

Chapter 2

Advanced Nuclear Technologies and Its Future Possibilities

Abstract Employment and supporting the use of nuclear energy for electricity generation suffered a significantly reduction in several countries after Fukushima Daiichi nuclear accident occurred in March 2011 in Japan due to the fear to a new nuclear disaster. Nowadays, nuclear energy has demonstrated that it is a secure energy source and its use for electricity generation is free of CO₂ emissions. It is also a mature technology that can assure an energy supply when needed and without interruption. For all that, nuclear energy has become again a secure energy source for many countries in all regions of the world. In order to increase the safe operation of nuclear power plants, there are now three lines of investigation for the development of new type of nuclear power reactors. These are: (a) European Pressurised Reactor (EPR), a Generation III⁺ reactor; Generation IV reactors with six different types of designs (GFR, LFR, SCWR, VHTR, MSR and SFR); and the so called “Small Modular Reactors (SMRs)”, with tens of different concepts and designs at various stages of development in several countries.

2.1 Introduction

Nuclear energy is an energy that guarantees the supply of electricity, curbs the emissions of greenhouse gases effect, reduces external energy dependence and produces continuously electricity with low, steady, and predictable costs.

The use of nuclear energy to produce electricity began at the end of 1950s and grew until the 1990s. The nuclear technology has been developed for over 60 years and currently continues to innovate in new concepts of nuclear reactors in which is being incorporated all development and knowledge achieved.

In the second half of 1960s, U.S. launched the first nuclear power programme focused to electricity generation. Although four years earlier, the former URSS began the operation of the APS-1 Obninsk, the first nuclear power plant in the world, according to IAEA sources. Little by little other developed countries

followed the example of the U.S. and the former URSS running their own nuclear power programmes of construction and operation of nuclear power plants. Economic stability, stronger growth in energy demand and promising economic prospects in many countries were the engine of development of this energy source.

In the early 1970s, the oil energy crisis provided the definitive boost to the use of nuclear energy for the electricity generation within the energy plans of several developed countries, such as Germany, Canada, France, and Japan, among others. Highlights the strong commitment to the development of nuclear energy for electricity generation made by France, on the basis of the use of graphite-gas reactors, the government decided to favour the use of the American pressured water technology. In addition, some developing countries, such as Mexico, Brazil, Taiwan and South Korea were prepared to begin the development of their nuclear power programmes (Mínguez 2015).

In the coming years the main developed countries will continue to increase their installed power capacity to face the industrial challenges, but much of the future expansion of the electricity needs will take place in developing countries, particularly in China and India, who have part of its population with very limited access to electricity or not access at all. These countries are expecting to have a quick growth in the energy demand in the coming years.

In 2016, a total of 31 countries was operating 447 nuclear power reactors with an installed capacity of 389,051 MWe distributed in all regions of the world. Nuclear energy is one of the main base-load electricity-generating sources available in the world today generating 11.5% of the global power production, according to WNA. The nuclear energy continues to be an interest energy source for many countries in spite of Fukushima Daiichi nuclear accident. Now, several alternatives are under consideration, either throughout increasing power and the lifespan of currently nuclear power plants up to 60 years of operation or with the development of new nuclear reactor designs, some already under construction and others in design phase. The Fukushima Daiichi nuclear accident, occurred on March 2011 in Japan, caused a slowdown in the nuclear renaissance, and forced many countries to review their approved nuclear power programmes, to adopt a decision to review the safety of their nuclear power plants and other nuclear facilities, such as Japan, the U.S., the EU, China, Russia, the Republic of Korea among others, and to adopt a moratorium on the construction of new nuclear power reactors or to delay the construction of new nuclear power plants. However, some of these countries are now thinking to start the construction of new nuclear power plants (Mínguez 2015).

Many operating nuclear power plants will reach 40 years of operation in the next 5–10 years. One way to make nuclear power more competitive is to take a decision to extend the life of these plants, from 40 to 60 years. Another way is to bet on the construction of new plants.

2.2 Generation III, III⁺ and IV Nuclear Power Reactors

Most nuclear power reactors that are now under construction or have recently started operation are of the so-called “Generation III or III⁺”. These type of nuclear power reactors were developed in the 1990s and in recent years several improvements, in both safety and fuel efficiency, have been incorporated. The new nuclear power reactors classified as Generation III⁺ are: ABWR, AP1000, ACR100, APWR, ESBWR and EPR. It is expected that this generation of reactors will be the ones to be built for much of this century, in its present form or with important improvements (Mínguez 2015).

The so-called “Generation IV” reactors, were designed to incorporate significant improvements in security, in the total cost of the nuclear power plant, in proliferation resistance and in the reduction of the production of radioactive waste. The aim is to make better use of fuel source, both the U and the Pu and to some extent the use of thorium. Another feature of this type of nuclear power reactors is the ability to perform certain transmutation of minor actinides, so that energy efficiency and the fuel recovery will be much higher than today (Mínguez 2015).

Ten countries make up the Generation IV International Forum or GIF.¹ The group was created in January 2000 with the mandate to develop new types of nuclear power reactors. It is important to highlight that this group of countries is now working in the development of six new types of reactors, from which its basic technology is well-known. However, there are certain areas that need an important development, especially in materials, safety, fuel, and the use of such reactors to produce not only electricity, but also heat and other products such as hydrogen or other industrial uses.

2.2.1 Generation IV Nuclear Power Reactors

Nuclear reactor technology has been under continuous development since the first commercial exploitation of civil nuclear power in the 1950s. This technological development is presented in a number of broad categories, or generations, each representing significant technical advance (either in terms of performance, costs or safety) compared to the previous generation. The first generation, Generation-I, advanced in the 1950s and 1960s with early prototype reactors (gas-cooled/graphite moderated or prototype water cooled and moderated). The second generation, Generation-II, began in the 1970s in the large commercial power plants that are still

¹These countries are: France, United Kingdom, United States, China, Japan, Canada, South Africa, Republic of Korea, Russia and Switzerland. The group has today 14 members with the incorporation of Argentina, Australia, and Brazil. The European Union is also a member of the GIF. The European Atomic Energy Community (Euratom) is the implementing organisation for development of nuclear energy within the European Union.

operating today. The advanced LWRs and other systems with inherent safety features and more favourable characteristics in the event of extreme events such as those associated with core damage, which are so-called “Generation-III”. These reactors have been designed in recent years. Generation-III+ offers significant improvements in safety and economics over Generation-III advanced reactors and are under development and are being considered for deployment in several countries, in fact, new nuclear power reactors built between now and in the coming decades will likely be chosen from this type of nuclear power reactor. A typical example is the EPR—the European Pressurised-Water Reactor.

While the current second and third generation of nuclear power reactors designs provide an economically, technically and publicly acceptable electricity supply in many countries, further advances in nuclear energy system design can broaden the opportunities for the use of nuclear energy for this specific purpose. The fourth generation of nuclear power reactors, Generation IV, is expected to start being deployed by 2030.

2.2.2 *GIF Initiative Goals*

GIF considers that nuclear energy is needed to meet future energy demand, and that international collaboration is required to advance nuclear energy into its fourth generation of systems, deployable after 2030.

A technology roadmap² was made to guide the Generation IV effort. This roadmap, defines and plans the necessary R&D and associated timelines to allow deployment of Generation IV systems after 2030. When preparations for the Generation IV Technology Roadmap began, it was essential to establish goals for these nuclear energy systems.

The goals have three purposes: First, they serve as the basis for developing criteria to assess and compare the systems in the technology roadmap. Second, they are challenging and stimulate the search for innovative nuclear energy systems—both fuels cycles and reactor technologies. Third, they will serve to motivate and guide the R&D on Generation IV systems as collaborative efforts get underway (GIF 2002) (Fig. 2.1).

Eight goals are defined for Generation IV in the four broad areas of sustainability, economic competitiveness, safety and reliability, proliferation resistance, and physical protection. These goals are the following:

- (a) **Sustainability:** Regarding sustainability, the main concern was the management of the environment through clean air restrictions, waste management restrictions and conservation of resources.

²A Technology Roadmap for Generation IV Energy Systems (GIF 2002).

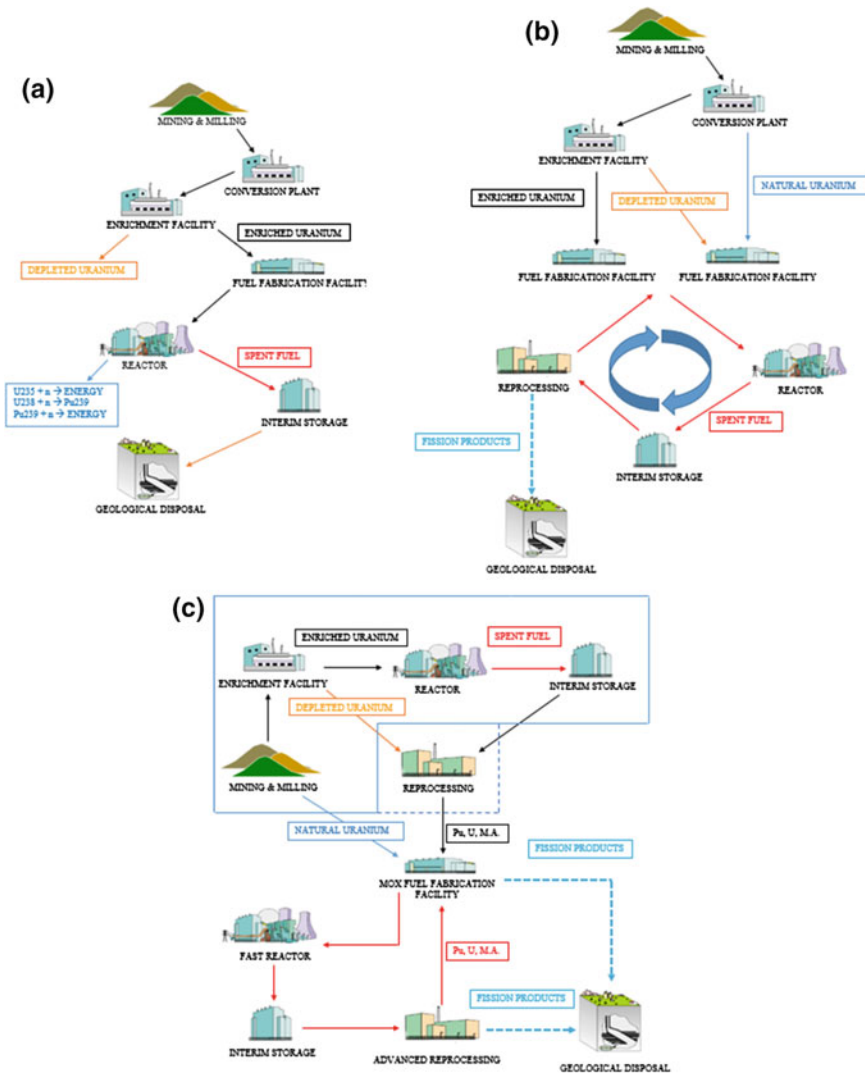


Fig. 2.1 Types of nuclear fuel cycles: **a** one through fuel cycle, **b** closed fuel cycle, **c** transuranic elements multi-recycling (González-Romero 2012)

- (i) Sustainability-1: Generation IV nuclear energy systems will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilisation for world-wide energy production.
- (ii) Sustainability-2: Generation IV nuclear energy systems will minimise and manage their nuclear waste and notably reduce the long-term stewardship burden, thereby improving protection for the public health and the environment (GIF 2002).

Four classes of nuclear fuel cycles were considered:

- The once through fuel cycle;
- A fuel cycle with partial bred fissile Pu recycling (closed fuel cycle);
- Full plutonium recycling (closed fuel cycle);
- A cycle with transuranic elements recycling (GIF 2002).

It appears that waste management is a major concern with the existing once through cycle because of the limited availability of repository space worldwide. Closed fuel cycles or recycling reactors allow some of the fuel to be reused so less of it has to be placed in a repository. Improvement in reactor performance can be achieved if thermal and fast reactors are operated in a coupled mode (Fig. 2.1).

- (b) **Economics:** Economic goals focus on competitive life cycle and energy production costs and financial risks. New nuclear power reactors must be competitive in a changing market place with energy demand.
- (i) Economics-1: Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.
 - (ii) Economics-2: Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects (GIF 2002).
- (c) **Safety and Reliability:** Safety and reliability goals focus on safe and reliable operation, improved accident management and minimisation of consequences, investment protection and essentially eliminating the technical need for off-site emergency response.

Active and passive safety feature against accidents are to be carefully considered. International safety and regulations for the handling of fissile materials in place are to be strictly enforced. To reduce the probability of radioactive elements leaking into the atmosphere or damaging to the nuclear power plants, there must be an emphasis on the human factor pertaining to plant operations.

- (i) Safety and Reliability-1: Generation IV nuclear energy systems will excel in safety and reliability.
 - (ii) Safety and Reliability-2: Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.
 - (iii) Safety and Reliability-3: Generation IV nuclear energy systems will eliminate the need for offsite emergency response (GIF 2002).
- (d) **Proliferation resistance and physical protection:** The proliferation resistance and physical protection goal focus on controlling and securing nuclear material and nuclear facilities.

New nuclear power plants are designed to cope with natural disasters such as earthquakes. Attention is to be devoted to the possibility of sabotage or acts of fissile material theft or dispersal by individuals or non-national groups.

Proliferation Resistance and Physical Protection-1: Generation IV nuclear energy systems 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 (GIF 2002).

These considerations resulted in six concepts for research and development.

2.2.3 Descriptions of the Generation IV Systems

The Technology Roadmap exercise was a two-year effort by more than 100 international experts to select the most promising nuclear systems. In 2002, GIF selected the six systems listed below, from nearly 100 concepts, as Generation IV systems:

- Gas Cooled Fast Reactor (GFR);
- Sodium Cooled Fast Reactor (SFR);
- Supercritical Water Cooled Reactor (SCWR);
- Very High Temperature Reactor (VHTR);
- Molten Salt Reactor (MSR);
- Lead Cooled Fast Reactor (LFR).

The Technology Roadmap defined, planned the necessary research, development and associated timelines to achieve the previously described goals to allow deployment of Generation IV energy systems after 2030. These timelines have suffered changes and updates since their creation in 2002 (GIF 2014) (Fig. 2.2).

System arrangements have been established for four systems (SFR, VHTR, SCWR and GFR) and Memoranda of Understanding (MoU) were agreed on for each of the remaining systems (LFR and MSR). The status of these arrangements and MoU as of January 2014 is the following (GIF 2014) (Fig. 2.3).

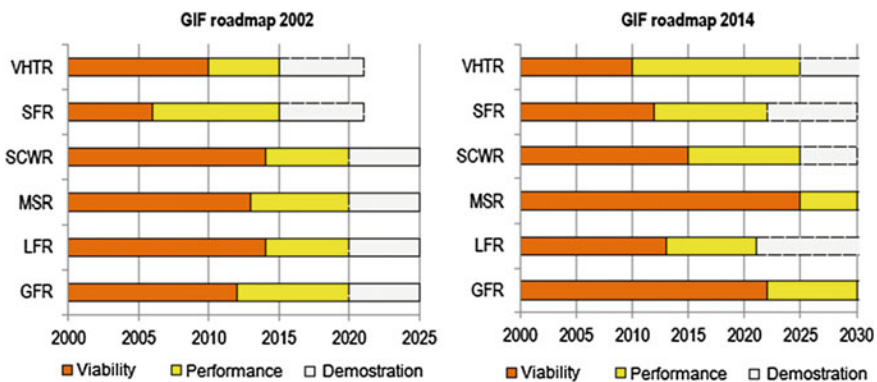


Fig. 2.2 System development timelines as defined in the original Roadmap in 2002 and in the 2014. *Source* Technology Roadmap Update for Generation IV Nuclear Energy Systems 2014 (These timelines are indicative and may change, depending on the components are not validated at the planned dates)

System	CA	CN	EU	FR	JP	KR	RU	CH	US	ZA
SFR		✓	✓	✓	✓	✓	✓		✓	
VHTR		✓	✓	✓	✓	✓		✓	✓	
SCWR	✓		✓				✓			
GFR			✓	✓	✓			✓		
LFR			P		P		P			
MSR			P	P			P			
✓ = Signatory to the System Arrangement; P = signatory to the Memorandum of Understanding; Argentina, Brazil, and the United Kingdom are inactive.										

Fig. 2.3 Status of the GIF system arrangements and MOU. *Source* Technology Roadmap Update for Generation IV Nuclear Energy Systems, 2014

2.2.3.1 The Gas Cooled Fast Reactor (GFR)

This concept features a fast-neutron-spectrum, a He-gas cooled reactor and a closed fuel cycle. A fast spectrum would efficiently convert fertile uranium into fissile fuel and manage the actinides by burning them for energy (GIF 2014).

The high outlet temperature of helium coolant makes it possible to deliver electricity, hydrogen or process heat with high efficiency. The reference design is a 2400-MWth/1100-MWe, helium-cooled system operating with an outlet temperature of 850 °C using three indirect power conversion systems with a combined cycle (an indirect cycle with helium on the primary, a Direct-Bryton cycle on the secondary circuit, and a steam cycle on the tertiary circuit) (GIF 2014).

Several fuel forms are candidates that hold the potential to operate at very high temperatures and to ensure an excellent retention of fissile products: Composite ceramic clad mixed actinide carbide fuel or advanced fuel particles. The core configurations may be based on prismatic blocks, pin or plate-based hexagonal fuel assemblies. The GFR reference design has an integrated, on site spent fuel treatment and refabricating plant (GIF 2009). It uses a direct-cycle helium turbine of electricity generation or can optionally use its process heat for thermochemical production of hydrogen. Hydrogen as a nuclear energy carrier is considered for a future no pollution energy economy with fuel cells directly producing electricity from hydrogen and releasing steam and water as a product.

Through the combination of a fast spectrum and full recycle of actinides, the GFR minimises the production of long-lived radioactive waste. The GFR’s fast spectrum also make it possible to use available fissile and fertile material including depleted uranium considerably more efficiency than thermal spectrum gas cooled reactors using the once-through fuel cycle (Fig. 2.4).

2.2.3.2 The Sodium Cooled Fast Reactor (SFR)

The SFR uses liquid sodium as the reactor coolant, allowing a low-pressure coolant system and high-power-density operation with low coolant volume fraction in the

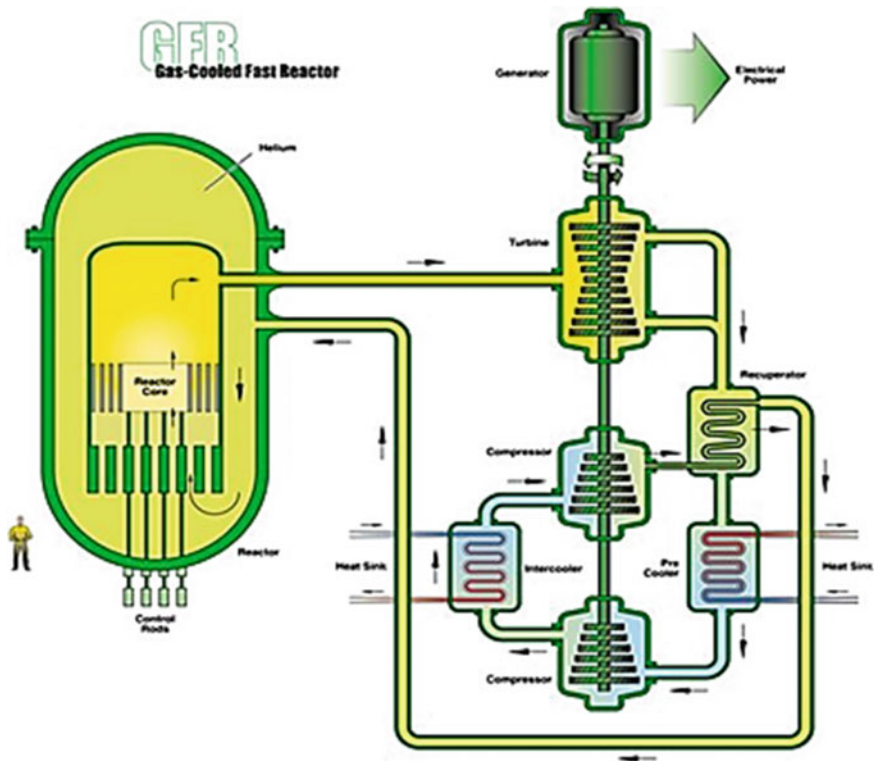


Fig. 2.4 Gas Cooled Fast Reactor (GFR). *Source* Technology Roadmap Update for Generation IV Nuclear Energy Systems, 2002

core. Nuclear power reactor size options under consideration range from small, 50 to 300 MWe modular reactors to larger units up to 1500 MWe. The outlet temperature range is 500–550 °C for the options under consideration (GIF 2014).

The fuel cycle employs a full actinide recycle with three configurations: pool, loop and modular. A large size (600–1500 MWe) loop-type reactor with mixed uranium-plutonium oxide fuel and potentially MA-bearing fuel, supported by a fuel cycle with advanced aqueous processing at a central location serving a number of units; an intermediate-to-large size (300–1500 MWe) pool-type reactor with oxide or metal fuel; and a small size (50–150 MWe) modular type reactor with metal-alloy fuel (uranium-plutonium-MA-zirconium), supported by a fuel cycle based on pyro-metallurgical processing in facilities integrated with the nuclear power reactor (GIF 2014).

The SFR is designed for management of high-level wastes and, in particular, management of Pu and other actinides. Important safety features of the system include a long thermal response time, a large margin to coolant boiling, a primary

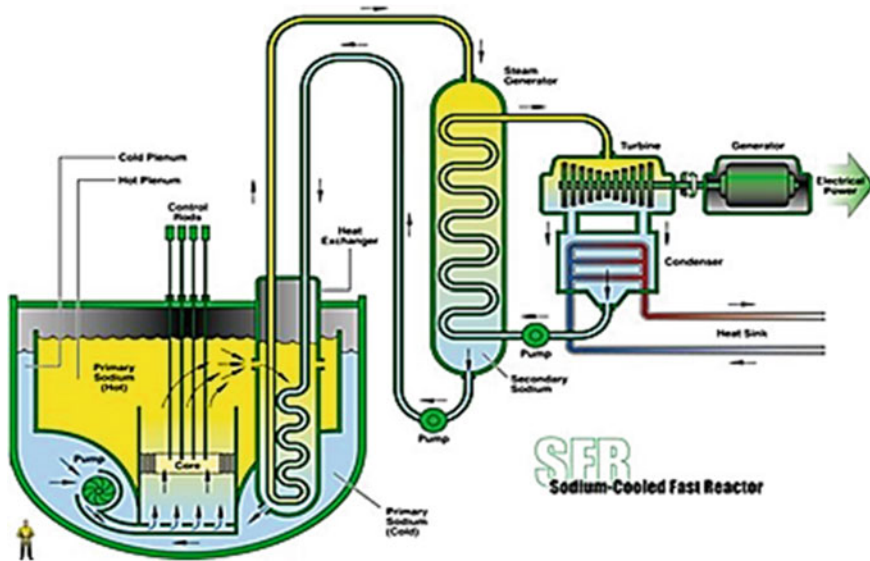


Fig. 2.5 Sodium Cooled Fast Reactor (SFR). *Source* Technology Roadmap Update for Generation IV Nuclear Energy Systems, 2002

system that operates near atmospheric pressure and intermediate sodium system between the radioactive sodium in the primary system and the power conversion system. Water/steam and alternative fluids are considered for the power conversion system to achieve high performance in terms of thermal efficiency, safety and reliability (GIF 2014).

With innovations to reduce capital cost, the SFR can competitively serve markets for electricity. Research must decide a choice between a metal alloy or a metal oxide fuel. An economic consideration is the choice of structural components for tubes and pipes. Ferritic steels with 12% Cr could be considered since they possess better strength at high temperature than austenitic steels (GIF 2002).

The SFR's fast spectrum also makes it possible to use available fissile and fertile materials, including depleted uranium, more efficiently than thermal spectrum reactors with once-through fuel cycles. The good management of the actinides is expected as well as good resource life (Fig. 2.5).

2.2.3.3 The Supercritical Water Cooled Reactor (SCWR)

The SCWR system is a high temperature, high-pressure water-cooled reactor that operates above the thermodynamic critical point of water at 374 °C, 22.1 MPa or 705 °F and 3208 psia (GIF 2002).

The supercritical water coolant enables a thermal efficiency about one-third higher than current light-water reactors, as well as simplification in the balance of

the nuclear power plant. The balance of the plant is considerably simplified because the coolant does not change phase in the reactor and is directly coupled to the energy conversion equipment. However, steam above the critical point is highly corrosive and requires special designed materials. The reference system is 1500 MWe with an operating pressure of 25 MPa and a nuclear power reactor outlet temperature of 510 °C, possibly increasing up to 625 °C. The fuel is UO₂. Passive safety features are incorporated similar to those of simplified boiling water reactors (SWBRs) (GIF 2009).

The SCWR system is primarily designed for efficient electricity production, with an option for actinide management based on two options in the core design: The SCWR may have a thermal or fast-spectrum reactor; the second is a closed cycle with a fast-spectrum reactor and full actinide recycle based on advanced aqueous processing at a central location. The concept may be based on current pressure-vessel or on pressure-tube reactors, and thus may use light water or heavy water as a moderator (Ragheb 2014) (Fig. 2.6).

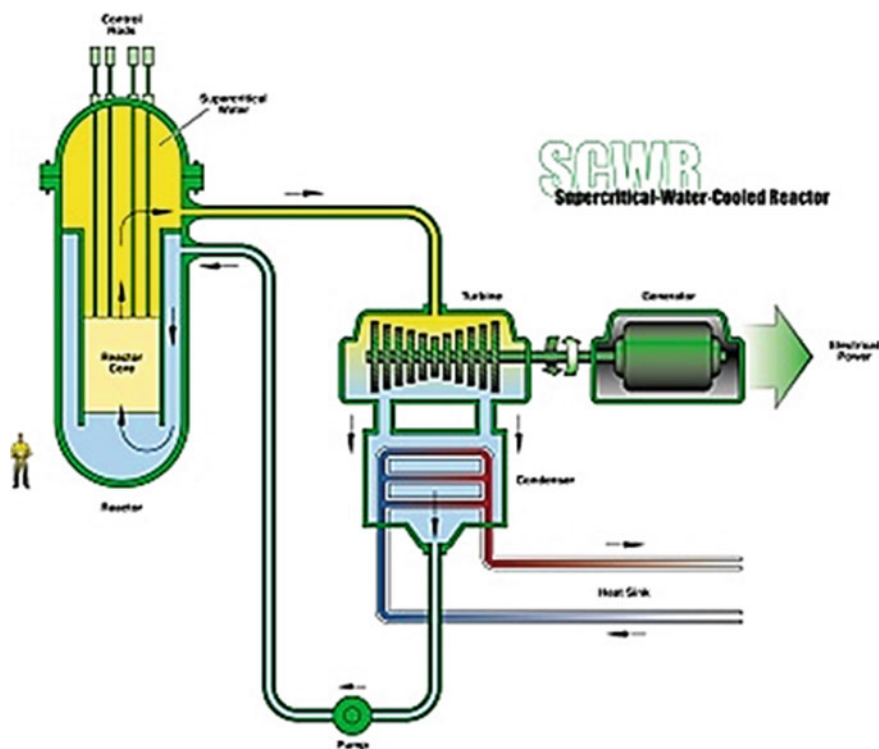


Fig. 2.6 Supercritical Water Cooled Reactor (SCWR). *Source* Technology Roadmap Update for Generation IV Nuclear Energy Systems, 2002

2.2.3.4 Very High Temperature Reactor (VHTR)

The VHTR is the next generation in the development of high-temperature nuclear power reactors and is primarily dedicated to the cogeneration of electricity, hydrogen and heat processes for industry. Hydrogen can be extracted from water by using thermo-chemical, electro-chemical or hybrid processes. The nuclear power reactor is cooled by helium gas and moderated by graphite with thermal neutron spectrum and a core outlet temperature greater than 900 °C, potentially more than 1000 °C in the future, to support the efficient production of hydrogen by thermo-chemical processes. The high outlet temperature also makes it attractive for the chemical, oil, and iron industries. The VHTR has for high burnup (150–200 GWd/tHM), passive safety, low operation and maintenance costs, and modular construction (GIF 2009).

Two baseline options are available for the VHTR core: The pebble bed type and the prismatic block type. The fuel cycle will initially be once-through with low-enriched uranium fuel and very high fuel burnup. The system has the flexibility to adopt closed fuel cycles and offers burning of transuranic. Initially, the VHTR will be developed to manage the back end of an open fuel cycle. Ultimately, the potential for a closed fuel cycle will be assessed (GIF 2009).

The electric power conversion may employ either a direct (helium gas turbine directly placed in the primary coolant loop) or indirect (gas mixture turbine) Brayton cycle. In the near term, the VHTR will be developed using existing materials, whereas its long term development will require new and advanced materials (GIF 2009) (Fig. 2.7).

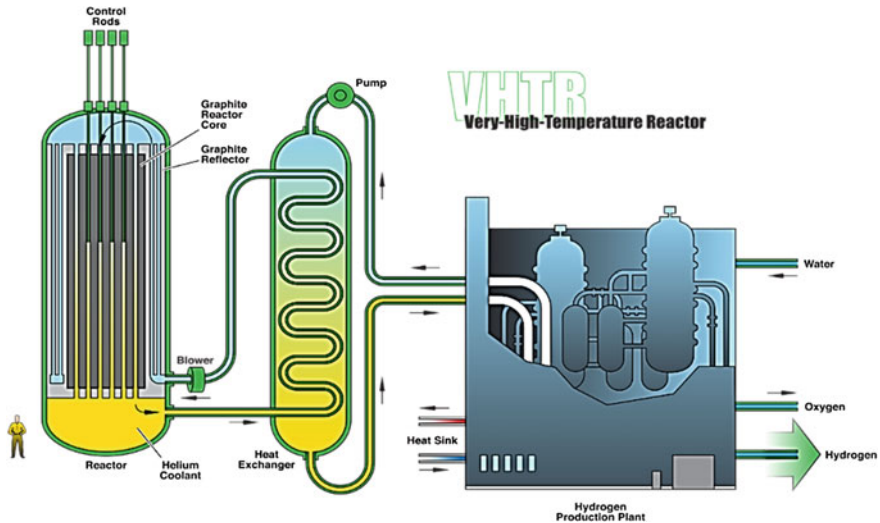


Fig. 2.7 Very High Temperature Reactor (VHTR). *Source* Technology Roadmap Update for Generation IV Nuclear Energy Systems, 2002

The reference pilot nuclear power reactor is a 600 MWth core connected to an intermediate heat exchanger to deliver process heat.

Some advantages of this type of reactor are: the benefit of the strong negative temperature coefficient of reactivity, the high heat capacity of the graphite core, the large temperature increase margin and the robustness of TRISO fuel introducing a nuclear power reactor concept that does not need off-site power to survive multiple failures or severe events.

2.2.3.5 Molten Salt Reactor (MSR)

The MSR system produces fission power in a circulating molten salt fuel mixture with an epithermal neutron spectrum reactor with graphite core channels, and full actinide recycle fuel. The MSR can be designed to be a thermal breeder using the Th-232 to U-233 fuel cycle (GIF 2002).

MSR can be divided into two subclasses. In the first subclass, fissile material is dissolved in a molten fluoride salt. In the second subclass, the molten fluoride salt serves as the coolant of a coated particle fuelled (GIF 2014).

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 an epithermal spectrum. The heat generated in the molten salt is transferred to a secondary coolant system through an intermediate heat exchanger, and then through a tertiary heat exchanger to power conversion system. The reference nuclear power plant has a power level of 1000 MWe. The system has a coolant outlet temperature of 700 °C, possibly ranging up to 800 °C, allowing improved thermal efficiency (GIF 2002).

The closed fuel cycle can be tailored to the efficient burn up of plutonium and minor actinides. The MSR's liquid fuel allows addition of actinides such as plutonium and avoids the need for fuel fabrication. Actinides, and most fission products, form fluorides in the liquid coolant. Molten fluoride salts have excellent heat transfer characteristics and very low steam pressure, which reduce stresses on the vessel and piping.

An Engineered Safety Feature involves a freeze plug where the coolant is cooled into a frozen state. Upon an unforeseen increase in temperature, this plug would melt and the liquid content of the reactor flow down into emergency dump tanks where it cannot continue the fission and ensure the safety in cooling. In absence of moderation by the graphite, the coolant would be in a subcritical safe state (Fig. 2.8).

2.2.3.6 Lead Cooled Fast Reactor (LFR)

The LFR system features a fast-spectrum Pb or Pb.Bi eutectic liquid metal-cooled nuclear power reactor and a close fuel cycle for efficient conversion of fertile uranium and management of actinides. The fuel is composed of fertile uranium and transuranic, and is metal or nitride based. The plant can be large and monolithic

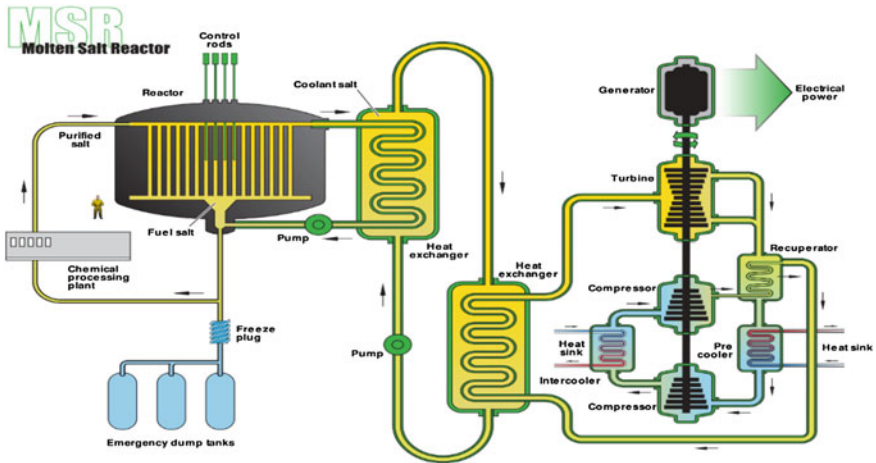


Fig. 2.8 Molten Salt Reactor (MSR). *Source* Technology Roadmap Update for Generation IV Nuclear Energy Systems, 2002

with a factory manufactured battery of 1200 MWe or it could be a modular system with 300–400 MWe or it could be a small battery of 50–150 MWe that features a very long refuelling interval (GIF 2002) (Fig. 2.9).

It has a fast neutron spectrum with a closed full actinide recycle fuel cycle with central or regional fuel cycle facilities. The LFR is cooled by natural convection

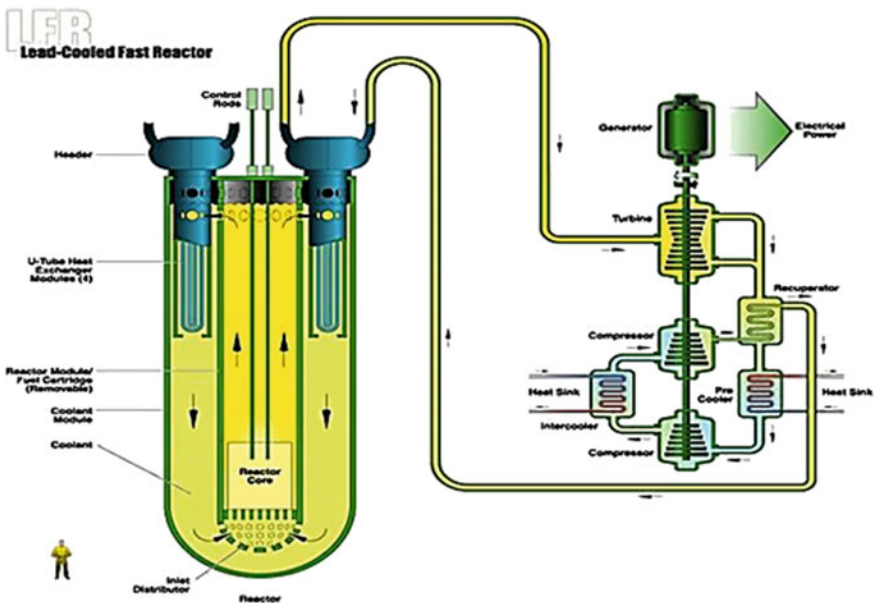


Fig. 2.9 Lead Cooled Fast Reactor (LFR). *Source* Technology Roadmap Update for Generation IV Nuclear Energy Systems, 2002

with a nuclear power reactor coolant temperature of 550 °C, possibly ranging up to 800 °C with advanced materials. The high temperature enables the production of hydrogen by thermo-chemical processes (GIF 2002).

The small size LFR is designed as a nuclear battery. It is a small factory-built turnkey nuclear power plant operating on closed fuel cycles with very long refuelling intervals of 15–20 years' cassette core or replaceable reactor module. Its features are designed to meet market opportunities for electricity production on small grids and for developing countries that may not wish to deploy an indigenous fuel cycle infrastructure to support their nuclear energy systems. The battery system is designed for electricity distrusted generation and other energy products, including hydrogen and fresh water obtained through sea water desolation (GIF 2002).

2.2.4 The Future of the Generation IV Nuclear Power Reactors

It is important to take into account that some Generation IV systems enjoy a more advanced state of development than others and each one need to make an effort in research and development for specific issues of each design, but there are areas in common which follow the same line of investigation, so it is possible to join efforts to improve common topics. The common areas encompass: Fuel cycles, fuels and materials choices, energy products, risk and safety, economics, proliferation, and physical protection concerns.

(a) Gas Cooled Fast Reactor: Specific Challenges and Possibilities

It is the only Generation IV design with no operating antecedent, so a prototype in not expected before 2022. However, a 75 MWt experimental technology demonstration GFR, ALLEGRO, is planned by Euratom to be built from 2018. It will incorporate all the architecture and the main materials and components foreseen for the GFR without the power conversion system. Euratom, France, Japan and Switzerland have signed on to System Arrangements (SA) for the GFR under the Framework Agreement (WNA 2016).

The General Atomics has team up with Chicago Bridge and Iron, Mitsubishi Heavy Industries and Idaho National Laboratory to develop the Energy Multiplier Module,³ according to WNA sources, but is not part of Generation IV programme or mentioned in the 2014 roadmap.

In their report 2014, GIF poses the following ten-year objectives to reach goals and to meet timelines:

³Energy Multiplier Modular is an advanced modular reactor expected to produce 265 MWe of power at 850 °C and be fully enclosed in an underground containment structure for 30 years without requiring fuel.

- Reference concept of 2400 MWth nuclear power reactor capable of breakeven breeding;
- Improving the design for the safe management of loss-of-coolant accidents including depressurisation and a robust removal of decay heat without external power supply;
- Advancing suitable nuclear fuel technologies with out-of-pile and irradiation experiments;
- Building experimental facilities for qualifying the main components and systems;
- Design studies for a small experimental nuclear power reactor.

There are four challenges need to be overtaken to achieve GIF objectives: Testing the structure for resiliency during radiation under temperatures up to 1400 °C; improvement in heat removal at low pressures of the He coolant in case of leakage and low thermal inertia; study of passive approaches of natural convection cooling; and the development of semi-passive heavy gas injectors and conduction paths.

(b) Lead Cooled Fast Reactor: Specific Challenges and Possibilities

A two-stage development programme leading to industrial deployment is envisaged by 2025 for reactors operating with relatively low temperature and power density, and by 2040 for more advanced higher-temperature designs.

In Japan, two basic design concepts have been developed: a small LFR called LSPR and a direct contact PBWFR. In parallel, accelerator driven system (ADS) activities have been performed. At present, the experimental activities are concentrated on basic research related to thermal-hydraulics, material corrosion, oxygen sensor, and oxygen control (GIF 2014).

The Russian Federation is carrying out design activities for the BREST-300, expected to be in operation after 2020. In parallel activities are carried out on SVBR-100, a Lead-Bismuth Eutectic Cooled Reactor, based on the previous experience developed for naval propulsion systems (GIF 2014).

In Europe, significant activities included projects aimed at the conceptual design of an industrial-size plant, the conceptual design of a 300 MWth demonstrator called “ALFRED” (Advanced Lead Fast Reactor European Demonstrator) and the activities on MYRRHA (an accelerator driven lead-bismuth cooled system) designed by SCK-CEN in Belgium (GIF 2014).

In the U.S., only limited development of the Small, Sealed, Transportable, Autonomous Reactor (SSTAR) has been carried out. However, private investors are considering possible modifications of this design to shorten its implementation phase, and there is some industrial interest in promoting a LFR concept (GIF 2014).

In China, the Chinese Academy of Sciences started in 2011 a new effort to develop an ADS. The China Lead-Based Reactor (CLEAR) was selected as the reference reactor. The CLEAR development plan includes three phases: The first being a 10 MWth LBE-cooled research device (CLEAR-I), with both critical and sub-critical modes of operation, expected to be built roughly 2017; the second phase

(CLEAR-II) is a 100 MWth ADS experimental reactor and expected to be built around 2022; and the third phase (CLEAR-III) is a 1000 MWth ADS demonstration reactor and envisaged to be built approximately 2032 (Bai et al. 2011).

In Republic of Korea, R&D activities on LFR are on-going since 1996. Helios, one of the largest test loops, has been operated in both forced and natural circulation conditions of PEACER (Proliferation resistance, Environment friendly, Accident tolerant, Continual, Economics Reactor). The results were published in the framework of the OECD/NEA Task Force on Benchmarking of Thermal-hydraulic Loop Models for Lead-Alloy-Cooled Advanced Nuclear Energy Systems (LACANES). Advanced corrosion-resistant materials have been developed and tested in both static and dynamic conditions. SMR design has been developed to explore their potential as distributed power/heat sources (GIF 2014).

The ten-year objectives by GIF are:

- Prototypes expected after 2020: Pb–Bi-cooled SVBR-100, Brest-300 in Russia;
- Proceeding with detailed design and licensing activities;
- Preliminary analyses of accidental transients, including earthquake and in-vessel steam generator pipe ruptures;
- Main R&D efforts will be concentrated on: Materials corrosion and development of a lead chemistry management system; core instrumentation; fuel handling technology and operation; advanced modelling and simulation; fuel development and possibly nitride fuel for lead-cooled reactors; actinide management (fuel reprocessing and manufacturing); In Service Inspection and Repair (ISI&R) (techniques for operate medium, seismic impact).

(c) Supercritical Water Cooled Reactor: Specific Challenges and Possibilities

Since the SCWR builds both on much BWR experience and that from hundreds of fossil-fired power plants operated with supercritical water, it can readily be developed, and the operation of a 30–150 MWe technology demonstration nuclear power reactor is targeted for 2022 (WNA 2016).

Japanese studies on a pressure-vessel design have confirmed target thermal efficiency of 44% with 500 °C core outlet temperature, and estimate a potential cost reduction of compared with present PWRs. Safety features are expected to be similar to ABWRs. Canada is developing a pressure-tube design with heavy water moderation. Euratom, Canada, and Japan have signed on a SA for the SCWR under the Framework Arrangement. In 2011, Russia joint them, followed by China in 2014. Project arrangement are pending for thermal-hydraulics and safety. Pre-conceptual SCWR designs include CANDU (Canada), LWR (Euratom) and Fast Neutron (Japan) (WNA 2016).

The ten-year objectives by GIF are:

- Two baseline concepts (pressure-vessel-based and pressure-tube-based);
- R&D: Advancing conceptual designs of baseline concepts and associated safety analyses; more realistic testing of materials to allow final selection and qualification of candidate alloy for all key components; out-of-pile fuel assembly

testing; qualification of computational tools; first integral component tests and start of design studies for a prototype; and in-pile tests of a small scale fuel assembly in a nuclear reactor;

- Definition of a SCWR prototype (size and design features).

Issues for development include corrosion and stress corrosion cracking, radiolysis as a function of temperature and fluid density, water chemistry, dimensional and micro-structural stability and strength, embrittlement and creep resistance. The effects of neutrons, gamma radiation and impurities introduced into the primary system on water radiolysis needs to be studied. Water flow could affect the criticality safety of the system, since cold water would have a higher moderating ability possibly leading to a power surge.

(d) **Very High Temperature Reactor: Specific Challenges and Possibilities**

While the original approach for VHTR at the start of the Generation IV programme focused on very high outlet temperature and hydrogen production, current market assessments have indicated that electricity production and industrial processes based on high temperature steam that require outlet temperatures of 700–850 °C, already have a great potential for applications in the next decade and also reduce the technical risk associated with higher outlet temperature. As a result, over the past decade, the focus of design studies has moved from higher outlet temperatures designs such as GT-MHR and PBMR to lower outlet temperature design such as HTR-PM in China and the NGNP in the U.S. (GIF 2014).

The ten-year objectives by GIF are:

- In the future, the main focus will be on VHTR with core outlet temperatures of 700–950 °C;
- Further R&D on materials and fuels should enable higher temperatures up to above 1000 °C and a fuel burnup of 150–200 GWd/tHM;
- Development of further approaches to set up higher-temperature process heat consortia for end-users interested in prototypical demonstrations;
- Development of the interface with industrial heat users-intermediate heat exchanger, ducts, valves and associated heat transfer fluid:
 - Advancing H₂ production methods in terms of feasibility and commercial viability to better determine process heat requirements for this application.
 - Regarding nuclear safety: Verify the effectiveness and reliability of the passive heat removal system; confirm fuel resistance to extreme temperatures (~1800 °C) through testing; and proceed with the safety analyses of couple nuclear processes for industrial sites using process heat.

(e) **Molten Salt Reactor: Specific Challenges and Possibilities**

For the MSR, no SA have been signed, and collaborative R&D is pursued by interested members under the auspices of a provisional steering committee involving France, Russia and Euratom (WNA 2016). There will be a long lead time to prototypes, and the R&D orientation has changed since the project was set up,

due to increased interest. It now has two baseline concepts: the Molten Salt Actinide and Transmuter (MOSART), and the Molten Salt Fast Reactor (MSFR) (GIF 2014).

In 2011, a European project called “EVOL” (Evaluation and Viability of Liquid Fuel Fast Reactor Systems) started, in parallel with a complementary Russian project named “MARS” (Minor Actinide Recycling in Molten Salt). The common objective of these projects was to propose conceptual design for the best MSFR system configuration (GIF 2014).

The ten-year objectives by GIF are:

- A baseline concept: MSFR;
- Commonalities with other systems using molten salts (FHR, heat transfer systems);
- Further R&D on liquid salt physical chemistry and technology, especially on corrosion, safety-related issues and treatment of used salts.

(f) Sodium Cooled Fast Reactor: Specific Challenges and Possibilities

After entering in 2000s, the nuclear energy caught people's attention again for its capacity of supplying suitable energy without giving harmful effects to the environment such as global warming. In France, Russia, India, China, the Republic of Korea and Japan, each country made a development plan for the realisation of the next generation SFR technology, which has an economic competitiveness in parallel with further enhanced built-in safety features.

In Russia, although they have faced the slow-down phase in the past, such as a postponement of the construction of BN-800 reactor, they are now attaining excellent capacity factor in the BN-600 reactor, have complemented the construction of the BN-800 reactor and achieved the first critically in 2014. The BN-1200 design has been in progress as the next generation reactor (Pioro 2016).

In China, an experimental fast reactor has been connected to the grid in 2011 as the result of vigorous R&D as a response to the foreseen large increase in the domestic energy demand. Then a prototype reactor, CFR-600 and the following commercial reactor, CFR-1000 are planned. India is also about to start a prototype fast breeder reactor (PFBR) (Pioro 2016).

France is proceeding a Generation IV SFR prototype project called “ASTRID” (Advanced Sodium Technological Reactor for Industrial Demonstration) and the Republic of Korea and Japan proceed in their design of Prototype Generation IV Sodium Cooled Fast Reactor (PGSFR) and the Japanese Sodium Cooled Fast Reactor (JFSR), respectively (Pioro 2016).

The U.S. is continuing a modular SFR development whereas 4S, PRISM and Travelling Wave Reactor-Prototype are being developed in the industry (Pioro 2016).

The ten-year objectives by GIF are:

- Three baseline concepts (pool, loop and modular configurations);
- Several sodium cooled reactors operational or under construction (e.g. in China, India, Japan and Russia);

- Develop advanced national SFR demonstrators for near-term deployment (France, Japan and Russia); proceed with respective national projects in China, the Republic of Korea, and India.

In the coming years, the main R&D efforts will be concentrated on: safety the operation (improving core inherent safety and instrumentation and control, prevention and mitigation of sodium fires, prevention and mitigation of severe accidents with large energy releases, ultimate heat sink, ISI&R consolidation of common safety design criteria; advanced fuel development (advanced reactor fuels, MA-bearing fuels); component design and balance plant (advanced cycles for energy conversion, and innovative component design); used fuel handling schemes and technologies; system integration and assessment; implementation of innovative options and economic evaluations, and operation optimisation.

2.3 European Pressurised Reactor (EPR)

The EPR is an advanced nuclear power reactor of evolutionary design and, as such, incorporates improvements arising from accumulated operating experience, new passive intrinsic safety systems of high reliability and an advanced technology instrumentation and control aimed at eliminating or mitigate operational human mistakes. Thus, the EPR offers great progress both in technology and economic, in addition, incorporates a high safety level and produces less high activity waste, reduces notably the energy cost, flexibility and availability operation, along with the better use of fuel.

2.3.1 EPR Design Project

In 1989, Framatome (France) and Siemens (Germany), the most experienced makers of European nuclear power plants, decided to cooperate to design a new PWR reactor generation, so they set up a joint venture, which they called “Nuclear Power International (NPI)”. At the end of 1991, the electrical French company Electricité de France (EdF) and the mainly part of the electrical German operators of nuclear power plants merged their development programmes with NPI, and in this manner, EPR design was started.

For this task, both manufacturers contributed the experience gained through the design, the construction, putting into service and the operation of existing PWRs in France and Germany, and they carried out an exhaustive analysis of technical solutions, comparing and evaluating them before their integration into EPR design.

Thanks to the successfully collaboration of all parties involved, it has been achieved very satisfactory results, not only for superposition of existing design

features, otherwise throughout a carefully reassessment and combination of the best alternatives. The goal was to achieve a design that merged both features French and German designs so that the optimal solution was reached with what the best of each one. As a result, the EPR is the direct descendent of the French N4 series nuclear power reactors, which are the most modern reactors in operation today, and Konvoi Germans, the latest design of Siemens before its merger with Framatone.

The EPR reactor, therefore, provides new technologies based on about a well-know and safe models like are N4 and Konvoi. For this reason, its design has been based on an update and improve of the above mentioned type of nuclear power reactors, giving rise to a safer, more efficient and more competitive reactor.

Main objectives assigned to EPR were twofold:

- After a carefully evaluation of specific passive safety systems features, it was decided to design the EPR following an evolutionary approach: The advantage of founding an advanced design on operational experience from approximately 100 nuclear power reactors constructed by Framatone and Siemens was deemed by the designers to be quite important;
- As important as the evolutionary feature, was the objective to ensure the competitiveness of nuclear power generation in comparison with other alternative energy sources. EPR was intended to provide a significant improvement in terms of power generation costs as compared to most modern nuclear power reactors, including gas power plants with combined cycles. To match this objective a large unit power size was selected, i.e. in the 1600 MWe range (Debontride 2006).

2.3.2 *EPR Description*

The EPR reactor is a PWR with a rated thermal power of 4500 MW and an electrical power output around 1630 MW depending on conventional island technology and heat sink characteristics (Ardron 2009).

The EPR evolutionary design is based on experience gained many years of operation of LWR worldwide. The EPR primary system design, loop configuration and other main components are similar of currently operating PWRs, giving a proven foundation for the design.

Relative to current generation PWRs, the EPR design philosophy has the following objectives:

- To reduce core damage frequency;
- To reduce the frequency of large releases of radioactivity;
- To mitigate severe accidents;
- To protect critical systems from external events such as aircraft impact;
- To achieve an improved plant availability factor (above 90%);

- To give extended flexibility for different fuel cycles lengths and capability for load following;
- To give increased saving on uranium consumption per MWh produced;
- To achieve further reduction in long-lived actinides generation per MWh through improved fuel management;
- To provide a plutonium recycling capability with a core able to accommodate up to 50% of MOX⁴ fuel assemblies (Ardron 2009).

The EPR operating design life of 60 years, reduced fuel consumption and waste production per unit energy output, contribute to long term sustainability. Economic viability is provided by the fact that (Ardron 2009):

- The investment and operating costs are balanced by a large power output;
- The large scale core with a low power density provides an efficient use of fuel;
- The high steam pressure leads to a high net efficiency;
- The high availability is ensured by the use of proven technology and Konvoi design features which allow short outages (Ardron 2009).

The EPR reactor is a four-loop PWR whose reactor coolant system (RCS) comprises a reactor pressure vessel (RPV) containing the fuel assemblies, a pressuriser (PSR) including control systems to maintain system pressure, one reactor coolant pump (RCP) per loop, one steam generator (SG) per loop, associated piping, and related control and protection systems. These components are standardised for all EPR projects.

In PWRs ordinary (light) water is utilised to remove the heat produced inside the reactor core by the thermal nuclear fission. The water in the core acts to slow down (moderate) the neutrons. Slowing down neutrons is necessary to sustain the nuclear chain reaction. The heat produced inside the reactor is transferred to the turbine through the steam generators. Only heat energy is exchanged between the reactor cooling circuit (primary circuit) and the secondary circuit used to feed the turbine. No exchange of cooling water takes place.

In the RCS, the primary cooling water is pumped through the reactor core and the tubes inside the SGs, in four parallel closed loops, by four RCPs powered by electric motors. The reactor operating pressure and temperature are such that the cooling water does not evaporate in the primary circuit, but remains in the liquid state, increasing its cooling effectiveness. A PSR, connected to one of the coolant loops is used to control the pressure in the RCS. Feedwater entering the secondary side of the steam generators absorbs the heat transferred from the primary side and evaporates to produce saturated steam. The steam is dried inside the steam generators then delivered to the turbine. After exist the turbine, the steam is condensed and returned as feedwater to the SGs. A generator, driven by the turbine, generates electricity (Fig. 2.10).

⁴Note with some plants modifications, 100% of the core could be composed of MOX fuel assemblies.

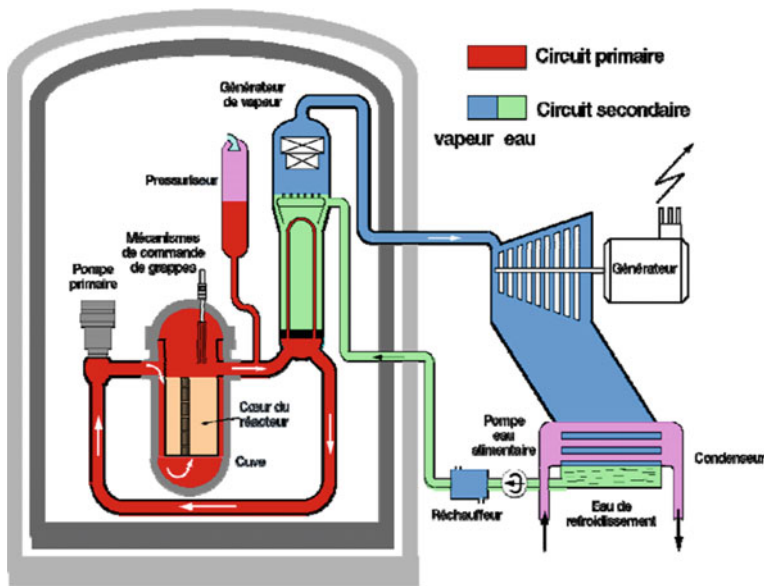


Fig. 2.10 European Pressurised Reactor (EPR). Source AREVA (2008)

The EPR plant layout is shown in Fig. 2.11. Referring to that figure, the EPR plant comprises a reactor building, a fuel building, four safeguard buildings, two diesel buildings, a nuclear auxiliary building, a waste building, a turbine building and C.I. electrical building.

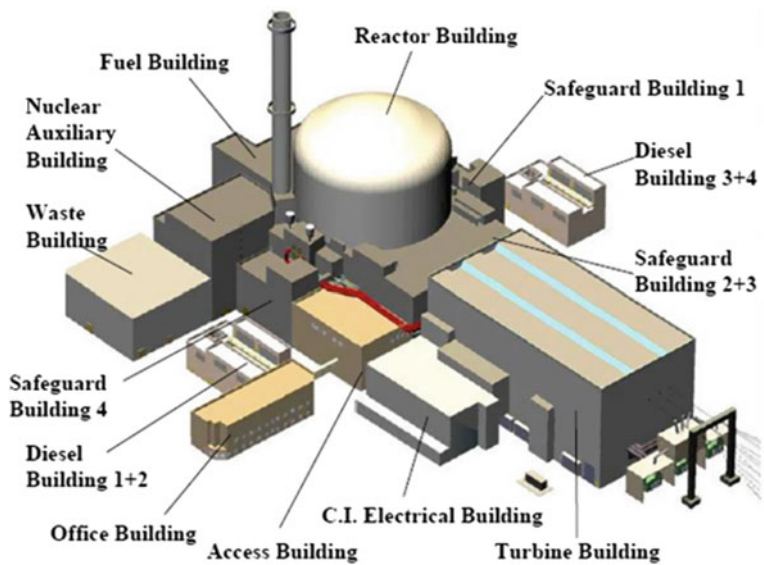


Fig. 2.11 Typical EPR layout. Source AREVA (2008)

The reactor building is surrounded by the four safeguard buildings and the fuel building. The internal structures and components within the reactor building, fuel building and two safeguard buildings (including the plant main control room) are protected against aircraft hazard and external explosions. The other two safeguard buildings are not protected against aircraft hazard; however, they are geographically separated by the reactor building, which prevents both buildings from being simultaneously affected by such a hazard.

2.3.3 Safety, Competitiveness and Flexibility

(a) Safety properties:

According to Areva data base (2016), a twofold strategy is pursued for the EPR safety requirements:

- Unrivalled level of safety: Resistance to plane crashes and seismic vibrations; quadruple safety device redundancy; core meltdown risk further reduced and minimisation of the consequences from such an accident thanks to a special compartment isolating the molten core;
- Active and passive safety systems: Designed as an extension of the Konvoi and N4 reactors, the EPR reactor combines active and passive safety systems to increase safety and provide better process control over plant operation.

These safety requirements are implemented by designing the nuclear power plant on a strong deterministic basis and, beyond this basis, by consideration of risk reduction measures.

Low probability events with multiple failures and coincident occurrences up to the total loss of safety-grade systems are considered, in addition to the deterministic basis design. Representative scenarios are defined for both mitigations of core melt and prevention of large releases, in order to provide a basis design for risk reduction features.

(b) Competitiveness:

Due to an early focus on economic competitiveness during the design process, the EPR offers significantly reduced power generation costs, estimated as being 10% lower than those of the most modern nuclear power reactors currently in operation, and about 20% less than those of large combined-cycle gas plants (Gerard et al. 2004).

According to Gerard et al. (2004), this competitiveness is achieved through:

- Unit power in the 1600 MW range, i.e. the highest unit power to date (a further power increase is possible without major changes, as the reactor equipment is already designed for a core thermal power of 4500 MWth);
- Between 36 and 37% efficiency depending on site conditions, the highest value ever for LWR;

- Construction time from pouring of the first concrete not exceeding 48 months;
- Service life increased to 60 years;
- Enhanced fuel utilisation;
- Up to 92% availability factor, on average, during the entire service life of the plant, obtained through long irradiation cycles, another shorter refuelling outages and increase maintenance;
- Environment protection. Reduction in fuel consumption per kWh and production of long-life waste products (−15%), through improved thermal efficiency and uranium utilisation (Areva source);
- An unrivalled experience on large projects.

(c) **Flexibility:**

Due to its considerable margins for fuel management optimisation, EPR core is designed for outstanding flexibility with respect to fuel cycle length and fuel management strategy: Reference cycle length is 18 months, but fuel cycle lengths up to 24 months, IN-OUT and OUT-In fuel management capabilities are offered. A great flexibility for using MOX (Mixed $\text{UO}_2\text{--PuO}_2$) fuel assemblies in the core, i.e. of recycling plutonium extracted from spent fuel assemblies is also provided (Debontride 2006).

In terms of operation, EPR is designed to offer the utilities a high level of manoeuvrability. It has the capacity to be permanent operated at any power level between 20 and 100% of its nominal power in a fully automatic way, with primary and secondary frequency controls in operation.

The EPR capability regarding manoeuvrability is a particular well adapted response to scheduled and unscheduled power grid demands for loads variations, managing of grid perturbations or mitigation of grid failures.

2.3.4 EPR Projects

(a) **Olkiluoto 3 (Finland)**

The construction of the Olkiluoto 3 nuclear power reactor in Finland commenced in August 2005. It was initially scheduled to go online in 2009, but the project has suffered many delays, and according to Areva operations are expected to start in 2018. The nuclear power reactor will have an electrical power output of 1600 MWe (net). The construction is a joint effort of French Areva and German Siemens through their common subsidiary Areva NP, for Finnish operator TVO. Initial cost estimates were about €3.7 billion, but the project has since seen several severe cost increments and delays.

Due to budget increase and delays in the construction of Olkiluoto 3, the Finland operator, TVO, has sue Areva for €2.6 billion and, at the same time, Areva has sue TVO for €3.52 billion. The claim includes payments delayed by TVO under the construction contract, and penalty interest totalling about €1.45 billion and

€135 million in alleged loss of profit. In May 2016 Areva NP called off arbitration negotiations for a settlement with TVO (WNA [2016](#)).

(b) Flamanville 3 (France)

First concrete was poured for the demonstration EPR reactor at the Flamanville nuclear power plant on 6 December 2007. As the name implies this will be the third nuclear power reactor on the Flamanville site and the second instance of an EPR being built. Electrical output will be 1630 MWe (net) and the project involves around €3.3 billion of capital expenditure from EdF. It is important to highlight that the construction of the Flamanville nuclear power plant also had suffered of certain delay and increase in the budget allocated to the construction of the EPR reactor on an estimated of €10.5 billion, three times its original estimate (WNN [2015](#)).

(c) Taishan 1 and 2 (China)

In 2006, Areva took part in the first bidding process for the construction of four new nuclear power reactors in China, together with Toshiba-owned Westinghouse and Russian Atomstroexport. However, Areva lost this bid in favour of Westinghouse's AP1000 reactors, in part because of Areva's refusal to transfer the expertise and knowledge to China. Following this Areva managed to win a deal in February 2007, worth about €8 billion for two EPRs located in Taishan, Guangdong Province in southern China, in spite of sticking to its previous conditions. The General Contractor and Operator is the China Guangdong Nuclear Power Company. As of December 2012, the two Taishan EPRs will cost about the same as the single EPR being built in the Finnish Olkiluoto estimated in €8.5 billion (WNN [2007](#)).

In November 2007, former French president Nicolas Sarkozy signed a US \$12 billion deal that will allow the third and fourth EPR units to be constructed in China (Nuclear Engineering International [2007](#)).

(d) Hinkley Point C (United Kingdom)

On March 2013, planning consent for Hinkley Point C nuclear power plant was given, and on October 2013, UK government have agreed with EdF, after more than two years of negotiation, that the French company will be guaranteed a strike price of £92.50 for every megawatt hour of power produced by the Hinkley Point C power plant for 35 years, around double the current market rate at the time (Gribben and Ronald [2013](#)).

Following an 11-month investigation into UK support for Hinkley Point C nuclear project, the European Commission approved a UK support package on October 2014. Because of that, Austria and Luxembourg, on June 2015, launched their appeal at the General Court of the European Union, challenging the European Commission's clearance decision (Buckworth et al. [2015](#)).

EdF and UK government were about to sign off the subsidiary deal for the £18 billion plant on 29 July 2016, after the board of EdF approved the project by ten votes to seven, Greg Clark, the new Business and Energy Secretary, announced

a new review that the final decision will now be delayed until in the early autumn. This announcement surprised EdF, whose directors were preparing to sign contracts with the government (Gosden and Swinford 2016).

(e) Possible future nuclear power plants

In February 2009, the Nuclear Power Corporation of India (NPCIL) signed a MoU with Areva to set up two 1650 MWe reactors at Jaitapur in Maharashtra. This was followed by a framework agreement in December 2010. NPCIL has ambitions to build up to 9900 MW at the Jaitapur site, equating to six EPRs, according to Areva sources. In July 2008, the French President announced that a second EPR would be built in France due to high oil and gas prices. Penly was chosen as the site in 2009, with construction planned to start in 2012. However, in 2011, following the Fukushima Daiichi nuclear accident, EdF postponed public consultations. In 2013, EdF confirmed there was no start plan for Penly, as expected demand did not warrant it (WNA 2016).

2.4 Small Modular Reactors (SMRs)

Further, a new group of nuclear power reactors, the so-called “Small Modular Reactors or SMRs” has been developed, with new important features, which do not fall into the above groups, but they can supply electricity to the market in countries without great financial resources, lack of well-trained work forces, relative small grid and moderate technology development.

The SMR systems adopt modularisation, by which the structures, systems, and components are shop-fabricated then shipped and assembled on site, thus the construction time for SMRs can be substantially reduced. Some of the SMRs are to be deployed as multiple-module power plants allowing utilities to add additional units and power conversion modules as demand for local power increases.

SMRs will use different approaches in comparison with large nuclear power reactors for achieving a high level of safety and reliability in their systems, structures, and components. These improvements will be the result of a complex interaction between design, operation, material, and human factors. Interest in SMRs continues to grow in several developed countries as an option for future power generation and energy security, but particularly in countries that are thinking to introduce, for the first time, the use of nuclear energy for electricity generation, according to IAEA sources.

2.4.1 Why the Interest in Small Modular Reactors

Small reactors and the modular construction of them are not a new concept. Historically, early reactors for commercial electricity production were of small size,

a consequence of the prudent engineering process of constructing plans starting at small ratings to gain the needed construction and operating experience necessary to move confidently to larger ratings. Now a half-century of experience, commercial civil reactors are being deployed with ratings of up to 1660 MWe. Additionally, small units were built for terrestrial deployment to provide electric power for remote, vulnerable military sites, for the propulsion of submarines, naval and commercial ships, and for aircraft propulsion.

However, starting from the mid-1980s, a new set of requirements have motivated the deployment of intentional smaller reactors in some countries aimed at the niche markets that cannot accommodate nuclear power plants with reactors of large capacity.

The main arguments in favour of SMRs are:

- Because of their size, construction efficiency and passive safety systems, the upfront capital investment for one unit is significantly smaller than for a large nuclear power reactor, and there is flexibility for increasing capacity. This reduces financial risks and could potentially increase the attractiveness of nuclear power to private investors and utilities (Kuznetsov and Lokhov 2011);
- Smaller nuclear reactors could represent an opportunity to develop new markets for nuclear power plants. In particular, SMRs could be suitable for areas with small electrical grids and for remote locations or, alternatively, in countries with insufficiently developed electrical infrastructure;
- Effective protection of plant investment from the potential to achieve a reactor design with enhanced safety characteristics;
- Possible reduction of the current emergency planning zone by virtue of smaller core inventory and potential for added safety design features. According to WAN, the emergency planning zone required is designed to be no more than about 300 m radius;
- Potential benefits regarding non-proliferation of nuclear material;
- Reduction of transmission requirements and a more robust and reliable grid;
- SMRs are better adapted to low growth rates of energy demand;
- Use of components which do not require the ultra-heavy forgings of today's gigawatt-scale nuclear power plants and are rail shippable (Carelli and Ingersoll 2014).

SMR often offer a variety of non-electrical energy products (heat, desalinated water, process steam, district heating mission or advanced energy carriers) via operation in a co-generation mode.⁵

⁵It is important to underline that co-generation is not unique to SMRs. However, the SMR power range corresponds well to the infrastructure requirements for non-electrical products (e.g. district heating) (Kuznetsov and Lokhov 2011).

2.4.2 Small Modular Reactors and Their Attributes

It is first worth defining what a “Small Modular Reactor (SMR)” is. SMR is defined as a reactor of advanced generation of nuclear power reactors to produce equivalent electrical power less than 300 MWe per unit, and designed to be built in factories in modular form and shipped to utilities for installation as demand arises.

The philosophy is to add an incremental number of small units at the same site as and when the electricity demand is there, or as and when the revenue from the previous units is such that another unit can afford to be built by the owners (National Nuclear Laboratory 2012).

The motivation in SMR design and potential implementation remains the same as the large nuclear power plants (i.e. reduced CO₂ emissions, energy security, and economics), but with additional proposed benefits, including safer new plant designs that require less investment.

The attributes of SMRs are:

- Small reactor size allowing transportation by truck (as well as by rail or barge) and installation in proximity to the users, such as residential housing areas, hospitals, military bases, or large government complexes;
- The compact architecture enables modularity of fabrication (in-factory), which can also facilitate implementation of higher quality standards. Factory assembly of the complete nuclear steam supply system and, therefore, short construction duration on site;
- Lower requirement for access to cooling water, therefore suitable for remote regions and for specific applications such as mining or desalination (WNA 2016);
- Small absolute capital outlay and an option of flexible capacity addition/removal through modular approach to plant design, deemed attractive to private investors (Kuznetsov and Lokhov 2011);
- Small power and compact architecture and usually employment of passive concepts. Therefore, there is less reliance on active safety systems and additional pumps, as well as AC power for accident mitigation;
- Individual containments and turbine generators for each of the reactor modules;
- Potential for sub-grade (underground or underwater) location of the reactor providing more protection and high level of safety and security from natural or man-made hazards;
- Lower power leading to reduction of the source term as well as smaller radioactive inventory in a reactor;
- Long refuelling interval and once-at-a-time whole core reloading on the site or at a centralised factory (as a future option) (Kuznetsov and Lokhov 2011);
- Ability to remove reactor module or in situ decommissioning at the end of the lifetime;
- Provision for flexible co-generation options (generating electricity with co-production of heat, desalinated water, synthetic fuels, hydrogen, etc.).

2.4.2.1 Safety Designs of Small Modular Reactors

In general, the engineering challenges of ensuring safety in SMRs are not qualitatively different from those of large nuclear power reactors. No matter the size, there must be systems in place to ensure that the heat generated by the reactor core is removed both under normal and accident conditions at a rate sufficient to keep the fuel from overheating, becoming damaged, and releasing radioactivity.

A major advantage of SMRs is their natural safety. No electrical supplies or pumps are required to cool the reactor following an incident, as this is achieved by natural convection and gravity coolant feed. This feature ensures the reactor will remain safe under severe accident conditions.

Natural (passive) safety systems reduce the capital and maintenance costs compared to large nuclear power reactors and fundamentally changes the economic equation in favour of SMR nuclear power generation, according to SMR Nuclear Technology.

Despite a large variety of SMR designs, they tend to share a common set of design principles to enhance plant safety (National Nuclear Laboratory 2012):

- Eliminate potential accident initiators if possible [e.g. avoid loss of coolant accident (LOCA)];
- Reduce probability of an accident occurring (e.g. reducing vessel dose during operations reduces likelihood of RPV fail);
- Mitigate consequences of potential accidents (e.g. increased volume of primary coolant slow down potential heat-up accidents).

Some of the typical features that enhance the safety, include (National Nuclear Laboratory 2012):

- Incorporation of primary system components into a single vessel;
- Increased relative coolant inventory in the primary reactor vessel;
- Smaller radionuclide inventory per reactor;
- Vessel and component layout that facilitate natural convection cooling of the core and vessel;
- More elective decay heat removal;
- Smaller decay heat per reactor;
- Enhanced resistance to seismic events.

It is also possible to enhance the security locating the reactor underground. This significantly reduces the potential impact of external events such as aircraft collision or natural disasters. Locating the reactor below ground also reduces the number of paths for fission product release following an accident. But this location could have a disadvantage as well, so in case of accident, emergency crews could have greater difficulty accessing underground reactors.

According to SMR Nuclear Technology, the key features depending on the type of SMRs are the following:

- Key features of Small Modular Light Water Reactors:
 - The most common power nuclear reactor type, with proven technology, and extensive accumulated operational experience;
 - Uses cheap demineralised water as the primary coolant;
 - Natural or pumped coolant circulation and passive back-up systems for safety;
 - Coupled to standard turbine/generator as used in fossil fuelled power plant;
 - In the PWR, the primary coolant water is kept under sufficient pressure to prevent it from boiling, and the heat extracted from the nuclear fuel is transferred to a secondary water circuit in a heat exchanger where steam is produced to drive a turbine.
- Key features of Small Modular Fast Neutron Reactors:
 - Very compact design due to high conductivity liquid metal coolant;
 - Higher efficiency than LWR due to higher operating temperature;
 - Very long operating time between refuelling (up to 30 years);
 - Inherent safety features.
- Key features of Small Modular Very High Temperature Gas Reactors:
 - Capable of operating at very high temperature for hydrogen production or high efficiency (50%) electricity generation;
 - Proven fuel technology;
 - Inherent safety features due to fuel type and gas coolant.

2.4.2.2 Proliferation Resistance and the IAEA Safeguards System

Proliferation resistance has become one of the primary topics to be addressed if new energy systems are going to be developed as any current nuclear system presents potential proliferation risks. SMR systems could raise specific proliferation concerns mainly because they could be deployed in: a) remote areas, b) small countries, c) in large numbers, d) in countries that are “newcomers” in nuclear industry, and e) can be used not only for electric generation, but also on potable water production, heat, industrial processes, among others. In this sense, the whole SMR system requires specific attention in order to reduce the attractiveness of fissile material that could be used for nuclear weapons. The strategies to increase proliferation resistance are presently oriented to prevent access to the fuel and/or develop reactor designs implying quite long refuelling periods (Polidoro et al. 2013).

The IAEA provides international verification of nuclear activities in a host state, through the implementation of nuclear safeguards that include inspections to verify facility design and nuclear inventory, and also instrumentation and other measures

that provide “continuity of knowledge” between inspections. The IAEA safeguards system is viewed as a key instrument of non-proliferation (Whitlock and Sprinkle 2012).

IAEA safeguards verify the operator’s declarations about activities involving nuclear material. These declarations address the receipts, shipment, storage, movement and production of nuclear material. Inspection intensity depends on the type of nuclear material used (depleted uranium, high enriched uranium, low enriched uranium, natural uranium, plutonium or thorium) and whether the material is irradiated.

Safeguards considerations take into account various aspects:

- Accessibility to the nuclear material;
- Whether the reactor facility is operated continuously;
- How the reactor facility is refuelled;
- Location and mobility of the reactor facility;
- Existence and locations of the other nuclear facilities in the state.

2.4.2.3 Economic Analysis of Small Modular Reactors

The full cost of electricity from SMRs have similar structures to large nuclear power reactors, and according to WNA, the economics of nuclear power involves consideration of several aspects:

- Capital costs, which include the cost of the establishment of a nuclear programme (for newcomers), cost of licensing, site preparation, construction, manufacturing, commissioning, and financing a nuclear power plant;
- Plant operation costs, which include the cost of fuel, operation, and maintenance (O&M), and a provision for funding the costs of decommissioning the plant, and treating and disposing of fuel and wastes;
- External costs to society for the operation, which in the case of a nuclear power plant is usually assumed to be zero, but could include the costs of dealing with a serious accident that is beyond the insurance limit and in practice need to be picked up by the government;
- Others costs such as taxes and levies, as well as grid and backup costs (transport, reserve capacity, etc.).

One of the main factors negatively affecting the capital cost of the SMRs is the lack of economy of scale. As a result, the specific (per MWe) capital costs of the SMR are expected to be tens to hundreds of percent higher than large nuclear power reactors (Lokhov et al. 2013).

The construction duration of the SMRs could, in principle, be significantly shorter than for large nuclear power reactors, especially in the case of factory-assembled reactors. This would result in important savings for financial costs, which are particularly significant if discount rate is high. Some SMRs could be fully factory-assembled, and transported to the deployment site. Factory fabrication is also subject to learning effects which could reduce the SMR capital costs.

The magnitude of this reduction is considered to be comparable or even higher to that of the effects for series build of plants constructed on site. In particular, full factory fabrication is possible for a barge-mounted plant.

According to the designers' estimates a full factory-fabricated, barge-mounted nuclear power plant could be 20% less expensive than land-based nuclear power plant with an SMR of the same type. Further decrease of SMR capital cost can be achieved due to learning effects of factory fabrication. However, to fully utilise this effect, series of at least 5–7 units are needed. In some advanced SMRs, significant design simplifications could be achieved through broader incorporation of size-specific inherent safety features that would not be possible for large nuclear power reactors. The designers estimate that these simplifications could reduce specific capital costs by at least 15%. Even if all of the above mentioned factors are taken into account where they are applicable, the investment component of the levelised cost for a SMR still appears to be higher than in the case of large nuclear power reactors (Lokhov et al. 2013).

The sum of the cost for O&M and the fuel cycle components for advanced SMRs is expected to be close to the corresponding value for a large nuclear power reactor (of similar technology). Lower O&M costs are expected for SMRs but, in contrast, the fuel costs could be higher in the case of a SMR than for large nuclear power reactors (in particular, because of lower fuel utilisation) (Lokhov et al. 2013).

The cost of electricity generation with SMRs might decrease for large scale serial production, which is very important for providing the competitiveness of SMRs. A large initial order of SMRs would be needed to launch the process and improve the economic competitiveness. On the other hand, to obtain a large order, one would already need to demonstrate the economic attractiveness of the SMR technology (Lokhov et al. 2013).

2.4.2.4 Waste Problem and Decommissioning

In addition to the problems mentioned above, there are other issues that need to be considered associated to the use of SMRs for electricity generation and other uses. These are:

- **Waste problem:** Proponents claim that with longer operation on a single fuel charge and with less production of spent fuel per reactor, waste management would be simpler. In fact, spent fuel management for SMRs would be more complex, and therefore more expensive, because the waste would be located on many more sites. In some proposals, the reactor would be buried underground, making waste retrieval even more complicated and therefore complicating retrieval of radioactive materials in the event of an accident (Makhijani and Boyd 2010);
- **Decommissioning:** The modular nature of the reactor components not only assists in the construction of the plant, but will also ease the decommissioning timescales. With smaller modules, the ability to dispose of the entire unit could

be feasible, including in the case of the cartridge type spent fuel. In addition, with many of the SMRs being based underground, there is the potential to back fill the site as is, simply removing the outer shell and buildings (National Nuclear Laboratory 2012).

2.4.3 Small Modular Reactor Designs

There are tens of SMR concepts and designs at various stages of development around the world. Some are being developed by universities as pure research and teaching projects, others by private investors looking to break into the new build market and several by the large international reactor vendors (Tables 2.1, 2.2, 2.3 and 2.4).

2.4.3.1 Light Water Reactors (LWR)

This type of SMRs is moderated and cooled by ordinary water and have the lowest technological risk, being similar to most operating power and naval reactors today. They mostly use enriched fuel to less than 5% U-235 with no more than six-year refuelling intervals, and regulatory hurdles are likely least of any small reactors.

They mostly have steam supply systems inside the reactor pressure vessel and others have conventional pressure vessels plus external steam generators. This type of SMR has enhanced safety features relative to a current LWRs and require conventional cooling steam condensers (Fig. 2.12).

Table 2.1 Small nuclear power reactors operating in 2016

Name	Capacity (MWe)	Type	Developer
CNP-300	300	PWR	CNNC, operational in Pakistan and China
PHWR-220	220	PHWR	NPCIL, India
EGP-6	11	LWGR	At Bilibino, Siberia (cogen)

Source WNA

Table 2.2 Small reactor designs under construction

Name	Capacity (MWe)	Type	Developer
KLT-40S	35	PWR	OKBM, Russia
CAREM	27	Integral PWR	CNEA & INVAP, Argentina
HTR-PM, HTR-200	2 × 105	HTR	INET, CNEC & Huaneng, China

Source WNA

Table 2.3 Small (25 MWe up) reactors for near-term deployment—development well advanced

Name	Capacity (MWe)	Type	Developer
VBER-300	300	PWR	OKBM, Russia
NuScale	50	Integral PWR	NuScale Power + Fluor, U.S.
Westinghouse SMR	225	Integral PWR	Westinghouse, U.S.
mPower	180	Integral PWR	Bechtel + BWXT, U.S.
SMR-160	160	PWR	Holtec, U.S.
ACP100	100	Integral PWR	NPIC/CNNC, China
SMART	100	Integral PWR	KAERI, South Korea
Prism	311	Sodium FNR	GE-Hitachi, U.S.
BREST	300	Lead FNR	RDIPE, Russia
SVBR-100	100	Lead-Bi FNR	AKME-engineering, Russia

Source WNA

Table 2.4 Small (25 MWe up) reactors designs at earlier stages

Name	Capacity	Type	Developer
EM2	240 MWe	HTR, FNR	General Atomics, U.S.
VK-300	300 MWe	BWR	RDIPE, Russia
AHWR-300 LEU	300 MWe	PHWR	BARC, India
CAP150	150 MWe	Integral PWR	SNERDI, China
ACPR100	140 MWe	Integral PWR	CGN, China
IMR	350 MWe	Integral PWR	Mitsubishi Heavy Ind, Japan
PBMR	165 MWe	HTR	PBMR, South Africa
SC-HTGR (Antares)	250 MWe	HTR	Areva, France
Xe-100	48 MWe	HTR	X-energy, U.S.
Gen4 module	25 MWe	FNR	Gen4 (Hyperion), U.S.
MCFR	Unknown	MSR/FNR	Southern Co, U.S.
TMSR-SF	100 MWt	MSR	SINAP, China
PB-FHR	100 MWe	MSR	UC Berkeley, U.S.
Integral MSR	192 MWe	MSR	Terrestrial Energy, Canada
Moltex SSR	c 60 MWe	MSR	Moltex, UK
Thorcon MSR	250 MWe	MSR	Martingale, U.S.
Leadir-PS100	36 MWe	Lead-cooled	Northern Nuclear, Canada

Source WNA

A brief technical description of the currently SMRs under development in several countries is included in the following paragraphs:

(1) **KLT-40S**

The KLT-40S nuclear power plant was developed on the basis of a standard KLT-40 type nuclear propulsion plant that has the experience of more than 250 reactor-years of failure-free operation. Components of the original plant have been

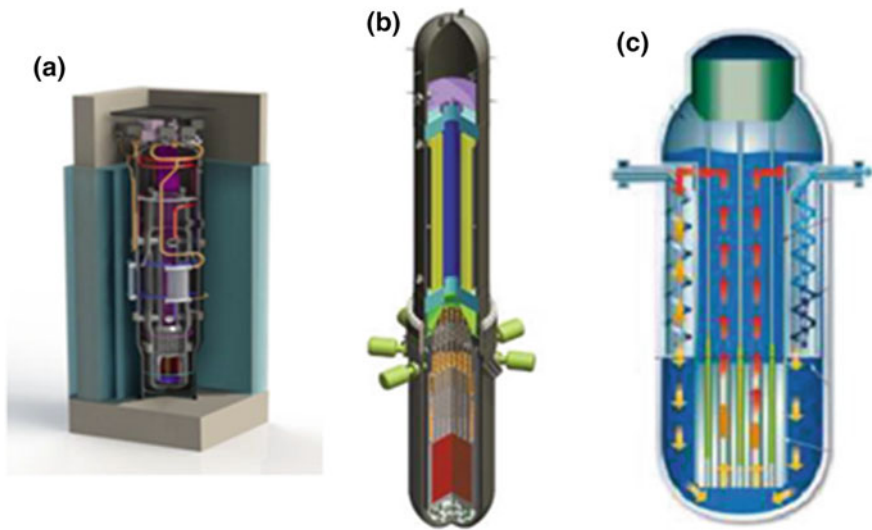


Fig. 2.12 LWRs: (a) NuScale, (b) IMR and (c) CAREM. *Source* (a) IAEA (2014), (b) and (c) ANSTO IAEA

modernised to increase plant reliability, to extend its service life and to improve the conditions of maintenance. The design of safety systems is based on safety regulations for marine nuclear power reactors and was updated to meet the requirements of the Russian Regulatory Authority—GAN RF—for nuclear power plants, according to IAEA (2004).

The KLT-40S is a PWR developed for a floating nuclear power plant to provide capacity of 35 MWe per module and 150 MWth. Floating power unit (FPU) has been developed to produce electricity and heat and to transfer them to customers making use of the coastal infrastructure. Safe positioning and retaining of FPU is provided by the hydraulic-engineering structures. The coastal infrastructure includes structures and special devices for reception and transmission of electric power and heat to users, and is operated co-jointly with an FPU (IAEA 2004).

The FPU is a smooth deck non-self-propelled ship. The FPU consists of a living module and a power module. The power module accommodates two KLT-40S nuclear power reactors, two steam turbine plants and electric power system. The FPU is manufactured at a specialised shipyard factory and transported to an operation site fully assembled (IAEA 2004).

The floating ship-type configuration SMR (KLT-40S) provides cogeneration capabilities for reliable power and heat supply to isolated consumers in remote areas without centralised power plants. Besides, this FPU can be used for seawater desalination complexes as well as for autonomous power supply for sea oil-production platforms.

KLT-40S nuclear power reactor are designed to run 3–4 years between refuelling with on-board refuelling capability and used fuel storage. At the end of a

12-year operating cycle the whole plant is taken to a central facility for overhaul and storage of used fuel. Two units will be mounted on a 20,000 tonnes barge to allow for (70% capacity factor). Although the reactor core is normally cooled by forced circulation (four-loop), the design relies on convection for emergency cooling. The fuel is uranium aluminium silicide with enrichment levels of up to 20%, giving to a four-year refuelling intervals. A variant of this is the KLT-20 (WNA 2016).

According to Afrikantov OKBM, FPU is constructed in the factory conditions that make it possible to reduce deadlines and cost of construction. Relatively small capital cost, short construction period (four years) and increased resistance to external impacts reduce the investment risk to minimum, and increase commercial attractiveness of power units.

The first FPU carrying the KLT-40S is the Akademik Lomonosov in the Chukotka region. The construction of this reactor start in 2007. The Akademik Lomonosov is expect to be completed by the end of 2016 and expected electricity production by 2017.

(2) RITM-200

The RITM-200 is being developed by OKBM Afrikantov as an integral nuclear power reactor for multipurpose nuclear icebreaker, floating and land-based nuclear power plants, with an electrical output of 50 MWe and a thermal power of 175 MWth (IAEA 2012). It incorporates the experience in design and operation of many of Russian marine propulsion reactors.

An integrated approach was adopted to determine the main parameters of the primary system, selection of equipment and layout, determining the optimal inventory and parameters of the safety systems for the RITM-200. Inherent safety characteristic of the RITM-200 is ensured based on the following principles: High thermal storage capacity, primary coolant natural circulation sufficient for reactor cool down, minimal length of the primary pipelines, leak stoppers in small nozzles, greater volume of primary coolant in the reactor vessel as compared with the modular arrangement increase the time margin until core drainage in loss of coolant accidents, and introduction of active and passive safety systems, according to Afrikantov OKBM.

The RITM-200 has four coolant loops and external main circulation pumps, use low-enriched fuel (<20%) and refuel every seven years a 65% capacity factor, over a 40-year total lifespan (WNA 2016).

The RITM-200 is designed to provide the shaft power on a typical nuclear icebreaker and can be used on a vessel of 150–300 tonnes displacement. The nuclear power reactor can also be considered for floating heat and power plants, power and desalination complexes, and offshore drilling rigs. The designers also claim that the overall size of the steam generating unit allows transport of the nuclear power reactor by rail. The nuclear power reactor plant in containment has a mass of 1100 ton (IAEA 2012).

The current status of RITM-200 concept is the two reactor plants for the first multipurpose icebreaker (complete delivery in 2016) are being manufactured. In 2020, two serial universal nuclear icebreakers will be commissioned, according to Afrikantov OKBM.

(3) **CNP-300**

This is based on the Qinshan 1 reactor in China as a two-loop PWR operating in Pakistan and with further units being built there. It is 1000 MWth, 325 MWe with a design life of 40 years. Fuel enrichment is 2.4–3.0%; fuel cycle 12 months. It is from the China National Nuclear Corp (CNNC) (WNA 2016).

Two vertically mounted external reactor coolant pumps circulate the primary coolant between the reactor pressure vessel and the two vertical U-tube steam generators. Because of CNP-300 uses a loop-type configuration, a large-break loss of coolant accident is possible and the multiple safety systems are incorporated to mitigate its consequences, including high-pressure injection systems (Carelli and Ingersoll 2014).

(4) **NuScale**

According to the IAEA (2014), NuScale reactor is made up of one to twelve independent reactor modules each producing a net electric power of greater than 45 MWe resulting in a plant output greater than 540 MWe for a twelve-module power plant. The reactor operates based on natural convection instead of using pumps to circulate water through reactor core and adopts fully passive safety features. The NuScale value proposition involves innovative design principles to achieve significant improvement in safety, reduces capital at risk, and flexibility/scalability in plant size and application. Each reactor module includes a high pressure containment vessel immersed underwater in a below-grade pool. The NuScale primary system and containment are prefabricated and transported by rail, truck or barge to the plant site, which shortens construction schedule to approximately 36 months. The integral modular design of NuScale allows new modules to be added to the plant or refuelled independently while the other modules continue to operate. NuScale reactor uses standard PWR fuel enriched to 4.95% in normal PWR fuel assemblies, with a 24-months refuelling cycle. Design life is 60 years (WNA 2016).

NuScale design is a modular reactor for electricity production and non-electrical process heat applications. The NuScale Integral System test facility is being used to evaluate design performance and improvements, and to conduct integral system tests for NRC certification.

(5) **mPower**

The mPower is designed by Generation mPower and its affiliated Babcock & Wilcox mPower, Inc. and Bechtel Power Corporation.⁶ According to Generation

⁶In 2011, members of B&W and Bechtel Power Corporation entered into a formal alliance called “Generation mPower to design, license and deploy mPower modular nuclear power plant”.

mPower, the BWXT mPower reactor design is a scalable, SMR, and iPWR in which the nuclear core and steam generators are contained within a single vessel. It utilises passive safety systems and is housed in an underground containment structure. With fewer components and systems, overall reliability is enhanced and affordability improved.

The BWXT mPower generates a nominal output of 195 MWe per module. In its standard plant design, each mPower plant is comprised of a ‘twin-pack’ set, or two mPower units, generating a nominal 390 MWe, but it is possible to use a multi-unit (1 to 10+) nuclear power plant depending on the energy demanded by customers (the facility structure is composed of reactor modules that are fully shop-manufactured on an as-needed basis to meet demand growth). The nuclear steam supply system is shippable by rail.

The generation mPower solution is expected to lower the overall capital cost of construction (three-year construction cycle) and optimise plant size to customers’ local power generation requirements. Also, the ability to bring increments of power online, while additional modules are under construction, will provide early returns on the investment.

The design adopts internal steam supply system components, once-through steam generators, pressurised, in-vessel control rod drive mechanisms, and horizontally mounted canned motor pumps for its primary cooling circuit and passive safety systems. The plant is designed to minimise emergency planning zone requirements.

The mPower reactor has a conventional reactor core and is standard fuel enriched to almost 5%, with burnable poisons, to give a four-year operating cycle between refuelling, which will involve replacing the entire core as a single cartridge. A 60-year service life is envisaged, as sufficient used fuel storage would be built on site for this (WNA 2016).

Two features of the Babcock design could cut down on operating costs. First, each nuclear power reactor will be housed in a containment structure big enough to store all of the waste generated by the plant during its 60-year life span, eliminating the need for a separate storage facility. That could be especially important, as nuclear power plant operators may have to store their own waste while they wait for the government to provide a permanent storage facility, which it is obligated to do by law. Second, the nuclear power reactors are also designed so that fuel has to be replaced only once every four years, instead of the usual two years. That will increase the amount of time that the plant can operate (Advanced Materials & Processes 2009).

The primary application for the mPower reactor is electricity production. The mPower design could be retrofitted to support other heat-requiring or cogeneration applications.

According to Generation mPower, the BWXT mPower reactor is expected to play a critical role in providing electric power while contributing to the overall reduction of greenhouse gas emission in the U.S. and around the world.

(6) International Reactor Innovative and Secure (IRIS)

The IRIS is a LWR with a modular, integral primary system configuration. The concept was originally pursued by an international group of organisations led by Westinghouse. Currently the IRIS related activities, especially those devoted to large scale integral testing, are being pursued by Italian organisations. IRIS is designed to satisfy enhanced safety, improved economics, proliferation resistance and waste minimisation. Its main characteristics are: Medium power of up to 335 MWe per module; simplified compact design where the primary vessel houses the steam generators, pressurisers and pumps; an effective safety approach of active and passive safety systems; optimised maintenance with intervals of at least four years; and fuel is similar to present LWRs, enrichment is 5% with burnable poison (IAEA 2014). IRIS is designed to accommodate a variety of core designs. Future core designs will include higher enriched UO_2 fuel and the capability to use MOX fuel.

The primary application of IRIS design is electricity production. However, this integrated PWR can support the heat producing process and seawater desalination options. Coupling with renewable energy parks and energy storage systems has been addressed as well.

The IRIS team has completed the design of the large scale test facility, currently under construction. R&D activities in the field of design economics, financial risk and SMR competitiveness are under way (IAEA 2014).

(7) Westinghouse SMR

The Westinghouse SMR is an integral PWR that improves on the concepts of simplicity and advanced passive safety that are demonstrated in the company's AP1000 nuclear power plant. The Westinghouse SMR delivers a thermal output of 800 MW_{th} and an electric output of greater than 225 MWe as a stand-alone unit, completely self-contained within a compact plant site. The entire plant is designed for modular construction with all components shippable by rail, truck, or barge.

The elimination of control rod drive mechanism penetrations through the RPV head prevents postulated rod ejection accidents as well as potential nozzle cracking. Eight seal-less canned motor pumps are mounted horizontally to the shell of the RPV and provide forced reactor coolant flow through the core. The steam generator is a straight tube configuration with the primary reactor coolant passing through the inside of the tubes and the secondary coolant on the outside. The integral pressuriser located above the steam generator within the RPV is used to control pressure in the primary system. Both the reactor vessel and the passive core cooling system are located within a compact, high pressure, steel containment vessel (CV). The CV operates at a vacuum, and is designed to be fully submerged in water to facilitate heat removal during accident events.

The Westinghouse SMR is an advanced passive nuclear power plant, where the safety systems are designed to mitigate accidents through the use of natural driving forces such as gravity flow and natural circulation flow. The plant is not reliant on AC power or other support systems to perform its safety functions. The seven day

minimum coping time following loss of offsite power is a fundamental advancement over the three day coping time of the current best in class plants. The below grade locations of the reactor vessel, containment vessel, and spent fuel pool provide protection against external threats and natural phenomena hazards. The nuclear power plant is designed to be standalone with no shared systems between adjacent units on the same site.

Three diverse decay heat removal methods are used by the Westinghouse SMR: 1) Gravity feed, 2) Passive decay heat removal heat exchanger, and 3) Bleed and feed enabled by a two-stage automatic depressurisation system.

The Westinghouse SMR uses a standard fuel product with enrichment at less than 5% in a 24-month refuelling cycle over a 60-year plant design life.

(8) Holtec SMR-160

The SMR-160 conceptual design has been developed by Holtec International (U.S.) as an advanced PWR-type, producing power of 525 MWth or 160 MWe adopting passive safety features.

SMR-160 achieves its supreme safety by eliminating vulnerabilities that have been the source of accidents in nuclear power plants, namely pumps and motors to run the plant's safety systems. Instead of motors or pumps, SMR-160 relies on Mother Nature's gravity to run all safety significant systems in the plant. Because gravity cannot fail, a SMR-160 plant is assured to remain safe under every operating and accident scenario. Replacing motors and pumps with gravity driven fluid flow systems not only hardens the plant against disasters, but leads to huge reductions in the plant's overnight, operating and maintenance costs (Holtec International 2015).

The SMR-160 uses two external horizontal steam generators and using fuel very similar to that in larger PWRs, including MOX. The 32 full-length fuel assemblies are in a fuel cartridge, which is loaded and unloaded as a single unit. The whole reactor system will be installed below the ground level, with used fuel storage. A 24-month construction period is envisaged for each unit. A modular construction plan for SMR-160 involves pre-assembling the largest shippable components prior to arrival at a site. Holtec claims a one-week refuelling outage every 42 months and an operational life of 80 years (WNA 2016).

The SMR-160 is designed to be an unconditionally safe reactor, which means it will not release radioactivity regardless of the severity of the natural or manmade disaster. Every conceivable catastrophic event—severe cyclones (hurricanes or typhoons), tsunamis, flood, fire and crashing aircraft—has been considered, with appropriate features incorporated in the design of SMR-160 to ensure that it will withstand these events without releasing radioactivity or pose any risk to public health and safety. Because of these, it can be sited next to population centers without any threat to the local environment or populace. Placing SMR-1600 close to cities and towns will reduce transmission losses and enable the plant's workers to live in the local community (Holtec International 2015).

A typical SMR-160 uses cooling water from a local natural source such as a lake, river or ocean to condense its exhaust steam. However, it can also be deployed in water-challenged regions by using air as the condensing medium (Holtec International 2015).

The primary application of SMR-160 is electricity production with optional cogeneration equipment (i.e., hydrogen generation, district heating, and seawater desalination). Target applications include distributed electricity production, repowering coal facilities, uprate existing nuclear facilities and providing electricity and low temperature process heat for commercial and military installations. Design optimisation includes air cooled condensation for no wet cooling.

The pre-application activities for the technology have started with the U.S. - NRC. The project baseline plan reflects realistic work scope, task durations and schedules to ensure Design Certification in time to support commercial operation of the first plant by 2025.

(9) VVER-300 (V-478)

According to ARIS IAEA (2011a), the design of the VVER-300 is based on the following concepts: Design is developed for the regions with smaller power grids; structure, materials, heat-engineering parameters of loop main equipment (steam generator, reactor coolant, pumps and main coolant pipelines) is optimally unified with similar equipment in the design of VVER-640; core is designed on the basics of fuel assemblies similar to VVER-1000 fuel assemblies with long-term experience of nuclear power plant operation; and isolation of primary-to-secondary leak without radioactive releases into atmosphere.

Enrichment of the fuel is 3.3%. The standard fuel cycle is not closed, length of the cycles is 12 months and the time of fuel residence in the core (fuel life) is eight years.

The nuclear power plant construction time, from the initial stage to commissioning for commercial operation, is expected by the designer to be four years (IAEA 2014). The VVER-300 is designed to generate a thermal power of 850 MWth or an electrical power of about 300 MWe. The design of the two loop reactor is based on the VVER-640 (V-407) design (WNA 2016). The VVER-300 design is to be deployed in the remote areas with power grids of limited capacity.

(10) VBER-300

The VBER-300 nuclear power reactor is a medium sized power source for ground-based nuclear power and cogeneration plants, as well as for floating nuclear power plants and having an electric power output of 325 MWe and thermal power output of 917 MWth. The thermal power increase is reflected in an increase in mass and overall dimensions, while the reactor's appearance and main design solutions are kept as close as possible to those of marine propulsion reactors (ARIS IAEA 2011b).

According to Afrikantov OKBM, the VBER-300 nuclear power reactor, designed on the basis of ship-based modular PWR, is an improved plant able to be a part of co-generation plants. It embodies use, to the maximum degree, mastered technologies of ship-based modular nuclear power reactors proved by long-term operation under heavy navigation conditions, and technology and operation experience of VVER-type reactors that increases its reliability and safety.

The distinctive feature of the VBER-300 nuclear power reactor design is maximum usage of proved engineering solutions based on ship reactor-building and VVER-type reactor experience.

VBER design solutions permit to create a wide power range of nuclear power plants on the basis of unified equipment. The developed designs of VBER nuclear power reactors are high consistency and cover the unit capacity range from 100 to 600 MWe. The possible nuclear power plant options are: Three-loop nuclear power reactor, four-loop nuclear power reactor and five-loop nuclear power reactor. A four-loop nuclear power reactor is accepted as a basic option.

The reactor is designed to have a 60-year life span and a 90% capacity factor. VBER uses standard VVER fuel assemblies enriched to 5%. The VBER-300 design concept allows a flexible fuel cycle for the reactor core with standard WWER fuel assembly. The refuelling interval can be pushed from two years out to five years (6–15 years fuel cycle) (WNA 2016).

The VBER-300 nuclear power reactors is intended to supply thermal and electrical power to remote areas where centralised power is unavailable, and to substitute capacities of available co-generation plants on fossil fuels. They are also proposed to be used as power sources for seawater desalination complexes. Nuclear power plants with the VBER-300 formed by two reactor units in the steam-condensing mode are capable of generating 600 MWe, equivalent to power demands of a city with 300,000 population.

The VBER-300 preliminary design was completed in 2002, and a technical and commercial proposal (a shorter version of technical and economic investigation) for construction of a land based or floating nuclear power plant with the VBER-300 was prepared.

(11) **VK-300**

According to NIKIET (2014), the VK-300 is a simplified integral single-circuit BWR with natural circulation of coolant. The reactor's key components are comparatively few: A vessel, a core, steam separators, and control. The vessel and the fuel elements have the same design as in VVER reactors. The steam separators were developed for the VVER steam generators as well. The VK-300 design relied a great deal on a long-term successful experience of operating VK-50, a BWR with natural coolant circulation based at NIIAR.

The core is cooled thanks to natural coolant circulation during normal operation and in any emergency. The VK-300 design uses an innovative reactor coolant circulation and multistep separation concept, which makes it possible to ensure the

required natural circulation rate and steam quality (humidity <0.1). Special emphasis is placed on ensuring the required safety level. The safety systems are passive, feature a simple design and have analogs.

The VK-300 is capable to produce 750 MWth or 250 MWe, it uses UO_2 fuel with an enrichment of 4% and with 18-month refuelling (WNA 2016). VK-300 reactor facility specially oriented to effective cogeneration of electricity and heat for district heating and for seawater desalination having excellent characteristics of safety and economics.

Research and development activities are currently under way for further validation and updating of the design approach adopted in the VK-300 design. In September, it was announced that six would be built and to start operating in 2017–2020.

(12) VKT-12

A smaller Russian BWR design is the 12 MWe transportable VKT-12, described as similar to the VK-50 prototype BWR at Dimitrovgrad, with one loop. The unit's core cooling system is passive. It has a ceramic-metal core with uranium enriched to 2.4–4.8%, and 10-year refuelling interval with the reactor's design life of 60 years (WNA 2016).

(13) ABV-6M

According to IAEA (2012), the ABV-6M installation is a nuclear steam generating plant with an integral pressurised LWR and natural circulation of the primary coolant. The ABV-6M design was developed using the operating experience of water cooled, water moderated nuclear power reactors and recent achievements in the field of nuclear power plant safety. The main objective of the project is to create small, multipurpose power sources based on proven marine nuclear reactor technologies, providing easy transport to the site, rapid assembly, and safe operation.

The ABV-6M reactor is designed to produce 45 MWth and 8.6 MWe in condensation mode, and 14 MWth and 6 MWE in co-generation mode. The core lifetime without reloading or shuffling of fuel is 10–12 years. The ABV-6M reactor has a service life about of 60 years.

The ABV-6M reactor installation is intended as a universal power source for floating nuclear power plants. The reactor is designed with the capability of driving a floating unit with a maximum length of 115 m, a beam of 26 m, a draft of 3.5 m and a displacement of 8000 tonnes. Depending on the needs of the region, the floating nuclear power plant can generate electric power or provide heat and power co-generation or can be used for other applications.

The stationary nuclear power plant (land based or underground) is fabricated as large, ready-made units, these units are transported to the site in a special truck or by water. The floating nuclear power plant is factory fabricated.

Currently, the development of ABV-6M is at the finishing stage. The ABV-6M installation is a nuclear steam generating plant producing 14 MWth or 6 MWe in co-generation mode. This integral PWR has a natural circulation of the primary coolant. The ABV-6M design was developed using the operating experience of water cooled, water moderated power reactors, and recent achievements in the field of nuclear power plant safety. The main objective of the project is to create a small, multipurpose power source providing easy transport to the site, rapid assembly, and safe operation for 10–12 years without refuelling at the berthing platform or on the coast. Plant maintenance and repair, refuelling, and nuclear waste removal are fulfilled at special enterprises suitable for that purpose (IAEA 2014). ABV-6M reactor has a service life about of 60 years.

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(14) Central Argentina of Modular Elements (Central Argentina de Elementos Modulares, CAREM-25)

CAREM-25 is a national SMR development project based on LWR technology coordinated by the Argentina National Atomic Energy Commission (CNEA) in collaboration with leading nuclear companies in Argentina. CAREM-25 reactor was developed using domestic technology, at least 70% of the components and related services were sourced from Argentinian companies.

According to Delmastro et al. (2011), CAREM-25 project consists on the development, design, and construction of a small nuclear power reactor with high economic competitiveness and high level of safety. It is a prototype with an electrical output of about 27 MWe, CAREM-25, will be constructed in order to validate the innovation of CAREM design (with 150–300 MWe) concept and the developed to commercial version.

The design basis is supported by the cumulative experience acquired in research reactors design, construction and operation, and PHWR nuclear power plants operation, maintenance and improvement, as well as the finalisation of the CAN-II and the development of advanced design solutions.

CAREM-25 is an indirect cycle nuclear power reactor with some distinctive and characteristics features that greatly simplify the reactor and also contribute to a higher level of safety. Some of the design characteristics of CAREM-25 are:

- Integrated primary cooling system;
- Primary cooling by natural circulation;

- Self-pressurised;
- Safety systems relying on passive features.

The most innovative feature of this design is that the entire primary coolant system is contained within the reactor pressure vessel. The integral reactor vessel contains the reactor core and support structures, steam generators, and the control rod system. The primary system is self-pressurised by the steam generated inside the vessel. The operating pressure is the steam pressure corresponding to the temperature of the coolant at the core exit. A steam chamber, located near the top of the reactor vessel, is used to regulate pressure against variations in the coolant temperature (U.S. Department of Energy 2001).

Using natural circulation instead of coolant pumps has a number of important benefits contributing to higher reliability and safety, better economic performance, and sabotage and proliferation resistance. An disadvantage is that the CAREM reactor is not highly modularised and requiring a substantial amount of on-site construction (U.S. Department of Energy 2001).

Fuel is standard 3.1 or 3.4% enriched PWR fuel in hexagonal fuel assemblies. With burnable poison, and is refuelled annually (WNA 2016).

CAREM-25 is designed as an energy source for electricity supply of regions with small energy demands. It can also support seawater desalination processes to supply water and energy to coastal sites.

The licensing process for the construction of CAREM-25 prototype was approved by the Argentina Regulatory Body (ARN) in 2010.

(15) System Integrated Modular Advanced Reactor (SMART)

According to the ARIS IAEA (2011c), SMART is a small-sized integral type PWR with a rated power of 330 MWth or 100 MWe. It is a nuclear power reactor with a sensible mixture of proven technologies and advanced design features.

SMART aims at achieving enhanced safety and improved economics; the enhancement of safety and reliability is realised by incorporating inherent safety improvements features and reliable passive safety systems. The improvement in the economics is achieved through a system simplification, component modularisation, reduction of construction time, and high plant availability. The preliminary analyses on the selected limiting accidents assure the reliability of the SMART reactor system.

By introducing a passive residual heat removal system, and an advanced mitigation system for loss of coolant accidents, significant safety enhancement is achieved. The low power density design, with about a 5% UO_2 fuelled core, will provide a thermal margin of more than 15% to accommodate any design basis transients with regard to the critical heat flux. This feature ensures core thermal reliability under normal operation and any design basis events. Design life is 60 years with a three-year refuelling cycle.

SMART as an integral-type nuclear power reactor contains major components within a single RPV. Eight modular-type once-through steam generators consist of

helically coiled tubes producing 30 °C superheated steam under normal operating conditions, and a small inventory of secondary side water sources as the steam line break accident. Four reactor coolant pumps with a canned motor, which has no pump seals, inherently prevents a loss of coolant associated with a pump seal failure. Four channel control rod position indicators contribute to the simplification of the core protection system and to an enhancement of the system reliability. The in-vessel pressurised is designed to control the system pressure at a nearly constant level over the entire design basis events (Koo Kim et al. 2014).

The integrated arrangement of reactor vessel assembly enables the large size pipe connections to be removed, which results in the elimination of large break loss of coolant accidents. Furthermore, an advanced man-machine interface system using digital techniques and equipment reduce the human error factors, and consequently improve the plant reliability.

SMART is applicable for electricity production suitable for small or isolated grids and heat district as well as processing heat for desalination purposes having the enough output to meet demands for a city of 100,000 population. SMART has been fully licensed in South Korea and a standard SMART design was approved by the Korean Nuclear Safety and Security Commission in July 2012.

(16) MRX

The Japan Atomic Energy Research Institute designed the MRX, a small (50–300 MWth) integral PWR reactor for marine propulsion or local energy supply (30 MWe). It has conventional 4.3% enriched PWR uranium oxide fuel and has a water-filled containment to enhance safety (WNA 2016).

The advantage of the MRX design comes from its assembly-line fabrication of the entire plant, and its transportability as a self-propelled ship. Refuelling every 3.5 years may be more frequent than desired, however, the capability of performing refuelling and maintenance activities improves diversion and proliferation resistance. The design lends itself to existing commercial LWR infrastructure for fuel fabrication and handling, spent fuel processing, and waste disposal (U.S. Department of Energy 2001).

The steam generator and pressurised are installed inside the pressure vessel, although there are other major components of the primary coolant system which are outside of the reactor vessel. A relatively large water inventory increases the thermal capacity of the primary systems and reduces radiation damage to the vessel.

(17) NP-300

Technicatome (Areva TA) in France has developed the NP-300 PWR design from submarine power plants and aimed it at export markets for power, heat, and desalination. It has passive safety systems and could be built for applications of 100–300 MWe or more with up to 500,000 m³/day desalination (WNA 2016).

According to UxC Company, the reactor's fuel is enriched to less than 5% U-235 and one third of the fuel is replaced every 18 months to two years. The

NP-300 is based on Techicatome's K15 naval nuclear power reactor of 150 MW and its land-based equivalent called "Réacteur d'essais à terre (RES)", a test version of which is under construction at Cadarache. Depending on local conditions, the NP-300 could be also modified to be used as a floating power unit.

(18) Nuclear Heating Reactor (NHR-200)

The Chinese NHR-200, developed by Tsinghua University's Institute of Nuclear Energy Technology (now the Institute of Nuclear and New Energy Technology), is a simple 200 MWth integral PWR design for district heating or desalination. It is based on the NHR-5. Used fuel is stored around the core in the pressure vessel (WNA 2016).

(19) ACP-100

ACP-100 is an innovative design developed by CNNC and producing power of 100 MWe. The ACP-100 is based on existing PWR technology adapting a passive safety system which uses natural convection to cool down the reactor in case of operational transients and postulated design basis accidents. The ACP-100 is an integrated PWR in which the major components of its primary coolant circuits are installed within the RPV. The ACP-100 plant design will allow the deployment of one to eight modules to attain larger plant output as demands arise (IAEA 2014).

Safety-grade batteries provide backup power for up to 72 h in case of a plant blackout and sufficient water is provided in the spent fuel pool to allow seven days' grace period before fuel is uncovered (Carelli and Ingersoll 2014).

The expected average fuel enrichment is about 2.4–3.0% with 24-month refuelling and 60-year design life.

The ACP-100 is a multipurpose nuclear power reactor designed for electricity production, heating, steam production or seawater desalination, and is suitable for remote areas that have limited energy options or industrial infrastructure. The operation of the first unit of this type of reactor is expected to begin in 2017.

(20) CAP-150

CAP-150 is an integral type SMR, which employs the most advanced PWR technology, developed by Shanghai Nuclear Engineering Research and Design Institute (SNERDI), a subsidiary of the State Nuclear Power Technology Corporation (SNPTC). It has more simplified systems and more safety than current operating PWRs, to generate an electric power of 150 MWe (IAEA 2016).

CAP-150 design is based on the state of art philosophy of advanced PWR reactor concepts. It aims to innovatively bring out a new Gen III⁺ LWR with higher safety, reasonable and competitive economy, and good engineering feasibility.

As a supplement to large reactors, the CAP-150 is mainly developed for providing a flexible way for remote electric supply and district heating, and also to replace the old thermal power plants near cities.

(21) CAP-200

The China Advanced Passive PWR 200MWe (CAP-200) is one of the serial research and development products of PWRs adopting passive engineered safety features initiated by SNERDI. The design of CAP-200 is based on the experience of the PWR technology R&D for more than 45 years, construction and safe operation for more than 20 years in China. It adopts safety enhancement measures based on lessons learnt from the Fukushima Daiichi nuclear accident (IAEA 2016).

Compared with large PWRs, CAP-200 has a number of advantages such as higher inherent safety, lower frequency of large radioactivity release, longer time without operator intervention, smaller environmental impact, lower site restrictions, shorter construction period and smaller financing scale as well as lower financial risk.

CAP-200 can be used as a supplement to large PWRs, this reactor is designed for multiple applications, such as nuclear cogeneration and replacing retired fossil power plants in urban areas.

(22) ACPR100

China General Nuclear Group (CGN) is developing the ACPR100 reactor with passive cooling for decay heat and 60-year design life. ACPR100 has standard type fuel assemblies and fuel enriched to <5% with burnable poison giving 30-month refuelling. The ACPR100 is an integral PWR, 450 MWth, 140 MWe. It is designed as a module in larger nuclear power plants and would be installed underground (WNA 2016).

(23) ACPR50S

The ACPR50S is a small modular offshore floating nuclear power reactor developed by the China General Nuclear Power Corporation (CGNPC)—aiming for high safety and adaptability, modularised design, and multi-purpose applications. It is intended as a potential optimal solution for combined supply of heat, electricity, and fresh water for marine resource development activities, energy supply and emergency support on islands and along the coastal area (IAEA 2016).

The ACPR50S adopts design simplification with less cost and lower investment risks in order to be competitive with conventional offshore energy sources. Modular design is adopted through standardised streamline manufacturing aiming for shorter construction period as well as less cost. Higher load factor to be attained by a long refuelling cycle.

According to the Lyncean Group (2016), the major components of the nuclear steam supply system (NSSS) are the reactor vessel, two steam generators and primary pumps, and one pressuriser. The primary system is housed within a containment structure that is protected against damage from a ship collision. Active and passive safety systems provide for core and containment cooling during an

accident. Severe (beyond design basis) accident mitigation measures include operating safety plugs to submerge the NSSS in seawater to ensure continued core cooling.

The floating nuclear power plant is designed for on-ship refuelling and pre-treatment of radioactive waste. When the floating nuclear power plant is deployed in a remote location, a visiting offshore engineering services vessel will provide logistics and maintenance services as needed.

An industrial demonstration plant of ACPR50S is being planned to be constructed in China, with a target of start-up commissioning in 2020.

(24) **Flexblue**

According to Lee et al. (2015), DCNS in France is developing a submerged type ocean nuclear power plant (ONPP) with an output capacity of 160 MWe, named Flexblue. Flexblue is a submerged, cylindrical, fully-modular, and transportable ONPP. Each module, with the length of 146 and 14 m diameter, is moored on a stable seafloor at a depth of up to 100 m and 15 km from the shoreline. Undersea cables would bring the electricity to customers.

Flexible is manufactured in factories, assembled in a shipyard using naval modular construction techniques, transported by the ship, and located at the operation site. Depending on the seismic hazard, the module is either anchored horizontally on the seabed, or suspended a few meters from the bottom of the ocean with positive buoyancy. The modules can be manufactured in different places and in parallel, allowing a shorter overall construction time.

Once the Flexblue is installed, it is monitored, protected, and operated from an onshore center. If a safety issue developed with the nuclear power reactor, it could be brought to the surface and taken to a regional support facility for repair and it could be refuelled the same way. At the end of its life, it could be repatriated to a shipyard for decommissioning, a process that would resemble the decommissioning of nuclear submarines.

During exploitation, maintenance teams gain regular access to the module with an underwater vehicle. This power unit can be used within a ‘nuclear farm’ that includes other modules. This allows the owner to increment the number of units that operated on the site, depending on the power needs. Water offers a natural protection against most of the possible external hazards and guarantees a permanently and indefinitely available heat sink (IAEA 2012).

The enrichment is kept below 5% and reactivity is controlled without soluble boron. This latter characteristic reduces the generation of radioactive wastes and simplifies the chemical control system. Flexblue is designed to supply electricity to coastal grids. Presentations of the concept have been made to the French safety authority. Technical discussions have been initiated with the French technical safety authority.

(25) UNITHERM

According to IAEA (2007), the UNITHERM system is developed based upon NIKIET's experience in designing marine nuclear installations. In the first design options, the core thermal power was defined as 15 MWth. Later, reactor power has been increased to 30 MWth as a result of the discussion with potential users, with an electrical output of 6.6 MWe. The land-based siting nuclear power plant or barges siting conditions are both viable for the UNITHERM reactor design.

No refuelling of the reactor core is envisaged during the plant service life. This would eliminate potentially hazardous activities related to core refuelling, simplify operating technologies, and could ensure enhanced proliferation resistance. The reactor core life can be equal to the plant lifetime and is estimated as 20–25 years at the capacity factor of 0.7.

The design assumes that most of the fabrication, assembly, and commissioning of the nuclear power plant modules can be done at the site. Nuclear power plant with UNITHERM may consists of a number of units depending on purpose and demand. To enhance security of supply, having at least two power units could be recommended. Each unit includes the reactor and turbine unit; the design of the latter varies depending on local demands.

The fuel is in the form of tiny blocks of UO_2 grains coated with zirconium and dispersed in a zirconium matrix.

The UNITHERM nuclear power plant can be used as a source of energy for electricity generation, district heating, seawater desalination and process steam production. In general, configuration and design of the UNITHERM is sufficiently flexible to be adjusted or modified for different target functions and user requirements, without compromising the underlying principles of the concept.

The UNITHERM nuclear power plant requires no major research and development for deployment. The detailed design stage would include qualification of the core, heat exchangers, and other components.

(26) SHELF

The N.A. Dollezhal Research and Development Institute of Power Engineering in the Russian Federation is currently developing a nuclear turbine-generator plant of 6 MWe as an underwater energy source. The plant comprises a two circuit nuclear power reactor facilities with a water cooled and water moderated reactor of 28 MWth, a turbine-generator plant with a capacity of 6 MW, and an automated remote control, monitoring and protection system by means of engineered features, including electricity output regulation, control and monitoring instrumentation (IAEA 2012).

It uses low-enriched fuel of UO_2 in aluminium alloy matrix with a fuel cycle of 56 months. It is intended as an energy supply for oil and gas developments in Arctic seas (WNA 2016). At the present, the SHELF reactor is in the early design phase and does not yet include a planned date of deployment.

(27) **Integrated Modular Water Reactor (IMR)**

The IMR is a medium sized nuclear power reactor with a reference output of 1000 MWth and producing electricity of 350 MWe. This integral primary system reactor set a potential deployment after 2020. The IMR design goals are to attain economic competitiveness with other electric power sources, including large-scale nuclear power reactors, and attain a high degree of reliance on intrinsic safety features, i.e., elimination of initiating events that might cause fuel failure, operator-free management of accidents, no need for external water and power during accidents, etc. To achieve these targets, IMR employs the hybrid heat transport system, which is a natural circulation system under bubbly flow condition for primary heat transportation, and no penetrations in the primary cooling system by adopting the in-vessel control rod drive mechanism. These design features allow the elimination of the emergency core cooling system (ARIS IAEA 2011d). It has a design life of 60 years, 4.8% fuel enrichment, and a fuel cycle of 26 months.

The IMR is primarily designed to generate electricity as a land-based nuclear power plant module. The capacity of the nuclear power plant can easily be increased and adjusted to the demand by constructing additional modules. Because of its modular characteristics, it is suitable for large-scale nuclear power plants consisting of several modules and also suitable for small distributed-power plants, especially when the capacity of grids are small. IMR also has the capability for district heating, seawater desalination, process steam production, and so forth. Validation testing, research and development for components and design methods, and basic design development are required before licensing (ARIS IAEA 2011d).

The project has involved Kyoto University, the Central Research Institute of the Electric Power Industry and the Japan Atomic Power Company and the target year to start licensing is 2020 at the earliest.

(28) **TRIGA**

The TRIGA power system is a PWR concept based on U.S. General Atomics' well-proven research reactor design. It is conceived as a 64 MWth, 16.4 MWe pool-type system operating at a relatively low temperature. The secondary coolant is perfluorocarbon. The fuel is uranium-zirconium hydride enriched to 20% and with a little burnable poison and requiring refuelling every 18 months. Used fuel is stored inside the reactor vessel (WNA 2016).

The primary coolant system of the TRIGA consists of the reactor core, primary circuit piping, pressurised, coolant pump, and a heat exchanger. The reactor vessel containing the core and the primary heat exchanger where heat is transferred from the primary circuit to the secondary circuit constitutes two large factory-fabricated modules to permit a transportable system. Not being an "integral PWR", there is a substantial amount of auxiliary equipment and piping systems needed to support the TRIGA reactor. The secondary system is housed in a room adjacent to the reactor room (U.S. Department of Energy 2001).

The spent fuel is stored inside the reactor vessel until it is cold enough to be removed and shipped from the reactor site.

(29) Fixed Bed Nuclear Reactor (FNBR)

FNBR is an early conceptual design from the Federal University of Rio Grande do Sul, in Brazil. It a PWR with pebble fuel, with a thermal power capacity of 134 MWth or 70 MWe, with flexible fuel cycle (WNA 2016).

The FNBR is suitable for both urban and remote locations and is designed to produce electricity alone or to operate as a co-generation plant producing simultaneously electricity, desalinated water, steam for industrial purposes, and heat for district heating.

According to UxC, this type of reactor uses spherical fuel elements that are at a fixed position within the core. FBNR has long fuel cycle and operates without on-site refuelling. FBNR has an integral primary system design and allows for an incremental capacity increase through modular approach.

(30) Small Modular Adaptable Reactor Technology (SMART)

The SMART from Dunedin Energy Systems in Canada is a 30 MWth, 6 MWe battery-type unit, installed below grade. It is replaced by a new one when it is returned to a processing facility for refuelling, at 83% capacity factor this would be every 20 years. Emergency cooling is by convection (WNA 2016).

(31) Double Modular Simplified and Medium Small Reactor (DMS)

According to the IAEA (2014), the concept design of this nuclear power reactor has been developed by Hitachi-GE Nuclear Energy under the sponsorship of the Japan Atomic Power Company from 2000 to 2004. The design is small-sized BWR, which generates thermal power of 840 MWth or 300 MWe. The heat from the core is removed by natural circulation so recirculation pumps and their driving power sources are eliminated. This feature allows for a simplified and compact RPV and containment. Due to the natural circulation feature, reactor internals and systems are also simplified. As a defence-in-depth measure, enhanced hybrid safety systems that combine passive and active methods are adopted. Like in other BWR, steam separation is performed inside the RPV. In DMS however, this mechanism is done through free surface separation in which the steam is separated from water by gravity force. Hence, no physical separator assembly is required. DMS has fuel enrichment of 4.3% and the refuelling period is 24 months.

A small-to-medium sized BWR is suitable for where budget for construction is limited and electricity transmission networks have not been fully constructed. DMS design also provides a nonelectric use of energy such as for district heating, mining, and desalination. At the moment, no domestic license or pre-license activities for SMR, since there is no SMR construction project in Japan. Some SMR design applied or will apply for a pre-licensing in U.S. or Canada.

(32) RUTA-70

According to Kozmenkov et al. (2012), RUTA-70 is a pool-type water-cooled water-moderated nuclear power reactor designed to work with forced convection of coolant at nominal power of 70 MWth, but has a natural convection capability at power below 30% of the nominal value. It is under development by Russia.

The reactor core and the core reflector are located at the lower part of the pool, while the most of the plant equipment, including the primary-to-secondary side heat exchangers reside at dry boxes outside the pool. The inner surfaces of the pool concrete walls are plated with stainless steel.

The basic design principles of the nuclear power reactor are simplicity of the design, high reliability and inherent safety features due to a low pressure and temperature of the primary coolant as well as integrated design of the reactor. Due to high safety features, the nuclear district heating plant using RUTA reactors could be constructed in maximum proximity to the consumers. The period of continuous operation of the reactor equipment without a need of maintenance is about one year.

Summing up can be stated that the RUTA concept and design is primarily developed to provide district heating in remotely isolated areas of Russia suffering from a lack of fossil fuels. The continuous increase of organic fuel costs in the country essentially broaden the area of competitive application of RUTA as a heating reactor. In addition, a promising way for the commercial application of low potential thermal energy generated by the RUTA reactor is the distillation process for seawater and brackish water desalination.

In this current status, RUTA is still in the conceptual design stages and a pilot RUTA-70 plant is planned to be built at the site of the Institute of Physics and Power Engineering (IPPE, Obninsk, Russia) (Kozmenkov et al. 2012).

(33) ELENA NTEP

According to IAEA TECDOC-1536 (2007), ELENA NTEP is a direct conversion water-cooled nuclear power reactor capable to supply electricity and heat over a 25-year life of the plant without refuelling. This is a very small nuclear power reactor with just 68 kWe in power generating capacity and another 3.3 MWth of heating capacity. The key aspect of this design is that it is meant to be an “unattended” nuclear power plant, requiring nearly no operating or maintenance personnel over the lifetime of the unit. The ELENA NTEP project was developed using the experiences in construction and operation of marine and space power plants and the operation experience of the GAMMA reactor. The concept has been developed by the Russian Research Centre Kurchatov Institute. The ELENA NTEP is a land-based nuclear power plant; however, in principle it is possible to develop versions for underground or underwater deployment. The nuclear power reactor and its main systems are assembled from factory-fabricated finished units, whose weight and dimensions enable any transport delivery for the complete plant, including helicopter and ship. Pellet type uranium dioxide fuel is used with the average U-235 enrichment of 15.2%.

(34) KARAT-45

According to IAEA (2016), KARAT-45 is a small BWR, with a rated power of 45 MWe designed by NIKIET as an independent co-generation plant for producing electric power, steam, and hot water. It is developed as the base facility for the economic and social development of the Arctic region and remote extreme northern areas of the Russian Federation.

The primary cooling mechanism for the reactor core is natural circulation for all operating modes and the reactor will be shop-fabricated in modular fashion to make it transportable. This SMR is designed for a long service life.

(35) KARAT-100

According to IAEA (2016), KARAT-100 is an integral type multi-purpose BWR with a power output of 360 MWth and a rated electrical output of 100 MWe. The design adopts engineering approaches proven at prototype and testing facilities. This SMR is designed for the production of electrical power, heat for district heating, and hot water in co-generation mode. The design adopts natural circulation for its primary cooling system core heat removal in all operational modes. The design configuration incorporates passive safety systems to enhance the safety and reliability.

KARAT-100 reactor is being built as the base reactor for the evolution of power generation in isolated or remote locations not connected to the unified grid. The key factor that makes this SMR a perfect choice for a nuclear co-generation plant is its economic competitiveness against other sources of thermal and electric power, achieved primarily due to a combined generation of heat (for district heating) and electricity.

2.4.3.2 Heavy Water Reactors

The heavy water, i.e. water in which the two hydrogen atoms are replaced by deuterium atoms, is an attractive coolant and moderator because it has a much lower tendency to absorb neutrons. This allows the reactor core to be fuelled by natural uranium rather than requiring the complex and expensive process of enriching the uranium. Because of the excellent neutron economy provided by the low neutron absorption of the heavy water, this coolant became a favourite option for several production reactors (Carelli and Ingerson 2014).

The following are some of the SMR heavy water reactors under development in some countries:

(1) Pressurised Heavy Water Reactor (PHWR-220)

According to the IAEA (2011), the Indian PHWR programme consists of the fabrication of 220 MWe, 540 MWe and 700 MWe units. India is operating sixteen 220 MWe units at five nuclear power plants. The PHWR uses heavy water as the

moderator and coolant and natural uranium dioxide as the fuel. The reactor consists of an integral assembly of two end shields and a calandria, with the latter being submerged in the water filled vault.

Unlike most nuclear power reactors that use batch refuelling, the PHWR is refuelled on a continuous basis using two refuelling machines—one on either end of the core. Refuelling is accomplished by inserting a fresh fuel bundle into one end of the pressure tube and collecting the spent fuel bundle that is forced out on the other end, which then transported to a spent fuel area (Carelli and Ingersoll 2014).

(2) Advanced Heavy Water Reactor (AHWR-300 LEU)

According to ARIS IAEA (2013), the Indian Advanced Heavy Water Reactor (AHWR) is a vertical pressure tube type, boiling light water cooled and heavy water moderated reactor. The nuclear power reactor incorporates a number of passive features and is associated with a closed fuel cycle having reduced environmental impact. At the same time, the nuclear power reactor possesses several features, which are likely to reduce its capital and operating costs.

The nuclear power reactor has been designed by Bhabha Atomic Research Centre (BARC) mainly to achieve large-scale use of thorium for the generation of commercial nuclear power plants. This nuclear power reactor will produce most of its power from thorium, with no external input of uranium-233 in the equilibrium cycle.

AHWR300-LEU is a land-based nuclear power plant. The nuclear power reactor is designed to produce 920 MW of thermal power, generating 300 MWe and 2400 m³/day of desalinated water. The nuclear power plant can be configured to deliver higher desalination capacities with some reduction in electricity generation. AHWR based nuclear power reactor can be operated in base load, as well as in load following mode. The target lifetime load factor and availability factors for AHWR are 80% and 90%, respectively.

According to BARC, AHWR300-LEU employs natural circulation for removal of heat from the reactor core under operating and shutdown conditions. The reactor physics design of AHWR300-LEU is optimised to achieve high burn-up with the LEU-thorium based fuel along with inherent safety characteristics like negative reactivity coefficients, among others.

The emphasis in design has been to incorporate inherent and passive safety features to the maximum extent, as a part of the defence in depth strategy. AHWR300-LEU design provides a grace period of seven days for absence of any operator or powered actions in the event of accident, according to ARIS IAEA (2013). One of the most important design objectives of AHWR300-LEU is to eliminate any significant radiological impact, and therefore, the need for evacuation planning in the public domain. This may facilitate siting of these nuclear power reactors close to population centres, according to BARC.

Site selection of several AHWR has been completed and the necessary clearance from competent authorities is underway (Fig. 2.13).

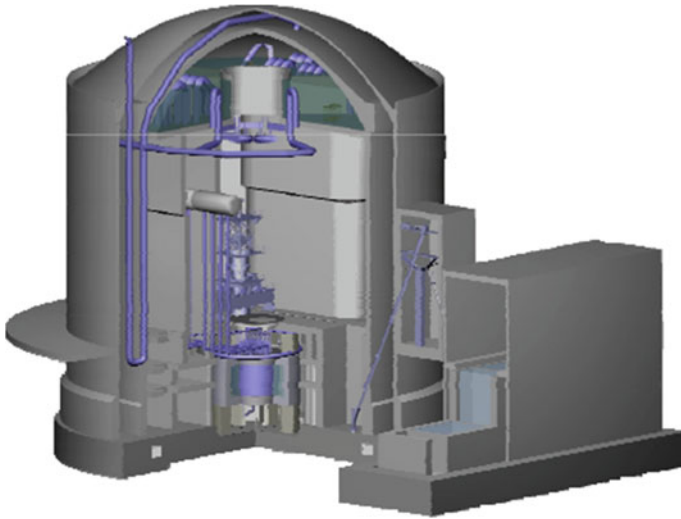


Fig. 2.13 Layout of AHWR-300-LEU. *Source* BARC

2.4.3.3 High-Temperature Gas-Cooled Reactors (HTRs)

After the operating experience of LWRs, gas cooled reactors have the next most operating experience, with a number of more advanced designs being looked at in the 1960s and 1970s.

HTRs are being developed which will be capable of delivering high temperature (700–950 °C and eventually up to about 1000 °C) helium either for industrial application via a heat exchanger, or to make steam conventionally in a secondary circuit via a steam generator, or directly to drive a Brayton cycle gas turbine for electricity with almost 50% thermal efficiency possible. Improved metallurgy and technology developed in the last decade makes HTRs 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 (tristructural-isotropic) particles less than a millimetre in diameter. Each has a kernel of uranium oxycarbide (or uranium dioxide), with the uranium enriched up to 20% U-235, though normally less. This is surrounded by layers of carbon and silicon carbide, giving a containment for fission products, which is stable to over 1600 °C.

There are two ways in which these particles are arranged: In blocks—hexagonal ‘prisms’ of graphite, or in billiard ball-sized pebbles of graphite encased in silicon carbide, each with about 15,000 fuel particles and 9 g uranium. There is a greater amount of used fuel than from the same capacity in a LWR. The moderator is graphite. HTRs can potentially use thorium-based fuels, such as highly-enriched or low-enriched uranium with Th, U-233 with Th, and Pu with Th.

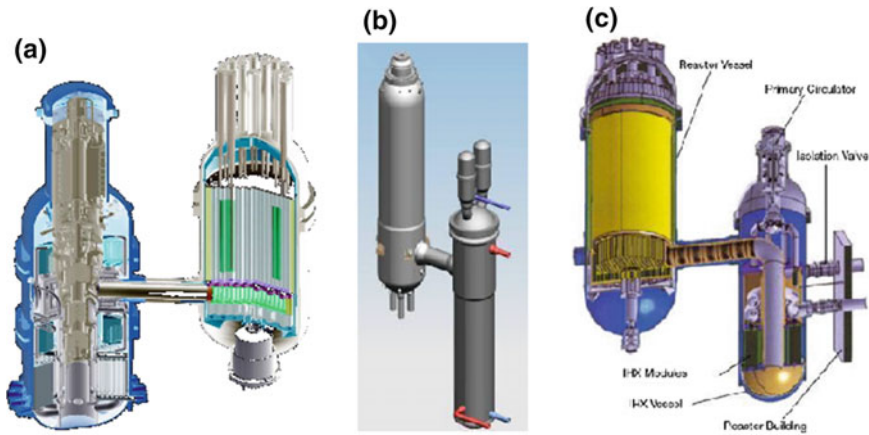


Fig. 2.14 High Temperature Gas Cooled Reactors: (a) GT-MHR, (b) PBMR-400 and (c) Antares. *Source* Oak Ridge National Laboratory (2011)

With negative temperature coefficient of reactivity (the fission reaction slows as temperature increases) and passive decay heat removal, this type of reactors is inherently safe. HTRs therefore do not require any containment building for safety. They are sufficiently small to allow factory fabrication, and will usually be installed below ground level (Fig. 2.14).

The following are some of the SMR High Temperature Gas Cooled Reactor under development in some countries:

(1) Gas Turbine High Temperature Reactor (GT-HTR 300)

According to Takizuka (2005), the GT-HTR300 is a JAERI proposed design concept of a gas-turbine high-temperature nuclear power reactor with 600 MWth power at 850–950 °C. Based on experience gained with HTTR. The objective of the project is to establish a feasible plant design and helium gas-turbine technology with ultimate goal for commercialisation in 2020s in Japan. High temperature capability enables a wider range of applications such as high temperature heat applications.

The GT-HTR300 design features modular nuclear power reactor, fully inherent and passive reactor safety, improved pin-in-block type fuel element, high burnup and long refuelling interval, conventional steel RPV, non-intercooled Brayton cycle, horizontal single-shaft turbo-machine, magnetic bearings to support turbo-machine rotor, and separate containment of turbo-machine and heat exchangers.

The direct-cycle helium gas turbine of the GT-HTR300 design generates about 300 MW electric power for 600 MW reactor thermal power at 45–50% thermal efficiency.

The design incorporates all ceramic fuel, low power density, but high thermal conductivity graphite core, and inert helium coolant to secure inherent reactor safety. The inherent safety permits siting proximity to customers, in particular to

industrial heat users so as to minimise the cost and loss of high temperature heat supply. Dry cooling becomes economically feasible due to the use of gas turbine. The waste heat from the gas turbine cycle is rejected from 200 °C, creating large temperature difference from ambient air and making dry cooling tower size per unit of power generation comparable to the wet cooling towers used in nuclear plants today. The economical dry cooling permits inland and remote reactor siting even without a large source of cooling water.

The nuclear power plant system consists of three basic subsystem modules including the reactor module, the gas turbine generator module, and the heat exchangers module. The functionally-oriented modules are contained in individual steel vessels situated in separate confinement silos. Partitioning the large nuclear power plant into properly sized subsystems and arranging them separately facilitates cost-effective modular construction and independently-accessed modular maintenance. The modules can be factory built in whole or in phase vessel subassemblies and transported to site for erection in parallel, followed by simple piping connection (Yan et al. 2002).

Typical applications of GT-HTR300 include electric power generation, thermochemical hydrogen production, desalination co-generation using waste heat only, and steelmaking. The maximum product output per reactor is 120 t/d hydrogen enough to fuel about one million cars, 280–300 MWe electricity generation with additional seawater desalination co-generation of 55,000 m³/d potable water for about a quarter of a-million population, and annual production of 0.65 million tons of steel. All these are produced without CO₂ emission.

(2) High-Temperature Gas-Cooled Experimental Reactor (HTR-10)

China's HTR-10, is a 10 MWt experimental nuclear power reactor. It has its fuel as a pebble bed of oxide fuel. Each pebble fuel element has 5 g of uranium enriched to 17% in TRISO-coated particles. The reactor operates at 700 °C (potentially 900 °C) and has broad research purposes. Eventually it will be coupled to a gas turbine, but meanwhile it has been driving a steam turbine (WNA 2016).

(3) High Temperature Reactor Pebble-Bed Module (HTR-PM)

Construction of a larger version of the HTR-10, China's HTR-PM, was approved in principle in November 2005, with preparation for first concrete in mid-2011, full construction started in December 2012 and is expected to conclude at the end of 2017. This was planned to be a single 200 MWe (450 MWth) unit, but it will now have twin reactors, each of 250 MWt driving a single 210 MWe steam turbine. The fuel is 85% enriched uranium (WNA 2016).

The HTR-PM is a commercial demonstration unit for electricity production. The twin reactor units driving a single turbine configuration was specifically selected to demonstrate its feasibility. HTR-PM commercial deployment based on batch construction is foreseeing, and units with more modules and bigger power size are under investigation. Standardised reactor modules with two, six or nine units with a single turbine (200, 600 or 1000 MW) are envisaged.

Development and research on process heat applications, hydrogen production and gas turbines are continuing for future application.

(4) **Pebble Bed Modular Reactor (PBMR-400)**

According to ARIS IAEA (2011), South Africa's Pebble Bed Modular Reactor (PBMR) is a High Temperature Gas Cooled Reactor based on the evolutionary design of the German AVB, THTR and HTR-Module design. It is being designed and marketed by PBMR (Pty) Ltd. The PBMR is designed in a modular fashion to allow for additional modules to be added in accordance with demand. In addition, the PBMR can be used as base-load power plant or load-following plant, and can be configured to the size required by the community it serves.

The term modular stems from the design intent that identical modules can be placed in a block of 4–8 units to make up a large nuclear power plant. The small size and modularity allow short (24 months) construction times and give flexibility to the utility to match generation capability more closely to demand than single large nuclear power plants allow. The small size is dictated by the design requirement that neither fuel nor major nuclear power plant systems may be damaged to the extent that there could exist a danger to the public at the exclusion zone of 400 m, following a total cessation of active system (Koster 2008).

The reactor core and power conversion unit components are contained within steel pressure vessels similar to those employed for LWR. Cross-vessels and external piping are provided to contain flow paths to and from the reactor core and between other components of the power conversion unit.

Full-scale production units had been planned to be 400 MWth (165 MWe), but more recent plans were for 200 MWth (80 MWe) (WNA 2016).

The PBMR-400 can produce electricity at high efficiency via a direct Brayton cycle employing a helium gas turbine.

(5) **Gas Turbine Modular Helium Reactor (GT-MHR)**

The GT-MHR is being developed by General Atomics in partnership with Russia's OKBM Afrikantov, supported by Fuji (Japan).

The use of modular helium nuclear power reactors makes the system flexible and allows the possibility to use various power unit schemes: With gas turbine cycle, steam-turbine cycle, and with the circuit supplying high-temperature heat to industrial applications (LaBar et al. 2003). The modular high temperature gas cooled unit possess salient safety features with passive decay heat removal providing a high level of safety even in case of total loss of primary coolant.

GT-MHR would be built as modules of up to 600 MWth, but typically 350 MWth, 150 MWe, nuclear power plant variation as a function of a module number. This provides good manoeuvring characteristics of the nuclear power reactor plant for regional power sources. Half core is replaced every 18 months and the fuel enrichment is about 15.5% (WNA 2016).

The GT-MHR reactor module is located in an underground containment building. The components of the primary cooling loop are housed within two

metallic pressure vessels that are connected by a cross-vessel. One of the vessels contains the modular high-temperature reactor nuclear heat source and the other the conversion power conversion unit. The power conversion unit design is based upon a recuperated direct gas-turbine cycle that is optimised for minimum cost and high efficiency. During normal operation, the heated, high-pressure helium leaving the core is routed via the hot duct inside the cross-vessel to the turbine where it is expanded to produce mechanical energy. The mechanical energy produced in the turbine is used to drive the generator, as well as two compressor gases located on the same shaft (Baxi et al. 2006).

The GT-MHR can produce electricity at relative high efficiency (approximately 48%). As it is capable of producing high coolant outlet temperatures, the modular helium nuclear power reactor system can also efficiently produce hydrogen by high temperature electrolysis or thermochemical water splitting.

Russia has selected the GT-MHR as an option for plutonium destruction, because of GT-MHR's efficiency in burning plutonium in once-through fashion, and because the technology is also readily convertible to a conventional low enriched uranium fuel cycle for commercial power applications.

Reactor plant preliminary design completed with the demonstration of key technologies are underway.

(6) Energy Multiplier Module (EM2)

According to the IAEA (2012), the EM2 is a 500 MWt, 240 MWe helium-cooled fast-neutron HTR operating at 850 °C. The EM2 design intended to burn used nuclear fuel and has a 30-year core without the need for refuelling or reshuffling. In a first generation plant, the fuel consists of about 22.2t of LEU starter and about 20.4t of used nuclear fuel. The used nuclear fuel is roughly 1% U, 1% Pu and mixed actinides and 3% fission products, the rest is U-238.

The nuclear power reactor design life is 60 years. The design has one loop and utilises two shutdown systems, control drums and separate shutdown rods. The design utilises the power conversion system for normal decay heat removal from the reactor vessel with the passive direct auxiliary cooling system. Specific design features include vented porous uranium carbide fuel, silicon carbide clad and a variable high speed turbine-generator set (TRP 2012).

EM2 would also be suitable for process heat applications. The main pressure vessel can be trucked or railed to the site, and installed below ground level (IAEA 2012).

(7) Antares

According to UxC Company, Antares is a concept being developed by Areva and belongs to the HTR/VHTR family of SMRs. This design could be used for hydrogen production at high temperatures, for industrial heat production, and for water desalination.

Antares is a modular design and uses TRISO fuel. In its standard HTR composition, each module has an output of 600 MWth and electricity production of

285 MWe with a reactor core outlet temperature of up to 850 °C. A very high temperature (VHTR) version of the Antares operates with a reactor outlet temperature of up to 1000 °C, and this design allows for hydrogen production among other co-generation features.

(8) Adams Engine

Adams Engine is a small HTR concept with a capacity of 10 MWe direct simple Brayton cycle plant with low-pressure nitrogen as the reactor coolant and working fluid, and graphite moderation. The initial units will provide a reactor core outlet temperature of 800 °C and a thermal efficiency near 25%. Power output is controlled by limiting coolant flow. A demonstration plant is proposed for completion after 2018. The Adams Engine is designed to be competitive with combustion gas turbines (WNA 2016).

(9) The Modular Transportable Small Power Nuclear Reactor (MTSPNR)

According to UxC Company, the MTSPNR design, which has recently become better known as GREM, has twin nuclear power reactors with a total thermal capacity of 4.8–5.2 MWth, producing 2 MWe el electricity. The unit is designed for co-generation of electricity and district heating and can supply 2×1.2 GJ/h in heat generation. The unit is meant to service remote regions, including potentially settlements of up to 2500 residents or strategically important industrial facilities.

MTSPNR is a high-temperature reactor with a single circuit. The unit has a closed cycle gas turbine and is air-cooled, eliminating the need for local sources of water. It will be factory fabricated and factory fuelled, ensuring lifetime core operation. It uses 20% enriched fuel and is designed to run for 25 years without refuelling (WNA 2016).

(10) X-Energy-100 (Xe-100)

The Xe-100 is a small-sized pebble bed high temperature gas-cooled nuclear power reactor with continuous thermal rating of 100 MWth. It features a continuous fuelling regime with low enriched fuel spheres of about 10% entering the top of the reactor going once through the core to achieve a final average burnup of 80,000 MWd/thm after a single passage. The relatively high burnup causes the bred fissile Pu to be utilised in situ by about 90%, thus rendering the spent fuel well depleted. Furthermore, the total used fuel inventory will be stored on-site in a designated interim storage facility for the life of the nuclear power plant. A major aim of the design is to improve the economics through system simplification, component modularisation, reduction of construction time and high plant availability brought about by continuous fuelling (IAEA 2014).

The Xe-100 is intended for electricity production suitable for small or isolated grids. It can also provide super-heated steam for co-generation, petro-chemical processes, etc. The Xe-100 also provides a scalable platform to increase total power generation from a single site by adding additional reactor modules. Site

configurations can consist of one to eight nuclear power reactor modules with a small operational staff.

(11) **Star Core HTR**

The Star Core HTR is a small (30 MWe) concept design of helium-cooled pebble bed reactor from StarCore Nuclear in Quebec, Canada, designed for remote locations (displacing diesel and propane) and with remote control system. The company said it is prepared to complete the design and detailed engineering, build, and begin operating at least two pilot plants in Canada by 2018 (WNA 2016).

(12) **MHR-T**

According to OECD-NEA (2009), the MHR-T reactor/hydrogen production complex makes use of the basic GT-MHR reactor design as the basis for a multi-module nuclear power plant for energy and hydrogen production. For the energy production (energy sector), MHR-T uses a four-module nuclear power plant including four reactor modules, as well as nuclear power plant systems and facilities supporting operation of these plants. For hydrogen production (chemical-technological sector) is achieved through the steam methane reforming process or high-temperature solid oxide electrochemical process from water is performed by coupled the plant with the modular helium nuclear power reactor(s). The basic operation mode of the MHR-T energy-technological complex is 100% power operation with parallel production of hydrogen and electric energy (combined mode).

The use of modular helium units makes the system flexible and allows the possibility to use various power unit schemes: with gas turbine cycle, steam-turbine cycle and with the circuit supplying high-temperature heat to industrial applications. The modular high temperature gas-cooled nuclear power reactor possess salient safety features with passive decay heat removal providing a high level of safety even in case of total loss of primary coolant.

(13) **High Temperature Modular Reactor (HTMR-100)**

The HTMR-100 pebble bed is a high temperature gas cooled reactor, graphite moderated and cooled by forced helium. The existing design of the module is to produce high quality steam which is coupled to a steam-turbine/generator system to produce 35 MW electric power. The steam can be used in a wide range of co-generation applications. The reactor is also suitable to provide direct high temperature energy for process heat. The design of the reactor is based on proven technology and therefore no new basic technology development is needed. The size of the reactor and the fuel cycle were chosen to simplify the design and operation of the module. The approach to small intrinsic safe modular units ensures continuous production, easy road transportability, skid mounted sub systems, wider range of manufactures, fast construction and a fairly easy licensing process (IAEA 2014).

The HTMR-100 exhibits the following excellent features:

- Fully ceramic fuel elements, which cannot melt even in extreme accidents, which may result in the total loss of active core cooling;
- Use of coated fuel particles (TRISO) effectively retaining the fission products within the fuel, and allowing for very high burn-up of the fuel;
- Use of helium as coolant, which is both chemically and radiologically inert and does not influence the neutron balance. It allows for very high coolant temperatures during normal operation;
- Use of fully ceramic (graphite) core internal structures, which enables operation at high temperatures;
- A reactor core with a low power density, providing a thermally robust design with a high heat capacity, renders the reactor thermally stable during all operational and control procedures;
- The reactor core can tolerate a loss of forced cooling event. Passive decay heat removal is possible and fuel temperatures stay below admissible values. Therefore, the fission products remain inside the fuel particles even in extreme accidents;
- Very strong negative temperature coefficients contribute to the excellent inherent safety characteristics of the reactor;
- Efficient retention of fission products in the coated particle fuel in normal operation allows for a clean helium circuit, resulting in low levels of contamination of the coolant gas, low release of radioactivity, and extremely low radiation dose values to the operation staff;
- Efficient retention of fission products in the coated particles under extreme accidents results in a nuclear power reactor without catastrophic release to the environment under these conditions.

The HTMR-100 is capable of supplying electric power to a large, medium and small grids, to standalone or isolated electric users as single module or multi-module plants and for medium temperature process heat applications. The HTMR-100 is a perfect fit for clients who want to progressively extend their generating capability. The unique safety characteristics of this type of reactor make it possible to introduce and construct these plants to non-nuclear countries. Developed countries that want to utilise their stock of plutonium for peaceful applications are also markets for HTMR-100 reactors. Conceptual design is completed and the design is in an early stage of the basic design phase.

(14) Steam Cycle High Temperature Gas-Cooled Reactor (SC-HTGR)

The SC-HTGR is a modular, graphite-moderated, helium-cooled, high temperature nuclear power reactor with a nominal thermal power of 625 MWth and a nominal electric power capability of 272 MWe. It produces high temperature steam suitable for numerous applications, including industrial process heat and high efficiency electricity generation. The safety profile of the SC-HTGR allows it to be collocated with industrial facilities that use high temperature steam. This can open a major new avenue for nuclear power use. The modular design allows plant size to be matched to a range of applications. The SC-HTGR concept builds on Areva's

past experience of HTGR projects, as well as on the development and design advances that have taken place in recent years for modular HTGRs. The overall configuration takes full advantage of the work performed on early modular HTGR concepts such as the MHTGR and the HTR-MODUL (IAEA 2016).

The HTGR steam cycle concept is extremely flexible. Since high pressure steam is one of the most versatile heat transport mediums, a single basic reactor module configuration designed to produce high temperature steam is capable of serving a wide variety of near-term markets. The steam cycle is also well suited to co-generation of electricity and process heat.

2.4.3.4 Fast Neutron Reactors

The major difference with fast reactors compared with LWRs, is that they are designed to use the full energy potential of uranium via a full reprocessing recycle route, i.e. closing the nuclear fuel cycle with management of plutonium and consumption of minor actinides. Typical coolants include liquid metal such as sodium, lead, or lead-bismuth, with high conductivity and boiling point, each of which carries its own challenges. They operate at or near atmospheric pressure and have passive safety features (most have convection circulating the primary coolant) (Fig. 2.15).

The following are some of the small fast nuclear power reactors under development in some countries:

(1) Power Reactor Innovative Small Module (PRISM)

General Electric with the U.S. national laboratories had been developing a modular liquid-metal-cooled inherently-safe reactor called “PRISM”, but today’s PRISM is a GE Hitachi (GEH) design for compact modular pool-type reactors with passive cooling for decay heat removal.

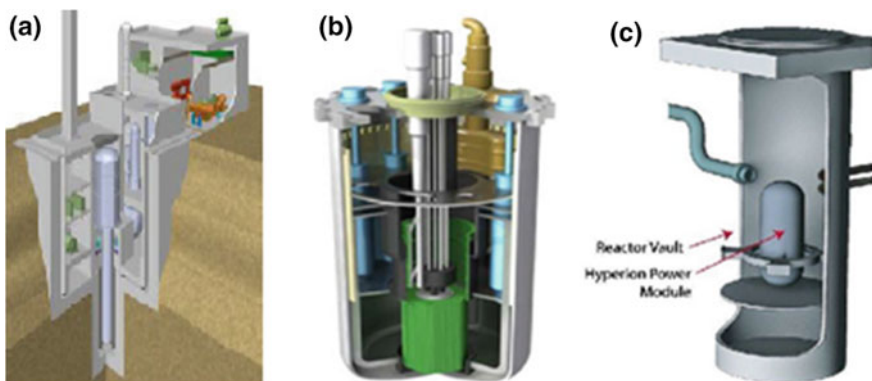


Fig. 2.15 Fast Neutron Nuclear Power Reactors: (a) 4s, (b) PRISM and (c) Gen4 (Hyperion). Source (a) IAEA (2014) (b) and (c) Oak Ridge National Laboratory (2011)

Each PRISM power block consists of two modules of 311 MWe (840 MWth) each, (or, earlier, three modules of 155 MWe, 471 MWth), each with one steam generator, that collectively drive one turbine generator. The pool-type modules below ground level contain the complete primary system with sodium coolant at about 500 °C. An intermediate sodium loop takes heat to steam generators. The metal Pu and DU fuel is obtained from used light water reactor LWR fuel. All transuranic elements are removed together in the electrometallurgical reprocessing so that fresh fuel has minor actinides with the plutonium and uranium. PRISM have two versions: For the LWR fuel recycle version, fuel stays in the reactor four years, with one-quarter removed annually and for breeder version fuel stays in the reactor about six years, with one-third removed every two years (WNA 2016).

The nuclear power plant design life is 60 years. Specific design features include the use of electromagnetic pumps. Transportability is enhanced by the modular construction sized for trucks and rail. Special benefits of the design are flexibility allowing use for either waste management or resource utilisation missions and the co-location of a small recycling centre (TRP 2012). In 2011, GE Hitachi announced that it was shifting its marketing strategy to pitch the reactor directly to utilities as a way to recycle excess plutonium while producing electricity for the grid.

(2) CEFR

The CEFR is a sodium cooled, 65 MWth experimental fast reactor with PuO_2 – UO_2 fuel, but with UO_2 as the first loading. It has been operating since 2010 and it is an important part of China's reactor development. The main objective of the CEFR is to accumulate experience in fast reactor design, fabrication of components, construction, pre-operational testing, and operation and maintenance (WNA 2016).

(3) Integral Fast Reactor (ARC-100)

Advanced Reactor Concepts LLC (ARC) is commercialising a 100 MWe sodium-cooled fast reactor based on the 62.5 MWth Experimental Breeder Reactor II (EBR-II).⁷

The ARC-100 system comprises a uranium alloy core submerged in sodium. The liquid sodium is passed through the core where it is heated to 510 °C, then passed through an integral heat exchanger (within the pool) where it heats sodium in an intermediate loop, which in turn heats working fluid for electricity generation. It would have a refuelling interval of 20 years. A 50 MWe version of the ARC is also under development (WNA 2016).

(4) Rapid-L

A small-scale nuclear power reactor design developed by Japan's Central Institute of Electric Power Industry is the 5 MWth, 200 kWe Rapid-L, using lithium-6 (a neutron poison) as control medium. The reactivity control system is passive, using lithium expansion modules (LEMs) which give burn-up

⁷The EBR-II was a significant fast reactor prototype at Idaho National Laboratory.

compensation, partial load operation as well as negative reactivity feedback. During normal operation, lithium-6 in the LEM is suspended on an inert gas above the core region. As the reactor temperature rises, the lithium-6 expands, moving the gas/liquid interface down into the core and hence adding negative reactivity. Cooling is by molten sodium. The refuelling would be every 10 years in an inert gas environment. Operation would require no skill, due to the inherent safety design features. The whole plant would be about 6.5 m high and 2 m diameter (WNA 2016). The larger RAPID reactor delivers 1 MWe and is U–Pu–Zr fuelled and sodium-cooled.

(5) **Super-Safe, Small and Simple (4S)**

The Super-safe, small and simple (4S) ‘nuclear battery’ system is being developed in Japan. It uses sodium as coolant (with electromagnetic pumps) and has passive safety features, notably negative temperature coefficient of reactivity. The whole unit would be factory-built, transported to site, installed below ground level, and would drive a steam cycle via a secondary sodium loop. It is capable of three decades of continuous operation without refuelling. Metallic fuel is uranium-zirconium enriched to less than 20% or U–Pu–Zr alloy with 24% Pu for the 30 MWth (10 MWe) version or 11.5% Pu for the 135 MWth (50 MWe) version. Both versions of 4S are designed to automatically maintain an outlet coolant temperature of 510–550 °C. After 30 years the fuel would be allowed to cool for a year, then it would be removed and shipped for storage or disposal (WNA 2016).

The design has one intermediate loop and one secondary loop and utilises two independent diverse shutdown systems: the drop of the annular reflector, and the insertion of a central shut-down rod. The design utilises passive decay heat removal from the reactor vessel. Transportability is via truck or rail (TRP 2012).

Toshiba planned a worldwide marketing programme to sell the units for power generation at remote mines, for extraction of tar sands, desalination plants and for making hydrogen. Eventually it expected sales for hydrogen production to outnumber those for power supply (WNA 2016).

(6) **BREST-300**

A significant Russian design from NIKIET is the BREST fast neutron reactor, of 700 MWth, 300 MWe, or more with lead as the primary coolant, at 540 °C, supplying supercritical steam generators. It is inherently safe and uses a U+Pu nitride fuel. Fuel cycle is 10 months. No weapons-grade plutonium can be produced (since there is no uranium blanket), and used fuel can be recycled indefinitely, with on-site facilities (WNA 2016).

BREST-300 is a reactor facility of pool-type design, which incorporates within the pool the reactor core with reflectors and control rods; the lead coolant circulation with steam generators and pumps; equipment for fuel reloading and management; and safety and auxiliary systems. The reactor equipment is arranged in a steel-lined, thermally insulated concrete vault. The BREST decay heat removal systems are characterised by passive and time-unlimited residual heat removal

directly from the lead circuit by natural circulation of air through air-cooled heat exchangers, with the heated air vented to the atmosphere (Alemberti et al. 2014).

A pilot unit was planned to be built at Beloyarsk, and 1200 MWe units are planned. It is at preliminary design stage.

(7) SVBR-100

A smaller and newer Russian design is the lead-bismuth fast reactor SVBR-100 of 280 MWth, 100 MWe, being developed. It is an integral design, with 12 steam generators and two main circulation pumps sitting in the same Pb–Bi pool at 340–490 °C as the reactor core. It is designed to be able to use a wide variety of fuels, though the pilot unit will initially use uranium oxide enriched to 16.3%. The refuelling interval is seven or eight years and 60-year operating life is envisaged. The SVBR-100 unit would be factory-made and transported by railway, road or waterway (WNA 2016).

According to UxC Company, the SVBR-100 nuclear power reactor is designed to be used in the remote regions of Russia and can have various power levels and purposes. The nuclear power reactor can be used for co-generation of electricity and process heat (and potentially serve as a power source for desalination) and will be placed in the immediate vicinity of populated areas. The nuclear power reactor can be also used for coastal and offshore nuclear power plant. The designers of the reactor also envisioned for it to be used as part of industrial facilities. One other interesting possible implementation of SVBR reactors is on the sites of the PWR units undergoing decommissioning using the existing infrastructure. Several modules can be used together if more power is required.

The plan is to complete the design development and put online a 100 MWe pilot facility by 2019.

(8) Gen4 (Hyperion) Power Module

The Gen4 Power Module is a 70 MWth/25 MWe lead-bismuth cooled reactor concept using 19.75% enriched uranium nitride fuel, from Gen4 Energy. The reactor was originally conceived as a potassium-cooled self-regulating ‘nuclear battery’ fuelled by uranium hydride. However, in 2009, Hyperion Power changed the design to uranium nitride fuel and lead-bismuth cooling to expedite design certification. This now classes it as a fast neutron reactor, without moderation. The company claims that the ceramic nitride fuel has superior thermal and neutronic properties compared with uranium oxide. Enrichment is 19.75% and operating temperature about 500 °C. The unit would be installed below ground level (WNA 2016).

The nuclear power reactor design life is 30 years. The design has one primary loop and one secondary loop and utilises two independent shutdown systems. The design utilises passive natural circulation for decay heat removal from the reactor vessel with water as the ultimate heat sink. Specific design features include containing the reactor in a sealed cartridge to avoid onsite refuelling, a primary shutdown system with inner and outer B₄C control rods and a secondary shutdown system having a central cavity into which a single B₄C control may be inserted. The

nuclear power reactor is transported via truck, ship or rail. Special benefits of the design include passive decay heat removal from the reactor vessel with a water jacket and the ability to operate in remote locations (TRP 2012).

This type of nuclear power reactor is designed to operate for electricity or process heat (or co-generation) continuously for up to 10 years without refuelling. Another unit could then take its place in the overall nuclear power plant.

In March 2012, the U.S. DoE signed an agreement with Hyperion regarding constructing a demonstration unit at its Savannah River site in South Carolina (WNA 2016).

(9) Encapsulated Nuclear Heat-Source (ENHS)

The ENHS is a liquid metal-cooled reactor concept of 50 MWe. The reactor core is at the bottom of a metal-filled module sitting in a large pool of secondary molten metal coolant, which also accommodates the eight separate and unconnected steam generators. The whole reactor sits in a 17-metre-deep silo. The fuel is a uranium-zirconium alloy with 13% enrichment (or U–Pu–Zr with 11% Pu) (WNA 2016).

The ENHS has a very long reactor core life (15–20 years), and it uses natural circulation to cool the reactor and to produce steam to drive the turbine. The ENHS concept relies on autonomous control that is, after the reactor is brought to full power, variation in power output follow the electricity generating needs automatically (load following) by using temperature feedback from the varying steam pressure and feedwater flow. The ENHS concept is based on the idea of encapsulating the reactor core inside its own vessel as a module, with no external piping connections (U.S. Department of Energy 2001).

The ENHS module is manufactured and fuelled in the factory, and shipped to the site as a sealed unit with solidified Pb (or Pb–Bi) filling the vessel up to the upper level of the fuel rods. With no mechanical connections between the reactor module and the secondary system, the module is easy to install and replace, similar to using a battery. After installation, hot coolant is pumped into the vessel to melt the solid lower part. At the end of its life, the ENHS module could be removed from the reactor pool and stored on site until the decay heat drops to a level that lets the coolant solidify. The module with the solidified coolant would then serve as a shipping cask. Its compact, sealed design combined with very infrequent refuelling provides high proliferation resistance (U.S. Department of Energy 2001).

The ENHS is designed for developing countries and is highly proliferation-resistant, but is not yet close to commercialisation.

(10) Secure Transportable Autonomous Reactor (START) STAR-LM, STAR-H2, Small STAR (SSTART)

START reactor is being designed by Argonne National Laboratory and its portfolio consists of three designs: 1) Small STAR; 2) STAR-LM, where LM stands for liquid metal; and 3) STAR-H2 design, where H2 stands for hydrogen

production. Main components of STAR-H2 vessel are presented in the work of Wade and Peddicord (2002).

According to WNA (WNA 2016), the STAR-LM is a factory-fabricated fast neutron modular reactor cooled by lead-bismuth eutectic, with passive safety features. Its 300–400 MWth size means it can be shipped by rail. It uses uranium-transuranic nitride fuel in a cartridge which is replaced every 15 years. Decay heat removal is by external air circulation. The STAR-LM was conceived for power generation with a capacity of about 175 MWe.

The STAR-H2 is an adaptation of the same nuclear power reactor for hydrogen production, with reactor heat at up to 800 °C being conveyed by a helium circuit to drive a separate thermochemical hydrogen production plant, while lower grade heat is harnessed for desalination (multi-stage flash process). Its development is further off.

SSTAR has lead or Pb–Bi cooling, 564 °C core outlet temperature and has integral steam generator inside the sealed unit, which would be installed below ground level. Conceived in sizes 10–100 MWe, main development was focused on a 45 MWth/20 MWe version as part of the U.S. Generation IV effort.

Some notable features of SSTAR include reliance on natural circulation for both operation and shutdown heat removal; a very long core life without refuelling (after a 20 or 30-year life without refuelling, the whole reactor unit is returned for recycling the fuel); and an innovative supercritical CO₂ (S-CO₂) Brayton cycle power conversion system (Alemberti et al. 2014).

The main mission of the SSTAR is to provide incremental energy generation to match the needs of developing nations and remote communities without grid connections. A prototype was envisaged for 2015, but development has apparently ceased.

(11) LBE Cooled Long-Life Safe Simple Small Portable Proliferation-Resistant Reactor (LSPR)

The Lead Bismuth Eutectic Cooled Fast Reactor of 150 MWth/53 MWe, the LSPR, is under development in Japan. The concept is intended for developing countries (WNA 2016).

According to UxC, the LSPR could be used for electricity generation, seawater desalination, hydrogen production, process steam production, among others. The LSPR is a factory fabricated design and it is designed to operate without on-site refuelling during the whole reactor life (30 years), then it is returned. The type of fuel used in this reactor is nitride with a fuel enrichment of 10–12%.

(12) Swedish Advanced Lead Reactor (SEALER)

SEALER is a lead-cooled reactor designed with the smallest possible core that can achieve criticality in a fast spectrum using 20% enriched uranium oxide (UOX) fuel. The nuclear power reactor is a 8 MWth, with a peak electric power of 3 MWe, leading to a reactor core life of 30 full power years (at 90% availability). The reactor vessel is designed to be small enough to permit transportation by aircraft (WNA 2016).

2.4.3.5 Molten Salt Reactors (MSR)

This type of reactor use molten fluoride salts as primary coolant, at low pressure. Lithium-beryllium fluoride and lithium fluoride salts remain liquid without pressurisation up to 1400 °C, in marked contrast to a PWR which operates at about 315 °C under 150 atmospheres pressure. In most designs (not the AHTR) the fuel is dissolved in the primary coolant.

In a normal MSR, the fuel is a molten mixture of lithium and beryllium fluoride (FLiBe) salts with dissolved enriched uranium—U-235 or U-233 fluorides (UF₄). The reactor core consists of unclad graphite moderator arranged to allow the flow of salt at some 700 °C and at low pressure. Heat is transferred to a secondary salt circuit and thence to steam.

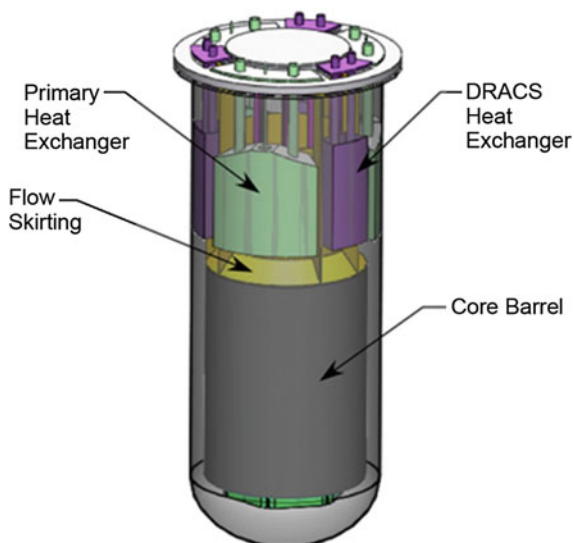
The fission products dissolve in the fuel salt and may be removed continuously in an on-line reprocessing loop and replaced with fissile uranium or, potentially, Th-232 or U-238. Actinides remain in the reactor until they fission or are converted to higher actinides which do so.

The liquid fuel has a negative temperature coefficient of reactivity and a strong negative void coefficient of reactivity, giving passive safety.

Potential roles for MSRs, in addition to electricity production include minor actinide and plutonium consumption and for those designs that can use higher temperature salts, then process heat applications, including hydrogen production could be feasible (Fig. 2.16).

The following are some of the Molten Salt Reactors under development in some countries:

Fig. 2.16 Molten Salt Reactor (SmATHR). *Source* Oak Ridge National Laboratory (2010)



(1) Liquid Fluoride Thorium Reactor (Flibe LFTR)

Flibe LFTR or LFTR is under development in the USA and is a 40 MW two-fluid graphite-moderated thermal reactor with a reactor outlet temperature of 450 °C. It uses lithium fluoride/beryllium fluoride (FLiBe) salt as its primary coolant in both circuits. Fuel is U-233 bred from thorium in FLiBe blanket salt. Fuel salt circulates through graphite logs. Secondary loop coolant salt is sodium-beryllium fluoride (BeF₂-NaF) (WNA 2016).

The nuclear power reactor design life is five to ten years. The design utilises passive decay heat removal from the reactor vessel with water in the underground silo as the ultimate heat sink. Specific design features include underground location of the reactor and primary heat exchanger is made of liquid-silicon-impregnated carbon-carbon composites. Transportability is via barge or truck. Many quantitative aspects of the design have not yet been determined (TRP 2012). A 2 MWth pilot nuclear power plant is envisaged to be built in the future (WNA 2016).

(2) Thorium Molten Salt Reactor (TMSR)

China is building a 5 MWe TMSR, essentially an LFTR. China claims to have the world's largest national effort on these and hopes to obtain full intellectual property rights on the technology. The target date for TMSR deployment is 2032 (WNA 2016).

(3) Fuji MSR

The Fuji MSR is a 100–200 MWe graphite-moderated design developed internationally by a Japanese, Russian, and U.S. consortium.

According to UxC Company, the basic technology of the nuclear power reactor includes the use of molten salt coolant in a graphite moderated core. The nuclear power reactor operates at a near-breeding level with a mix of Th and U-233 fuel, although it can also use U-Pu MOX fuel.

Several variants have been designed, including a 10 MWe mini Fuji. Thorium Tech Solutions Inc (TTS) plans to commercialise the Fuji concept and is working on it with the Halden test reactor in Norway (WNA 2016).

(4) Advanced High-Temperature Reactor (AHTR/FHR)

Research on molten salt coolant has been revived with the AHTR. This uses a coated-particle graphite-matrix fuel and with molten fluoride salt as primary coolant. It is also known as the Fluoride High Temperature Reactor (FHR). While similar to the gas-cooled HTR it operates at low pressure (less than 1 atmosphere) and higher temperature, and gives better heat transfer than helium. The FLiBe salt is used solely as primary coolant (WNA 2016).

The AHTR pursues three design goals: High reactor-coolant exit temperatures (700–100 °C) to enable the efficient production of hydrogen by thermochemical cycles and the efficient production of electricity; passive safety systems for public acceptance and reduce costs; and competitive economics. To achieve competitive economics, the AHTR is a large nuclear power reactor with passive safety systems

and high temperatures. The high temperatures minimise the size of the safety systems and power conversion equipment per kilowatt (electric) output (Forsberg 2004).

A 5 MW thorium-fuelled prototype is under construction at Shanghai Institute of Nuclear Applied Physics in China with a target for operation by 2020. A 100 MWth demonstration pebble-bed power plant with open fuel cycle is planned by about 2025.

(5) **IMSR**

This simplified IMSR integrates the primary reactor components, including primary heat exchangers to secondary clean salt circuit, in a sealed and replaceable core vessel that has a projected life of seven years. The IMSR will operate at 600–700 °C, which can support many industrial process heat applications in Canada. The fuel-salt is a eutectic of low-enriched uranium fuel (UF₄) and a fluoride carrier salt at atmospheric pressure. Emergency cooling and residual heat removal are passive. Each plant would have space for two nuclear power reactors, allowing seven-year changeover, with the used unit removed for off-site reprocessing when it has cooled and fission products have decayed (WNA 2016).

According to IAEA-ARIS (2016), the reactor core of the ISMR is manufactured in a controlled factory environment and then brought to the nuclear power plant site where, following final assembly, it is lowered into a surrounding, annular buffer salt tank, which itself sits in a below grade reactor silo. There, the reactor core is connected to secondary piping, which contains a non-radioactive coolant salt.

The fuel is separately brought to the nuclear power plant site as a solid, where it is melted and added to the ISMR core. This allows the ISMR operates with online fuelling. Additionally, and unlike solid-fuel nuclear power reactors, there is no need to remove a proportion of old fuel during refuelling. All of the fuel stays inside the closed ISMR core during the entire power operations period of the reactor core. Unlike other nuclear power reactor systems, the ISMR core needs never be opened at the nuclear power plant site, either during start-up fuelling or during refuelling.

The basic design approach to safety in the ISMR is to achieve an inherent, walk-away safe nuclear power plant. No operator action, electricity, or externally-powered mechanical components are needed to assure the most basic safety functions.

The IMSR is scalable and three sizes are under development: 80 MWth, 300 MWth and 600 MWth, ranging 30 MWe–300 MWe, but a 2016 report from the company gives 400 MWth and 192 MWe (WNA 2016). It is expected that the first commercial nuclear power reactor is ready by the early 2020s.

(6) **Transatomic TAP**

Transatomic Power Corp is a new U.S. company partly funded by Founders Fund and aiming to develop a single-fluid MSR using very low-enriched uranium fuel (1.8%) or the entire actinide component of used LWR fuel. The TAP nuclear power reactor has an efficient zirconium hydride moderator and a LiF-based fuel

salt bearing the UF_4 and actinides, hence a very compact reactor core (WNA 2016). After a 20 MWth demonstration reactor, the envisaged first commercial plant will be 1250 MWth/550 MWe.

(7) ThorCon

This is a single-fluid thorium converter reactor in the thermal spectrum, graphite moderated with a capacity of 250 MWe. It uses a combination of U-233 from thorium and U-235 enriched from mined uranium. Fuel salt is sodium-beryllium fluoride ($\text{BeF}_2\text{-NaF}$) with dissolved uranium and thorium tetrafluorides (Li-7 fluoride is avoided for cost reasons). There is no on-line processing—this takes place in a centralised plant at the end of the reactor core life—with off-gassing of some fission products meanwhile. All components are designed to be easily and frequently replaced. It is expected an operating prototype by 2020 (WNA 2016).

(8) Molten Stable Salt Reactor (MSSR)

MSSR is a conceptual UK nuclear power reactor design with no pumps and relies on convection from vertical fuel tubes in the reactor core at the center of a tank holding the primary coolant, while a secondary salt coolant conveys heat to the steam generators. Reactor core temperature is 500–600 °C, at atmospheric pressure. Decay heat is removed by natural air convection. A 150 MWth pilot plant is envisaged in the coming years (WNA 2016).

(9) Seaborg Waste Burner—SWAB

Seaborg Technologies in Denmark has a thermal-epithermal single fluid reactor design for 50 MWth pilot unit with a view to 250 MWth commercial modular units fueled by spent LWR fuel and thorium. Fuel salt is Li-7 fluoride with thorium, plutonium and minor actinides as fluorides. Fission products are extracted on-line (WNA 2016).

(10) Molten Salt Thermal Wasteburner (MSTW)

The Seaborg Technologies' MSTW is a thermal spectrum, single salt, MSR, operated on a combination of spent nuclear fuel and thorium. It is envisioned to produce 100 MWe, or 115 MWe with a two stage turbine, from 270 MWth. The reactor core outlet temperature is 700 °C, but can go as high as 900 °C for special uses, such as hydrogen production. The MSTW is designed around inherent safety features; no active measures are required to control the reactor under abnormal circumstances (IAEA 2016).

According to Seaborg Technologies (2015), one novel safety feature of the MSTW is the use of an overflow system in addition to the commonly used salt plug system. This safety system prevents meltdowns, hinder accidents from human operator error, automatically shuts down in case of out of scope operation conditions, and flushes the fuel inventory to a passively cooled and sub-critical dump tank below the core vessel in case of loss of operation power.

The reactor relies on a novel on-board chemical fluoridation flame reactor, which can continually extract fission products from the salt during operation. The flame reactor is also used to adjust the fuel levels in the salt such that no absorbing control rods are needed during normal operations; this facilitates a better neutron economy in the reactor.

The fully modularised MSTW is suitable for mass production. As a module reaches the end of its lifecycle, it will be extracted and returned for recycling in a central production facility after it has cooled down. The reactor core, including the graphite-based moderator, is projected to have a lifecycle of seven years, while the power plant will operate on the same batch of spent nuclear fuel for the 60 year facility lifetime. The MSTW is in the early design phase and Seaborg Technologies is focused primarily on neutronics, radiative transfer, computational fluid dynamics, and the physics of the design (IAEA 2016).

The MSTW is designed for electricity production, district heating/cooling, sea water desalination, among others. Due to the high outlet temperature it is well suited for synthetic fuel, as well as industrial process heat applications. Its high burnup and the fact that it is fuelled directly with spent nuclear fuel makes it a good option for spent nuclear fuel stockpile reduction. However, it can, without modification, operate on a wide array of different fuels.

(11) **SmAHTR**

According to Greene et al. (2010), SmAHTR is a 125 MWth, integral primary system FHR concept. The design goals for SmAHTR are to deliver safe, affordable, and reliable high-temperature process heat and electricity from a small nuclear power plant that can be easily transported to and assembled at remote sites. The initial SmAHTR concept is designed to operate with a nuclear core outlet temperature of 700 °C, but with a system architecture and overall design approach that can be adapted to much higher temperature as higher-temperature structural materials become available. The SmAHTR reactor vessel is transportable via standard tractor-trailer vehicles to its deployment location.

SmAHTR employs a “two-out-of-three system” philosophy for operational and shutdown decay-heat removal. Transition from operational power production to shutdown decay-heat removal is accomplished without active components, employing passive systems relying on natural convection, and the reactor core is designed with large negative reactivity feedback coefficients.

The SmAHTR concept has been developed with three potential operating modes: 1) Process heat production, 2) Electricity production, and 3) A combined co-generation mode in which both electricity and process heat are produced.

The nuclear core and all primary components are contained in the reactor vessel (integral design). This design eliminates the large break loss-of-coolant accident scenario. The use of an innovative liquid-salt thermal energy storage system or “salt vault” expands the flexibility and applicability of the SmAHTR reactor for all applications. The salt vault offers the potential to combine multiple SmAHTR reactor modules to meet thermal and electricity demands, a robust capability to

buffer the reactors and processes heat load from transients, and the ability to buffer multi-reactor module installations from upset within a single reactor.

As a high-temperature system, SmAHTR is potentially compatible with several highly efficient power conversion technologies. The most attractive options for power conversion systems are Rankine and Brayton cycle technologies.

(12) Mark 1 Pebble-Bed Fluoride-Salt-Cooled High-Temperature-Reactor (Mk1 PB-FHR)

The Mk1 PB-FHR design is the first integrated FHR design to propose driving a nuclear air-Brayton combined cycle for base-load electricity generation.

The purpose of the Mk1 PB-FHR design is to provide efficient and highly flexible power output and grid support services, and therefore to enable a new value proposition for nuclear power. The 236 MWth Mk1 PB-FHR uses a General Electric 7FB gas turbine, modified to introduce external heating and one stage of reheat, in combined-cycle configuration to produce 100 MWe under base-load operation, and with natural-gas co-firing to rapidly boost the net power output to 242 MWe to provide peaking power (Andreades et al. 2014).

The Mk1 PB-FHR is designed so that all components (including the reactor vessel, gas turbine, and building structural sub-modules) can be transported by rail, enabling modular construction. Sub-module fabrication and delivery occur in parallel with module assembly and plant civil construction at the reactor site (Andreades et al. 2014).

The configuration of the Mk1 PB-FHR places the reactor and coiled tube air heaters slightly below grade, and allows the reactor building to be separated from the power conversion system so the reactor can be located inside the plant protected area, while the power conversion system is in the owner-controlled area that requires a lower level of physical protection (Hong et al. 2014).

The Mk1 PB-FHR is designed with advanced passive safety features and intrinsic fuel and coolant properties, which make the consequences of severe accidents much easier to manage.

2.4.3.6 Others Types of SMR

LEADIR-PS100

This is a new design from Northern Nuclear Industries in Canada, combining a number of features in unique combination. LEADER-PS100 is a 100 MWth, 36 MWe reactor with graphite moderator, TRISO fuel in pebbles, lead (Pb-208) as primary coolant, all as integral pool-type arrangement at near atmospheric pressure. It delivers steam at 370 °C, and is also envisaged as an industrial heat plant. The coolant circulates by natural convection. Passive decay heat removal is by air convection (WNA 2016).

2.4.3.7 Hurdles that Must Be Overcome for the Deployment of the SMRs

There are various factors that need to be overcome in order to facilitate the introduction of SMRs in a particular country, but many of these factors affect also any new technological product in several sectors. For this reason, specific needs must be identified, markets must be developed, and the product has to prove itself.

Some of the main factors that are impeding the deployment of the SMRs in several countries are the following:

- Economic competitiveness of SMRs, especially the higher specific construction cost of SMRs with respect to larger nuclear power reactors (Kuznetsov and Lokhov 2011);
- Economies of scale often advantage the construction of large nuclear power reactors, which fit well into long-term programmes for countries with centralised energy supply and well-developed distribution networks;
- Differences in the regulations among different countries could require various design changes, which increase the cost of developing SMRs, according to Nuclear Energy Agency source;
- Potential concerns about the possibility of constructing SMRs in sites close to end-users, based on the current regulatory norms and practises established to support the deployment of nuclear power plants with large nuclear power reactors (Kuznetsov and Lokhov 2011);
- Legal and institutional issues regarding the possibility of international transport of nuclear power plants with factory fabricated and fuelled reactors from one country for deployment in another (Kuznetsov and Lokhov 2011);
- The first-of-a-kind nature of this new type of nuclear power reactors usually implies the need to demonstrate the main new features;
- Management of nuclear waste;
- The multitude of similar SMR designs currently being proposed results in a splitting of efforts and capital;
- There is a lack of capital development. The uncertainty of market conditions in the medium term favours investments in well-established technologies rather than in riskier R&D efforts, according to Nuclear Energy Agency sources;
- Heat generation faces additional problems. Nuclear heat is currently not cost-competitive with fossil fuels in some countries, and district heating is additionally burdened by the distribution cost, according to Nuclear Energy Agency sources.

There are other factors that may affect the deployment of SMRs. Part of the population is generally opposed to any form of nuclear energy uses (specialty after Fukushima Daiichi nuclear accident), although the opposition may be less in the case of SMRs. There is also general uneasiness with regard to advanced technology, as well as fear of radiation. Opposition to SMRs in particular could also come from the concern that, because of their smaller size, there could be more of them on more sites, and that these sites could be closer to home.

2.5 Nuclear Fusion Reactors

The main part of the current nuclear power reactors is based in the so-called “fission reaction”, but there is another type of reaction to get energy, this is the so-called “fusion reaction”. Nuclear fusion is occurring in the Sun to generate heat, which allows us to live, and for this reason it can be said that the fusion energy would be an almost inexhaustible source of electricity in the future. However, nowadays the fusion nuclear energy is not feasible to generate electricity due to different technical reasons; this type of reaction is now in pre-development phase. For this reason, there will be no possibility to build a fusion nuclear power plant, at least until the middle of this century.

Nuclear fusion occurs when the nuclei of atoms collide with one another and bind together. This releases large amounts of energy, which can be converted to heat and used to generate electricity as with other thermal power plants. The most efficient fusion reaction to use on the earth is that between the hydrogen isotopes, deuterium and tritium, which produces the highest energy at the ‘lowest’ (although still extremely high) temperature of the reacting fuels.⁸

For the fusion reaction to occur, the nuclei need to be brought very close together. If the atoms of a gas are heated, the motion of the electrons and the nuclei will increase until the electrons have separated from the nuclei. This state, where nuclei and electrons are no longer bound together, is called “plasma”. Heating the plasma further to temperatures in the range of 100–200 million °C, results in collisions between the nuclei being sufficiently energetic to overcome the repulsive force between them and to fuse.

Although nuclear fusion is unlikely to be ready for commercial power generation in the coming decades, it remains nevertheless an attractive energy solution and arguably, as a sustainable option for large-scale baseload supply in the long-term. If the research and development in fusion energy deliver the advances predicted, then it will continue on a steady course to achieve this aim in the second half of the century (Tzimas 2011).

Fusion energy’s many benefits include an essentially unlimited supply of cheap fuel, passive intrinsic safety and no production of CO₂ or atmospheric pollutants. Compared to nuclear fission, it produces relatively short-lived radioactive products, with the half-lives of most radioisotopes contained in the waste being less than ten years, which means that within 100 years, the radioactivity of the materials will have diminished to insignificant levels (Tzimas 2011).

⁸There are others possibilities, but the conditions are more demanding to produce nuclear fusion reaction. These possibilities are: deuterium-deuterium reaction and deuterium-3 helium. In the latter, it is necessary tenfold temperature for deuterium-tritium reaction.

Other benefits are the following:

- The fuel for fusion is abundantly available. Deuterium is available from sea-water and it is expected that tritium can be produced within a fusion power plant from a small quantities of lithium;
- Fusion has a low environmental impact. The only radioactive wastes produced from a fusion power plant would be from the intermediate fuel, tritium, and any radioactivity generated in structural materials. The radioactivity of tritium is short-lived, with a half-life of around 10 years, and if chosen appropriately the structural materials have a half-life of around 100 years;
- Fusion is inherently safe in that it does not rely on a critical mass of fuel;
- Fusion power plants would present no opportunity for terrorists to cause widespread harm owing to the intrinsic safety of the technology;
- Fusion power plants would provide energy at constant rate, making them suitable for baseload electricity supply;
- Fusion power plants would not produce fissile materials and make no use of uranium and plutonium, the elements associated with nuclear weapons (Nuttall 2008).

At the same time, nuclear fusion has some drawbacks:

- It is an underdeveloped technology, therefore initial cost will be expensive and it will take several decades to be consolidated;
- It is necessary to use initially a lot of energy to get fusion of the nuclei.

Fusion energy production has already been demonstrated by the European flagship experiment, the Joint European Torus (JET). The next step on the path to fusion energy is the International Thermonuclear Experimental Reactor (ITER), which is under construction at Cadarache (France). It aims to carry out its first experiments before the end of the decade and in the following years it should demonstrate the scientific and technical feasibility of fusion energy.

The successful operation of ITER is expected to lead to the go-ahead for the following step, a demonstration power plant (DEMO), which would aim to demonstrate the commercial viability of fusion by delivering fusion power to the grid by 2050.

2.5.1 Plasma Confinement and Its Devices

Given the tendency of plasma to diffuse itself, splitting the nuclei each other at high speed, it is necessary to confinement it in a closed space where cannot run away. In addition, plasma cannot touch the walls of confinement vessel due to its high temperatures, because if that happens the walls would be destroyed and the erosion would contaminate the plasma, stopping the reaction and the plasma.

The plasma should be held for a long enough time to occur a lot of fusion reactions and a very high temperature. There are several options to achieve the plasma confinement:

(a) **Magnetic confinement:**

Plasma is an electrically conducting fluid electrically neutral from the outside in which ions and electrons move practically independently of each other. Submerged in a magnetic field, ions and electrons will follow helical trajectories winding around field lines and will be forced to move along the field. This is the magnetic confinement principle. Initially, straight topologies were examined.

However, these featured a drawback in that they let the plasma escape at extremities. To avoid this, the cylinder was closed on itself, resulting in a toric configuration, frequently characterised by the ratio between its major radius and its minor radius, referred to as the aspect ratio. However, in this type of configuration, the curve and lack of homogeneity of the field give rise to a drift of charged particles. Ions and electrons tend to separate; some move to the top and the others move to the bottom and end up leaving the magnetic trap. To compensate for this effect, the field lines were modified and made helicoidal. Thus, particles successively cross at the top and then at the bottom of the magnetic configuration. Therefore, the drift effect, which is always in the same direction, is on average compensated. This is achieved by adding another magnetic field (poloidal field) perpendicular to the toric field (toroidal field).

The method implemented to generate these helicoidal field lines gives rise to two types of machines:

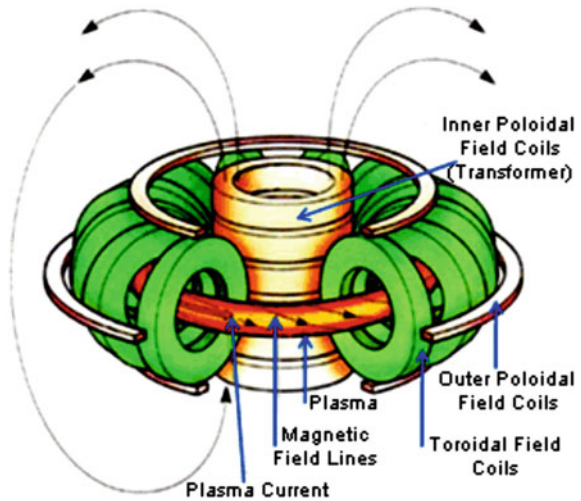
- **TOKAMAK:** The word “Tokamak” come from Russian and is an acronym of Toroidalnaya KAMERA MAGnetiK (toroidal chamber with magnetic coils), first developed by Soviet research in the late 1960s, according to ITER sources. It is a doughnut or torus-shaped vacuum chamber surrounded by magnetic coils, which create a toroidal magnetic field. A second set of coils is centred on the axis or pole of the torus (the hole in the donut). This poloidal magnetic field adds a vertical component to the magnetic field, which has the effect of giving the magnetic field throughout the vessel a twist. This circulates the particles that have drifted towards the outside of the ring back into the centre, preventing the plasma from escaping.

Deuterium and tritium are injected into a doughnut chamber and heated to the point at which its electrons break free (Fig. 2.17).

- **STELLARATOR:** In a Stellarator, unlike in a Tokamak, the field coils alone provide an induced helicity to the plasma. There is no transformer action with a sweeping driven current, so the machine operates in a steady-state mode, with plasma confinement arising solely from the geometry of the external magnetic field.

Several different configurations of Stellarator exist, including: Torsatron (continuous helical coils (Pastor 2013)), Heliotron [helical coils together with a pair of poloidal field coils (Pastor 2013)], Modular Stellarator [with a set of modular

Fig. 2.17 Tokamak principle. *Source* European Nuclear Society (2016)

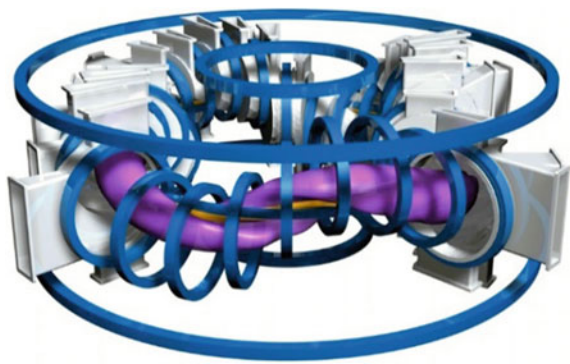


coils and a twisted toroidal coils (Wakatani 1998)], Heliac [the magnetic axis follows a helical path to form a toroidal helix rather than a simple ring shape (Pastor 2013)] and Helias [using an optimised modular coil set designed to simultaneously achieve high plasma, low Pfirsch-Schulter currents and good confinement of energy particles (Wakatani 1998) (Fig. 2.18)].

(b) **Inertial Confinement:**

The method called “inertial confinement” provides the best hope for a workable fusion reactor. This uses bombardment by high-energy photons—X-rays—to confine and compress a pellet of hydrogen and its isotopes. Successive X-ray pulses emanate from a large number of lasers completely surrounding the pellet, doing the work of heating, ionising and compressing the hydrogen to the point where it can fuse. The biggest barrier to a working model lies in the X-ray lasers, which require a lot of energy to operate.

Fig. 2.18 Stellarator principle. *Source* CIEMAT



(c) **Muonic fusion:**

Muonic fusion, which seemed very promising in the beginning, is now only investigated in a few laboratories. The idea is to produce muons, which are the heavy sisters of the electron. The muon is injected into a deuterium-tritium gas mixture. There is a finite probability that the muon will be captured by a tritium or deuterium atom and form a deuterium-tritium molecule. Since the muon is very heavy, the dimension of such a molecule are much smaller than those of a normal molecule with bound electrons. Therefore, the nuclei will be much closer to each other and there is a greater likelihood that they will undergo a fusion reaction. The problem of this scheme is that the production of muons costs too much energy and that the muon will only “catalyse” about two hundred fusion reactions (Hamacher and Bradshaw 2001).

2.5.2 *Fusion Reactor Projects*

The main projects are:

- **Joint European Torus (JET):** It is a Tokamak. The JET investigates the potential of fusion power as a safe, clean and virtually limitless energy source for the future generations. The largest Tokamak in the world, it is the only operational fusion experiment capable of producing fusion energy. Advanced facilities are being developed for the future project, such as ITER and DEMO;
- **International Thermonuclear Experimental Reactor (ITER):** ITER is one of the most ambitious energy projects in the world today. In southern France, 35 nations are collaborating to build the world’s largest Tokamak, a magnetic fusion device that has been designed to prove the feasibility of fusion as a large-scale and carbon-free source of energy based on the same principle that power our Sun and stars, according to ITER sources. The experimental campaign that be carried out at ITER is crucial to advancing fusion science and preparing the way for the fusion power plants of tomorrow.

ITER will be the first fusion device to produce net energy; it will be the first fusion device to maintain fusion for long periods of time, and will be the first fusion device to test the integrated technologies, materials, and physics regimes necessary for the commercial production of fusion-based electricity.

- **Demonstration Power Station (DEMO):** The know-how gained with ITER is to yield the principles underlying construction of a demonstration power plant that affords all the functions of the power plant. This could be followed up with the first commercial fusion power plant. The objectives set for DEMO are still generally formulated for the time being: The device is to supply the power grid with a few hundred MW of electric energy, demonstrate sufficient reliability and availability and work with a closed fuel cycle. This latter means that the device must itself produce beforehand the tritium it needs, according to ITER sources;

Table 2.5 Total number of fusion experimental devices by confinement method

FUSION EXPERIMENTAL DEVICES BY CONFINEMENT METHOD			
MAGNETIC	TOKAMAK	INTERNATIONAL	ITER-DEMO
		AMERICAS	STOR-M, Alcator C-Mod, DIII-D, UCLA ET-LTX, NSTX, Pegasus, PBX-M, TEXT, TFTR, ETE, Novillo
		ASIA AND AUSTRALIA	LT-1, CT-6, CFETR-EAST, HL-1(M), HL-2A, HT-6(B,M), HT-7(U), KT-5, SUNIST, ADITYA, SST-1, IR-TJ
		EUROPE	JET, COMPASS, GOLEM, T3-I, Tore Supra, TFR, ASDEX Upgrade, TEXTOR, FTU-IGNITOR
		AMERICAS	RFP, ISTOK, T-3, T-4, T-10, T-15, TVC, START, MAST, MAST-U
		ASIA AND AUSTRALIA	ATF-CAT, HSX, NCSX, QPS, SCR-1
	STELLARATOR	EUROPE	H-1NF, Lingyun, CHS, HeliotronJ, UHD, JETTU-Heliac
	RFP	IRFX, TPE-RX, EXTRAP T2R, MST	
	OTHER	INTF, LDJ, SSPX, MFTF, MCX, Polywell, Dense plasma focus, ZETA	
	LASER	AMERICAS	NIF-OMEGA, Nova, Nike, Shiva, Argus, Cyclops, Janus, Long path
INERTIAL	LASER	ASIA	SG-I, SG-II, SG-III, SG-IV, GEKKO XII
		EUROPE	HIPER, Asterix IV (PALS), LMJ, LULI2000, ISKRA, Vulcan
	NON-LASER	Qiangguang-1, PTS, Z machine, PACER	

- Wendelstein 7-X: Following nine years of construction work and one year of technical preparations and tests, on 10 December 2015 the first helium plasma was produced in the Wendelstein 7-X device at the Max Planck Institute for Plasma Physics (IPP) in Greifswald, Germany. The production of the first hydrogen plasma followed on 3 February 2016 and marked the start of the experimental operation of the device. The purpose of the Wendelstein 7-X, the world’s largest Stellarator type fusion device, is to investigate the suitability of this configuration for use in a power plant (Max-Planck Institute 2016) (Table 2.5).

2.6 Conclusions

In the coming years several countries will need more electrical power installed to satisfy the foreseeable increase in the electricity demand necessary to support the new industrial development in developed countries, and for the support of the economic boost in developing countries to continue their deployment. The increase use of fossil fuel to satisfy the growth in electricity demand in several countries enhance the concern about climate change and emergency requirement of curbing the emissions of greenhouse gases effect.

In this future situation, nuclear energy has a great opportunity to increase its participation in the energy mix in developed and developing countries alike. The other alternative energy source that can play an important role in the structure of the energy mix of many countries is the so-called “renewable energies”. However, this type of energy source cannot assure a continuous energy supply due to its dependency of weather conditions and storages equipment to stock up the excess energy generated when the power plants are in full operation.

With the intention of enlarging its participation in the energy mix in several countries, nuclear energy industry and institutions are carrying out a great effort in the development of new technologies and type of nuclear power reactors. Fruits of

this effort three lines of investigation can be identified: European Pressurised Reactors (EPRs), Generation IV of nuclear power reactors, and SMRs.

Today several EPR reactors (Generation III⁺ reactor) are under construction and although much is expected of the EPRs, great size of the nuclear power reactors with evolutionary design and improved safety characteristics, and better use of fuel, continuous delays and overruns in their construction are affecting their acceptance in the energy global market.

On the other hand, SMRs have drawn attention because of their size (300 MW or less compared to large reactor 1100–1600 MW); their many useful applications, including generating emission-free electricity in remote locations, where there is little or no access to the main power grid, and in providing heat for industrial applications; their modular design, which means they can be manufactured in a factory and delivered and installed at the sites in modules as needed based on electricity demand growth; and their construction on a smaller scale, reducing the upfront costs of building a nuclear plant power plant and shrinking the time needed to build it.

But if the world really want to change the economics of nuclear energy, then it will need to fundamentally change how nuclear power plants operate. That is where Generation IV reactors come in. These new designs alter the kinds of fuel and coolant that would be used, experimenting with mixes that potentially offer inherit safety, greater efficiency, and less waste.

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