CHAPTER

12

Nuclear reactors of the future

Om Pal Singh[a,b,c]

[a]Formerly at Indira Gandhi Centre for Atomic Research, Kalpakkam, Tamil Nadu, India
[b]Atomic Energy Regulatory Board, Mumbai, Maharashtra, India
[c]Indian Institute of Technology Kanpur, Kanpur, Uttar Pradesh, India

12.1 Introduction

Technologies evolve with time in terms of their safety, economics, operational ease, and utility aspects. This has happened in the case of nuclear power also. The first generation reactors were developed in the beginning of nuclear era with primary emphasis on the demonstration of nuclear technology. Reactors such as boiling water reactors (BWRs) Dresden, pressurized water reactors (PWRs) Shippingport, and Magnox Calder Hall come under this category. In parallel with nuclear power technologies, the development of enrichment technology led to the new types of nuclear reactor development. These are BWRs and PWRs. General Electric and Westinghouse companies, respectively, developed and deployed different types of BWRs and PWRs. These and the Canadian and Indian pressurized heavy water reactors (PHWRs) come under the category of Gen II reactors. The Gen II reactors technologies provided advantages of technology simplification and economics. After the Three Mile Island (TMI) and Chernobyl Accidents, emphasis shifted on development of technologies to resolve safety issues. Therefore next generation (Gen III) reactors were provided with advanced safety features. The current reactors such as LWRs, CANDU-6, and AP-600 are the reactors which come under Gen III reactors. Gen III+ reactors came into existence along with the advanced passive safety and inherent safety designing features. ABWR, APWR, AP-1000, and EPR are the reactors that come in this class. These reactors are intrinsically safe and less dependent on the active components such as pumps and valves. Considering the emerging international scenario on future energy needs, nuclear technology and availability of nuclear fuel in 21st century, a need was felt to develop Generation IV reactors. Therefore in the year 2000, Generation IV International Forum (GIF) was set up by the US Department of Energy (DOE) for the purpose of defining the goals of Gen IV reactors, frame work of International co-operation and coordinating R&D work aimed to

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deploy these reactors by the second half of this century. The challenges behind the design of these reactors are to improve nuclear safety and economics of the plants, promote resistance to proliferation and minimize wastes generation so that these are competitive to other energy sources. Other challenges that are to be met are safe management of rare events, protection of capital investment and assurance to the public and the Government that the above issues have been addressed satisfactorily and robust safety of Gen IV systems would assuage public anxiety. The new concepts, design and methodologies used would also be complying with the requirements of Regulatory Bodies. The above aspects are covered in detail in several references [1–7].

Besides Gen IV reactors, there is interest and considerable emphasis in Small and Modular Reactors (SMRs) development and deployment owing to the emerging advantages such as low initial investment, short construction time and interest during construction, modular fabrication possibility, and suitability for deployment of the reactors in remote areas. These reactors can also be called advanced reactors (ARs).

Apart from the improvement in the current technology, new concepts are in vogue to make the reactor technology highly economical by making increased utilization of fuel by high burnup (BU) and high coolant temperature, inherent safety that does not demand operator intervention, automatic protection of the reactor without any powered system and self-regulation without any regulatory device such as control rods. These features are embraced in a new concept known as traveling wave reactors (TWRs) where Breed and Burn (B&B) concept is used. The TWRs are also known by different names like nuclear burning waves (NBW), self-regulating waves (SRW) and CANDLE (Constant Axial shape of Neutron flux and nuclide densities During Life of Energy) reactors.

In this chapter, we would be discussing Gen IV Reactors, SMRs, and TWRs.

12.2 Generation IV reactors

As a part of responsibility, the GIF, consisting of nine members, identified six major concepts from more than 100 competing concepts suggested using a screening methodology [3–7]. The reactors are as follows: very high temperature reactor (VHTR), sodium-cooled fast reactor (SFR), Lead-cooled fast reactors (LFR), molten salt reactor (MSR), gas-cooled fast reactor (GFR), and supercritical water-cooled reactor (SCWR).

The criteria of selection of these reactors includes improvements relative to Gen III light water reactors, their economic competitiveness and safety, increase in fuel utilization, reduction in radioactive waste generation; especially the high level and long-lived ones and great protection of nuclear material against diversion and theft.

The important goals of Gen IV reactors, the approach followed and other aspect of these reactors are discussed below.

- *Economics:* The economics of the reactors is to be improved by better fuel utilization. This could be achieved through designs such as reactors with high coolant temperature to achieve high thermal to electrical conversion efficiency and fast neutron energy spectrum of the reactor for breeding fissile material, burning actinides, and achieving high burnup.

Reduction in construction cost and construction time using innovative design would also lead to better economics.

- *Safety:* Safety is to be improved by incorporating in the design of the reactors, the intrinsic safety features such as low power density, small reactivity reserve, strong negative reactivity feedback coefficients, and passive safety features in critical safety systems, that is, shutdown systems (SDS), decay heat removal system (DHRS), and reactor containment building. The Doppler reactivity feedback to be increased by increasing the epithermal neutron flux.
- *Proliferation resistance:* Current international safeguard regimes are inadequate to meet the security challenges of the expected expansion in nuclear development program in the world. Therefore the nuclear material proliferation resistance should be increased so that the possibility of theft of the fuel and its diversion is reduced. This could be possible by measures like implanting the reactor underground, co-locating of the reactor and reprocessing and fuel fabrication plants, long fuel cycle, and multi-recycling of actinides.
- *Waste generation reduction:* This is to be achieved by high thermal efficiency, high fuel burnup and actinides burning, and long fuel cycle.

Here in Section 12.2.1, the main features of all the six types of Gen IV reactors and challenges faced in technology development are described. In Section 12.2.2, the challenges of reactor physics are discussed. In Section 12.2.3, Indian Gen IV reactors are introduced and Section 12.2.4 provides the conclusions.

12.2.1 Gen IV reactor types

12.2.1.1 Very high temperature reactor

(i) *Parameters:* In very high temperature reactors (VHTRs) [1,8–10], coolant temperature are high and can be 1000 K. As a consequence, the thermal efficiency of the order of 50% is possible. For every 50°C temperature increase, thermal efficiency increases by 1.5%. For temperatures above 1123 K, the reactor, besides producing electricity, can be used for many other applications such as hydrogen production, desalination of water, coal gasification, and blast furnace steel making. Therefore the design, fuel and structural material and other requirements of VHTRs are governed by the high temperature of VHTRs. The VHTRs are thermal reactors. In these reactors, fuel is uranium oxy carbide (UOC) or uranium dioxide (UO_2) as TRISO (tristructural-isotropic) particle. The fuel is enriched (<20%). The use of thorium-based fuels (^{233}U with ^{232}Th and ^{239}Pu with ^{232}Th) is possible. The moderator is graphite which is stable at high temperatures. BeO also can be used as moderator if higher neutron moderating capability is needed. Graphite reaction with water is chemical and endothermic and with air, the oxidation reaction is small. It possesses high heat capacity and therefore core transients are slow and easily controllable. Helium is used as a coolant which is inert. No chemical reaction takes place with any material and no radiation activation occurs. The hot gas can be used to produce electricity through steam cycle or Brayton cycle.

(ii) *Fuel:* The *TRISO* fuel particle is generator of nuclear energy and the container of the fission products. Therefore TRISO particle physical integrity should be exceptionally high. Thus TRISO particle fuel kernel (of about 0.60-mm thickness) is covered by multiple layers of materials that can withstand high temperatures. The pebble is of 6-cm diameter and

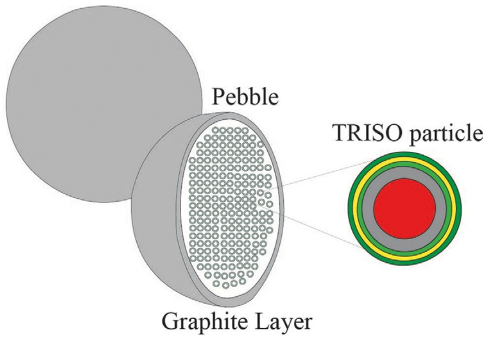

FIG. 12.1 Schematic of a pebble and TRISO fuel particle.

constitutes about 13,000 TRISO particle of 0.92-mm diameter embedded in graphite in the central part. This central part is covered with a graphite layer of about 5 mm (Fig. 12.1). The fuel particle, commonly of less than 1 mm in diameter and the kernel consisting of UO_2 or UCO is surrounded by first layer of pyrolytic carbon (PyC) buffer (thickness 0.06 mm) of low density. The buffer accommodates fission product gases resulting from fuel fission. This layer is followed by three layers of structural coating. The coating consists of silicon carbide layer (SiC) of 0.025-mm thickness and is sandwiched between dense PyC layers of thickness 0.03-mm, termed inner PyC layer (IPyC) and outer PyC layer of 0.045-mm thickness (OPyC). The TRISO fuel particle is structurally more resistant to high temperature, corrosion, neutron irradiation, and oxidation compared with conventional fuels. Each particle acts its own containment system owing to its triple-coated layers.

(iii) *Core configuration:* For these types of reactors, prismatic type and pebble bed type configurations are used all over the world. The schematic of these types, in different forms, are shown very well in the literature [1]. Moreover, in Section 12.2.3, the prismatic configuration is expressed in Figs. 12.2 and 12.3 and for pebble bed type configuration in Fig. 12.4.

Prismatic/block type: Prismatic or block type reactors generally use prismatic block core with conventional stationary system, that is, cylindrical fuel always remains in stationary condition. These reactors use TRISO fuel particles embedded in cylindrical fuel “compacts” which are then embedded in hexagonal graphite blocks to form fuel assemblies. These graphite blocks are stacked to fit in a pressure vessel. The hexagonal graphite block core moderates the neutrons to thermal level. These blocks contain various sized bore holes. Some of these bore holes are filled with fuel rod containing TRISO fuel particles, the others provide the channel for helium coolant to directional flow and some specific holes are reserved for neutron absorbers and control materials. The refueling process can be done by replacing the depleted fuel blocks by fresh fuel blocks periodically. These reactors generally use batch refueling method. The major advantages of this configuration are as follows: high outlet temperature and no water ingress problem.

Pebble bed type: Pebble bed reactors (PBRs) generally consist of the annular cylindrical core. The core shape is right circular cylinder; the bottom of core being cone shaped. The central cylindrical ring acts as reflector and consists of graphite pebbles. The central reflector can be static and dynamic. After central reflector, the annular cylindrical part is fueled zone core and contains fuel pebbles. The fuel zone is surrounded by the side reflector. The number of pebbles used in these reactors depends on the thermal power of the reactor. For example, for a 250 MWt power reactor, there are 11,000 TRISO fuel particles in a pebble and 360,000 (with 80,000 graphite) pebbles in the reactor. A 6-cm diameter spherical fuel pebble consists of a fueled zone (5-cm diameter) and a nonfueled zone (0.5-cm thick). In a 400 MWt power reactor, the number of pebbles in the reactor are 441,000 (with 110,000 graphite pebbles). All the high temperature reactors (HTRs) use fuel and graphite pebbles of same size. The fueling scheme in PBR is in the range of 6–10 fuel cycles. The feeding of the pebbles are from the top of core, they flow slowly downward under the action of gravity and exit at the bottom of the core. Every time at the exit point of the pebbles, the burn-up level and failure status of the pebbles are checked. The failed pebbles are separated out. The pebbles with lower burnup are recycled back into the core for additional burnup. This process of PBR enables the reactor to better fuel utilization and achieving higher burnup. This too enables the reactor to operate with very low excess reactivity.

(iv) Features as per goal required [10–14]

Safety: The temperature limit of the TRISO fuel particle is 1873 K. The low power density is adjusted to cap the 1873 K temperature. The quantity of graphite that is maintained in the core is governed by the extent and the rate at which temperature excursion can occur. The inherent safety in the reactor is achieved by strongly negative temperature reactivity coefficient. The temperature coefficient, on any abnormal temperature rise, shuts the reactor down. The decay heat removal is then by conduction heat transfer process. The core meltdown does not occur even if all cooling systems are turned off and no operator action is taken into account. The high temperature reactors therefore may not require any containment building. These reactors, being small, are suitable for factory fabrication and can be installed below ground level. The size of the fissile and fertile particles in the TRISO fuel particle is less than the critical size so that heat transfer from fissile particle to fertile particle is almost instantaneous and the Doppler feedback from the fertile material is almost instantaneous. Helium coolant does not change phase, is inert chemically and stable under neutron irradiation. Therefore application of helium as coolant reduces the consequences of leakage of radioactive coolant. Hydrogen gas generation is eliminated in the case of loss of coolant accident (LOCA). Thus the core meltdown and significant radioactive release to the environment is prevented. This type of safety has been checked on tests conducted on HTR-10 under anticipated transient without safety (ATWS) control rod axe man SCRAM. The fuel cycle of these reactors prevents generation of weapon grade plutonium and minor actinides get transmuted during reactor operation. The burnup achieved in VHTR is of the order of 700 GWd/ton. Such a large burnup provides deep burn and thus reduces quantity of fuel to be reprocessed and consequently the cost of reprocessing.

Economics: The VHTR system is a multipurpose system. The VHTR system is rated highly in economics owing to its high thermal efficiency, cogeneration process, and safety because of fuel and reactor inherent safety features.

Proliferation resistance: It is extremely difficult to access the isotopes of spent nuclear fuel (SNF) inside the TRISO particle illicitly and SNF is of poor quality. Thus HTRs are rated significantly good from the consideration of proliferation resistance, physical protection, and neutral in sustainability.

Waste reduction: Features such as high efficiency, high burnup, and destruction of actinides enables this system to reduce the waste generation significantly.

Past experience: The experience gained on HTGR plants, in terms of design, construction, and operation has made the HTR technology relatively mature [1–3, 10]. As an example, HTGR plants such as the prototypes at US Fort Saint Vrain (prismatic core, 842 MWt power, 1050 K outlet temperature, 1974 criticality) and the German AVR (Pebbles core, 46 MWt power, 1223 K outlet temperature, 1967–88 operation) provided significant feedback. The DRAGON (prismatic core, 20 MWt power, 1023 K outlet temperature, 1966–76 operation) from UK was the first reactor to pioneer the TRISO-coated particle fuel. Further, technology is being advanced through the projects such as GT-HTR [573 K (multipurpose reactor, Japan)], HTR-PM (China), NuH2 (Korea), and NGNP (USA). The reactors, HTTR-30 (Prismatic core, 30 MWt power, 1223 K outlet temperature, in 1999 criticality) in Japan and HTR-10 (pebble core, 10 MWt power, 1173 K outlet temperature, in 2000 criticality) in China are operational now a days. These reactors have demonstrated achieving high coolant outlet temperature and even 1223 K by HTTR30. The process heat of temperature 1136 K is also achieved.

12.2.1.2 Sodium-cooled fast reactor

Sodium-cooled fast reactors are described briefly in Chapter 3. Some more details are discussed below [15–18].

(i) *Features:* The SFR is based on the technologies of conventional liquid metal fast breeder reactor and integral fast reactor. The SFR follows fast neutron spectrum and closed fuel cycle. These reactors are homogeneous, pool type, sodium coolant with outlet temperature of 823 K (550°C). The coolant high temperature results in relatively higher (>40%) thermal efficiency. The important missions of the SFR are better fuel utilization by transmuting fertile isotopes such as ^{238}U to fissile isotopes ^{239}Pu and ^{232}Th to ^{233}U. This is also utilized for burning actinides resulting in reduction of radioactive waste. Normally the fuel used is MOX fuel $(U,Pu)O_2$. This provides the advantage of applying the existing technologies developed in connection with thermal reactors, for fuel fabrication, reprocessing, and good feedback on operating experience. Besides MOX fuel, metallic, carbide, and nitride fuels can be used if required to improve breeding ratio (BR) and doubling time (DT). For example, utilizing the metallic fuel (U-Zr alloy, sodium bonding with cladding tubes), some of the design objectives of SMRs are achieved. New cladding and subassembly materials have been developed and are under development to increase the burnup. For example, while with the use of SS 316 and D9 (Austenitic SS) materials in SFRs, the burnup achieved is 50 and 100 GWd/ton, respectively. The use of Ferritic Steel (9Cr.-1Mol.) burnup achieved is 150 GWd/ton. The oxide dispersion strengthened steels (ODS) would give burnup up to 200 GWd/ton.

The schematic of pool type FBR systems and fuel subassembly are depicted in Figs. 3.12 and 3.13 of Chapter 3.

(ii) *Reactor safety:* Besides the use of standard safety principles in the technology of SFR, the current emphasis has been to innovative ideas and concepts that enhance safety. That is, incorporate passive safety features in shutdown systems (SDS) and decay heat removal system (DHRS). The increase in the resistant of proliferation is increased by co-locating the nuclear plant and the reprocessing and fuel fabrication plants. The following variety of options are under study and development to improve reactor shutdown system reliability.

Self-actuated SDS (SAS): The control rods (CRs) of SDS and control road drive mechanism (CRDM) are locked by temperature sensitive Curie point magnet so that as the temperature of the coolant reaches a critical (Curie) temperature, the CRs drop under gravity in the core. This device applies to protect the event propagation initiated by transient over power accident (TOPA) and loss of flow accident (LOFA).

Control rod enhanced expansion device (CREED): Another approach is the automatic insertion of the CRs in the core when the rods shaft gets heated and there is thermal expansion of the shaft. This expansion is enhanced by the use of bi-metallic expansion in the shaft.

Hydraulically suspended rods: The CRs float in the coolant under normal flow during reactor operation. Under loss of flow conditions, these rods fall in the core and shuts down the reactor.

Hydraulically suspended neutron absorber balls: These balls also float in the coolant under normal reactor operation but drops in the core under LOF conditions.

Stroke limiting device: The control rods inadvertent withdrawn is stopped by a stroke limiting device. Therefore the rod withdrawal is limited and TOPA is eliminated.

FAIDUS (fuel assembly with inner duct structure): The concept is under study to eliminate the recriticality of the core after its meltdown. By design, if the core melts, the molten fuel is directed to move out of core through the hole in the fuel pellet.

(iii) *Passive decay heat removal:* The decay heat is removed by two diverse systems. That is, operation Grade decay heat removal system in which secondary circuit and steam water system are used to remove decay heat. The second system is safety grade decay heat removal system. The system consists of thermosiphon loops that work purely on natural convection mode. In small-size reactors, reactor vessel auxiliary cooling system (RVACS) is provided which is completely passive and removes heat from safety vessel (SV) surface by natural convection of air. The RVACS also serves as emergency core cooling system (ECCS). In some of the designs, the containment is cylindrical/spherical. This system contains SV and a dome which covers the upper region of reactor vessel (RV), a shielding plug and the equipment located in the plug. For the mitigation of sodium fire, nitrogen gas is provided in the dome.

It can be summarized that SFR technology provides large measures of reactor safety, thermal efficiency is large and fuel Breeding Ratio (BR) is high. This leads to better fuel utilization and therefore better economics in power production. The SFR technology also burns the minor actinides which leads to reduction in waste generation. There are measures that increases the resistance to proliferation. The technology is mature and therefore the deployment of the technology is easy.

12.2.1.3 Lead-cooled fast reactors

The Lead-cooled fast reactors (LFRs) are cooled by either pure molten Lead or its alloy; Lead and bismuth eutectic [19–21]. This is known as LBE. These two coolant-based reactors are also referred to as heavy metal-cooled reactors (HMRs). The boiling point of Lead and LBE are 2010 K and 1943 K, respectively. The melting point for Lead is 600 K and for LBE is 400 K. The advantages of Lead and LBE-cooled fast reactors are higher safety margins, operation at high temperatures, the coolants are inactive chemically with air or water and the voiding reactivity coefficient of coolant is not positive. In case of loss of coolant event, Lead-cooled reactors have higher thermal inertia due to its high density and volume. Lead also provide better shielding to energetic neutrons and gamma rays. The actinides have better solubility with Lead. This, in the events of core melting, minimizes the potential of re-criticality.

In 1950s, Sodium, Lead and LBE were investigated as coolant for fast reactors. Sodium became a preferred choice as coolant because of high power density achievable with sodium as coolant. This resulted in increased BR which was the objective at that time. Between Lead and LBE, LBE is a better coolant as it has low melting point and the reactor is not to be maintained at high temperature, as in the case of Lead. However, LBE is of higher cost due to the higher cost of Bi and higher radioactivity level associated with the Polonium production from bismuth. Therefore for large size reactors, there is inclination towards Lead as a coolant. A Lead-cooled and nitride fuel reactor, the coolant outlet temperature of 1023–1073 K can be achieved. Thus, the production of hydrogen and other heat applications are possible with LFR. It is rated exceedingly good in safety as well as in economics.

The HMRs with closed fuel cycle offers a very wide range of optional designs: This includes: 50–150 MWe power plants, long refueling interval, factory fabricated core and easy transportability and a large plant operation at 1200 MWe. The metal or nitride based fuels containing fertile uranium and transuranic are used. The reactor is cooled by natural convection. The thrust area is on development of three types of design; that is, large size European Lead cooled Fast Reactor, medium size, Russian BREST-OD-300 reactor and small size SSTAR Transportable Reactor.

There are a few drawbacks of Lead or LBE relative to sodium as a coolant. The Lead and LBE are relatively more corrosive to steel and pumping power is high due to high density of Lead. The coolant is highly toxic and radioactive ^{210}Po is formed. It provides environmental pollution risk. The problem of corrosion is to be resolved by selecting the materials that are corrosion resistant. The structural material can be protected by addition of inhibitors like zirconium to the coolant and coating refractive metals such as W, Mo, Nb and nitrides, carbides and ceramics on the tubes.

R&D focus is on the development of corrosion and erosion free material, appropriate chemistry of the coolant, MOX, nitride, MA bearing fuel, fuel handling technology and advance modeling and simulation.

The reactors comply with many goals of Gen IV reactors. The thermal energy efficiency of these reactors is more than 40%. These follow integrated design approach and installed mostly below the ground level. There is reduction in number of components. These are inherently safe, have high passive cooling potential, factory production is possible and the disposal of spent fuel is with entire unit and there is no separation of spent fuel or fuel removal for reprocessing.

12.2.1.4 Gas-cooled fast reactor

In fast reactors, another coolant, that is, a gas, was also considered as an alternative to the sodium [22–24]. It is because of certain good characteristics of gaseous coolant such as gas coolant does not change phase, is inert chemically and stable under neutron irradiation. Therefore application of helium as coolant reduces the consequences of leakage of radioactive coolant. In GFRs (initially known as GCFRs), several potential gas coolant that include helium, steam, carbon dioxide, and steam were considered as coolant for fast reactors. However, steam was rejected due to its in-compatibility with the cladding and positive coolant reactivity effects. The carbon dioxide was rejected due to the effects that include higher pressure drop and increased acoustic loadings. In 1960s, these types of reactors were conceived and development work continued as an alternative to the SFRs (see next paragraph). In 2002, when Gen IV reactors were being conceived and their characteristics were being specified, the concept of GCFR was revisited by GIF assessment and so the GCFR was changed to Gen IV Fast Reactor (GFR).

The first GCFR originated in 1962 in USA at General Atomics. The studies, for a decade or so, were mostly concentrated on the design of 300 MWe demonstration reactor and 1000 MWe commercial plant. The fuels considered were oxides and carbides. In 1960s the work on GCFR expanded to other countries that include Germany, France, Russia, and Switzerland. Prior to 1978, all studies were on a reference design with downward flow of the coolant in the core. However, by that time passive safety was gaining importance and in 1979, the coolant flow in the core was made upward. This was done so that the decay heat removal system can be made reliable by incorporating heat removal by natural circulation. In GFR, the objective is to achieve the goals of Gen IV reactors. Thus the design of GFR followed to achieve the goal of Gen IV reactors.

The neutron spectrum of GFR is fast spectrum for efficient fissile material breeding and proper management of actinides by burning them in the reactor system. This results in better economy and reduction in waste. GFR is helium-cooled, coolant outlet temperature is 1123 K and direct Brayton cycle gas turbine is used. These features are for high thermal efficiency and cogeneration of electricity, hydrogen or process heat for other applications. There are several fuel forms that hold the potential to operate at very high temperature and pressure and can ensure excellent retention of fission products. These are composite ceramic fuel, advanced fuel particles, or ceramic clad elements of actinide compounds. GFR core configuration may be pin- or plate-based fuel assemblies or prismatic blocks.

The GFRs have not been deployed so far though they have great potential. The basic reasons are that the enabling technologies/solutions of these reactors are to be fully developed/solved and demonstrated. The R&D is to be carried out on structural and cladding materials and fuel that is capable of withstanding high temperature. The control and instrumentation and other design features should be such that reliability and safety is ensured under extreme operational conditions. These challenges have resulted in global R&D which is giving promising results. But the demonstration of viability and qualification of components of GFR is a challenging R&D.

12.2.1.5 Molten salt reactor

The molten salt reactors (MSRs) [25–27] use liquid fuel and the circulating fuel also plays the role of coolant. The potential advantages of these reactors over solid fueled reactors are

(i) flexibility in composition of fuel and its online reprocessing, (ii) no difficulty in composing fuel with large amounts of TRU elements, and (iii) better fuel utilization potential as TRUs remain in the liquid fuel and undergo fission or transmutation to fissile isotopes. The advantages of circulating fuel as coolant are (i) no heat transfer delays in the coolant, (ii) no fuel loading plans, and (iii) the passive fuel geometry configuration is settled by gravitational draining.

MSRs development work started in the United States for military jet aircraft propulsion in late 1940s. In 1954, a 2.5 MWt Aircraft Reactor Experiment (ARE) was conducted and the operation of the reactor at 860°C was demonstrated. This established the performance of circulating molten salt ($NaF\text{-}ZrF_4$) with uranium fuel dissolved in it. After 1960, the ARE started a program to develop breeder reactor with 8 MWt power. The ^{235}U tetrafluoride enriched to 33% was dissolved in LiBeZrF at 1133 K.

The MSR operated successfully at full power between 1965 and 1968. However, after that the MSR program was terminated due to the corrosion problem. Now the interest in MSR concept has arisen again.

The fuel of normal MSR is enriched mixture of uranium fluoride (UF4) with lithium and beryllium fluoride (LiBeF) salts coolant. The LiBeF salts remains liquid up to 1673 K and also serve as moderator. The use of lithium (^{6}Li) in primary salt produces tritium. Therefore ^{6}Li is enriched with ^{7}Li. The moderator is graphite and the salt is arranged to flow through the moderator at low pressure and at about 973 K and even at higher temperatures it is possible. The dissolved fission products in molten salt can be removed online in the reprocessing circuit. The actinides are not taken out from the reactor where they undergo fission or get transmuted to other isotopes or elements.

The high outlet temperature (1123–1373 K) of MSR is useful for high heat applications such as production of thermochemical hydrogen, coal gasification and in blast furnace steel making and provides high thermal efficiency in electricity production. The good neutron economy provides actinides burning along with the fission and high transmutation. The low vapor pressure of molten fluoride salt reduces stresses on the vessel, pipes, and other components. It offers the online refueling, reprocessing and fission product removal. The low inventory of volatile fission products in fuel and their fail-safe drainage and passive cooling are the main sources of inherent safety in MSRs.

The experiments conducted on water-based liquid fuel has established MSRs intrinsic stability and tests on liquid fluoride-based fuel at 923 K established corrosion-free circulation. Continuous reprocessing has also been tested. Continuous extraction of fission gases, that is, Kr and Xe product has been possible before these decay to Rb and Cs. Besides this, MSRs could not be constructed because MSRs requires removal of the fission products (FPs) and protactinium to eliminate neutron capture and formation of fissile material, ^{233}U in mass production (proliferation issue).

The inherent and passive safety is provided in MSRs by well-known methods such as negative temperature and void coefficient of reactivity. These reactors have significant load following capability. In MSRs, reactivity control is done through the pumps of secondary coolant salt circuit or the circulation that can change the temperature of fuel salt in the core. Criticality can be prevented by dumping primary salt in dump tank by gravity force draining. It allows online refueling, reprocessing and fission product removal.

The development work on MSR is going on in many countries such as Europe, France, Russia, Canada, japan, Korean republic, and the United States. European Union is working on EVOL (evaluation and viability of liquid fuel fast reactor system) project with focus on studying the design and safety and many other aspects of this type of system. France, in internal frame, is working on 1000 MWe fast spectrum reactor. Russia is working on MOSART (Molten Salt Actinides Recycler and Transmutation) and designed to incinerate TRU. The facility is 2240 MWt with epithermal neutron spectrum. China is working on Thermal MSR, Japan is conducting several projects and the United States is carrying out work on salt-cooled HTR.

The Molten Salt Fast Reactor (MSFR) is 3000 MWt (1300 MWe) power, 43% efficiency and power density is 330 MW/m^3. The core reactivity depends on the temperature and amount of fissile material in central cavity. Fuel BU is uniform. It is hard to control the distribution of power and fissile material. This may result in hot spots in the system. Safety characteristics of MSFR form combination of fast reactor safety and the one associated with liquid fuel. However, these are affected by the drawbacks of salts, like high solidification temperature and low thermal inertia. Reactor design, simulation and other studies are ongoing to verify whether the goals of Gen IV reactor are being satisfied. It is observed that for MSFR, the above features are more favorable when compared with the solid state fuel fast reactors. The MSFR concept is recognized as a long-term alternative to solid fuel fast reactors.

12.2.1.6 Super critical water-cooled reactor

The reference design of the reactor constitutes thermal neutron spectrum, UO_2 fuel, austenitic or ferritic martensitic steel cladding material, 1700 MWe power, 44% thermal efficiency, 553 K coolant inlet and 783 K outlet temperatures and fuel BU is 45 GWd/t HM. The Super Critical Water Reactor (SCWRs) [28–30] main advantage arises from high temperature which results in high efficiency and improvement in economics. The research focus is on improving the safety, increase resistance to proliferation and other aspects of Gen IV reactors. Canada, Japan, Russia, Europe, and China have SCWR design. India also is involved in the development of SCWR technology.

In Japanese model once through cycle is followed and direct Rankine cycle is used. The power output is in the range of 600–1700 MWe. In European SCWR, the efficiency is 43.5% and power output is 1000 MWe. The coolant enters in the inlet nozzles and comes out through outlet nozzles. In Canadian approach pressure is 25 MPa and outlet temperature is 625°C. Thermal efficiency is 48%. Superheated steam directly goes to the turbine and so the need of steam generator is eliminated.

The SCWR operates above the thermodynamic critical point of water (647 K and 22.1 MPa). These reactors can be of open fuel cycle with thermal neutron spectrum and closed fuel cycle with fast neutron energy spectrum. The full recycle of actinides can be done using central fuel cycle facilities. At high temperature, the coolant enthalpy increase is significantly high in the core. This leads to lower mass flow rate in the core for a given thermal power. Higher enthalpy permits decrease in the size of the turbine system. Comparably this results in reduction of the size of several components such as the reactor, coolant pumps, and piping. Low coolant inventory may lead to the possibility of smaller containment building. The reduction in the capital cost is expected 30% due to reduction in size of containment, safety systems and simplicity in the design. Since coolant remains in the single (fluid) state in a reactor, the discontinuous heat transfer regimes get avoided. The dry out problems and departure from nucleate boiling of PWRs are not present in this concept. The steam is directly fed to turbine and therefore some components such as steam generators, steam dryers, and recirculation pumps are eliminated.

In SCWR the R&D focus is on several aspects. Some of these are water chemistry and radiolysis in supercritical state of water and thermodynamic behavior of water in pseudocritical region is to be understood. As in depressurization accident, water, and steam phases get separated. This affects the heat exchange. Thus the consequences of this process are to be understood. Safety demonstration for SCWR by experiment is difficult because of close coupling between neutrons and thermal hydraulics. Constraints in fuel reprocessing is to be accounted in fuel design stage.

12.2.1.7 The Gen IV parameters

Generation IV reactor parameters are summarized in Table 12.1. A very comprehensive description, covering all aspects of Gen IV reactors, is given in Ref. [1–3, 10].

TABLE 12.1 Typical Generation IV reactor parameters [3, 7, 10, 24].

Parameter	VHTR	SFR	HMR	GFR	MSR	SCWR
Power (MWe)	250–300	50–1500	20–1200	1200	1000	300–1500
Neutron spectrum	Thermal	Fast	Fast	fast	Thermal/fast	Thermal
Power density (MWt/m^3)	5–10	300	100	100	330 fast spectrum	100
Fuel cycle	Open	Closed	Closed	Closed	Closed	Open/closed
Fuel	^{235}U–^{238}U, Pu–^{238}U, ^{233}U–^{232}Th	Oxide/metal Pu- ^{238}U ^{235}U–^{238}U	Oxide/ nitride/ carbide Pu–^{238}U	Nitride/ carbide Pu–^{238}U	Fluorides of ^{233}U–^{232}Th, ^{235}U–^{232}Th, Pu–^{232}Th	MOX Pu–^{238}U
Coolant	Helium	Sodium	Lead/LBE	Helium	Fluoride salt	Water
Temperature (°C)	900–1000	500–550	480–800	400–850	700–800	510–625
Moderator	Graphite	–	–	–	Graphite	Water/heavy water
Efficiency	55	40	40–50	50	50	40–50
Experience	7 HTR operated	400 ry	No civilian reactor	None	Two thermal reactors built	None
R&D and issues such as	Low power, passive safety acceptance, interface with heat users	Enhanced safety demo., core is not most required configuration	Corrosion, nitride fuel development, simulation modeling	Limited studies, long schedule, lack of interest	Corrosion, salt processing, fuel cycle development	Positive void reactivity, stability of core and primary circuit, gap in heat transfer and critical flow data

12.2.2 Reactor physics challenges

It can be observed that the goals of Gen IV reactors put constraints and contradictory requirements on reactor physics design. For example, fast neutron spectrum is needed for better fuel utilization. However, for favorable value of Doppler reactivity coefficient which is an important safety parameter, needs epithermal neutron spectrum. In fast reactors, achieving intrinsic safety by passive heat removal by natural convection process requires higher volume fraction of the coolant. However, the requirement in fast reactor technology is of high fuel fraction. This is to be compensated by higher enrichment. If the core is not pan-caked (core height $<$ diameter), coolant void fraction would add higher positive reactivity. A pan-caked core results in neutron leakage. In HTGRs, the decay heat removal by heat conduction process limits core volume and hence the power rating. Therefore the tall annular configuration is desirable to reduce radial heat conduction length. This leads to higher neutron leakage that requires high enrichment than the HTGRs of earlier years. Therefore none of the above six types of reactors can be designed with all the requirements mentioned earlier.

The new concepts of nuclear reactors generate reactor physics issues like availability of nuclear data for different materials including minor actinides at high temperatures is a problem. It is discovered that ^{238}U cross-sections set differ widely. Current mathematical methods and models may fail in some especial cases. For example, in high quantity graphite-moderated reactors such as HTRs, the neutron spectrum is more epithermal. Thus the self-shielding treatment model of resonances may not be appropriate.

The fuel in the form of particles is dispersed and the pebbles in PBRs are stacked randomly. Therefore in the Monte Carlo calculations, stochastic approach is to be followed. In the HTGR concept, a wide range of diverse fuel cycles (different fuel types, fraction of fertile material, and TRISO in graphite) can be achieved. This leads to strongly heterogeneous core configurations and the neutron spectrum becomes highly space dependent. Triso-coated fuels also create double level of heterogeneity in the system. The pebbles do not follow the same path every time they pass from top to bottom through the reactor core. Thus the isotopes in the pebble are space dependent. This makes the task of determining the actual reactor power history of a particular fuel pebble highly uncertain. The validation of newly developed methodologies and nuclear data used is a challenge. Managing the long fuel cycle is complex. Any new concept always generates issues and so it is hard in obtaining licenses from the nuclear regulatory bodies.

However, despite the various challenges, the solutions are being obtained by making improvements in data base, computational methodologies and using advances in parallel computation.

12.2.3 Indian high temperature reactors

High temperature reactor [31–34] development work is in progress in India. The basic objective is to develop reactors with high thermal to electrical conversion efficiency, produce hydrogen as fuel for transport sector and use process heat for desalination of sea water and other purposes. Two reactors are under development, that is, one is termed as compact high temperature reactor (CHTR) with 100 kWt power, and the other is an innovative high temperature reactor (IHTR) with 600 MWt power.

12.2.3.1 Compact high temperature reactor

This reactor is under development for demonstration of HTR-associated technologies. Achieving high burnup, inherent safety, passive safety systems, and heat removal by natural circulation is the design goal of the reactor. The reactor can be deployed at remote locations as a power pack and therefore the reactor size is small and compact in weight. The reactor configuration is prismatic. The reactor power is 100 kWt, coolant is LBE and its outlet temperature is 1273 K. Such high temperatures are needed for producing hydrogen. The fuel is $^{233}UC_2 + ^{232}ThC_2$ or U-Th enriched with ^{235}U and is made of TRISO-coated fuel particle. The Beryllia (BeO) is chosen as moderator as it is a better moderator than graphite. In Beryllia, energy loss per collision is high, number of collisions to achieve thermalization are less, scattering cross-section is high and absorption cross-section is low. Furthermore, its melting point is high and vapor pressure is low at high temperature. The BeO is in the form of hexagonal shaped blocks. At the center of each block, graphite fuel tubes are located which carry fuel in the bores. The coolant flows through the central bore of fuel tubes (the schematic is shown in Figs. 12.2 and 12.3 [32]. Blocks of each BeO reflector surround the moderator blocks which are surrounded further by graphite blocks. Reactor vessel material is Nb-1% Zr-0.1% C alloy.

The coolant flows upward to the upper plenum through the fuel tube and comes down through the down-comer tubes. Heat utilization vessels, set up above the upper plenum, act as an interface to systems of high temperature process heat applications. A set of sodium heat pipes passively remove the heat from upper plenum to these heat utilization vessels. Another set of heat pipes transfers the heat to the atmosphere in case of an accident.

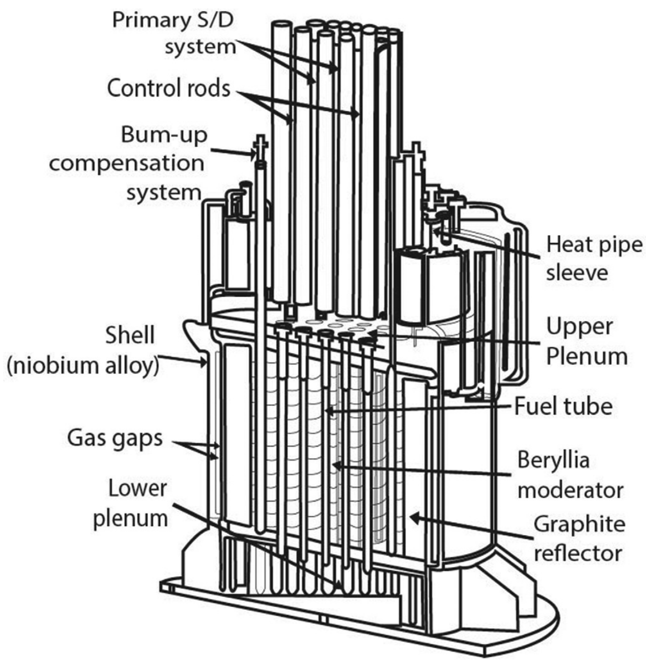

FIG. 12.2 Components layout of compact high temperature reactor (CHTR).

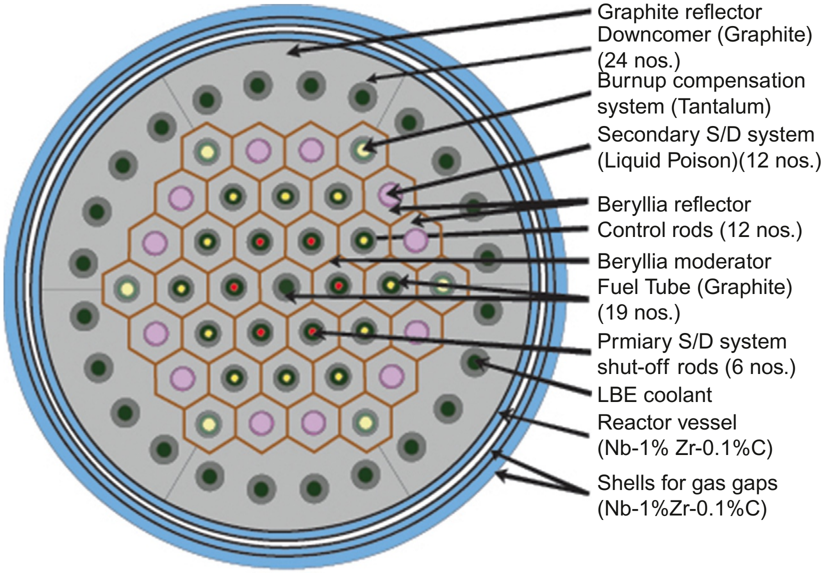

FIG. 12.3 Core cross-section of the core of compact high temperature reactor (CHTR).

The fuel development technology is in progress. The coating technology has been developed, applied on uranium oxide and tested under irradiation in fast breeder test reactor (FBTR) at IGCAR Kalpakkam. The fuel manufacturing technology work and development on fuel characterization is also in progress. The thermal hydraulic design is being tested by thermal hydraulic loops and computer codes are validated against experimental tests.

The CHTR core materials face extreme environmental conditions such as high temperatures, neutron fluence, and corrosive environment of LBE coolant. For certain components, carbon-carbon composites have been developed and are in further process of qualification. Oxidant-resistant coating is being used for carbon-based components. Many techniques of different types of coating such as plasma spray have been developed. For high temperature components, Niobium alloy (Nb-1%Zr-0.1%C) is a very good refractory metal alloy. It is suitable for several structural applications in advanced nuclear reactors where temperatures exceed 1273 K. The alloy has been developed and further test and qualifications are in progress.

The CHTR is designed with many inherently safe features common to high temperature reactors such as strong negative Doppler reactivity coefficient, high thermal inertia, a very large temperature margin to boiling point of LBE coolant, and low pressure. The passive systems include power regulation and shutdown mechanisms, decay heat removal by convection heat transfer and passive heat transfer to secondary side using sodium heat pipes.

The *reactor physics*: The physics design methodology is standard but by applying needed improvements. However, some challenges were faced due to compact size of the reactor. Due to this, in CHTR, the initial fissile fuel requirement is high. This results in high excess reactivity. The compact size also leads to limitation of space for neutron absorbers of control and shutdown system rods. Thus the reactivity worth of each rod is high to compensate the high excess reactivity and has safety implications. This issue is resolved by using Gadolinium as Burnable Poison (BP) in the central part of the fuel assembly. However, it resulted in reduction of negative fuel temperature coefficient of reactivity. This problem is resolved with the help of special absorber rods, named burnup compensation rods (BCR) by introducing them in fixed BeO reflectors. All BCR are left completely inserted in the operation phase of the reactor.

The power regulation and control of the reactor has been designed with the passive engineering design features to remove the excess reactivity from the system. In this approach, the absorber in annular shape floats on LBE in liquid form in the center of hexagonal BeO reflector blocks surrounding the fuel for control and regulation. A secondary shutdown system consisting of a set of seven shut off rods of tungsten has been provided which fall by gravity in the central seven coolant channels. The passive control cum shutdown system is slow in response. In place of control devices, axial or radial movement of the reflector blocks has been studied. It is found that both the axial or radial options provide sufficient reactivity margin and any of the two can be used. But the axial movement of the reflector appears to be better in terms of saving the fissile material.

The loss of regulation accident (LORA)analysis showed that even in case of no shutoff rods falling in, the fuel temperatures are within the permissible limits and power stabilizesto a value of about 2–3 times the initial value.

12.2.3.2 *Innovative high temperature reactor*

The proposed IHTR is a 600 MWt power (18 MWe) reactor, production of 80,000 Nm^3 per hour hydrogen and 375 m^3 per hour generation of drinking water. This is a pebble bed reactor. The pebble bed concept is chosen because of its advantages such as better fuel utilization and fuel handling. Here, fuel loading and removal and burnup measurements are online. This option is used to send the pebbles back to the core in case burnup is not as per design value. Molten salt (FLiNaK), with feasibility to provide better natural circulation cooling, has been chosen as coolant. Fuel is in the form of $^{233}UO_2$ and $^{232}ThO_2$ based high burn-up TRISO fuel particle. Graphite would be used as moderator and reflector. Coolant outlet and inlet temperatures are 1273 and 873 K, respectively. Reactor control is by passive power regulation and hydrogen production is by high efficiency thermochemical high temperature steam electrolysis processes.

The reactor is a long right circular cylinder with annular core. One of the design considerations is passive decay heat removal from the surface of the reactor vessel. For this the reactor height to diameter is kept high. Fig. 12.4 depicts the schematics of the reactor. The graphite reflectors are in the center, on the top, in the bottom and outside of the annulus. For control elements, vertical bores are provided in the central and outer reflectors.

The control and shutdown devices are located around the outer radius of pebble annulus and the secondary shutdown device is located around the central reflector (inner radius of the pebble annulus).

To develop and demonstrate the technology, a new IHTR of 20 MWt capacity is under design. Analytical modeling of circulating fuel is being developed.

A molten salt coolant loop has been set up to study the natural coolant circulation and thermal hydraulics and material compatibility aspects. Experimental facility is being designed to study the corrosion behavior of FLiNaK salt on structural material and many other studies.

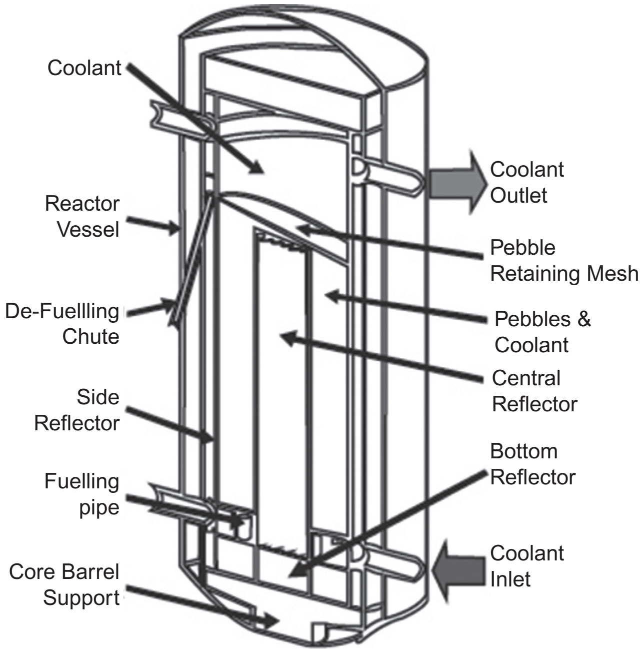

FIG. 12.4 Schematic of innovative high temperature reactor (IHTR).

The physics design envisaged to obtain maximum energy with a higher degree of safety; the target being to obtain a burnup of 900 Full Power Days (FPDs) in continuous operation of the reactor. To obtain this burnup, the value of initial k_{eff} turned out to be very high. Controlling this much high $k_{eff,}$ without any sacrifice of burnup is a challenge. The challenge has been resolved by controlling the initial reactivity and by optimizing the pebble configuration with respect to packing fraction of TRISO particles in the pebble and the percentage of ^{233}U in the fuel kernel. Based on parametric studies, it has been discovered that 900 FPD burnup is possible if packing fraction is 8.6% for which ^{233}U enrichment is 7.3%. In this case, the diameter of the pebble is 10 cm and the diameter of inner zone is 9 cm. The reactivity control is being studied by applying options such as by using dummy balls or thorium balls and maintaining fuel temperature reactivity coefficient negative.

12.2.4 Conclusions

In this part of the chapter, the features and technology of six reactors of Gen IV reactors are described briefly; especially bringing out the extent to which the four goals of Gen IV reactors are met and challenges to be resolved in these six types of reactors. Based on the experience of operating high temperature reactors in the past and recent R&D performed, it can be concluded that VHTR technology is intrinsically safe as, LOCA is ruled out and decay heat removal system is not needed. However, some other features such as fuel burnup and waste generation, and management are to be investigated further and their performance is to be demonstrated.

The HTR technology is indeed relatively better developed for early deployment. SFR technology is relatively mature and has performed well in various aspects. Better fuel utilization and reduced waste generation, fuel burnup, actinide burning, and others are positive intrinsic features of this technology. However, whether the safety of this technology is better than the current advanced reactors is to be established. Early deployment of SFRs is indeed possible. Some operating experience of civilian MSR and HMR (used in submarine) exists. For GFR and SCWR, no operating experience exists. For the four types of reactors, considerable and diverse technology development work and its successful demonstration is to be carried out.

12.3 Small and modular reactors

12.3.1 Introduction

Over the years, nuclear reactors have been developed from a size of few tens of MWe to up to 1650 MWe. The reactors are classified as small reactors with power under 300 MWe, medium reactor with power up to 700 MWe, and large size reactors over 700 MWe. The large size reactors provide economy of scale. But it requires high capital cost (higher initial investment) and long construction time (higher Interest During Construction (IDC) and late return in the investment). In the case of Small and Modular Reactors (SMRs), cost reduction is possible by modularization process and factory fabrication. These features have created a considerable interest in simple, small, medium and modular reactors [35, 36]. Initially SMR stood for small and medium size reactors but these days it is referred to as small and modular reactor. In these reactors, the reduction in the initial construction cost and construction time leads to better economic affordability. The "new comers" in the nuclear industry can afford the investment in SMR. The other benefits of SMRs include the following.

- Feasibility of providing power in remote areas away from large-size reactors and grid systems.
- Easy to transportation of the components and products to remote areas. In remote areas, electricity demand is less and variable. So SMRs can be designed for load following.
- Flexible power generation for wider ranges of power and applications.
- Some SMR designs can serve for specified objective such as burning of nuclear waste.
- SMRs can be placed in sites of aging fossil fuel units which are rarely large.

The SMRs are designed to have, as far as possible, Gen IV features [2, 3, 17]; that is, inherent safety features, low waste generation, high resistance to nuclear weapon proliferation, better fuel utilization and measures to make nuclear power more economical and environment friendly. Further, there is emphasis to develop nuclear reactor designs "without-on-site-refueling" and high burnups. This reduces/eliminates the frequency of fresh fuel manufacture and reactivity management by operation.

In Section 12.3.2, the general features of SMRs are described that are based on size, neutron energy spectrum, coolant, fuel, and design-based aspects. In Section 12.3.3, classification of SMRs and their characteristics are presented. The impact of SMRs on cost, safety, reduction in waste generation and increase in proliferation resistance are discussed in Section 12.3.4. Section 12.3.5 provides conclusions.

12.3.2 General features

The following are the characteristics of SMRs arising from "small size" and also by the "need-based design" considerations.

12.3.2.1 Size based

- In the case of small size reactors, the area to volume ratio of the core is high. Therefore in SMRs, effective control and safety required for large-size core and operator free scenario is not needed. The SMR design is therefore relatively simple. The production is possible in factories where there are almost completely controlled factory settings. This improves level of construction quality and efficiency. The modular fabrication is easily possible.
- The small reactors have lower requirement of water. These reactors have potential for implanting underground or under-water and leading to high resistance to outside threat and protection against external events.
- Compact architecture and small size enables modular small size construction, use of less number of components, better construction efficiency, series production reduces the cost, and therefore better economy.
- Unlike the requirement of emergency core cooling in conventional design, for small units, such core cooling may not be needed.
- Lower power of SMRs leads to reduction in source term and radioactive inventory in the reactor.

12.3.2.2 Design based

(i) Physical parameters

- *High temperature:* High temperature is desirable for high thermal efficiency. This leads to better fuel utilization and reduction in the waste generation. Besides this, high temperature also specifies the extent to which the energy generated can be used as process heat in the industry; that is, district heating; sea water desalination, pulp and paper manufacturing, methanol production, and petroleum refining.
- *Linear power and power density:* In certain designs, linear averaged power and power densities are reduced to a "lower level" so that low power density leads to low centerline temperature of fuel. This results in larger margin to departure from nucleate boiling (DNB). This improves safety significantly.

(ii) Compact and integrated design

In the compact and integral design, the primary circuit components such as steam generator (SG), pump, and pressurizer are housed in the reactor vessel (RV). This results in compact configuration which eliminates the need of designing a large housing of each equipment, reduction in piping and loss of coolant accident. In such designs, design problems like leakage in pumps and seal are eliminated. There is corresponding reduction in the cost and maintenance of sealing in pumps. In some designs, SGs and RV are directly connected. This improves the economics and safety of SMRs.

(iii) Reactor safety

A1. Inherent safety

- The reactor safety is achieved by increasing the resistance to accident, preventing the accident and core damage and mitigating the consequences of accident. The resistance to accident is increased by applying the best materials and best coolant chemistry and high level of quality control and assurance in design, construction and operation of the reactor. The accident prevention is achieved by using measures such as design with low excess reactivity, negative reactivity feedbacks, and use of highly reliable safety systems; that is, shutdown and decay heat removal systems and containment by active and/or passive provisions.

- The reactivity control is done during normal operation, by actively moving the control rods (CRs) and by dissolving neutron absorber in the coolant/moderator. This makes the moderator temperature reactivity coefficient more negative. Main reactivity control can be done by internally driven control rods (ICRDM). This eliminates the uncontrolled ejection of rods. The shutdown system rods drop freely under gravity or the drop is spring assisted. Curie magnet detachment of CRs from CRDM is used as a passive method of detachment. Often, it is backed by diverse system; the liquid neutron poison injection.
- The core meltdown/damage prevention is achieved by ensuring adequate cooling of the core during normal and abnormal operation and after shutdown of the reactor.

A2. Passive safety

- Passive shutdown by injection of neutron poison and passive cooling of containment.
- Passive automatic depressurization and passive core submergence after LOCA.
- Removal of decay heat by natural heat convection process.
- Maintain the integrity of the containment and prevent escape by leakage or other routes of radioactive material in the environment.
- The steel containment is designed to achieve design pressure and size is reduced by applying pressure suppression in the containment.
- The containment system is cooled and fission products are retained in the containment by passive means by submerging the containment in water pool.
- Hydrogen combustion problem is very well addressed. During normal operation, nitrogen is inserted in the Primary Containment Vessel (PCV) so that early hydrogen combustion is eliminated in PCV. For long-term elimination of hydrogen combustion, hydrogen re-combiners are planted in the PCV at appropriate locations.
- On hydrogen release in the containment, hydrogen control system monitors the hydrogen and ensures that hydrogen is not accumulated more than specified amount (10% under dry conditions for an oxidation of 100% active fuel cladding).
- For steam line breaks, the heat released gets transferred to the containment and then to the water pool. The provision made is such that the water inventory can last for seven days or more after the accident. This happens without any assistance of the operator or any external means. This reduces the inside pressure and so the leakage of radioactivity to the environment is reduced. Further, the fission products are retained in the containment of steel wall by passive means such as the gravitational deposition, thermophoresis and condensation.

- The risk of outside attack and external natural event is addressed by implanting the containment below the ground level. This makes the containment a difficult target for aircraft attack/crash by the terrorists/enemies. The need of offsite emergency response is eliminated.

(iv) Fuel cycle

Reactors are designed to have long fuel residence time in the reactor. It is often achieved by in situ burning of the neutron poison in thermal reactors and breeding the fissile material in fast reactors. This results in better fuel utilization and so reduction in waste generation.

12.3.3 Classification of SMRs

Most of the SMRs follow the principal technologies that have been developed in the past such as LWR, BWR, PHWR, FNR (SFR, LFR, and LBER), GFR, and MSR. These adopt advanced technologies in the form of single or multi-module system plants.

The SMR design is optimized on several factors that include requirements of the reactor size, geometry, power, fuel type and its enrichment, refueling time interval, spent fuel composition, site location and technology development in the country interested in nuclear power for peaceful applications. Besides, it is not straight forward to evaluate the design of different families of SMRs against the design criteria connected with cost, safety, waste generation, resistance to proliferation, and others. Therefore it is hard to design the SMRs with all the desired criteria. Considerable amount of efforts have been made across the world to design the SMR_S with opposing objectives and diverse goals.

12.3.3.1 Temperature and power range

The reactors classified as per their temperature range and for nonelectricity use, are presented in Table 12.2 [10]. This shows that as the outlet temperature of SMRs increases, more nonelectricity applications of SMRs become possible. The SMRs classified based on power range are given in Table 12.3 [35–37]. The types and the country of the reactor is also mentioned. The list is not comprehensive but it provides a good flavor of the SMRs that are operating, under construction and in design stages.

In the following section, the details of only some SMRs are explained. The details of all other SMRs are available in detail in Refs. [35–37].

12.3.3.2 LWR/PHWR water reactors

Both the types of technologies, that is, LWRs and PHWRs are well established and the plants are operating for several decades. Therefore the technology risk is minimum, the regulatory clearance of the smaller version would be easy and so the deployment of these reactors would be easy and faster. The LWR/PHWR advanced technologies developed for Gen III/III+ reactors are also well established. So the application of advanced technologies is possible with lesser R&D efforts in this case. Some of the additional aspects addressed in some of these reactor designs are the following.

- The design and technology is standardized, licensing is easier, capital cost is reduced and there are less chances of facing operational upsets.

TABLE 12.2 Coolant outlet temperature and nonelectricity applications of SMRs.

S. no.	Temperature range (°C)	SMRs	Nonelectricity application
1	<380	PWR, HMR, SFR, MSR, GFR, HTR	• District heating • Water desalination • Pulp and paper manufacture • Methanol production • Petroleum refining
2	380–600	VHTR, HMR, SFR, MSR, GFR	• Heavy oil desulfurization • Methane reforming • Hydrogen production
3	600–1000	VHTR, MSR GFR	• Methane reforming • Hydrogen production
4	800–900	VHTR and GFR	• Coal gasification • Thermo-chemical hydrogen production
5	800–1000	VHTR and GFR	• Coal gasification • Thermo-chemical hydrogen production • Blast furnace steel making
6	900–1200	VHTR	• Blast furnace steel making

- The availability of reactors is high and operational life time is large (approximately of the order of 60 years).
- The core damage frequency (CDF) is about 1×10^{-05} and the large release frequency is one-tenth of the CDF. These are within the limits specified by regulatory bodies.
- There is strong reinforcement against air craft impact.
- Higher fuel burnup leads to reduction in waste generation.
- Inherent and passive safety designs bring reduction in mechanical, electrical, and other components.

There is considerable support to SMRs-based LWRs/BWRs/PHWRs from countries such as USA, UK, Russian Federation, China, South Korea, India, and others. The interest is to develop land-based and marine-based SMRs. Currently, among land and Marine-based LWRs/BWRs/PHWRs, three (CNP-300, PHWR-220, and EGP-6) are in operation, four (KLT-40S, ACPR 50S, RITM-200, and CAREM-25) are under construction and others are in different stages (advanced and conceptual) of design. Table 12.3 provides list of SMRs of different types of reactors in different ranges of power. The list is indeed not comprehensive and some of the designs might have been shelved.

In most of the SMRs, integral design is followed. As mentioned earlier, in the integral design, the primary circuit components are housed in the reactor vessel (RV). This feature in SMRs is a major change in the design of current water reactors. The features of SMRs that

TABLE 12.3 The SMR units in different power ranges.

S. no.	Power range (MWe)	SMRs
1	0–50	PWR: CAREM (Argentina) UC, PWR: RITM (Russian Fed.) UC, PWR: KLT-40S (Russian Fed.) UC, PWR: ACP 100 (China) AD, LWGR: EGP-6 (Russian Fed.) OP, FNR (LBER): G4M (Hyperion) (USA) CD, FNR (SFR): 4S (Japan) AD, FNR (SFR): CEFR (China) CD, FNR: LFR-TL-X (Luxembourg) CD, **HTR:** HTR-10 (China) OP, **HTR:** HTTR-30 (Japan) OP, **HTR:** A-HTR (S. Africa) CD, **HTR:** HTMR-100 (S. Africa) CD, **HTR:** Xe-100 (USA) CD, **MSR:** CA Waste Burner (Denmark) CD, **MSR**: MCSFR (USA) CD, **MSR:** Moltex SSR global (UK)
2.	51–100	PWR: ACPR50S (China) UC, PWR: SMART (Rep. of Korea) AD, PWR: Nu Scale (USA) AD, PWR: ACP100 (China) AD, FNR: (LBER): SVBR (Russian Fed.) AD, FNR (SFR): ARC-100 (USA), **HTR:** MHR-100 (Russian Fed.) CD, **MSR**: CMSR (Denmark) CD, **MSR**: PB-FHR (USA), **MSR:** TMSR (China)
3.	101–150	PWR: IRIS (international), FNR: Superstar (USA) CD, **HTR:** GT-MHR (USA) CD, MSTW (Denmark), Mk1PB (USA), SmAHTR (USA)
4.	151–200	PWR: Flexblue (France) CD, PWR: SMR-160 (Canada) AD, PWR: CAP200 (China) CD, PWR: mPower (USA) AD, FNR: LFR-AS-200 (Luxembourg) CD, **HTR:** PBMR-400 (South Africa) AD, **MSR**: IMSR (Canada) AD, **MSR**: FUSI (Japan) AD
5.	201–250	PWR: W-SMR (USA) CD, **PHWR-220** (India) OP, BWR: VK-300 (Russian Fed.) AD, **HTR:** HTR-PM (China) UC, HTR: MHR-T (Russian Fed.), FNR: EM^2 (USA) CD, **MSR:** Thorcon (USA) AD, **MSR:** LFTR(USA) CD
6.	251–300	PWR: VBER-300 (Russian Fed.) AD, BWR: DMS (Japan) AD, FNR (LDR): BREST-OD-300 (Russian Fed.) AD, **HTR:** GTHTR-300 (Canada) AD, **HTR:** GT-MHR (Russian Fed.) AD, **HTR:** SC-HTGR (USA) CD, **MSR:** Moltex SSR (UK)
7.	>300	PWR: CNP (China) OP, PWR: IMR (Japan) CD, PWR: IRIS (Int. Consortium) CD, PWR: UK-SMR (UK) AD, **PHWR:** AHWR (India), **PHWR:** EC6 (Canada) CD, FNR (SFR): PRISM (USA) AD, FNR (LDR) ELSY (European Union) CD. FNR: WLFR (USA) CD

Abbreviations: AD, *advanced developed design*; CP, *conceptual design*; FNR, *fast neutron reactor;* LBER, *lead bismuth eutectic cooled reactor;* LFR, *lead-cooled reactor;* LWGR, *light water cooled graphite moderator reactor;* SFR, *sodium-cooled fast reactor;* OP, *in operation;* UC, *under construction.*

are attempted to put in these reactors due to their size and design requirements from various considerations including the safety have been explained in the previous section. In different reactors, the coolant outlet temperature is less than 350°C (623 K), the thermal to electric conversion efficiency is in the range of 25% to 35%, thermodynamic cycle is steam Rankine and circulation in the reactor pressure vessel is natural and forced.

Here some features of Indian PHWR-220 [32, 38] and AHWR [32, 39] are described. The details of other different types of reactors can be seen in the literature where these are discussed comprehensively.

A1. PHWR-220 (NPCIL, India)

In India, 16 PHWRs with power of 220 MWe and two reactors of 540 MWe are in operation and one unit of 700 MWe is in operation and several units are under construction. The PHWR-

220 are natural uranium-fueled and heavy water-cooled and moderated, reactor. The reactors are small but do not have "modular" features. The reactor has a horizontal Calandria made of SS-316 L, with 306 calandria tubes made of Zircalloy-2 and 306 pressure tubes made of Zirconium-2.5% Niobium alloy. Fuel material is UO_2 in pellet form, placed in a square lattice. Each pressure tube houses 12 fuel bundles and consists of 19 fuel elements per bundle. The total number of bundles in the core are 3672. The fuel cladding material is Zircalloy-2. The average and peak linear heat ratings are 35.3 and 50.2 kW/m, respectively. The maximum clad temperature is 574 K, maximum central line temperature is 2053 K and average fuel discharge burnup is 6700 MWd/tHM. The coolant channel inlet temperature is 522 K and outlet temperature is 566 K. The PHWR system schematic view is shown in Fig. 3.8 of Chapter 3.

The reactor shutdown system (SDS) constitutes primary SDS containing rods made of cadmium sandwiched in SS. The secondary SDS is highly diverse and is made of hollow tubes filled with lithium-penta borate neutron absorber solution. The SDS 1 is primary system to shut down the reactor on demand. In this system, on receiving the trip signal, the rod drops on de-energization of electromagnetic clutch. The rod drops under gravity though initially the rods are driven by a spring thrust. In the case of SDS 2, on demand, the high pressure neutron poison liquid gets injected in the reactor. Both the systems provide needed reactivity addition rates and total worth of reactivity needed even if one or two rods/tubes fail to operate. To provide a long-term subcriticality to the reactor, a third protection system, known as liquid poison injection system (LPIS), is provided. This caters to slow reactivity effects such as xenon decay.

The reactor regulation devices constitute regulation rods (RR) for regulating the power, absorber rods (ARs) for overriding Xenon and Shim Rods (SR) for reducing power (power setback). The material used in these rods is cadmium in aluminum tubes. Emergency core cooling system is available. For removal of heat from calandria vaults and other systems, process water system is provided. Demineralized light water is used in plate type heat exchanger.

The performance of these reactors has been ranked among the best in the world. The capacity factor being 65% in 1995–96 increased to 90% in 2002–03. The safety measures are diverse and robust. The safety systems, that is, SDS, DHRS, and containment have latest safety features that are followed in the industry. These SMRs are economically viable and environmentally benign source of energy.

A2. AHWR-300 (BARC, India)

The Indian advanced heavy water reactor (AHWR) is a 304 MWe (920 MWt) power, heavy water–moderated, vertical tube type light water cooled, pressure tube type reactor. In AHWR, the advantage is taken from the technology of PHWRs, especially that of pressure tube and low temperature moderator. The design goal of the reactor is large scale utilization of thorium. The reactor uses many passive safety features. Some of these features would reduce the capital cost and operating cost and are close to the goals of Gen IV reactors.

The reactor, besides producing electricity will also be used for desalination of sea water. The quality of water can be adjusted by adjusting the power production. The reactor design provides the flexibility to operate the reactor in "base load" and "load-following" mode. There is also flexibility in fuel utilization. The fuel of the reactor can be (Pu-Th) MOX and (Th-^{233}U) MOX or (Th-LEU) MOX. Two variants of AHWR have been developed. One design uses plutonium as external fissile feed and other uses LEU. The plutonium version works with Th-^{233}U fuel in a closed cycle. Another concept using Th-LEU fuel was developed in line with the Gen IV features where the fuel is operated in an open fuel cycle. The aim of the fuel cycle is that in the equilibrium cycle operation, the power from Th-^{233}U is maximum.

The reactor has several advanced safety features such as negative coolant void coefficient of reactivity, heat removal through natural circulation. Some of the other safety features of the reactor are the normal and decay heat removal during normal operation and hot shutdown is by natural circulation. In case of a LOCA, the reactor cavity is flooded with water. The containment cooling is also passive. The reactor is designed with a double containment and moderator is used as heat sink. The reactor also is provisioned for end shield cooling and passive decay heat removal by isolation condensers (IC) in case of Station Black Out. Inherent safety is by negative reactivity coefficient of fuel and by coolant void. Passive Poison Injection System (PPIS) provides diversity in shut down system.

The inlet/outlet coolant temperatures are 531/557 K with average steam quality of 19.7%. The system pressure is 7.0 MPa. The configuration of reactor coolant system is integrated. The coolant channels are shop assembled and the pressure tubes can be replaced without affecting any other installed coolant channel components. Design life expected is 100 years and capacity factor 90%. Power conversion process is Rankine cycle. Approach to engineering system is passive, refueling is on-power and seismic design is based on 0.20 g.

From the angle of safety objective, the core damage frequency of AHWR is lower in comparison with the goal of present day reactors design. Large radioactivity release frequency is set at 10^{-07} per reactor year. Simple design, extensive use of passive safety features, elimination of primary and circulation pumps, use of light water as coolant, higher burnup, and others would help to achieve high gain in the economics of the reactor.

12.3.3.3 Fast neutron spectrum reactor

These reactors are classified in two categories; namely sodium cooled and heavy metal (lead or lead bismuth eutectic) cooled fast reactors. The details of these reactors are discussed under the title, "Gen IV" reactors, in Sections 12.2.2 and 12.2.3. SMRs based on Fast Neutron Reactors (FNRs) technology are normally small, simpler, possess large number of inherent safety features, better fuel utilization, burning of actinides and increased resistance to the proliferation. Fuels are mostly with an enrichment less than 20% and can be (U-Pu)O_2, Uranium Nitride (U-N), Uranium transuranic nitride (U-Pu)N, Uranium Zirconium (U-Zr), or (U-Pu)Zr.

FNR-based SMRs design includes all or some of the features such as factory production, installed below the ground level, integrated design, reduction in large number of components, inherently safe, high passive cooling potential, the disposal of spent fuel is with the entire unit, and there is no separation of spent fuel or fuel removed for reprocessing.

Co-generation of industrial heat is possible and can be applied to district heating, water desalination, pulp and paper manufacture, methanol production, petroleum refining, and heavy oil desulfurization. The SMRs based on FNR and that are under development and deployment are that include sodium-cooled (PRISM of 311 MWe and ARC-100 of 100 MWe in USA), Lead-cooled (BREST, 300 MWe, Russian Federation), and LBE-cooled (SAVBR-100, 100 MWe, Russian Federation and EM^2, 240 MWe, USA) (see Table 12.3). The coolant outlet temperature is in the range of 753–823 K, Thermal energy efficiency is mostly more than 40%. Coolant pressure is near the ambient. Thermodynamic cycle is Steam Rankine [40].

A brief description is provided on sodium-cooled PRISM reactor, LBE-cooled SVBR-100, and Lead-cooled BREST reactor.

A1. GE-Hitachi (PRISM)

PRISM (power reactor innovative small module) [36]: This is being developed by GE and funded by US Department of Energy. It is compact, modular and pool type. The pool type module is below the ground level. It has two power block, each of 311 MWe (840 MWt) and one turbo generator (TG). The sodium temperature is about 500°C. Intermediate heat exchanger (IHX) takes the heat to SGs. The fuel (Pu and depleted U) is obtained from used fuel of LWRs. All transuranic elements are removed and MA remain with the plutonium and uranium. Reprocessing is done using pyrochemical process. Commercial scale plant would use three power blocks (six reactor modules). The design is simplified with passive safety features and using modular construction techniques.

A2. SVBR – 100

This is a small, multipurpose, fast, lead-bismuth (LBE) cooled, 280 MWt (100 MWe) power reactor being developed in Russia. The technology of SVBR is based on 80 reactor years of operational experience of LBE-cooled nuclear submarines. By and large, the design follows the aims of Gen IV reactors. Coolant inlet/outlet temperatures are 613/758 K. The operating pressure is small (0.1 MPa). Energy conversion efficiency is about 36%. The nuclear heat removal in primary coolant circuit is by forced circulation. Reactor coolant system is integral type where all the components, that is, reactor, SG modules, main circulating pumps are housed in the same vessel. Power conversion process is indirect Rankine cycle. Reactor fuel is UO_2 with enrichment 16.1% and MOX fuel can also be used. It follows closed fuel cycle. Fuel cycle is 8 years. Enrichment is below 20% and therefore the resistance to proliferation increases. Reactivity control is by control rod drive mechanism. Approach to engineering safety system is passive. Overall approach is to follow enhanced safety features prescribed for SMRs.

A3. BREST-OD-300

This is a Russian reactor. It is Lead (54% ^{208}Pb in naturally occurring Lead) cooled fast reactor with 300 MWe (700 MWt) power and is for experimental and demonstration purpose. The basic goal is to develop alternative fast reactor technology with Lead as a coolant. The fuel is plutonium uranium mono nitride (PuN–UN) with enrichment of 13.5%. The fuel burnup is 61.45 GWd/ton and fuel cycle is 60–70 months with partial refueling being one year. BR is about 1.05 and co-generation capability is provided in the design. The coolant inlet/outlet temperatures are 693/813 K. The system pressure is low and heat removal in primary coolant circuit is by forced circulation. The design follows integrated configuration. The core sits in a pool of lead at near atmospheric pressure. The reactor is inherently safe. As there is no blanket surrounding the core, no weapon grade plutonium is produced and fuel can be recycled indefinitely. Power conversion process is indirect Rankine cycle. The plant uses two circuit transport system to deliver heat to a supercritical steam turbine (811 K) and generate electricity.

12.3.3.4 High temperature reactors

The high temperature gas-cooled reactors are discussed in detail in Section 12.2.1.1. Mostly the HTRs are with all features of SMRs and Gen IV reactors. The HTTR-30 of Japan and HTR-10 of China are in operation and HTR-PM of China is under construction. Other reactors in different countries are in design stage (refer Table 12.3). A brief discussion on HTR-PM is given below.

HTR-PM (Tsinghua University, China)

HTR-PM is a pebble bed modular HTR and is under construction. The reactor power is 210 MWe (2×250 MWt), cooled by helium and uses graphite as moderator. The feedback from the operation of HTR-10, a pebble bed reactor that reached full power in 2003, has been utilized in designing HTR-PM. The reactor is modular. The core and the steam generator are housed in two steel vessels. The coolant enters the vessel from the bottom, flows up through the channels in the reflector to the upper reflector and then flows down through the pebbles. The coolant inlet/outlet temperatures are 523 and 1023 K, steam pressure/temperature are 13.25 MPa/840 K. The inner graphite reflector and outer carbon brick layers of the core are surrounded by the ceramic structure. The whole ceramic structure is inside a metallic core barrel which is supported by RPV.

There are two independent shutdown systems in the reactor for reactivity control. The first system is a control rod system and the second one is a Small Absorber Sphere (SAS) system. Both systems are located in radial reflector. The number of rods in the first SDS are 24 and in second SDS, there are six SAS. The shutdown rods are used for regulatory purpose and hot shutdown. SAS is used for long-term shutdown. The fuel burnup is 90 GWd/t and refueling is online. The fundamental safety functions are made effective through several inherent and passive safety factors explained in the previous section. The system designs are balanced.

The technical goals of HTR-PM are demonstration of proven technology, economics competitiveness, inherent safety and standardization and modularization.

12.3.3.5 Molten salt reactors

The MSR are explained in detail in Section 12.2. Some of the SMRs that are under design in this category include Integral MSR (192 MWe, Canada), PB-FHR (USA), TMSR (100 MWe, China), Moltex SSR (300 MWe, UK), Molten SSR Global (40 MWe, UK), and Thorcon (250 MWe, USA). Here, we explain some aspects of IMSR (Integral Molten Salt Reactor).

Integral molten salt reactor (Terrestrial Energy, Canada)

Integral Molten Salt Reactor-400 (IMSR-400) is a 400 MWt (185–192 MWe), small modular and molten salt fueled reactor. The reactor vessel is completely integrated with heat exchangers, pumps and shutdown rods. This vessel, called IMSR unit, is fabricated at factory level and has high level of quality and control and there is no need to open and service the vessel at the plant site. The unit can be replaced at the end of its life. The reprocessing is done off site when the reactor is cooled and fission products have decayed. Therefore space is provided for two reactors at each plant. This allows 7 years exchange of used unit by fresh fuel unit. The features such as inert and stable properties of salt, inherent safety features of the core, passive systems, containment cooling systems and an integral reactor architecture, provide ultimate safety to the reactor. Thus there is no dependence on operator intervention, powered mechanical components, coolant injection, etc. in dealing with abnormal conditions. The expected capacity factor of the reactor is 90%, and design life is 60 years.

The fuel salt is an eutectic of UF4 and carrier fluoride salt (FLiBe). The use of thorium-based fuel and breeding has been deliberately avoided due to its technical and regulatory complications. The coolant of secondary loop is ZrF4-KF. Fuel enrichment is low and depends on type of fuel cycle. Fuel cycle is 84 months. The moderator is unclad graphite and arranged in hexagonal geometry. The reactor uses forced circulation cooling on the primary side. System pressure is less than 0.4 MPa. Core inlet/outlet temperatures are in the range of 898–933 K and 943–973 K, respectively. As the coolant temperature is high, the process heat from reactor would be used for district heating, hydrogen production, ammonia production, mineral resource extraction, and petrochemical refining. The reactor is designed to meet different load needs of the users from base load to load following. Main reactivity control mechanism is by inherent negative temperature reactivity coefficients for short time reactivity compensation and long term is by addition of liquid fuel online.

Inherent and passive safety features are many. Some of these are large heat capacity, negative temperature reactivity coefficient, flow-driven control and heat removal by natural means. Residual heat removal is based on large heat capacity of salt and graphite and heat loss through guard vessel and surrounding air. Power conversion process is by supercritical steam turbine and two intermediate salt loops. The suitability of Chalk River as site of IMSR is being evaluated. The total levelized cost is expected to be competitive to natural gas. The reactor site would be off grid remote location.

12.3.4 The impact of SMRs

The impact of SMRs, as reported in some of the studies, on cost, safety, reduction in waste and proliferation resistance, is as follows [41, 42].

Cost: The cost of one reactor of 1000 MWe is less than the cost of four 250 MWe reactors. However, to a certain degree it gets compensated by economy of mass production. The effect of lower fuel burnup, use of special materials for fuel and special manufacturing techniques of fuel needs to be investigated.

Safety: The safety measures implemented in SMRs increase the safety of these reactors compared with present day reactors. Small power with smaller amount of fissile material benefits the safety. The compact architecture enables modularity concept application which leads to implementation of higher quality standards. Smaller power leads to smaller radioactive inventory in the reactor. This results in improvement of radiological safety as reactor module can be removed or in situ decommissioning at the end of the lifetime can be executed. The construction of the reactor in underground location provides better protection from natural (e.g., seismic or tsunami) or man-made (e.g., aircraft impact) hazards. However, in reactors such as FRs, besides many good safety factors, the potential of core disruptive accident is there due to increase in reactivity if core collapses. The practical measures, to remove above concerns, are under development and explained in Section 12.2 on Gen IV reactors.

Rad waste: The longer life of SMR core and high burnup results in lower waste. The lower amount of waste generation is an essential requirement of advanced reactors. However, from the angle of waste disposal and siting the repository, the level of resistance in SMRs to the public may not decrease. The level of resistance is to the radwaste and not to the amount of rad waste to be disposed in the repository.

Nuclear proliferation resistance: In nuclear industry, there is a grave concern for nuclear proliferation. The SMR design addresses this concern. The use of LEU fuel in low quantity is less desirable for nuclear device production. The irradiated fuel containing radioactive fission products is highly radioactive and so its handling requires special measures. It has access to a few number of people and so the risk of theft or diversion of nuclear material is reduced. The fuel burnup is high and so fuel residence time is long. Long fuel residence time decreases the number of fuel loading in the life of SMRs. This increases the nuclear fuel proliferation resistance.

In some cases, increasing the proliferation resistance imposes contradictory requirements. For example, decreasing the fuel cycle frequency demands increase in the enrichment of the fuel. This increases the risk of the diversion of the fuel from elsewhere. Further, the processing of spent fuel provides easy access to the plutonium transmuted in iPWRs or fast reactors and therefore the enhanced risk is to be investigated in current LWRs.

Licensing challenges: For any device with new design or change in the design makes licensing a challenge. For SMRs, the issues that attract attention of regulatory bodies are load following capability, human factor engineering for multimodule plants, development of new computer codes and standards, elimination of public evacuation during accident, and lower upfront cost.

It can be mentioned that as on today, by and large, the cost of different types of SMRs is increased, waste generation and proliferation resistance are mixed and safety has increased.

12.3.5 Conclusions

In this part of the chapter, the merits and demerits of small and modular reactors are discussed. The SMRs features and their classification in terms of their temperature, power, nonelectric applications, countries, and status is presented. The features of some reactors (LWRs, PHWRs, HTRs, SFRs, HMRs, and MSRs) are described. Though many SMRs use proven technology, reduction in size, modularity and long life cores have imposed challenges in material and designing of systems for inherent safety. The general features of SMRs arising from their size and design requirements are described. In these reactors, mostly the modular design and shop fabrication practices are followed. These bring quality in fabrication of equipment and in construction. Besides this, the intention of designers is to incorporate the features of Gen IV reactors. Therefore generic emphasis is on high temperature, integral design, high burnup, inherent safety, and passive safety systems. The impact of the small size and modular concept of SMRs on the goals of Gen IV reactors is discussed. The attractive features of SMRs are ease of fabrication, ease of operation, and simplified design. Some of the operating SMRs and new designs are highlighted. However, there are some technology issues which require detailed R and D and demonstration.

12.4 Traveling wave reactors

12.4.1 Introduction

Besides the various features of Gen IV reactors, there is a new uncommon reactor concept called B&B concept [43]. This concept is different from current reactors in the way the fuel burns in these reactors. In current reactors, the fuel burns in the entire volume of the reactor. In the B&B concept, the fissile fuel is bred from the transmutation of fertile material mainly in fast reactors while burning of fissile material is in situ. The concept can be applied to thermal reactors using burning of burnable absorber (BA) approach. In this text, we would always mean to refer to fast reactors unless and until it is mentioned otherwise. The concept leads to propagation of fuel burnup region like a wave and therefore B&B reactors are classified under traveling wave reactors (TWRs) and standing wave reactors (SWRs) when in the large-standing wave, fuel is moved by its shuffling. In TWRs fuel burnup is high which results in better economics through larger fuel utilization, and consequently lesser waste generation and lesser fuel fabrication. Further the need of spent fuel reprocessing is avoided. As explained latter the concept leads to better inherent safety as the reactor is self-regulated and therefore there is reduction in systems/equipment/components. The concept is also known as Nuclear Burn-up Wave (NBW), Slow Nuclear Burning (SNB) wave, and CANDLE reactor. The CANDLE stands for Constant Axial shape of Neutron flux and nuclide densities During Life of Energy producing reactor. The CANDLE nomenclature also suits as these reactors burn like a candle.

In Section 12.4.2, the TWR genesis mechanisms of fuel burnup (BU) wave formation and wave characteristics are explained. In Section 12.4.3, the requirements for self-sustained development of fuel BU wave, in terms of necessary conditions and neutronics needed and constraints to be handled are discussed. The Section 12.4.4 describes the initial stage mathematical methodology used in studying the feasibility of fuel BU wave development and its sustainability are discussed. The influence of various parameters on the wave characteristics is also reported. In Section 12.4.5, the results of the studies carried out on different types of B&B concept-based reactors such as normal thermal reactor, high temperature reactor, CANDLE reactor, and fast reactors based on standing wave concept are reported. Section 12.4.6 provides the conclusions.

12.4.2 The fuel burnup wave

12.4.2.1 The genesis

The concept emerged owing to the limitation of the current reactors and new challenges to this technology. In current reactors, particularly LWRs and PHWRs which are expected to provide a major contribution to nuclear power in future [44, 45], the fuel utilization is low and thermal to electrical conversion efficiency is poor. The fast breeder reactors (FBRs) attracted attention as in these reactors there is breeding of fissile material (^{239}Pu and ^{233}U) from the transmutation of fertile materials (^{238}U and ^{232}Th), Breeding Ratio (BR) and fuel BU are higher; the BU being 10% in FBRs and 5% in LWRs. These reactors operate at high temperature and therefore the thermal to electrical conversion efficiency is relatively high. However, as the value of microscopic fission cross-section in fast reactors is quite low, the enrichment needed in fast reactors is high relative to thermal reactors and the neutron flux in fast reactors is two orders of magnitude higher. Thus a factor of two increases in BU is not considered significant. The spent nuclear fuel (SNF) is to be reprocessed to extract the fissile and fertile materials from this and recycled back to fast reactors. The Partitioning (separation of fuel and waste) and Transmutation (P&T) of uranium is to be managed and the waste is to be stored in deep repositories and new fuel is to be fabricated and stored. All these steps increase the cost of nuclear power and so impact economics of nuclear power generation. The fissile fuel generation add to the risk of nuclear weapon proliferation. So the fast reactor technology needs to be changed for better fuel utilization, increase the resistance to weapon proliferation and decrease in the amount of waste generation.

Regarding reactor safety, the three severe accidents (TMI, Chernobyl and Fukushima) and the discovery of a cavity of a football size in reactor vessel head of unit-1 in Davis-Besse Nuclear Power Station [46], Ohio, USA, gave credence to the fact that everything cannot be anticipated and therefore if something can go wrong against nuclear safety, it can happen. Therefore the need arose for giving a new thinking on inherent safety of the nuclear reactors. So, new innovative concepts have been proposed and are being developed in which the emphasis is placed on drastic improvement of inherent safety of reactors. Inherent safety means that any type of failures in the reactor and the accidents that can lead to reactivity initiated accidents are to be prevented/protected by passive and passively designed reactor response and not by operator intervention and/or by automatic systems powered by some mechanisms [47, 48]. Therefore the accident prevention is desirable by self-regulation that is fully automatic. This would avoid the errors and any misuse by anyone. The resistance to proliferation should be increased by increasing the fuel cycle time and waste generation is to be reduced by increasing the fuel burn-up and burning the FPs in situ. Currently normal fast reactors and accelerator-driven subcritical system (ADSS) are planned for use to burn the FPs.

12.4.2.2 The mechanism

One of the concepts that addresses all the above points are addressed in the B&B concept used in TWRs with fast neutron spectrum reactors is, nonseparation of plutonium and minor actinides (MA) from the spent fuel. In this concept the core constitutes an “ignition zone/ignitor” which is followed by “breeder zone/blanket.” The ignitor provides neutrons to the breeder zone. This consists of Enriched Uranium (EU) $<20\%$ and depleted/natural uranium or TRansUranium (TRU) from LWRs Used Nuclear Fuel (UNF). The breeder zone (BZ) consists of fertile material (natural/depleted uranium or thorium). The ignitor zone (IZ) constitution should be such that it provides criticality at the beginning with power flattening. The source of neutrons, besides being from “ignitor zone,” can be external as from the ADSS. The neutrons entered in BZ from IZ generate fissile material from fertile material and the neutron irradiation of fissile material consumes (burns) fissile material. The slow rate of fuel burning arises due to the process of breeding ($2.4/\ln 2 = 3.5$ days for uranium fuel) of the fissile material. Here, 2.4 is half-life of neptunium produced in the transmutation process of ^{238}U to ^{239}Pu. Thus the slow process of breeding of fissile material prevents runaway reactions in the reactor. This physical law, that is, the slow rate of fissile material generation, is a big constraint

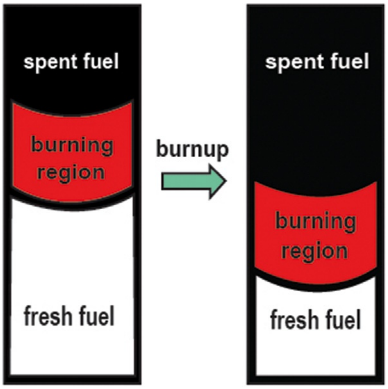

FIG. 12.5 CANDLE fuel burnup strategy.

to explosive energy release in this new concept. The B&B concept results in successive criticality. This process drives automatically a neutron flux wave phenomena (like a water wave) and a self-sustaining soliton type fuel BU wave gets established. This wave propagates in the long breeder zone and an extremely long time fuel cycle (large BU) emerges in the reactor system and the step of reprocessing of spent fuel can be eliminated. Besides neutron flux, the traveling wave develops in power density and the concentrations of associated nuclides such as fertile and fissile materials [44–48]. As an example, the CANDLE fuel BU is depicted in the Fig. 12.5 [49].

12.4.2.3 Characteristics

It is essential to specify parameters that characterize the fuel burnup waves so that the properties of the waves can be ascertained qualitatively and quantitatively. The characteristics of the waves are expressed for transient phase (build up) and for equilibrium phase (self-sustained). In equilibrium state, the burnup is characterized by burnup wave propagation speed, the full width at half maximum (FWHM) of the region under fuel burnup and the FW10M/FW1M (width at 10%or 1% of the maximum). The FW10M/FW1M parameters have been coined by a recent study by Anoop et al. [50] and Anoop and Singh [51] so that the measure of the spread of the wave in lower part of wave is also determined. The characteristics of wave development during the transient period are expressed in terms of two parameters; that is, (i) transient time (TT) and (ii) transient length (TL). The TT and TL are the time and length the fuel BU wave takes in attaining the equilibrium state. The recent studies [50, 51] focused on the detailed study of the buildup of the transient phase of fuel burnup and introduced new wave characteristic parameters, TT and TL.

12.4.3 Requirements for sustained fuel burnup wave

12.4.3.1 The necessary conditions

The necessary and sufficient conditions for buildup and propagation of fuel BU wave investigated by Feoktistov [47, 48, 52], Van Dam [53, 54], Sekimoto and his team [49, 52, 55–57], Greenspan [58], and Heidet [59] are discussed later.

A1. Feoktistov approach

The buildup of sustained fuel burnup wave occurs under certain conditions only. In the breeder and burner zone, plutonium accumulates and burns. The processes of criticality and breeding go together. The necessary condition for burnup wave development, that is, the composition of breeder zone should be such that

$$\eta = m_{pueq}/m_{pucrit} > 1 \tag{12.1}$$

where

$$m_{pueq} = \frac{\sigma_{c8}}{\sigma_{cPu} + \sigma_{fPu}} \text{ and } m_{pueq} = \left(\frac{N_{pu}}{N_8}\right)_{eq} \tag{12.2}$$

$$m_{pucrit} = \frac{\sum_i \sigma_{ci}\left(\frac{N_i}{N_{tot}}\right)}{(U_f - 1)\sigma_{fPu}} \text{ and } m_{pucrit} = \left(\frac{N_{pu}}{N_{tot}}\right)_{pucrit} \tag{12.3}$$

The Eq. (12.1) represents necessary condition for self-sustained fuel burnup wave. Here σ represents microscopic cross-section, N represents the nuclide density and subscripts c stand for capture, f for fission, Pu for plutonium and "i" for ith nuclide, 8 for ^{238}U and U_f for neutrons emitted per fission. It may be noted that the necessary condition for sustained development of wave is independent of initial concentration of plutonium, m_0. If $m_0 > m_{pueq}$, plutonium burns and tends to reach n_{pueq} and if $m_0 < m_{pueq}$ plutonium concentration accumulates and tends to reach, m_{pueq}.

The Eqs. (12.1)–(12.3) have been derived from basic neutron balance diffusion equations and nuclear fuel burnup equations in Section 12.4.4.1.

A2. The CANDLE fuel burnup

Sekimoto and his co-authors suggested a new fuel burnup strategy [49] in which, in equilibrium, spatial shape of neutron flux, nuclides densities, and nuclear power densities do not change as a function of fuel burnup in reactor axial direction and fuel burnup region moves with constant speed over the entire life of the reactor. The feasibility study to find the equilibrium state revealed that CANDLE fuel burnup strategy is possible only if the multiplication factor, k_∞, of the reactor varies with fuel burnup (fluence) as shown in Fig. 12.6. For fresh fuel, the value of k_∞ should be less than unity. At this stage, k_∞ increases due to breeding of fissile material from the transmutation of ^{238}U to ^{239}Pu and after a while, k_∞ rises sufficiently above unity. Thus the breeder zone produces more neutrons than required for sustaining a chain reaction and therefore these neutrons build up fissile material inventory in fertile material. Further the burn-up of fissile material and generation of fission products (that absorbs neutrons) increases and fissile material generation being slow (~3.5 days = the average life time of β-decay of ^{239}U to ^{239}Pu), the k_∞ value starts decreasing. The profile of k_∞ in space at a moment in time appears as shown in Fig. 12.7 and the profiles in time repeats. The quantitative values are determined by the mathematical modeling and simulation. It is worth emphasizing that the value of k_∞ for fresh fuel should be less than one otherwise the reactor would become supercritical and power would become high.

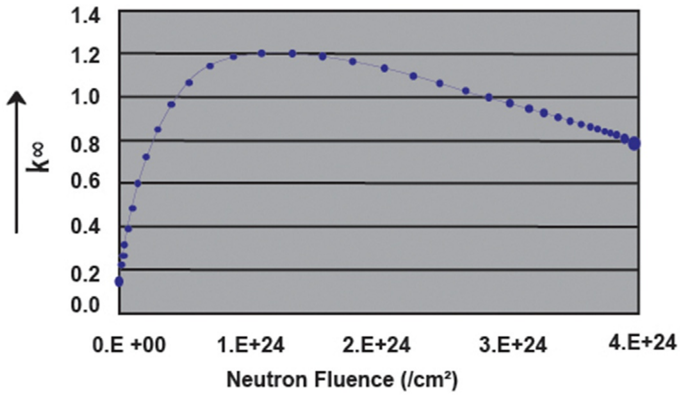

FIG. 12.6 Variation of k_∞ with neutron fluence at a given location.

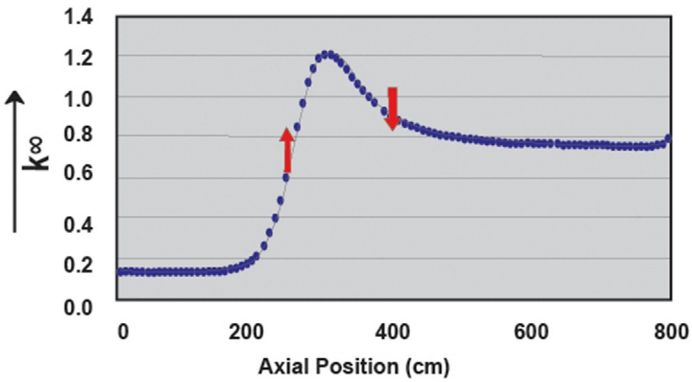

FIG. 12.7 Variation of k_∞ with axial position at a given time.

A3. The van Dam approach

van Dam [53] has expressed the condition for sustained fuel burnup wave in terms of k_0 and k_{max} where k_0 and k_{max} are initial and maximum values of k_∞ when the multiplication factor changes as a function of neutron fluence or space in the burning zone. The condition for sustained fuel burnup is

$$2k_{max} + k_0 > 3 \tag{12.4}$$

The analytical expression for neutron flux is expressed, in 1D plane geometry, as

$$\psi(x,t) = \psi(x-ut) \equiv \Phi(z) \text{ and } z = x - ut, \tag{12.5}$$

$$\Phi(z) = \Phi_m(z)\ \text{sech}^2\,(az) \tag{12.6}$$

where Φ_m is the peak flux; a is a measure of FWHM; x being the direction of wave propagation with speed u; and t is time. The mathematical expressions for these parameters are as follows:

$$a^2 = (1 - k_0)/\left(4L^2\right) \text{ and FWHM} = 1.76a, \tag{12.7}$$

$$u = \Phi_m/aF_m \text{ and } \Phi_m = (1.5 - 0.5k_0 - k_{max})/\mu \tag{12.8}$$

where L is diffusion length, F_m is the maximum fluence, and μ is feedback coefficient and represents the flux derivative of k_∞. It may be noted that the parameters; k_0, k_{max}, and μ carry great significance. The k_0 influences flux shape, FWHM amplitude and wave velocity. The k_{max} and μ influence only the flux magnitude and wave velocity. Some minimum value of k_{max} is needed for ignition of the fuel BU wave. The magnitude of k_{max} depends on initial k_∞. The numerical simulation study carried out by van Dam showed that the values calculated using analytical method and numerical methods are quite close to each other.

A4. UC Berkeley approach

For sustained operation of TWRs, it is necessary that minimum fuel burnup takes place in the breeder zone. The theoretical minimum burnup occurs when the total number of neutrons produced in the burnup (BU) process (since the beginning of life) (in FIMA) is equal to the total number of neutrons absorbed (in FIMA) [58, 59]. The FIMA is defined as fissions per initial heavy-metal atom. Thus

$$\int_0^{BU} \text{neutron production rate}\,(BU)\,d(BU) = \int_0^{BU} \text{neutron absorption rate}\,(BU)\,d(BU) \tag{12.9}$$

$$\int_0^{BU} \left[1 - \frac{1}{k_\infty\,(BU)}\right] \nu(BU) d(BU) = 0 \tag{12.10}$$

It may be noted that net number of neutrons produced per unit volume as a function of burnup (in FIMA) is the difference between,

(i) Number of fission neutrons generated per unit volume per unit burnup (in FIMA) and
(ii) Number of neutrons absorbed per unit volume per unit burnup (in FIMA).

The integration over this difference provides the left-hand side of Eq. (12.9). At the end of certain BU (minimum burnup) in the breeder zone (say, Segment 1), the total number of neutrons are available which can achieve same burnup in another zone, (say Segment 2). The Segment 2 is getting neutrons from Segment 1 and does not need external neutrons.

It is found that minimum theoretical BU required for sustained fuel BU is 20% (~200 MWd/kg HM). This applies for metallic fuel with 10% Zirconium and structural material is reconditioned at a stage the irradiation limit of the material is reached.

12.4.3.2 The neutronics

It is apparent from the earlier discussion that the type of reactor needed for the B&B concept should be the one that has high BR. This requirement is met by fast neutron spectrum reactors. Further the BR can be increased by high density of fuel, lower ratio of pitch to diameter (P/D) which results in high fuel volume fraction. From this consideration, metallic carbide or nitride with 10% Zr fuel; (Pu-U)C or (Pu-U)N or Pu-U-10Zr and metallic coolant with high thermal conductivity such as sodium, lead, and lead-bismuth-eutectic are preferred as coolant. However, from the consideration of the higher coolant temperature for better thermal efficiency and better coolant void coefficient of reactivity, Lead and LBE coolants are preferable. Fuel volume fraction needed is about 50%. However, the need of high fuel fraction leads to lower coolant fraction which results in decrease in cooling capability. To improve the cooling capability, power density should be reduced but this affects the economics of nuclear power. This concern is addressed by increasing the uranium density and flattening the power shape by shortening the core height. The shorter core height is favorable as it leads to lower core pressure and higher negative coolant void coefficient. Small core height results in increase in neutron leakage. HTRs are suitable for B&B concept. In HTRs, the coolant flows through the holes in the block. This improves the cooling even with small amount of the coolant. Thus the reactor with LBE, metallic fuel and high fuel fraction (~50%) is effective. Besides, fuel type, fuel volume fraction and uranium loading, the other considerations that affect the design are (i) core dimension which affects the neutron leakage, (ii) the fraction of the neutrons that are to be captured to reduce the reactivity for reactivity control, and (iii) structural material and its volume fraction. The maximum discharged fuel burnup is dictated by reactivity of the fuel.

12.4.3.3 The constraints

In current reactors, there are number of constraints such as the power density is to be less than 450 MWt/cm^3, the structural material can withstand maximum radiation damage of 200 dpa (displacement per atom) while expected dpa in TWR is ~400 dpa. Other constraints include the values of coolant velocity, pressure drop across the core, and temperatures of cladding and fuel [60, 61].

Reconditioning of clad: The issue of cladding material losing its characteristics under neutron irradiation is being addressed by using/developing new materials like "oxide dispersion strengthened (ODS)" steels. However, it takes time to develop a new material. Therefore the neutron irradiation problem can be addressed to a great degree by reconditioning of the fuel by its recycling (ensuring that in this process, actinides are not separated) and replacing the cladding. Some of the efforts that are going on in reconditioning the cladding are mentioned below.

DUPIC technique: This is a dry fuel handling process. In this process, the actinides and fission products are not separated and the irradiated cladding damaged up to a limit is replaced by the new one [60].

AIROX process: The gaseous fission products in the fuel leads to the swelling of the fuel that results in the development of stress and strain in the clad. Therefore removal of the gaseous fission products from the fuel is highly desirable before replacing the damaged cladding by fresh cladding. This is achieved by AIROX process which is also a dry process. In this process the irradiated UO_2 is transformed to fine powder form by oxidation process at 673 K and then back to UO_2 by reduction process (exposure of hydrogen) at 873 K several times. In this way, large number of fission products such as 100% of Kr, Xe, and I, 90% of Cs and Ru and 75.5% of Te and Cd get removed.

Melt refining process: In this process, the gaseous fission products are removed by melting the de-cladded fuel at 1573 K in zirconia crucible for several hours in argon atmosphere. The solid fission products are removed by oxidizing them in zirconia crucible. It is observed that

100% of fission products such as Br, Kr, Rb, Cd, I, Xe, and Cs and 95% of Sr, Y, Te, Ba, and rare earths (lanthanides) gets removed in this process.

12.4.4 Mathematical methodology

12.4.4.1 Mathematical approach

The mathematical method [62–67] used to study the buildup and sustainability of the equilibrium process of wave propagation, the neutron balance and nuclides burnup, time, and space-dependent partial differential equations are to be used. The equations used can be simplified in the form of neutron diffusion equations or more accurate neutron transport equations. For demonstration purpose, we will use the neutron diffusion equations for neutron balance. The reactor geometry is considered as a cylinder of finite size with azimuthal symmetry as depicted by Fig. 12.8. The total length of the cylinder extends from z_i to z_r. The left part of the cylinder over a length of d constitutes plutonium-fueled ignition zone and the rest to the right part is uranium-fueled B&B zone. Here J is the neutron current entering B&B zone. The radial neutron leakage is accounted as the cylinder is of finite size with radius R.

The generalized neutron balance and nuclides burnup equations in two-dimensional cylindrical geometry can be written as

$$\frac{\partial N}{\partial t} = vD\frac{\partial^2 N}{\partial z^2} + vD\frac{\partial^2 \mathrm{N}}{\partial \mathrm{r}^2} + vD\frac{1}{\mathrm{r}}\frac{\partial N}{\partial r} - vN\Sigma_k\sigma_{ck}N_k + \left(U_f - 1\right)(1-\beta)\nu N\sum_k \sigma_{fk}N_k + \Sigma_k\,\Sigma_j^m\lambda_{kj}C_{kj} \tag{12.11}$$

$$\frac{\partial Ckj}{\partial t} = -\lambda_{kj}C_{kj} + \beta_{kj}\nu N\sum_k \sigma_{fkj}N_k \tag{12.12}$$

$$\frac{\partial N_k}{\partial t} = \left[-v\left(\sigma_{ck} + \sigma_{fk}\right)N + \Lambda_k\right]N_k + (v\sigma_{ck}N + \Lambda_{k-1})N_{k-1}, \quad k = 1,2,\ldots,8 \tag{12.13}$$

$$\frac{\partial N_9}{\partial t} = -\Lambda_6 N_6 \tag{12.14}$$

$$\frac{\partial N_{10}}{\partial t} = \sum\nolimits_k^{1,4,5,6,7} \nu\sigma_{fk}NN_k \tag{12.15}$$

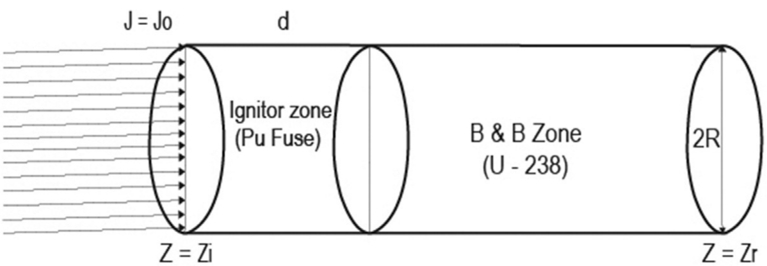

FIG. 12.8 A cylindrical system.

Here N represents neutron density. The neutron speed and diffusion coefficient of the medium are represented, respectively, by v and D. The microscopic capture and fission cross-sections are represented, respectively, by σ_c and σ_f and U_f represents the number of neutrons emitted per fission. β is fraction of delayed neutrons, m is number of delayed neutrons group, λ_{kj} and C_{kj} are decay constant and concentration of delayed neutron precursor of jth group and for nuclide number is k. The N_k represent nuclide density.

For ^{238}U to ^{239}Pu transmutation chain, the isotopes numbers for ^{238}U and ^{239}U is 1 and 2, for neptunium (^{239}Np) is 3, for plutonium isotopes (^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu) is 4, 5, 6, and 7 respectively, for Americium isotopes (^{243}Am, ^{241}Am) is 8, 9, and for all FPs is 10. k is a subscript and stand for 1, 2, 3, 4, 5, 6, 7, and 8. The summation (Σ_k) stands for nuclides represented by 1–8. Here $\Lambda_k = \log 2/T_{1/2}$ is beta decay constant and $T_{1/2}$ is half-life of the kth nuclide. $1/\Lambda_k = \tau_{\beta k}$ = mean life time of the β-decay of the kth nuclide. Here, in Eq. (12.11), the second and third terms represent radial neutron leakage. Eqs. (12.13)–(12.15) represent nuclide burnup equations. The average life time of β-decay for ^{238}U-^{239}Pu cycle is 3.4 days while it is 39 days for ^{232}Th-^{233}U cycle. The neutron balance and nuclides burnup equations can be solved numerically. The Eq. (12.11), if required can be simplified by using Transverse Buckling Approximation to the neutron leakage terms. That is, the neuron leakage terms, $vD\frac{\partial^2 N}{\partial r^2} + vD\frac{1}{r}\frac{\partial N}{\partial r}$, can be replaced by $= -B_r^2 vN$, where $B_r^2 = \left(\frac{2.405}{R}\right)^2$ is geometrical buckling and involves Bessel function for radial neutron flux. Thus the neutron balance Eq. (12.11) becomes a one-dimensional equation and can be written (without delayed neutron generation term) as

$$\frac{\partial N}{\partial t} = vD\frac{\partial^2 N}{\partial z^2} - vN\Sigma_i \sigma_{ci} N_i - B_r^2 vN + \left(U_f - 1\right) v\Sigma_k \sigma_{fk} N N_k \tag{12.16}$$

The earlier-mentioned equations can be modified by incorporating energy dependence of neutrons and the microscopic cross-sections. Presently, multigroup standard computer codes based on diffusion/transport codes or Monte Carlo codes and burnup equations prepared from standard methods and tested on realistic cases are in use.

It may be noted that partial differential equations (PDE) (refer to Eqs. 12.11–12.15) that describe the physical phenomena of neutron balance and burning of fuel are diffusive and nonlinear. The neutron diffusion term in the PDE is dispersive (Laplace term) and fuel burnup terms containing neutron flux gives rise to nonlinearity. The two terms, that is, neutron balance and nuclide burnup terms, balance each other and in such a way that the soliton (reinforcing wave) behavior of neutron flux and material concentrations take place. Further, the wave propagation velocity depends only on the amplitude of the wave which is phenomena of nonlinear wave physics. In this case, the principle of superposition of linear theory does not apply.

In the initial stages of studying the B&B concept workability and to gain insight in the concept and achieve computational economy, the earlier discussed equations have been further simplified. In this approach, in some of the studies [51, 62], only the density of four components (neutron, U^{238}, U^{239}, and Pu^{239}) (for U-Pu fuel cycle) are considered and the delayed neutrons contribution is not accounted. Also the radial neutron leakage term is dropped which is tantamount to consideration of an infinite system. Such simplifications are not suitable for design calculations but are enough to make relative qualitative and quantitative evaluations. Just for illustration purpose, this methodology is explained and results obtained from this and other improved methodologies [53, 54, 63–67] are presented.

The Eqs. (12.11)–(12.15) can be simplified and written as:

$$\frac{\partial N}{\partial t} = vD\frac{\partial^2 N}{\partial z^2} - vN\Sigma_i \sigma_{ci} N_i + \left(U_f - 1\right) v\sigma_{fPu} N N_{Pu} \tag{12.17}$$

$$\frac{\partial N_8}{\partial t} = -v\sigma_{c8} N N_8 \tag{12.18}$$

$$\frac{\partial N_9}{\partial t} = \frac{-1}{\tau_\beta} N_9 + v\sigma_{c8} N N_8 \tag{12.19}$$

$$\frac{\partial N_{Pu}}{\partial t} = \frac{1}{\tau_\beta} N_9 - v\left(\sigma_{cPu} + \sigma_{fPu}\right) N N_{Pu} \tag{12.20}$$

A1. Steady state

Considering steady state of Eqs. (12.19) and (12.20), one can write,

$$\frac{1}{\tau_\beta}(N_9)_{eq} - v\left(\sigma_{cPu} + \sigma_{fpu}\right) N \left(N_{pu}\right)_{eq} = \frac{1}{\tau_\beta}(N_9)_{pueq} - v\sigma_{c8} N (N_8)_{eq} \tag{12.21}$$

$$\left(\frac{N_{pu}}{N_8}\right)_{eq} = m_{pueq} = \frac{\sigma_{c8}}{\sigma_{cPu} + \sigma_{fPu}} \tag{12.22}$$

For critical condition,

$$\left(\frac{N_{pu}}{N_{tot}}\right)_{pucrit} = m_{pucrit} = \frac{\sum_i \sigma_{ci}\left(\frac{N_i}{N_{tot}}\right)}{\left(U_f - 1\right)\sigma_{fPu}} \tag{12.23}$$

For sustainability of fuel BU wave propagation, the necessary condition is that $\eta = m_{pueq}/m_{pucrit}$ is greater than one (see Feoktistov approach discussed earlier).

For some typical data of thermal and fast reactors given in Table 12.4, the value of m_{pueq} is 7% for fast reactors and 0.26% for thermal reactors. These values provide orders of magnitude of these parameters. For the used data, the value of m_{pueq} is more than one for a fast reactor but not for the thermal reactor. Fast spectrum reactors are a better choice for development of TWR technology.

A2. Transient state

The solution of Eqs. (12.17)–(12.20) would provide the propagation of fuel BU wave of velocity, u. For this, a self-similar variable is introduced, that is, $x = (z + ut)/\ell$, where $\ell = \sqrt{(D\tau_a)}$ is diffusion length and D is diffusion coefficient and τ_a is neutron life time due to absorption. For performing time-dependent analysis, the parameters are converted to dimensionless form. These are as follows:

TABLE 12.4 The values of m_{pueq} for thermal and fast reactors.

Parameters	Thermal (0.025 eV)	Fast (0.8–1.4 MeV)
σ_{c8} (barn)	2.71	0.13
σ_{cPu} (barn)	286	0.06
σ_{fPu} (barn)	742	1.8
$U_f\,(Pu)$	2.87	3.01
n_8	0.95	0.95
n_{pu}	0.05	0.05
m_{pueq}	0.0026	0.070
m_{pucr}	0.21	0.06
η	0.012	1.17

τ_a = neutron life time due to absorption by all nuclides $= 1/(v\Sigma_i N_i \sigma_{ai})$; n_i = nuclide atomic concentrations $= N_i/N_{tot}$; the total nuclide density, N_{tot} is constant; V = normalized velocity $= u\tau_\beta/\ell$; and n = neutron density $= vN\tau_\beta\sigma_{a8}$.

With these dimensionless parameters, the Eqs. (12.3.17)–(12.3.20), can be written as,

$$\beta\frac{\partial n(x,t)}{\partial t} = \frac{\partial^2 n(x,t)}{\partial x^2} + n(x,t)\left[\frac{n_{pu}(x,t)}{n_{pucrit}} - 1\right] + q(x,t) \tag{12.24}$$

$$\frac{\partial n_8(x,t)}{\partial t} = -n(x,t)[n_8(x,t) - n_9(x,t) - n_{Pu}(x,t)] \tag{12.25}$$

$$\frac{\partial n_9(x,t)}{dt} = n(x,t)[n_8(x,t) - n_9(x,t)] - n_9(x,t) \tag{12.26}$$

$$\frac{\partial n_{Pu}(x,t)}{\partial t} = n_9(x,t) - \frac{1}{n_{eq}} n(x,t) n_{Pu}(x,t) \tag{12.27}$$

As the constant $\beta = \tau_a/\tau_\beta$ ($\sim 0.6 \times 10^{-12)}$ in the left-hand term of Eq. (12.24) is almost zero, the time derivative term can be ignored. With this, the problem becomes quasistatic. In the right-hand side, $q(z,t)$ represents the external neutron source. As per the problem in hand, the boundary conditions can be used. In this example, the boundary conditions to be used are as follows:

(i) At the right boundary no neutron current is entering the cylinder.

$$\left\{\frac{\partial n(z,t)}{\partial z} + \frac{1}{2\gamma} n(z,t)\right\}_{z=z_r} = 0, \text{The constant}, \gamma = \ell/(v\tau_a). \tag{12.28}$$

(ii) On left boundary, a neutron current density is J which in normalized condition is $j_0 = \tau_\beta J / N\ell$.

Hence

$$\left\{ \frac{\partial n(z,t)}{\partial z} - \frac{1}{2\gamma} n(z,t) + j_0 \right\}_{z=z_l} = 0 \tag{12.29}$$

In ignition zone, over a distance d is given by,

$$n_{Pu}(z,t) = a_0 n_{critpu} \frac{\left\{ \cos\left(\frac{\pi z}{d}\right) + 1 \right\}}{2} \tag{12.30}$$

The numerical solutions of the above equations can be obtained by using standard methods [68, 69]. MATLAB programming environment can be used. The spatial mesh size and time steps can be optimized by trial and error methodology. In the following, some interesting results providing insight of the B&B concept are presented.

12.4.4.2 General results

Some very interesting and useful results have been obtained by numerical simulation studies carried out in many investigations. However, we would present here some results pertaining to the B&B concept and obtained by Pilipenko et al. [62], Fomin et al. [63–67], and van Dam [53, 54] on the verification of analytical results, determination of wave velocity and distribution of neutron flux and atomic densities of fissile and fertile materials and the neutron source effect on the fuel BU characteristics of TWRs.

The necessary condition for the establishment of self-sustaining burnup wave, that is, $\eta > 1$ ($m_{pueq} > m_{pucrit}$) is confirmed by the numerical calculations. The wave velocity calculated for different values of m_{pucrit} and fixed value of m_{pueq} (=0.1; a typical value for fast reactors) shows gradual decrease in the velocity with increasing values of n_{pucrit} (means decreasing values of η). The cut off value of η (designated as η_{cut}) below which no self-sustained wave develops is 1.58. The maximum value of η is about 2.0 and is fixed based on the acceptable high value of wave speed for practical purpose. The region, $\eta < \eta_{cut}$ is called "dead region." The spatial shape of neutron density and concentrations of materials remains practically same in all the cases of different values of η. The absolute values of neutron distribution increases rapidly with η. The external neutron source used to ignite the fuel burnup wave enables the wave propagation even in the dead region. For $\eta > \eta_{cut}$, the wave velocity is sensitive to the source strength; larger the source strength, lager is the wave velocity. The residual values of uranium and plutonium remains the same for all values of the source strength. The control of nuclear burnup is done by regulating the burning rate of fuel by an intensity variation of external neutron source.

Anoop and Singh [51], using the same data as in Ref. [62], calculated the transient length (TL) representing the length covered and transient time (TT) elapsed in establishing the equilibrium part of the wave in the range of $\eta = 1.59$; to $\eta = 2.00$. The parametric study with respect to the source strength and ignition zone width have also been carried out. Qualitatively the results are similar to the one mentioned earlier. However, quantitatively results show large variation. For example, for $\eta = 1.59$; TT = 1111 days; and for $\eta = 2.00$, TT is only 76 days. TL decreases from 124 to 49 cm when η is varied from 1.59 to 2.0.

The results of studies carried out by more accurate methodology such as using multigroup neutron balance equation with delayed neutrons and for one- dimensional plane geometry (with metallic fuel, sodium coolant, and HT9 as structural material in a fast reactor) showed that the self-sustained wave gets established over a certain period and sustains a stage close to criticality with no control. The average BU of about 40% appears to be attainable. The development of fuel BU wave is sensitive to radial neutron leakage and no wave develops if the leakage is more than a certain value. Other studies carried out on neutron leakage effect on neutron BU waves include Chen et al. [70] and Chen and Maschek [71] and Kumar and Singh [64].

The neutron energy density and neutron flux production of the above non-realistic fast reactor model is rather high. The results can be improved by suitably modifying the fuel composition and geometrical dimensions of the ignition zone of the model. The studies carried out, by and large, indicate the possibility of B&B concept being a reality in future.

Important contributions have been made by Seifritz in initiating and understanding fuel burnup wave propagation in different media. The studies were first carried out on propagation of neutron poison burnup in pure neutron absorbing media [72] and fast and thermal spectrum multiplying media [73, 74].

van Dam [53, 54] verified the validity of the condition for self-sustained development of nuclear burnup wave for symmetrical slab of 2-m half width. The external source was positioned in the middle of the slab; homogeneously over 10 cm. Calculations were done for different values of k_0 in between 0.80 and 0.95. The self-sustained flux forms in space, for different times, (from 10 to 100 days) is shown in Fig. 12.9. Once the burn-up wave is fully established, the wave propagates with constant velocity without any external intervention. The power of the reactor can be controlled by changing the value of μ, the feedback coefficient representing the flux derivative of k_∞. With a larger source strength, the reactor is slightly subcritical at criticality. This requires continuous source feeding. This situation can be avoided by switching off the source when the subcriticality is less than the power defect; the reactivity change in raising the power from zero level to full power.

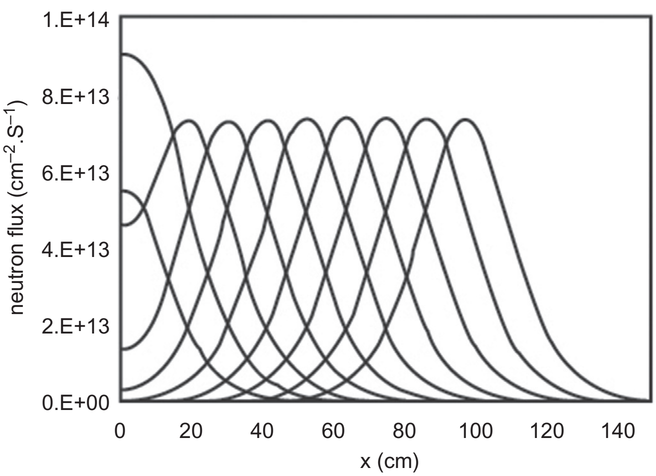

FIG. 12.9 Evolution of spatial flux wave with time (days).

The results presented on numerical simulation studies have established (i) the buildup of nuclear burning wave and their self-sustained nature, (ii) the conditions for establishment of fuel BU wave and their verification, (iii) the role and significance of external neutron source, and (iv) the significance of neutron leakage on fuel BU wave development.

12.4.5 Different types of B&B reactors

In the earlier section, the focus has been on the study of the feasibility of the B&B concept. In this section, some results of simulation studies are presented on studies carried out on different types of reactors.

12.4.5.1 Edward Teller

Edward Teller [44, 45] studies on TWR are very innovative. The B&B concept is followed independently. The basic motivation in the concept and design of the reactor includes (i) avoidance of reactor refueling and periodic outages, (ii) inaccessibility of reactor products after the reactor starts and is in operation, (iii) reduction in cost and utilization of fuel efficiently, (iv) personnel safety, (v) resistance to diversion of reactor products, and (vi) elimination of reactor accidents. Indeed, the concept and design essentially address and respond to the issues that are raised on nuclear energy.

The reactor core is cylindrical with a diameter of 3 m and length of 10 m and constitutes reactor ignitor (with enriched fuel) zone in the center, B&B region with thorium fuel, neutron reflector, thermostatic function on the reactivity control and automatic heat dumping. The reactor power is 1 GWe. The fuel burning waves move from center to left and right in the thorium fuel. The coolant is helium and the reactor is proposed to be implanted underground. The neutron source from ignition zone is needed only up to the start of the operation of the reactor after which it is a fully automated reactor. The human operator participation is needed only in the beginning and end. The regulation of the reactor core temperature is performed by uniformly distributed, functionally redundant thermo state module. The module consists of ^{6}Li (a neutron absorber) reservoir and its branches in different directions. When the material temperature rises and exceeds the design value, the module absorbs the neutrons and lead to reduction in local power. This assures thermal uniformity on every portion of the core. Each module, situated outside neutron reactor, acts reversibly.

The Monte Carlo method has been used in the reactor simulation studies, standard neutron cross-section data and computer tools were used in solving the neutron and material burnup balance equations. Multigroup cross-section sets, dozen or more isotopes, coolant, structural material and reflector, and others were accounted in the simulation studies. Important conclusion of the study is that self-sustained B&B wave is feasible with wave velocity less than 1 m/year. The expected burnup is 50% over a period of three decades. This shows that in the fuel cycle, the burning of actinides without fuel reprocessing, fuel fabrication, transport of fission products and other processes is feasible and the use of enriched fuel is in a very small quantity relative to the fuel burnt in B&B region.

At the end of cycle, the residual radioactivity is sealed in the core and allowed to decay. Though the occurrence of the accident is very unlikely but if it occurs the radioactivity will remain buried underground. The observation of the tests of nuclear devices done underground shows that the buried activity does not come out. So the earth surface contamination is prevented. Heat generated is of high grade. Thus the reactor can be used for space heating and in industrial processes.

12.4.5.2 HTGR TWR

Hugo van Dam [53, 54], to explore the feasibility of igniting criticality wave in a thermal reactor, studied criticality waves in a pebble bed HTGR; the fuel pebbles containing uranium with 19.75% enrichment. The pebble bed was considered as filled with fuel pebbles containing homogeneously distributed natural boron as Burnable Poison (BP). The radial neutron leakage into and from core were taken into account. The waves were ignited with a central critical core (ignition zone) with 3 m axial width filled with un-poisoned fuel elements. The waves got ignited in two opposite direction over a total axial length 20 m in one direction. The focus of the study was to study the properties of the asymptotic waves outside the igniting core.

The study has been carried out using standard computer code and two energy group, one dimension diffusion theory. All the isotopes of uranium and plutonium and all fission products (the capture cross-section being sum of all the FPs) were considered. The temperature reactivity coefficient was calculated by performing k_∞ calculations at 573 and 1173 K. Xenon effect was taken into account. The results obtained are: wave velocity = ~0.5 m/year, the specific power (per unit of radial cross-section) = 8.8 MW/m^2, maximum BU is 106 MWd/kg U and FWHM = 4.7 m. The sensitivity of the results to BP was checked. It was observed that increasing the natural boron results in reduction of k_{eff}, specific power and usual power density of HTGRs. The burn-up wave characteristics can also be influenced by "tailoring" burnup function by changing the BP or the shape of BP by selecting heterogeneous medium. The calculations done by artificially doubling the absorption cross-section of boron lead to faster burnup of BP and to a steeper initial increase of the burnup function. As a result, ignition can be realized with lower k_{eff}. The FWHM gets reduced by about 23% and the wave velocity increases by about a factor of two. The burn-up also increases more than a factor of two.

By and large the conclusion was that a stable and controlled fuel burnup wave in a thermal HTGR can be ignited by proper design of BP inventory of the fuel elements.

12.4.5.3 The CANDLE reactor

The strategy of fuel burnup of a CANDLE reactor [55–57, 75–77] can be applied to any reactor when the conditions of the type explained through Figs. 12.6 and 12.7 are applicable. However, as the condition demands excellent neutron economy, the reactor considered is a fast reactor, lead bismuth eutectic (LBE) cooled, metallic fuel [90% U-10% Zr (in weight)] with volume fraction of 50%. The results are discussed for a large size (3000 MWt) cylindrical reactor with cladding material HT-9. The core radius is 2 m and radial reflector thickness is 0.5 m.

As the speed of fuel burning region is small (~4 cm/year) for the present design of CANDLE reactor, the size of the reactor can be set small for a long-life reactor. However, in small reactors, the neutron leakage is relatively high. This creates difficulty in achieving criticality, leads to lower BR and neutron population is relatively small.

In the mathematical simulation, three-dimensional multigroup neutron diffusion and nuclides balance partial differential equations are used. In the calculations, delayed neutrons, actinides, and fission products are accounted. The JENDL-3.2 nuclear data library is used. In the equations, the Galilean transformation is used and therefore the coordinate system moves along with fuel burnup region.

The results are $k_{eff} = 1.012$, burning region speed = 4.40 cm/year. The average BU of spent fuel is 38.2% and half width of axial power shape is 150. The computation results showed that while it is easy to satisfy the condition of $k_{eff} < 1$ initially but it is a challenge to raise the $k_{eff} > 1$ for the entire system.

Many other studies [75–77] have been conducted by Sekimoto and his team. The important conclusion is that it is feasible to achieve CANDLE fuel burnup strategy for large- and small-size reactors and with metallic fuel and any type of coolant (sodium, helium, lead, and LBE). As the fuel is rarely critical and redistribution of fuel usually reduces its criticality performance, the reactor is free from core disruptive accident. All these features make the reactor safe and reliable and operation and maintenance become simple. All the above factors also lead to significant improvement in the economics. The fuel enrichment is needed only in ignition zone of initial core. In burnup zone, only fertile material is used in initial and succeeding cores. So enriched fuel requirement is very limited in CANDLE reactor.

At the end of the life of the reactor, the burning zone is suitable for serving as "ignition zone" in CANDLE and therefore can be used as ignition zone for the succeeding core (stacked core scheme). So no enrichment of fuel or reprocessing of spent fuel is required. The CANDLE characteristics do not change with burnup. Therefore certain physics parameters of the reactor such as power peaking and reactivity coefficients remain constant with burnup. This shows that reactor operation strategy remains same as a function of fuel burnup stage.

The studies carried out on HTGR technology showed that CANDLE burnup strategy can be applied, without much technology development. Among HTGRs, the reactor with "block type fuel" is better suited as no drastic change is necessary for this case. For CANDLE reactor, one block of spent fuel is removed and one block of fresh fuel is loaded. The fuel burnup achieved is small but it is much more than conventional HTGRs. The burnup can be improved if burning poison is mixed in graphite and not in the kernel of TRISO fuel particle. In high burnup, the fraction of plutonium in waste is reduced substantially.

12.4.5.4 The TerraPower standing wave reactor

The TerraPower [78–80] is a US-based company largely funded by the Microsoft Founder; Bill Gates. The TWR concept was patented by "Intellectual ventures" and from this spun out the "TerraPower" company. The first practical design efforts on the development of TWR were initiated by the company in 2006 and the design basis was chosen on the technology of sodium-cooled fast reactor.

In TWR design, two things were observed. First, coolant temperature becomes very high on power peaking in the core. This can be handled by reducing the power density of the reactor. But this leads to the increase in the cost of reactor power production. Second, the B&B zone is too long, and this leads to high pressure drop and resulting in high pumping power. Therefore the concept of standing burnup wave emerged. The standing waves can be achieved by shuffling of fuel assemblies periodically from and in the B&B zone. The fuel shuffling leads to controlling the spatial distribution of power and fuel burnup and maintain the core material within operating safe limits. This also supports, along with control rods, managing the excess reactivity and increasing the life of the reactor core.

In 2009 the designers decided to design a reactor, namely the "TerraPower-1 (TP-1)" of 1200 MWt and 500 MWe power and using the standing wave concept [79, 80]. The reactor core is approximately cylindrical; the fuel assemblies (fissile and fertile) being hexagonal. The fissile assemblies fuel is enriched uranium metallic alloy and fertile assemblies constitutes depleted uranium. The cladding material is ferritic martensitic steel. The reactor core constitutes two zones; the first zone (core) contains fissile assemblies and produces initial criticality and power. The second zone (Blanket) contains fertile assemblies and breeds plutonium from transmutation of uranium. The initial core is provided with some excess reactivity to raise the reactor power to full operational level after startup. Excess reactivity slowly increases owing to breeding of fissile material and is controlled by control rods. The fuel shuffling is done when certain number of specified assemblies attain specified burnup. In the shuffling process, high BU assemblies are moved to the blanket near the core periphery. The moved assemblies are replaced by depleted uranium assemblies.

The studies performed on TWRs and the present design show that by the shuffling process, high burnup of 40% can be achieved by recladding of the fuel when fuel radiation damage limit is reached. The first generation TWRs design discharge their fuel at an average BU of about 15%; the initial heavy metal atoms with axial peaking BU being 28%–30%. The k-infinity remains critical to over 40% average BU, even without any FPs removal by melt-refining process; the BU crosses 50% if the FPs are removed by the refining process. This shows that sufficient potential fuel is available in the discharged fuel which can be used in recladding.

The TP-1 uses the proven technology. Some exceptions are as follows: (i) the pins are designed to ventilate the fission gases in a controlled manner so that deep BU required sustains for 40 years and pin cladding failure probability is greatly reduced. (ii) A robust safety barrier is provided in IHX between the primary and secondary sodium. (iii) The decay heat removal system is backed by natural convection-based passive system. Thus decay heat removal is ensured even under station blackout condition. (iv) A probabilistic safety assessment is applied to assess the probability sequences.

12.4.5.5 UC Berkeley SWR

UC Berkeley studies on the B&B concept [58, 59] are quite significant. The guideline has been to study the concept by remaining close to the conventional fast reactor design and assess the parameters pertaining to fuel burnup wave such as minimum BU needed for sustained fuel burnup, and fuel utilization and their values with respect to LWRs and the conventional fast reactors. Indeed, the high fuel utilization is to be achieved by reconditioning of the fuel and changing the cladding when the radiation damage limit of the cladding is reached. The standing wave feature of B&B concept is applied in increasing the burnup and the fuel utilization. A most efficient fuel shuffling scheme is followed.

The studies have been carried out for large size/small size reactors; the reactor power being 3000/1200 MWt, ternary fuel with 6% (wt) zirconium, sodium coolant, and HT9 cladding. The fuel density is same for both types of reactors. The volume fraction of fuel and gap, HT9 and coolant are taken as 50/45.5%, 22/26.6%, and 28/28.0%, respectively, for large-sized/small-sized reactors. The reactor core is divided in various axial and radial zones. The central part contains enriched fuel with enrichment less than 20% or depleted uranium and TRU from LWR. Radial blanket contains fertile material and is followed by reflector. For small size reactor there is no radial blanket but radial reflector.

The mathematical simulation studies are carried out using standard Monte Carlo computer codes. The modeling of radial zones is done with concentric cylinders which are of uniform composition. Axial and radial zones are of the same volume so that the variation of the axial power and different burnup rates can be accounted. Fuel shuffling is done subassembly wise but the core is modeled in a coarse spatial resolution.

In the case of LWRs, calculations are done for once through fuel cycle, the core is divided in four fuel batches and fuel is taken out with maximum BU and is loaded for fresh fuel. The B&B calculations are done with eight batches for large size core and 15 batches for small size core. The calculations done for the conventional fast reactor are upper most estimates as actinides loss is not considered in fuel reconditioning process.

It is observed that the large- and small-sized B&B concept-based fast reactors can be designed and operated in standing wave fuel BU mode. The minimum fuel BU needed for sustained B&B concept of operation is 20%. This is achieved in about 13 years. At this BU, the discharge fuel is sufficient to serve as new ignitor. The maximum achieved BU with oxide, metallic, and thorium-metallic fuels, respectively, are 10%, 41%, and 38% FIMA (fissions per initial heavy-metal atom). The value of k remains close to 1.2 over BU in the range of 12%–22%, and k profile over fuel burnup remains above one in between 6% and 45% FIMA. The number of extra neutrons in discharged fuel are of the order of 7.5×10^{22} per cm^3 which are enough to start a new reactor.

The fuel (uranium) utilization in B&B concept reactor with respect to LWRs (per unit thermal power generation) is 93 times for large size reactor and 71 times for small size reactors. For a conventional large size fast reactor, the fuel burnup in B&B concept reactor is smaller but fuel utilization in B&B is 12 times more than a conventional fast reactor. The burnup in B&B is smaller because the conventional reactor is loaded with enriched fuel and B&B reactor is loaded with natural/depleted uranium fuel.

The waste generated for B&B at BU of 55% FIMA with respect to LWR discharged fuel, per unit electricity generation and after one year of the discharge is: TRU mass and decay heat are 40% and 12%, respectively, ^{237}Np inventory and its precursors is 10%, radiotoxicity is 28% and neutron emission rate is 7%. In the discharged plutonium, the fraction of fissile isotopes is comparable with LWR case. However, the neutron emission rate and the decay heat per unit mass of dischargeable plutonium are nearly half.

12.4.6 Conclusions

Traveling wave reactor concept is based on *B&B* concept and arises from the need of improving the (i) economics of nuclear power production by eliminating the need of fuel reprocessing by achieving high fuel burnup, (ii) inherent safety to the extent that uncontrolled supercriticality is eliminated and the reactor is self-regulated. The success in achieving these two goals results in reduction of waste generation and increase in resistance to nuclear material proliferation.

Some of the important aspects of TWRs such as necessary conditions for the ignition and self-sustainability of fuel burnup waves, mathematical modeling for design and analysis, dependence of the characteristics of burnup wave (velocity, FWHM and transient time and length in development of equilibrium wave) on external source and design are briefly discussed. The works done on different types of B&B-based TWRs and standing wave reactors (SWRs) are presented.

By and large the conclusions are as follows: the ignition and self-sustained fuel burnup conditions are established. The fuel utilization in B&B concept is several times higher as compared with LWRs and conventional fast reactors. The waste generation is low comparatively. For specified electricity generation, the TWR fuel in the "ignition zone" is enriched but in "*B&B* zone" the fuel is not enriched. The need of reprocessing of spent fuel and fabrication of fresh fuel is reduced. Thus, the overall cost is reduced.

The challenges regarding the initial core configuration availability of the structural material that can withstand high fluence (equivalent to 500 dpa) and/or by reconditioning the fuel and replacing the irradiated cladding and other issues such as large size core are understood and are possible to resolve. However, the verification and validation of the technology is a big challenge.

The high fuel utilization in TWRs provide long-term energy security.

12.5 Review questions

A. Gen IV reactors

1. What are the new features of Gen III/III$^+$ reactors relative to the Gen I and Gen II reactors?
2. What are the features that foster to innovate over Gen III/III$^+$ reactors and what new goals are specified for Gen IV reactors?
3. The challenges arising in designing the reactor with core outlet temperature of the order of 1273 K?
4. What are the common and unique features of six types of Gen IV reactors?
5. The fuel invented to withstand high temperature in HTRs and its special characteristics?
6. Explain the configurations in vogue in the design of HTRs.
7. Advantages of heavy metal-cooled fast reactors and helium gas–cooled fast reactors over sodium-cooled fast reactors?
8. Advantages and challenges in developing the molten salt reactors?
9. Challenges in the design of super critical water reactors?
10. What are the objectives of developing HTRs and MSR in India?
11. List the reactor physics design challenges in Gen IV reactors?

B. Small and modular (medium) reactors

1. List the major incentives in developing SMRs.
2. The goals to be achieved by SMR design?
3. The applications of nonelectric power generation in nuclear reactors in different types of industry?
4. Major change in the design of light and heavy water reactors to achieve design characteristics of SMRs.
5. Explain inherent safety and passive safety in the reactors.

6. How the cost, safety, waste generation, and proliferation of resistance of SMRs can be compared with large-sized reactors.
7. List the design features that are affected by the “size” and the “design” of the SMRs?

C. Traveling wave reactors

1. What is Breeder and Burn (B&B) concept?
2. The genesis of TWRs and the mechanism that establishes the TWRs?
3. The parameters that characterize the TWRs?
4. What are the necessary conditions for development of TWRs, SWRs, CANDLE, and UCB SWRs?
5. What way the TWRs are different than the SWRs?
6. What are the reactor physics requirements for sustainability of TWRs?
7. What is the strategy of fuel utilization in a CANDLE reactor?
8. What are the advantages of CANDLE reactors?
9. Major challenges in developing B&B concept based reactors?

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