Chapter 6
GEN-IV Reactors

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Glossary

Breeding ratio	Ratio of fission atom production to fissile atom destruction during a certain time interval in a nuclear system.
Closed fuel cycle (full recycle)	One of the nuclear fuel cycle options, in which all actinides in the used nuclear fuel are separated and recycled to reduce the radiotoxicity of a geological repository while enhancing uranium utilization.
Energy sustainability	Ability to meet the energy needs of the present generation while enhancing the ability of the future generation. In GEN-IV, the sustainability is measured by utilization of uranium resource without creating any weakness in economics and environmental goals.
GFR	Gas-cooled Fast Reactor, which features a fast reactor and closed fuel cycle.
GIF	Generation IV international forum, which is a cooperative international endeavor organized to carry out the R&D needed to establish the feasibility and performance capabilities of GEN-IV nuclear systems.
LFR	Lead-cooled Fast Reactor, which features a fast reactor and closed fuel cycle.
MSR	Molten Salt Reactor, which features thermal, epithermal, or fast reactor and closed fuel cycle.

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Open fuel cycle (once-through cycle)	One of the nuclear fuel cycle options, in which the used nuclear fuel discharged from a nuclear system is stored for some period of time and disposed in a geological repository isolating from environment.
Pyroprocessing	The complete set of operations developed in USA. Integral Fast Reactor program based on the pyrometallurgical and electrochemical processes for recovering actinide elements from the used nuclear fuel and recycling them.
SCWR	Supercritical Water Reactor, which features either thermal or fast reactor and open or closed fuel cycle.
SFR	Sodium-cooled Fast Reactor, which features a fast reactor and closed fuel cycle.
Uranium utilization	Ratio of uranium mass used in a nuclear system for energy generation to the uranium mass required by the nuclear system in a nuclear fuel cycle option.
VHTR	Very-High-Temperature Reactor, which features a thermal reactor and open fuel cycle.

Definition of the Subject

Generation-IV reactors are a set of nuclear reactors currently being developed under international collaborations targeting sustainability, safety and reliability, high economics, proliferation resistance, and physical protection of nuclear energy. Nuclear systems have been developed over a number of decades and have evolved to the third generation from the first generation of prototypes constructed in 1950s and 1960s, via the second generation of the commercial reactors operated worldwide after 1970s. While the third generation nuclear systems are currently proposed to the potential customers and under constructions with significant evolutionary in economics and safety based on lessons learnt through plenty reactor operations, nuclear experts from around the world began formulating the requirements for a generation IV of nuclear systems concerning over energy resource availability, climate change, air quality, and energy security. Six systems have been selected for further R&D as generation IV nuclear systems by Generation IV International Forum (GIF), which is a cooperative international endeavor organized to carry out the R&D needed to establish the feasibility and performance capabilities of Generation IV systems. The six systems are Gas-cooled Fast Reactor, Lead-cooled Fast Reactor, Molten Salt Reactor, Sodium-cooled Fast Reactor, Supercritical-Water Reactor, and Very-High-Temperature Reactor.

Introduction

Nuclear energy systems have evolved up to the third generation: a first generation of prototypes constructed in 1950 and 1960; a second generation of commercial nuclear power plants built from 1970, most of which are in operation today; and a third generation of advanced nuclear reactors, called Generation III/III+, which incorporate technical progress based on lessons learnt through more than 10,000 reactor-years of operation. While the generation III/III+ nuclear systems are currently proposed to the potential customers and under constructions with significant evolutionary in economics and safety, nuclear experts from around the world indicated that further advances in nuclear energy systems are required to better meet the rapid growth of environment friendly, highly economic, and secure nuclear energy in both industrialized and developing countries. In particular, it is now globally recognized that the nuclear energy is the practically available massive energy source without greenhouse gas emission among numerous options. To meet these needs, the international nuclear community has engaged in a wide-range discussion on the development of next generation nuclear energy systems known as *Generation IV* (GEN-IV) targeting the deployment around 2030. Figure 6.1 shows the evolution of the nuclear energy systems.

Nine countries, Argentina, Brazil, Canada, France, Japan, the Republic of Korea, the Republic of South Africa, the UK, and the USA, have initially joined together to form the Generation IV International Forum (GIF) [1] for developing GEN-IV nuclear systems that can be licensed, constructed, and operated in a manner that will provide competitively priced and reliable energy products while satisfactorily addressing nuclear safety, waste, proliferation, and public perception concerns. Now, the GIF consists of 13 membership countries added by China, Euratom, Russia, and Switzerland, and two permanent observers of International Atomic Energy Agency (IAEA) and the Organization for Economic Cooperation and Development Nuclear Energy Agency (OECD/NEA).

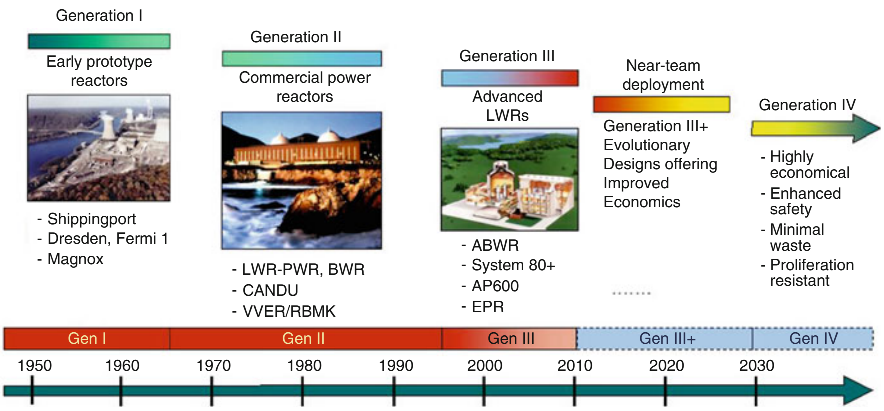

Fig. 6.1 Evolution of nuclear systems

Table 6.1 Goal for generation IV nuclear energy systems

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 utilization for worldwide energy production.
Sustainability 2	Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long-term stewardship burden, thereby improving protection for the public health and the environment.
Economics 1	Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.
Economics 2	Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.
Safety and Reliability 1	Generation IV nuclear energy systems operations will excel in safety and reliability.
Safety and Reliability 2	Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.
Safety and Reliability 3	Generation IV nuclear energy systems will eliminate the need for off-site emergency response.
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.

Beginning in 2000, more than 100 of nuclear experts from the countries constituting the GIF began to discuss for development of the GEN-IV technology roadmap in order to select the GEN-IV nuclear systems. As the first effort in the technology roadmap project [2], eight goals for the GEN-IV were defined in the four broad areas as shown in Table 6.1.

Since the eight goals are all equally important, the promising GEN-IV systems should ideally advance each and not create a weakness in one goal to gain strength in another. Under this central feature of the technical roadmap project, a series of GIF meeting was held in 2002 to conduct the selection process of the GEN-IV nuclear energy systems. The candidate systems were screened by the GIF expert group and six nuclear systems were selected on a consensus of the GIF membership countries such that the systems are the most promising and worthy of collaborative developments. The selected six systems for further R&D are alphabetically

– Gas-cooled Fast Reactor System (GFR),
– Lead-cooled Fast Reactor System (LFR),
– Molten Salt Reactor System (MSR),
– Sodium-cooled Fast Reactor System (SFR),
– Supercritical-water-cooled Reactor System (SCWR),
– Very-High-Temperature Reactor System (VHTR).

GEN-IV Nuclear Systems

In Table 6.2, the primary characteristics of the GEN-IV nuclear systems are summarized. In the roadmap project, it was recognized that the GIF countries would have perspectives on their priority missions for GEN-IV nuclear systems, which can be summarized as electricity generation, hydrogen production, and high-level radioactive material management. All six GEN-IV nuclear systems have electricity applications, while the high temperature and fast neutron spectrum are required for the hydrogen generation and high-level radioactive material management, respectively. The high temperature systems such as VHTR, GFR, LFR, and MSR have potential applications in hydrogen production. By reprocessing and recycling of actinides, the fast reactor systems such as SFR, GFR, and LFR would provide a significant reduction in radiotoxicity of all wastes.

GFR – Gas-cooled Fast Reactor

The Gas-cooled Fast Reactor system features a fast-spectrum helium-cooled reactor and closed fuel cycle. Figure 6.2 shows the schematic of the GFR, which uses a direct-cycle helium turbine for electricity. Like thermal-spectrum helium-cooled reactors such as the Gas Turbine-Modular Helium Reactor (GT-MHR [3]) and the Pebble Bed Modular Reactor (PBMR [4]), the high outlet temperature of the helium coolant makes it possible to deliver not only electricity, but also process heat for hydrogen production with a high conversion efficiency. Through the combination of a fast-neutron spectrum and closed fuel cycle options, the GFR can manage the high-level radioactive waste isotopes.

The technology base for the GFR includes a number of thermal-spectrum gas reactor plants, as well as a few fast-spectrum gas-cooled reactor designs. Past pilot and demonstration projects include decommissioned reactors such as the Dragon Project [5] built and operated in the UK, the AVR [6] and the Thorium High-Temperature Reactor (THTR [7]) built and operated in Germany, and Peach Bottom and Fort St Vrain [8] built and operated in the USA. Ongoing demonstrations include the High-Temperature engineering Test Reactor

Table 6.2 Summary of GEN-IV nuclear systems

	Coolant	Neutron spectrum	Coolant exit temp. (°C)	Fuel cycle	Size (MWe)
GFR	Helium	Fast	850	Closed	1,200
LFR	Lead	Fast	480–800	Closed	50–1,200
MSR	Fluoride salt	Fast/thermal	700–800	Closed	1,000
SFR	Sodium	Fast	550	Closed	30–2,000
SCWR	Water	Thermal/fast	510–625	Open/closed	300–1,500
VHTR	Helium	Thermal	900–1,000	Open	250–300

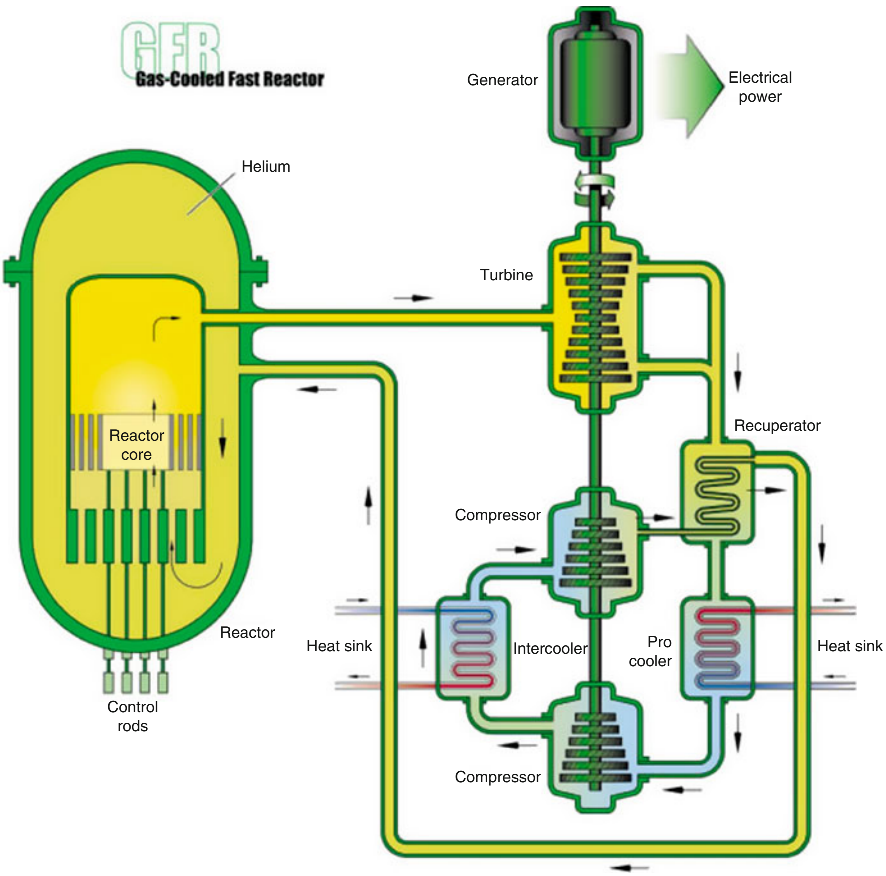

Fig. 6.2 Gas-cooled fast reactor

(HTTR [9]) in Japan, which reached full power (30 MWth) using prismatic fuel compacts in 1999, and the High-Temperature Gas-cooled Reactor (HTR-10 [10]) in China, which reached 10 MWth in 2002 using pebble fuel.

A 300-MWth pebble bed modular demonstration plant is being designed by PBMR Pty for deployment in South Africa and a consortium of Russian institutes is designing a 300-MWth GT-MHR in cooperation with General Atomics. The design of the PBMR and GT-MHR reactor systems, fuel, and materials are evolutionary advances of the demonstrated technology, except for the Brayton-cycle helium turbine and implementation of modularity in the plant design. The GFR may benefit from development of these technologies, as well as development of innovative fuel and very-high-temperature materials for the VHTR.

Spent fuel treatment for the GFR can be accomplished with aqueous processes similar to those of the SFR but qualified for the unique GFR fuel form. A composite ceramic–ceramic fuel (CERCER) with closely packed, coated (U, Pu)C kernels or fibers is considered as the primary option for fuel development. Alternative fuel options for development include fuel particles with large (U, Pu)C kernels and thin coatings, or ceramic-clad, solid-solution metal (CERMET) fuels. The need for a high density of heavy metal elements in the fuel leads to actinide-carbides as the reference fuel and actinide-nitrides with 99.9% enriched nitrogen as the backup.

The reference material for the structure is reinforced ceramic comprising a silicon carbide composite matrix ceramic. The fuel compound is made of pellets of mixed uranium-plutonium-minor actinide carbide. A leaktight barrier made of a refractory metal or of Si-based multilayer ceramics is added to prevent fission products' diffusion through the clad.

Neither experimental reactors nor prototypes of the GFR system have been licensed or built; therefore, the construction and operation of a first experimental reactor – 50 MWth Experimental Technology Demonstration Reactor (ETDR [11]) – is proposed with an extended performance phase to qualify key technologies. A technology demonstration reactor would qualify key technologies and could be put into operation by 2025.

Unlike the VHTR, which uses its considerable thermal mass to limit the rise of core temperature during transients, the GFR requires the development of a number of unique subsystems to provide defense in depth for its considerably higher power density core. These include a robust decay heat removal system with added provisions for natural circulation heat removal, such as a low-pressure-drop core. The secondary circuit uses a He–N2 gas mixture with an indirect combined (Brayton and bottoming steam) power cycle to achieve more than 45% thermal efficiency.

A gastight envelope acting as additional guard containment is provided to maintain a backup pressure in case of large gas leak from the primary system. It is a metallic vessel, initially filled with nitrogen slightly over the atmospheric pressure to reduce air ingress potential. This unique component limits the consequence of coincident first and second safety barrier rupture (i.e., the fuel cladding and the primary system). Dedicated loops for decay heat removal (in case of emergency) are directly connected to the primary circuit using cross duct piping from the pressure vessel and are equipped with heat exchangers and blowers.

Many of the structural materials and methods are being adopted from the VHTR, including the reactor pressure vessel, hot duct materials, and design approach. The pressure vessel is a thick metallic structure of martensitic chromium steel, ensuring negligible creep at operating temperature. The primary system is comprised of three main loops of 800 MWth, each fitted with compact intermediate heat exchangers and a gas blower enclosed in a single vessel.

As a high-temperature and high-power density system, the GFR gives special attention to safety and materials management for both economics and nonproliferation. During the viability phase that is underway now, there is special interest in examining the use of pin-type fuel with a small diameter, fuel and core performance optimized for a simplified GFR having no minor actinide recycle, but with limited Pu breeding and low fuel burnup, core outlet temperature optimized to balance efficiency with materials limits, and the potential of prestressed concrete vessel technology to replace the guard vessel.

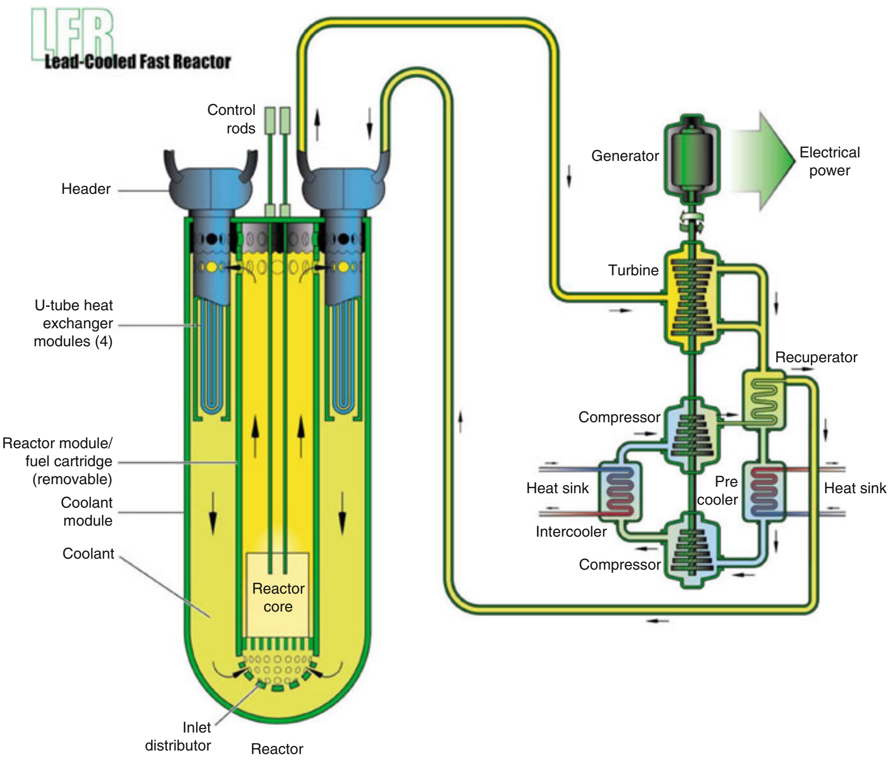

Fig. 6.3 Lead-cooled fast reactor

LFR – Lead-cooled Fast Reactor

The Lead-cooled Fast Reactor is similar to the sodium-cooled fast reactor in terms of neutron spectrum, fuel cycles, and the missions, but the coolant materials are changed to lead (Pb) or lead–bismuth (Pb–Bi). The lead coolant exhibits very low parasitic neutron absorption in fast neutron spectral environment, and this enables the sustainability and fuel cycle benefits traditionally associated with SFR. However, lead does not react readily with air, water, or carbon dioxide, which can eliminate the concerns about vigorous exothermic reactions. It has a high boiling temperature. The need to operate under high pressure and the prospect of boiling or flashing in case of pressure reduction are eliminated. Figure 6.3 shows the schematic of the LFR.

There are several potentials for advances compared to state-of-the-art liquid metal fast reactors. Innovations in heat transport and energy conversion are a central feature of the LFR options. Innovations in heat transport are afforded by natural circulation, lift pumps, in-vessel steam generators, and other features. Innovations in energy conversion are afforded by rising to higher temperatures than liquid sodium allows, and by reaching beyond the traditional superheated Rankine cycle to supercritical Brayton cycle or process heat applications such as hydrogen production and desalination. The favorable neutronics of coolant enable low power density, natural circulation-cooled reactors with fissile self-sufficient core designs that maintain criticality over 15-year refueling interval. For modular and large units, more conventional higher power density, forced circulation, and shorter refueling intervals are used, but these units benefit from the improved heat transport and energy conversion technology. The favorable properties of lead coolant and nitride fuel, combined with high-temperature structural materials can extend the reactor coolant outlet temperature up to 800°C, which is potentially suitable for hydrogen manufacture and other process heat applications.

Two types of LFR reactors were used in Russian submarines of the 1970s with the 155 MWth LFR reactors, OK-550 and BM-440. Recently, Russian joint venture AKME Engineering announced to develop a commercial LFR called SBVR-100 [12]. The core is based on the former LFR reactors used in the submarines and will produce 100 MWe electricity from gross thermal power of 280 MWth, about twice that of the submarine reactors. The coolant is 495°C and 16.5% enriched uranium oxide fuel is used with the refueling schedule of 7–8 years. The small lead-cooled fast reactor concept known as the small secure transportable autonomous reactor (SSTAR [13]) has been under ongoing development as part of the US advanced nuclear energy systems programs (see Fig. 6.4). It is a system designed to provide energy security to developing nations while incorporating features to achieve nonproliferation goals. A 600 MWe European Lead-cooled system (ELSY [14]) has been under development since 2006. The ELSY project aims at the demonstration that it is possible to design a competitive and safe fast power reactor using simple technical engineered features.

The LFR is mainly envisioned for electricity and hydrogen production and high-level radioactive material management. The proposed LFR options include a long refueling interval battery ranging from 50 to 150 MWe, a modular system from 300 to 400 MWe, and a large monolithic plant at 1,200 MWe. The LFR battery option (like SSTAR) is a small factory-built turnkey plant operating on a closed fuel cycle with very long refueling interval (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. Its small size, reduced cost, and full support fuel cycle services can be attractive for these markets. It had the highest evaluations to the GEN-IV goals among the LFR options, but also the largest R&D needs and longest development time.

The options in the LFR class may provide a time-phased development path: the nearer-term options focus on electricity production and rely on more easily developed fuel, clad, and coolant combinations and their associated fuel recycle and refabrication technologies. The longer-term option seeks to further exploit the inherently safe properties of lead and raise the coolant outlet temperature sufficiently high to enter markets for hydrogen and process heat, possibly as merchant plants.

The technologies employed are extensions of those currently available from the Russian submarine lead-bismuth alloy-cooled reactors, from the Integral Fast

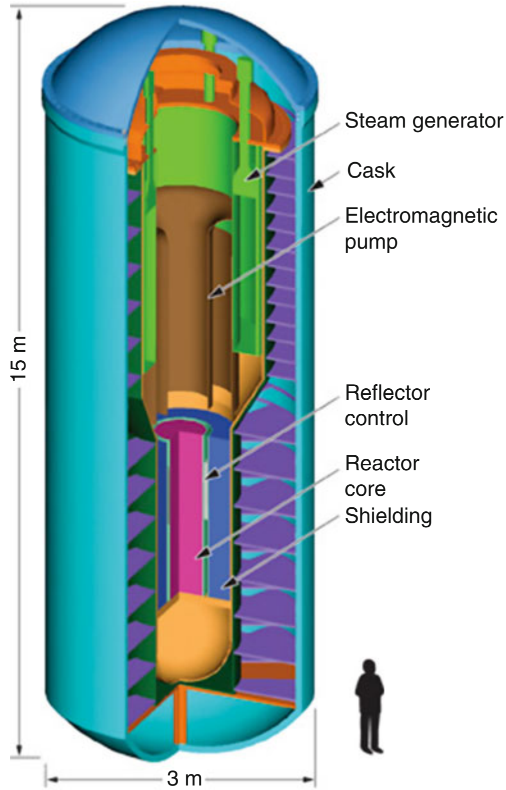

Fig. 6.4 SSTAR–A US lead-cooled fast reactor

Reactor (IFR [15]) metal alloy fuel recycle and refabrication development, and from the Advanced Liquid Metal Reactor (ALMR [16]) passive safety and modular design approach. Existing ferritic stainless steel and metal alloy fuel, which are already significantly developed for sodium fast reactors, are adaptable to lead-bismuth-cooled reactors at reactor outlet temperatures of 550°C.

Corrosion of structural materials in lead is one of the main issues for the LFR. Recent experiments confirm that corrosion of steels strongly depends on the operating temperature and dissolved oxygen. Indeed, at relatively low oxygen concentration, the corrosion mechanism changes from surface oxidation to dissolution of the structural steel. Moreover, relationships between oxidation rate, flow velocity, temperature, and stress conditions of the structural material have been observed as well. The compatibility of ferritic and austenitic steels with lead has been extensively studied and it has been demonstrated that generally below 450°C, and with an adequate oxygen activity in the liquid metal, both types of steels build up an oxide layer which behaves as a corrosion barrier. However, above about 500°C, corrosion protection through the oxide barrier appears to fail and is being addressed with various candidate materials. The prospects for extending much above this temperature are not proven at this time.

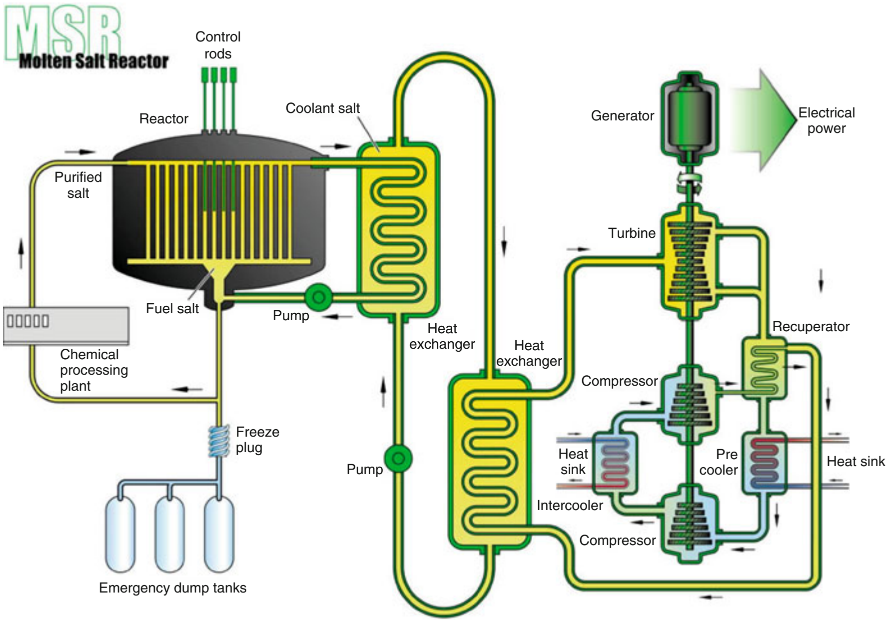

Fig. 6.5 Molten salt reactor

MSR – Molten Salt Reactor

The Molten Salt Reactor uses a molten salt mixture as a primary coolant. Systematic analysis of parameters such as reprocessing time, moderation ratio, core size, and content of heavy nuclei in the salt has resulted in several attractive reactor configurations, in thermal, epithermal, or fast neutron spectrum. The use of a molten salt coolant in a solid-fuel system has been investigated, known as the Advanced High-Temperature Reactor (AHTR [17]), which adapts VHTR fuel form and heat exchanger technology. However, in most MSRs, the fuel is dissolved in the molten salt coolant. Thus, the MSR has unique characteristic compared to other GEN-IV systems: i.e., online refueling and reprocessing are allowed without reactor shutdown because the fuel can move. In addition, the MSR have the following characteristics, which may afford advances: good neutron economy and alternatives for actinide burning or conversion, potential for hydrogen production with high operating temperature, low stresses on the vessel and piping with a very low vapor pressure, enhanced safety by fail-safe drainage, passive cooling, and a low inventory of volatile fission products, etc. Fig. 6.5 shows the schematic of the MSR concept with dissolved fuel.

The MSR was first developed in the late 1940s and 1950s for aircraft propulsion. The Aircraft Reactor Experiment (ARE [18]) was a 2.5 MWth nuclear reactor experiment designed to attain a high-power density for use as an engine in a nuclear powered bomber. One experiment used the molten fluoride salt $NaF\text{-}ZrF_4\text{-}UF_4$ (53-41-6 mol%) as fuel, was moderated by beryllium oxide, used liquid sodium as a secondary coolant, and had a peak temperature of 860°C. It operated for a 1,000 h cycle in 1954. The 8 MWth Molten Salt Reactor Experiment (MSRE [19]) was operated from 1965 to 1969 to demonstrate many features, including lithium/beryllium fluoride salt, graphite moderator, stable performance, off-gas systems, and use of different fuels such as U-233, U-235, and plutonium.

Recently, two MSRs were proposed: Thorium Molten Salt Reactor (TMSR [12]), and FUJI mini-MSR [12]. Figure 6.6 shows the 1,000 MWe TMSR with graphite moderator. Its operating temperature is 630°C and its thermodynamic efficiency is 40%. The salt used is a binary salt, $LiF\text{-}(HN)F_4$, with the $(HN)F_4$ content set to 22%, corresponding to a melting temperature of 565°C. The U-233 enrichment is about 3%. A graphite radial blanket surrounds the core to improve breeding performance. The reprocessing time of the total salt volume is specified to be 6 months, with external storage of the Pa and complete extraction of the fission products and TRU. It is assumed that the U-233 produced in the blanket is also extracted every 6 months. The FUJI mini-MSR is a 100 MWe molten-salt-fueled thorium fuel cycle thermal breeder reactor being developed internationally by Japanese, Russian and US consortium. Like all molten salt reactors, the core is chemically inert under low pressures to prevent explosions and toxic releases.

There are four fuel cycle options: (1) maximum breeding ratio (up to 1.07) using a Th and U-233 fuel cycle, (2) denatured Th and U-233 converter with minimum inventory of nuclear material suitable for weapons use, (3) denatured once-through actinide burning (Pu and minor actinides) fuel cycle with minimum chemical processing, and (4) actinide burning with continuous recycling. The fourth option with electricity production is favored for the GEN-IV MSR. Fluoride salts with higher solubility for actinides such as NaF/ZrF4 are preferred for this option. Salts with lower potential for tritium production would be preferred if hydrogen production was the objective. Lithium and beryllium fluorides would be preferred if high conversion was the objective. On-line processing of the liquid fuel is only required for high conversion to avoid parasitic neutron loses of Pa-233 that decays to U-233 fuel. Off-line fuel salt processing is acceptable for actinide management and hydrogen or electricity generation missions.

The reactor can use U or Th as a fertile fuel dissolved as fluorides in the molten salt. Due to the thermal or epithermal spectrum of the fluoride MSR, Th achieves the highest conversion factors. However, before sufficient fissile is bred for maintaining the criticality, the MSR requires low-enriched uranium or other fissile materials. The operating temperature ranges from the melting point of eutectic fluorine salts (about 450°C) to below the chemical compatibility temperature of nickel-based alloys (about 800°C).

The R&D will focus on fuel salt cleanup, including pyrochemical separation technologies, extraction of gaseous fission products and noble metals by gas bubbling, tritium speciation and control, and conversion of various waste streams into final waste forms. The research will gradually advance from laboratory scale to larger and more integrated demonstrations. MSR burner and breeder fuel cycles will

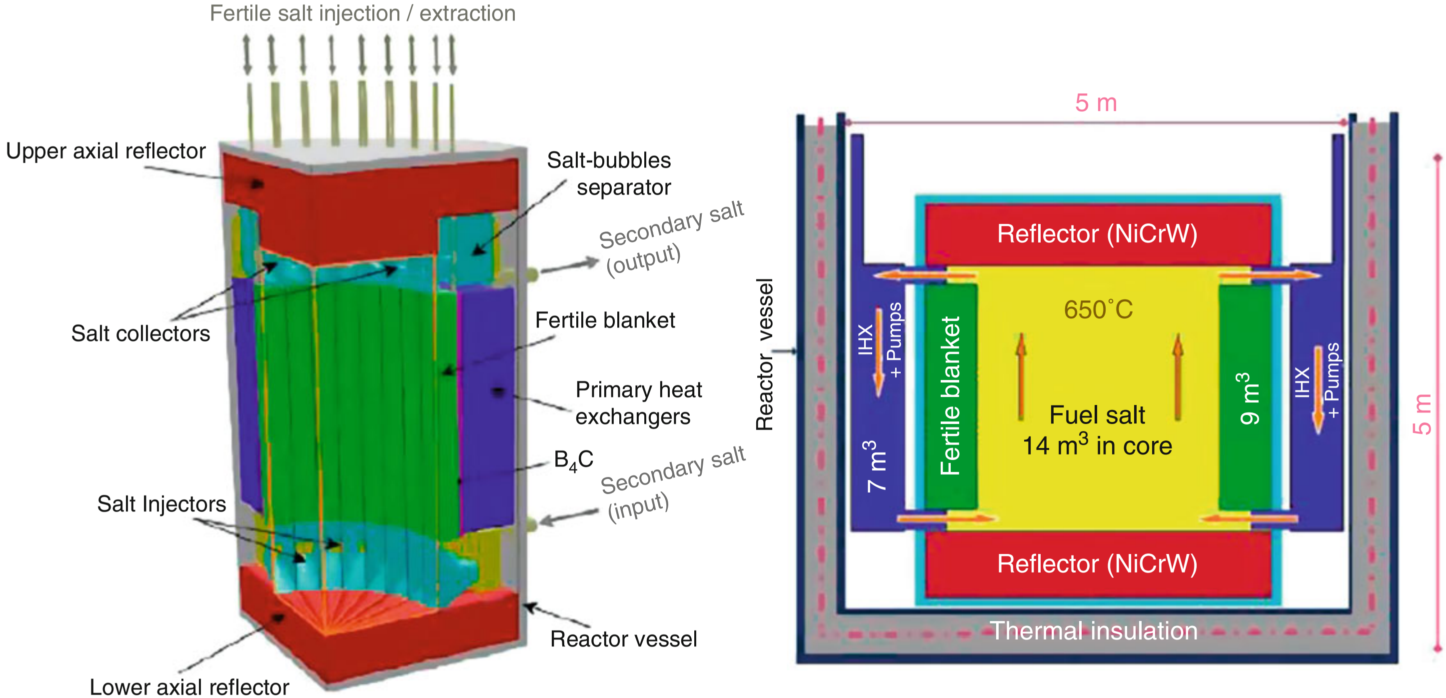

Fig. 6.6 Thorium molten salt reactor

be evaluated and compared with other nuclear systems. This includes examination of the burning of actinides from other nuclear systems, startup of MSRs on various actinides, avoidance of the generation of most actinides by use of thorium fuel cycles, and alternative breeder reactor fuel cycles.

The MSR also addresses research related to the compatibility of fuel and coolant salts with core and structural materials and challenging MSR subsystem integrity: reactor components and reprocessing unit regarding mechanical and corrosion resistance. The high temperature, salt reduction-oxidation potential, radiation fluence, and energy spectrum pose a serious challenge for any structural alloy in an MSR. The design of a practical system demands the selection of salt constituents such as LiF, NaF, BeF_2, UF_4, ThF_4, and PuF_3 that are not appreciably reduced by available structural metals and alloys whose component Fe, Ni, and Cr can be in near equilibrium with the salt. Small levels of impurities in the salt may also aggressively corrode the metallics.

Circulating fuel raises challenges within the core such as the loss of delayed neutrons, temperature differences between the salt, reflectors, and moderator, which requires the coupling between neutronics, thermal-hydraulics, salt composition, and properties of the MSR.

SFR – Sodium-cooled Fast Reactor

The Sodium-cooled Fast Reactor features a fast-spectrum reactor and closed fuel-recycle system. Including electricity generation, the primary mission for the SFR could be either enhancement of the uranium resource utilization or high-level radioactive material management, which depends on the SFR designs. Historically, the enhancement of the uranium resource utilization was the primary mission of the SFR by achieving a high breeding ratio, but the mission was recently shifted for consuming transuranics (plutonium and other long-lived radioactive material) in a very low breeding ratio core. The latter has been studied under the Global Nuclear Energy Partnership (GNEP), which was initiated to seek worldwide consensus on enabling expanded use of economical carbon-free nuclear energy to meet growing electivity demand. The GNEP adopted a fully closed nuclear fuel cycle option that enhances energy security while improving proliferation risk management. One of the major goals of the GNEP is to design and demonstrate a SFR for actinide management like the Advanced Burner Reactor (ABR, [20]).

Based on the arrangement of the primary coolant pump and intermediate heat exchanger (IHX), there are two options for the SFR systems: pool type and loop type (see Figs. 6.7 and 6.8). The primary pump and IHX are placed inside the reactor vessel in the pool type, while these two components are located outside reactor vessel by connecting them trough pipes. A hybrid option [21] of the pool and loop types has also been proposed.

The experiences on design, construction, and operation provide important input into the design process and have the potential to influence the maturity of the

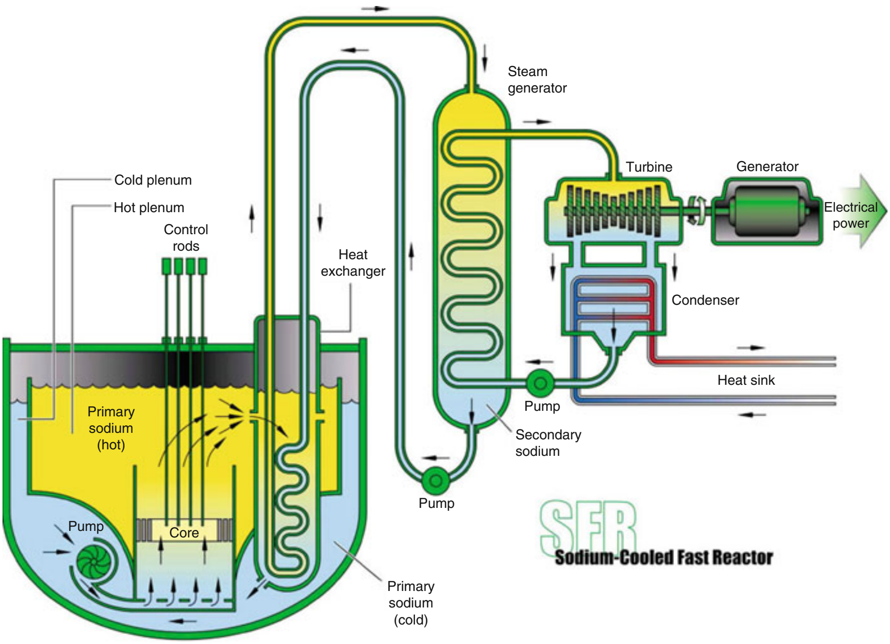

Fig. 6.7 Pool-type sodium-cooled fast reactor

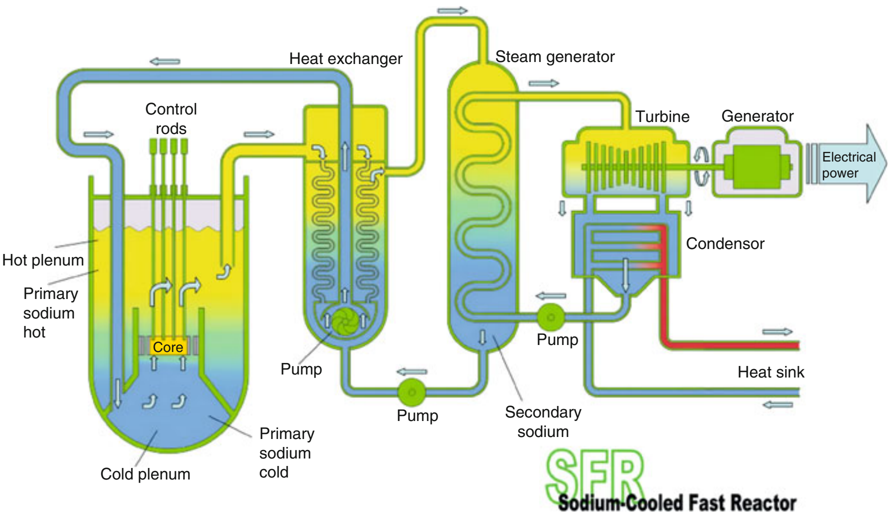

Fig. 6.8 Loop-type sodium-cooled fast reactor

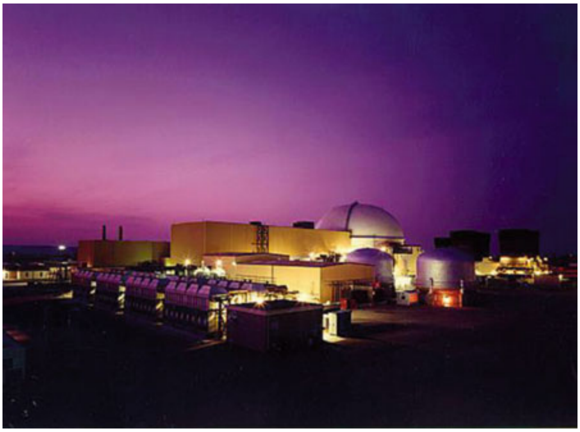

Fig. 6.9 Fast Flux Test Facility

various fast reactor concepts. The greater the number of operating experience years, the greater the opportunity to modify the design based on operating lessons learned. The SFR relies on technologies already developed and demonstrated for sodium-cooled reactors and associated fuel cycles that have successfully been built and operated in worldwide fast reactor programs. Overall, approximately 300 reactor years of operating experience have been logged on SFRs including 200 years on smaller test reactors and 100 years on larger demonstration or prototype reactors. Thus, the technical readiness level, which indicates how soon a system could be deployed, of the SFR is most matured among the six GEN-IV systems.

In the USA, the SFR technology was employed in the 20 MW-electric Experimental Breeder Reactor II (EBR-II [22]) that operated from 1963 to 1994. EBR-II R&D included development and testing of metal fuel and passive safety tests. The 400 MWth Fast Flux Test Facility (FFTF [23]) was completed in 1980 (Fig. 6.9). The FFTF operated successfully for 10 years with a full core of mixed oxide (MOX) fuel and performed SFR materials, fuels, and component testing. The US SFR development program stalled with cancellation of the Clinch River demonstration reactor in 1983, although US-DOE research for advanced SFR technology continued until 1994. The SFR experience also extends to the commercial sector with the operation of Detroit Edison's FERMI-1 plant from 1963 to 1972.

Significant SFR research and development programs are being conducted in China, France, India, Japan, Russia, and Republic of Korea. The most modern fast reactor construction project was the 280 MWe MONJU (Japan) that was completed in 1990, which will be restarted soon. The construction of 20 MWe Chinese Experimental Fast Reactor (CEFR) and coolant sodium loading was completed in 2009, and the full power operation is expected in 2010. India operates 40 MWth Fast Breeder Test Reactor (FBTR) since 1985 and 500 MWe Prototype Fast Breeder Reactor (PFBR) is under construction. The only current fast reactor for electrical generation is the Russian BN-600 that has reliably operated since 1980, and the BN-800 is under construction.

A range of plant size options are available for the SFR, ranging from a battery type systems of a hundred MW-thermal to large monolithic reactors of 3,500 MW-thermal. The sodium coolant outlet temperature is limited by the material properties. Coolant outlet temperatures are typically less than 550°C; however, further increase is considered.

A large margin to coolant boiling is achieved by design, and is an important safety feature of these systems. Another major safety feature is that the primary system operates at essentially atmospheric pressure, pressurized only to the extent needed to move fluid. Sodium reacts chemically with air, and with water, and thus the design must limit the potential for such reactions and their consequences. To improve safety, a secondary sodium system acts as a buffer between the radioactive sodium in the primary system and the steam or water that is contained in the conventional Rankine-cycle power plant.

Metallic and oxide fuel forms are available for the SFR. The metallic fuel was originally chosen in the early fast reactor programs because of its high density, compatibility with the liquid metal coolant, relative easiness to fabricate, and excellent thermal conductivity. In the late 1960s, before the full potential of metallic fuels were established, the interest worldwide for fast reactor fuel turned toward the oxide fuel, because the achievable burnup is limited by a large irradiation swelling. However, the development and irradiation test of metallic fuels continued though the 1970s and it was discovered that the metallic fuel can achieve a high burnup by allowing room for fuel to swell. In addition, the metallic fuel was focused again in the recent fast reactor programs because of its potential passive safety benefits.

The high burnup potential, rich experiences in commercial water-cooled reactors, and the existence of established industry for manufacturing were the critical factors that motivated interest in oxide fuel for the liquid-metal-cooled fast reactors. However, the low heavy metal density and low thermal conductivity are the principal disadvantages of the oxide fuel. The low density is unfavorable to implement a compact core and increase the breeding ratio or cycle length. The low thermal conductivity leads to high temperature gradient from fuel to coolant. As a result, the oxide fuel stores significant amount of Doppler reactivity in the normal operation condition and it provides the unfavorable positive reactivity feedback during an unprotected severe accident.

Recently, the mixed carbide and nitride fuels have been given attention as the alternative fuels for sodium-cooled fast reactor on the basis of their high density, compatibility with sodium coolant, high melting temperature, and excellent thermal conductivity although they are ceramic fuel like a mixed oxide fuel.

The SFR require a closed fuel cycle to enable their advantageous actinide management and fuel utilization features. There are two primary fuel cycle technology options: an advanced aqueous process and the pyroprocess [15] which derives from the term, pyrometallurgical process. Both processes have similar objectives: recovery and recycle of more than 99.9% of the actinides, inherently low decontamination factor of the product, making it highly radioactive, and never separating plutonium at any stage for nonproliferation. These fuel cycle technologies are adaptable to thermal spectrum fuels in addition to serving the needs of the SFR. Thus, the reactor technology and the fuel cycle technology are strongly linked.

Due to the flexibility of the conversion ratio depending on the core design options, the SFR can be operated in three distinct fuel cycle roles. A conversion ratio less than 1 ("burner") can reduce long-lived radioactive waste. A conversion ratio near 1 can increase the uranium utilization without feeding additional enriched uranium. A conversion ratio greater than 1 ("breeder") affords a net creation of fissile materials. An appropriately designed fast reactor has flexibility to shift between these operating modes; the desired actinide management strategy will depend on a balance of waste management and resource extension considerations.

Regarding economics, the reduction of the plant capital costs is crucial. A number of innovative SFR design features have been proposed: configuration simplifications, improved Operations & Maintenance (O&M) technology, advanced reactor materials, advanced energy conversion systems, fuel handling, etc.

With regard to reactor safety, technology gaps center around two general areas: assurance of passive safety response and techniques for evaluation of bounding events. The advanced SFR designs exploit passive safety measures to increase reliability. The system behavior will vary depending on system size, design features, and fuel type. R&D for passive safety will investigate phenomena such as axial fuel expansion and radial core expansion, and design features such as self-actuated shutdown systems and passive decay heat removal systems. The ability to measure and verify these passive features must be demonstrated. Associated R&D will be required to identify bounding events for specific designs and investigate the fundamental phenomena to mitigate severe accidents.

Finally, the development of SFR technology provides the opportunity to design modern safeguards directly into the planning and building of new nuclear energy systems and fuel cycle facilities. Incorporating safeguards into the design phase for new facilities will facilitate nuclear inspections conducted by the International Atomic Energy Agency (IAEA). The goal of this oversight is to always have an accurate grasp of the current inventory through the utilization of advanced technologies to verify the characteristics of the security system (accountancy, detection, and promptness) and the physical protection characteristics (physical protection measures, the monitoring level, and security measures) and for ensuring robust design to guarantee these characteristics. It is also necessary to maintain transparency and openness in terms of information to more effectively and efficiently monitor and verify nuclear material inventories.

SCWR – Supercritical Water-cooled Reactor

The Supercritical Water-cooled Reactor is a water-cooled reactor like Light Water Reactor (LWR) operated commercially, but the SCWR is operated above the thermodynamic critical point of water (374°C, 22.1 MPa). Figure 6.10 shows the SCWR system.

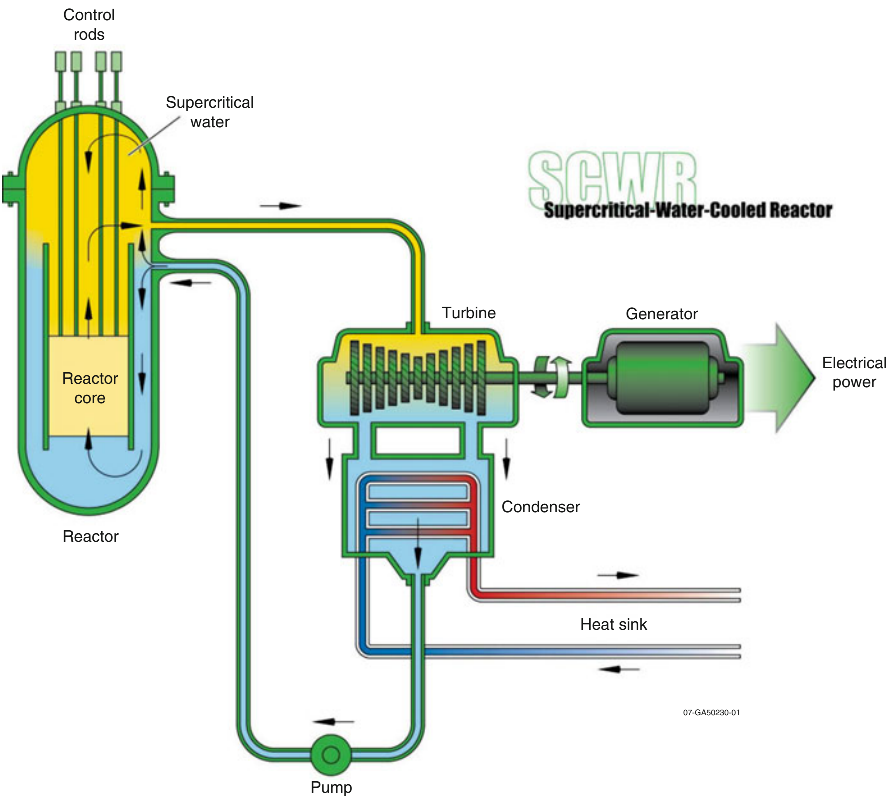

Fig. 6.10 Supercritical water-cooled reactor

The specific heat increases drastically and the water density decreases without boiling of water around the thermodynamic critical point. As a result, the SCWR has unique features that may offer advantages compared to state-of-the-art PWRs: Higher plant thermal efficiency compared to LWRs due to the higher operating temperature. Low density of water without boiling allows the direct cycle like Boiling Water Reactor (BWR), but steam dryers, steam separators, recirculation pumps, and steam generators are not necessary, and as a result, the SCWR can be a simpler plant with fewer major components. Lower-coolant mass flow rate per unit core thermal power results from the high heat capacity of the supercritical water. This offers a reduction in the size of the reactor coolant pumps, piping, and associated equipment, and a reduction in the pumping power. Lower-coolant mass inventory results from the once-through coolant path in the reactor vessel and the lower-coolant density. This opens the possibility of smaller containment buildings. No boiling crisis (i.e., departure from nucleate boiling or dry out) exists due to the lack of a second phase in the reactor, thereby avoiding discontinuous heat transfer regimes within the core during normal operation.

The SCWR systems may have a thermal [24], fast [25], or mixed-neutron spectrum [26] depending on the core design. The Japanese supercritical light water reactor (SCLWR) with a thermal spectrum has been the subject of the most development work in the last 10–15 years. The SCLWR reactor vessel is similar in design to a PWR vessel (although the primary coolant system is a direct-cycle, BWR-type system). High-pressure (25.0 MPa) coolant enters the vessel at 280°C. The inlet flow splits, partly to a downcomer and partly to a plenum at the top of the core to flow down through the core in special water rods. This strategy provides moderation in the core. The coolant is heated to about 510°C and delivered to a power conversion cycle, which blends LWR and supercritical fossil plant technology; high-, intermediate-, and low-pressure turbines are employed with two reheat cycles.

The SCWR can also be designed to operate as a fast reactor. The difference between thermal and fast versions is primarily the amount of moderator material in the SCWR core. The fast spectrum reactors use no additional moderator material, while the thermal spectrum reactors need additional moderator material in the core. The mixed-spectrum SCWR was proposed not only to achieve all advantages of SCWR but also the actinide management. The core uses two coolant flow paths: outer zone with high density water and inner zone with low density water (see Fig. 6.11). Thus, the inner zone features fast neutron spectrum, while the outer zone features thermal spectrum. By recycling TRU in the fast zone, the mixed-spectrum SCWR is capable of keeping all TRU in the reactor.

Much of the technology base for the SCWR can be found in the existing LWRs and in commercial supercritical-water-cooled fossil-fired power plants. However, there are some relatively immature areas. There have been no prototype SCWRs built and tested. For the reactor primary system, there has been very little in-pile research done on potential SCWR materials or designs, although some SCWR in-pile research has been done for defense programs in Russia and the United States. Limited design analysis has been underway over the last decade in Japan, Canada, and Russia. For the balance of plant, there has been development of turbine generators, piping, and other equipment extensively used in supercritical-water-cooled fossil-fired power plants.

The ability to use proven uranium oxide fuel greatly simplifies the application of fuel and fuel cycle technology to the SCWR. However, the supercritical water is known to challenge the corrosion/erosion performance of current cladding technology, and R&D is focused on advanced cladding materials.

There are several unique components needed for the SCWR, including the reactor pressure vessel or pressure tubes and its internal structural components, moderator channels, control rods and drives, the condenser and high-pressure pumps, valves, and seals. The reactor pressure boundary must operate above the high pressure (22.1 MPa) of supercritical water. This may be addressed with thicker sections, and thermal stresses can be avoided with a thermal sleeve for the outlet nozzle.

Zirconium-based alloys, common in water-cooled reactors, may not be a viable material without thermal and/or corrosion-resistant barriers. Based on available data for other alloy classes, there is no single alloy that has received enough study to

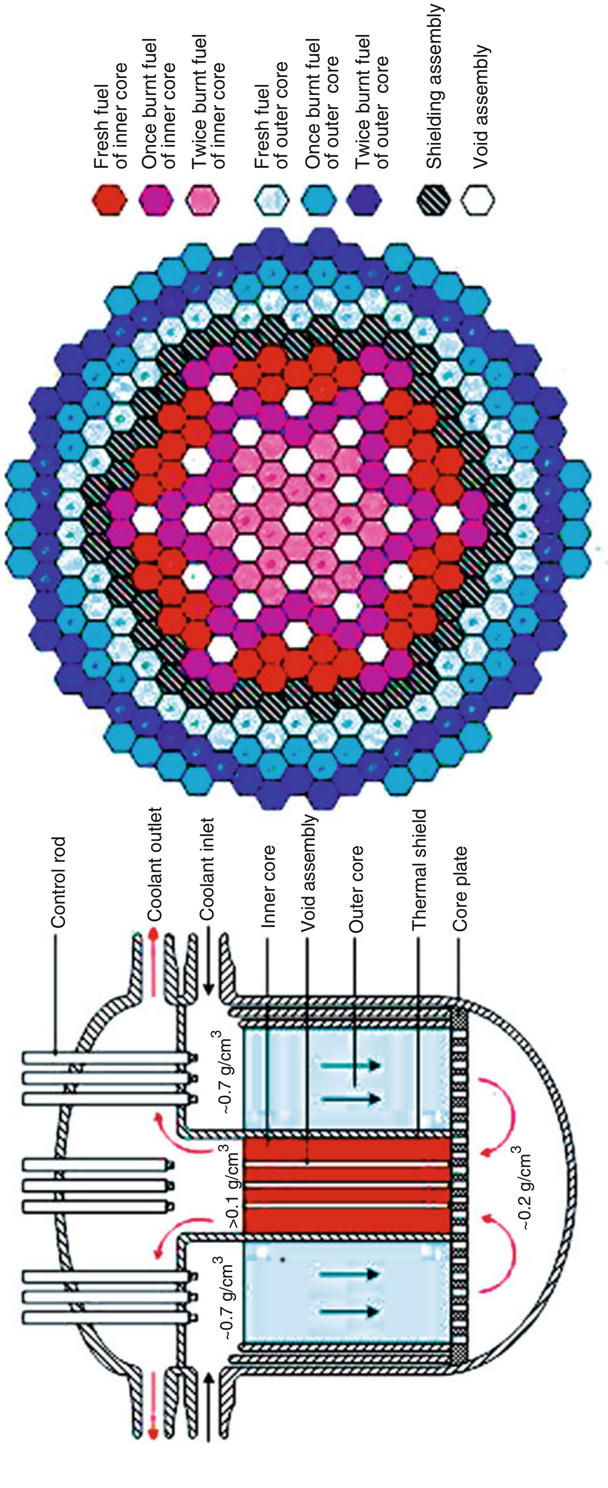

Fig. 6.11 Core layout of mixed-spectrum supercritical water reactor

unequivocally ensure its performance in an SCWR. Another key need of this system will be an enhanced understanding of the chemistry of supercritical water. Water above its critical point is accompanied by dramatic changes in chemical properties. Its behavior and degradation of materials is further accelerated by in-core radiolysis, which preliminary studies suggest is markedly different than what would have been predicted by simplistic extrapolations from conventional reactors.

The approach to development of materials and components will build on evaluation of candidate materials with regard to corrosion and stress corrosion cracking, strength, embrittlement and creep resistance, and dimensional and microstructural stability; the potential for water chemistry control to minimize impacts as well as rates of deposition on fuel cladding and turbine blades; and measurement of performance data in an in-pile loop. All of these are critical to establishing viability of the SCWR.

The SCWR leads the way among GEN-IV systems in the development of advanced materials for water coolant. In fact, the diffusion of this technology into current generation light and heavy water reactors seems assured. However, much remains to be done: the thermal-hydraulic performance during normal and off-normal operation, as well as postulated accidents, needs to be addressed both with advances in the design and safety approach as well as the analysis tools. Issues to be addressed include the basic thermal-hydraulic phenomenon of heat transfer and fluid flow of supercritical water in various geometries, critical flow measurements, the strong coupling of neutronic and thermal-hydraulic behavior, leading to concerns about flow stability and transient behavior, validation of computer codes that reflect these phenomena, and definition of the safety and licensing approach as distinct from current water reactors, including the spectrum of postulated accidents, flow instability, etc.

VHTR – Very-high-temperature Reactor

The Very-high-temperature Reactor is a graphite-moderated, helium-cooled reactor like GT-MHR and PBMR capable of generating electricity, but the coolant output temperature is significantly increased up to 1,000°C. In Fig. 6.12, the schematic of the VHTR is depicted. The higher temperatures of this reactor open the door for industrial heat processing opportunities, in particular, for hydrogen production. The annual US demand for hydrogen is over 12 million tons, and expected to grow to over 30 million tons by 2030. Industry uses hydrogen for fossil fuel refining, treating metals, and food processing. Hydrogen is currently produced primarily from steam methane reforming using fossil fuel as a heat source. Hydrogen can also be produced by various processes using a high-temperature gas-cooled reactor as the primary energy source.

Use of nuclear energy as the heat source of a large-scale hydrogen production operation would result in substantially lower carbon emissions over a natural

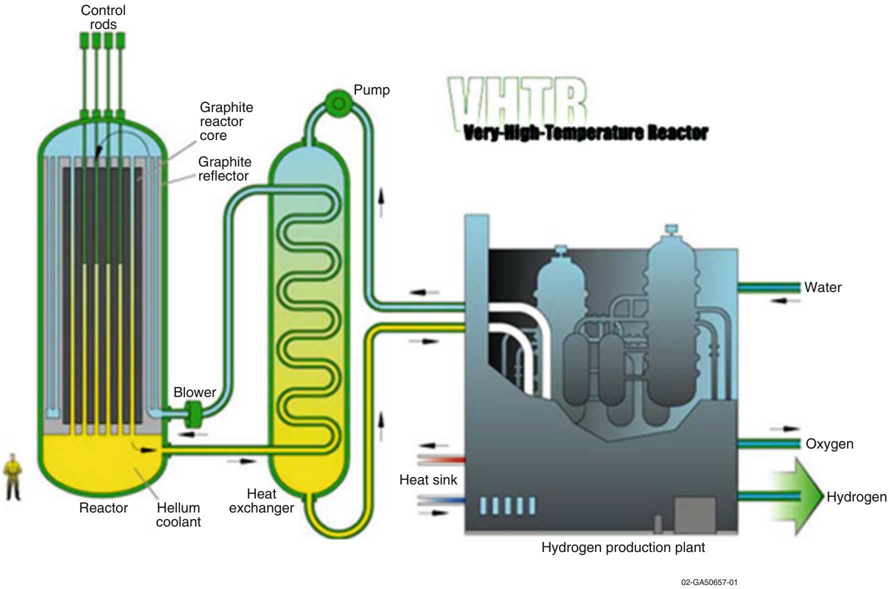

Fig. 6.12 Very-high-temperature reactor

gas-fired steam methane reforming operation. A 600 MWth VHTR dedicated to hydrogen production can yield over 2 million normal cubic meters per day. The VHTR can also generate electricity with high efficiency, over 50% at 1,000°C.

The VHTR has been evolved from gas-cooled reactor experiences and extensive international databases that can support its development. The basic technology for the VHTR has been well established in former gas-cooled reactors, such as DRAGON, Peach Bottom, AVR, THTR, and Fort St Vrain, and is being advanced in concepts such as the GT-MHR and PBMR. The ongoing 30-MWth HTTR project in Japan is intended to demonstrate the feasibility of reaching outlet temperatures up to 950°C coupled to a heat utilization process, and the HTR-10 in China will demonstrate electricity generation at a power level of 10 MWth. The former projects in Germany and Japan provide data relevant to the VHTR development.

The VHTR core uses TRISO particles to form a pebble bed or prismatic fuel element (see Fig. 6.13). The TRISO particle, which has a small diameter of less than 1.0 mm, has a fuel kernel in the form of uranium oxide. The enrichment of the uranium is dependent on the core design purposes. The kernel is subsequently coated with a porous carbon layer (to hold fission gases), a dense pyrolytic carbon layer, a silicon carbide layer, and finally another pyrolytic carbon layer. The coatings surrounding the kernel of TRISO particles produce a very robust fuel form by acting as the containment boundary for the radioactive material. These coatings work in much the same way as the massive reinforced concrete structure surrounding the light water reactors currently in service.

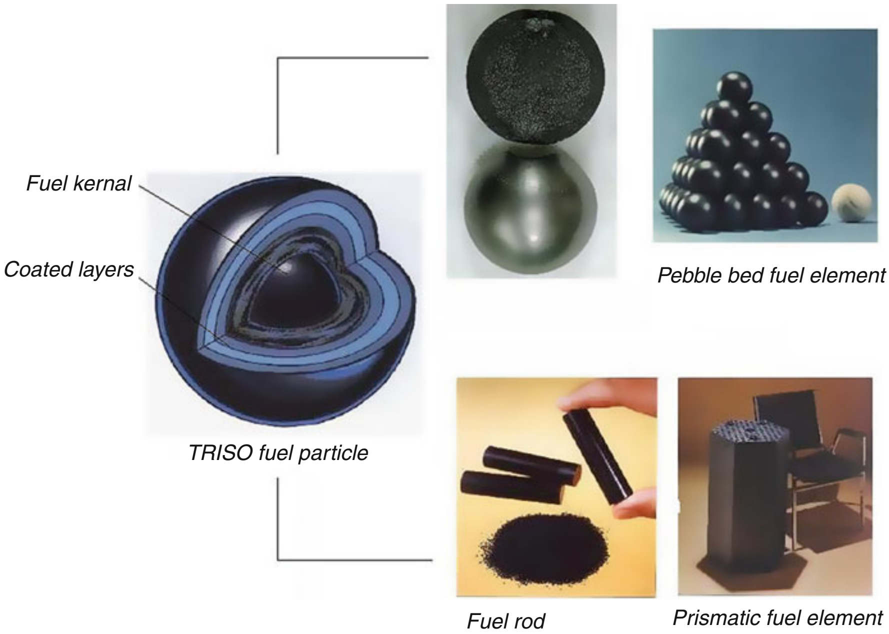

Fig. 6.13 VHTR fuel elements

The reactor core type of the VHTR can be a prismatic block core such as GT-MHR and Japanese HTTR, or a pebble-bed core such as PBMR and Chinese HTR-10. Despite of the alternate fuel element designs (pebble bed versus prismatic), the two baselines have many technologies in common that allow for a unified R&D approach. The well-known TRISO particle fuel with a UO_2 kernel and SiC/PyC coating may be used in either, or it may be enhanced with a different fuel kernel form such as UCO or an advanced ZrC coating through additional research. For electricity generation, the helium gas turbine system can be directly set in the primary coolant loop, which is called a direct cycle. For nuclear heat applications such as process heat for refineries, petro-chemistry, metallurgy, and hydrogen production, the heat application process is generally coupled with the reactor through an intermediate heat exchanger (IHX), which is called an indirect cycle.

The fuel cycle will initially be a once-through fuel cycle specified for high burnup (15–20 atom-%) using low enriched uranium. The operation with a closed fuel cycle will be assessed and solutions to better manage the fuel cycle back end will be developed. The possible use of TRU as a fuel will be studied conceptually for actinide management [27].

The primary emphasis in fuel development is on its performance at high burnup, power density, and temperature. The R&D broadly addresses its manufacture and characterization, irradiation performance, and accident behavior. Irradiation tests will provide data on coated particle fuel and fuel element performance under irradiation as necessary to support fabrication process development, to qualify the

Table 6.3 Anticipated deployment of GEN-IV systems

System	Deployment timelines Viability phase	Performance phase	Demonstration phase
GFR	2012	2020	2025
LFR	2014	2020	2025
MSR	2013	2020	2025
SFR	2006	2015	2020
SCWR	2014	2020	2025
VHTR	2010	2015	2020

fuel design, and to support development and validation of models and computer codes on fission product transport. They will also provide irradiated fuel and materials samples for postirradiation and safety testing. The performance expected for the fuel must be verified for all normal, transient, or accident conditions as well as certain severe accident conditions (beyond design basis). A key claim of the fuel is its ability to retain fission products in the fuel particles under a range of postulated accidents with temperatures up to 1,600°C.

Future Directions

The objective for Generation IV nuclear energy systems is to have them available for wide-scale deployment before the year 2030. The anticipated deployment dates for the six GEN-IV systems are provided in Table 6.3 in terms of R&D phases. The deployment dates of the SFR and VHTR are expected to be earlier than other GEN-IV systems because of their matured technical readiness level.

In the viability R&D phase, the feasibility of key technologies of the GEN-IV systems will be examined. The performance R&D activities undertake the development of performance data and optimization of the system. Assuming the successful completion of viability and performance R&D, the demonstration R&D phase activities involve the licensing, construction, and operation of a prototype or demonstration system in partnership with industry and perhaps other countries. Thus, the detailed design and licensing of the system will be performed during the demonstration phase. The R&D projects and milestones anticipated in each phase were defined in GEN-IV roadmap project [2].

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