

[16] World Nuclear Association, 2010, “Small nuclear power reactors”.

[17] Penner S.S., Seiser R., Schultz K.R., 2008, "Steps toward passively safe, proliferation-resistant nuclear power", Progress in Energy and Combustion Science, 34, 275-287.
[18] Hansen U., 1996, "Restructuring the East German energy system", Energy Policy, 24, 553-562.
[19] Adamovich L.A., Grechko G.I., Shishkin V.A., 1997, "Self-contained co-generation nuclear power plant with integral nuclear reactor designed for remote and difficult-to-access areas", Nuclear Engineering and Design, 173, 175-182.
[20] General Electric, "ESBWR nuclear reactor design technology fact sheet".
[21] Reuters, 2007, "TXU to order Mitsubishi's APWR", NEI Nuclear Notes.
[22] npjonline.com (Weistinghouse 's BWR 90+ -Key features).
[23] Rumiantseva I., von Hirschhausen C., 2005, "Economics of nuclear power development in Belarus", Energy Sector Reform in Eastern Europe working papers, WP-EE-11.
[24] World Nuclear Association, 2011, "Advanced nuclear power reactors".
[25] Cheweb.tamu.edu (Alternate energy sources-Solution of major problems).
[26] World Nuclear Association, 2011, "The Economics of nuclear power".
[27] World Nuclear Association, 2011, "Nuclear power in Finland".
[28] Mitenkov F.M., Polunichev V.I., 1997, "Small nuclear heat and power co-generation stations and water desalination complexes on the basis of marine reactor plants", Nuclear Engineering and Design, 173, 183-191.
[29] www.reak.bme.hu (Проект АЭС-2006).
[30] Department of Energy, 2001, "Study outlines reactor designs that may be ready for deployment by decade's end", Nuclear News.
[31] World Nuclear Association, 2011, "Nuclear power in Russia".
[32] www.redorbit.com (Kazakh nuclear centre proposes building nuclear stations in small towns).
[33] Gabaraev B. A., Kuznetzov Yu. N., Romenkov A. A., Mishanina Yu. A., 2004, "Nuclear desalination complex with VK-300 boiling-type reactor facility", Proceedings of World Nuclear Association Annual Symposium, 8-10 September, London.
[34] International Atomic Energy Agency, 2005, "Innovative small and medium sized reactors: Design features, safety approaches and R&D trends", IAEA-TECDOC-1451.
[35] Cho, B.H., Jung, H.H., Hwang, J.D., Kim, C., Choi, H.O., 2012, "Analytical estimation of structural integrity for a thermocouple in nuclear power plant", Advanced Materials Research, 569, 572-575.
[36] International Atomic Energy Agency, 2005, "Optimization of the coupling of nuclear reactors and desalination systems", IAEA-TECDOC-1444.
[37] Goliath.ecnext.com (S. Korea to build pilot desalination reactor).
[38] Nuclear Desalination, 2002, "Summary and conclusions", Proceedings of 1st International Conference, 16-18 October, Marrakech, Morocco.
[39] Jamison E., 2004, "Nuclear: A viable energy source to fuel the oil sands", Buec 463, Joseph Doucet.
[40] World Nuclear Association, 2010, "Advanced nuclear power reactors".
[41] www.defence.pk (India to begin construction of AHWR)
[42] Dubberley A., 2003, "S-PRISM fuel cycle study", Proceedings of ICAPP '03, May 4-7, Córdoba, Spain.
[43] www.nti.org (12/98: Construction licensed, requires sponsor).
[44] International Nuclear Safety Center, 2006, "Overview of fast reactors in Russia and the former Soviet Union".
[45] Ghosh C., 2003, "The great Indian nuclear dream: Is a thorium-based nuclear program sustainable?", Institute of peace and conflict studies.
[46] World Nuclear Association, 2005, "Plans for new reactors worldwide".
[47] Bratt, D., 2005, "Implementing kyoto in canada: The role of nuclear power", The Energy Journal, 26, 107-121.
[48] Michanek G., Söderholm P., 2009, "Licensing of nuclear power plants: The case of Sweden in an international comparison", Energy Policy, 37, 4086-4097.
[49] Vivianne H.M. Visschers V.H.M, Keller C., Siegrist M., 2011, "Climate change benefits and energy supply benefits as determinants of acceptance of nuclear power stations: Investigating an explanatory model", Energy Policy, 39 3621-3629.
[50] www.mi.infn.it (Considerations related to specific concepts: Water cooled reactors, gas cooled reactors, metal cooled reactors, non-conventional reactors).

[51] International Atomic Energy Agency, 2002, “Status of design concepts of nuclear desalination plants”, IAEA-TECDOC-1326.

[52] Wang Q., 2009, “China needing a cautious approach to nuclear power strategy”, Energy Policy, 37, 2487-2491.

[53] Denholm P., King J.F., Kutcher C.F,, Wilson H.P.H., 2012, “Decarbonizing the electric sector: Combining renewable and nuclear energy using thermal storage”, Energy Policy, 44, 301-311.

[54] Beerten J., Laes E., Meskens G., D’haeseleer W., 2009, “Greenhouse gas emissions in the nuclear life cycle: A balanced appraisal”, Energy Policy, 37, 5056-5068.

[55] McConnell, M.W., 2012, “Enhancement of NRC station blackout requirements for nuclear power plants”, Int. Congress on Advances in Nuclear Power Plants 2012, ICAPP 2012, 2, 1438-1442.

[56] World Nuclear Association, 2011, “Fast neutron reactors”.

[57] Fast neutron reactors, 2005, UIC nuclear issues briefing, Paper 98.

[58] Dautray R., 2001, “The long-term future for civilian nuclear power generation in France: The case for breeder reactors: Breeder reactors: The physical and physical chemistry parameters, associate material thermodynamics and mechanical engineering: Novelties and issues”, Comptes Rendus Mécanique, 339, 369-387.

[59] Cammi A., Finzi E., Lombardi C., Ricotti M.E., Santini L., 2009, “A “Generation III+” nuclear reactor for space needs”, Progress in Nuclear Energy, 51, 347-354.

[60] Jackson, J.D., Anderson, M., Cheng, X., Zahlan, H., Bae, Y.Y., Yamada, K., 2012, “Progress on the development of new correlations under the framework of the IAEA Coordinated Research Programme on heat transfer in SCWR's”, International Congress on Advances in Nuclear Power Plants 2012, ICAPP 2012, 1, 157-169.

[61] Abu-Khader M.M., 2009, “Recent advances in nuclear power: A review, Progress in Nuclear Energy”, 51, 225-235.

[62] Kinnunen M., Korppoo A., 2007, “Nuclear power in Northern Russia: A case study on future energy security in the Murmansk region”, Energy Policy, 35, 2826-2838.

[63] US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, 2002, “A technology roadmap for Generation IV nuclear energy systems”.

[64] Keksis A.L., Chadwick M.B., Selby H.D., Mac Innes M.R., Barr D.W., Meade R.A., Burns C.J., Wallstrom T.C., 2012, “Los Alamos National Laboratory Fission Basis”, ASTM Special Technical Publication, 1550 STP, 371-377.

[65] Revankar, S.T., Zhou, W., Wolf, B., Oh, S., 2012, “Condensation model for the ESBWR passive condensers”, International Congress on Advances in Nuclear Power Plants 2012, ICAPP 2012, 3, 2145-2152.

[66] Herranz L.E., Linares J.I., Moratilla B.Y., “Power cycle assessment of nuclear high temperature gas-cooled reactors”, Applied Thermal Engineering, 29, 1759-1765.

[67] Chung, W.-S., Yun, S.-W., Lee, D.-S., Go, D.-Y., 2012, “Renovated Korean nuclear safety and security system: A review and suggestions to successful settlement”, International Congress on Advances in Nuclear Power Plants 2012, ICAPP 2012, 2, 1433-1437.

Andreas Poullikkas holds a Bachelor of Engineering (B.Eng.) degree in mechanical engineering, a Master of Phlosophy (M.Phil.) degree in nuclear safety and turbomachinery, a Doctor of Philosophy (Ph.D.) degree in numerical analysis and a Doctor of Science (D.Sc.) higher doctorate degree in energy policy and energy systems optimization from Loughborough University, U.K. He is a Chartered Scientist (CSci), Chartered Physicist (CPhys) and Member of The Institute of Physics (MInstP). His present employment is with the Electricity Authority of Cyprus where he holds the post of Assistant Manager of Research and Development; he is also, a Visiting Professor at the American University of Sharjah and at the University of Cyprus. In his professional career he has worked for academic institutions such as a Visiting Fellow at the Harvard School of Public Health, USA. He has over 20 years experience on research and development projects related to the numerical solution of partial differential equations, the mathematical analysis of fluid flows, the hydraulic design of turbomachines, the nuclear power safety, the analysis of power generation technologies and the power economics. He is the author of various peer-reviewed publications in scientific journals, book chapters and conference proceedings. He is the author of the postgraduate textbook: Introduction to Power Generation Technologies (ISBN: 978-1-60876-472-3), of the book Renewable Energy: Economics, Emerging Technologies and Global Practices, ISBN: 978-1-62618-231-8, of the book: The Cyprus Energy Future (ISBN: 978-9963-9599-4-5) and of the book: Sustainable Energy Development for Cyprus, ISBN: 978-9963-7355-3-2. He is, also, a referee for various international journals, serves as a reviewer for the evaluation of research proposals related to the field of energy and a coordinator of various funded research projects. He is a member of various national and European committees related to energy policy issues. He is the developer of various algorithms and software for the technical, economic and environmental analysis of power generation technologies, desalination technologies and renewable energy systems. E-mail address: apoullik@eac.com.cy