Chapter 3
Components and Its Production

Abstract Materials are used for components of plants. According to its function they can differ considerably in size and complexity. The sizes range from heavy and thick-walled (e.g. pressure vessel) to wall sizes of below one millimeter (claddings and compact heat exchangers). For protection against environmental attack components can have coatings on the surface. The production of components needs semi-finished goods, welding and shaping which requires different techniques depending on kind of component. In the first part of the chapter the major components used in nuclear plants will be introduced. The second part will deal with production technologies. Melting, forging, bonding but also powder metallurgy and layered structures will be covered for metallic parts. Production methods for graphite and structural ceramics will also be briefly introduced.

3.1 Components of Nuclear Plants

Nuclear reactions used for power generation are either fission or fusion based. In contrast to traditional fission reactors which have been in operation since the 1950s, advanced reactors and fusion plants are still in its research and development phase. Structural materials for both technologies must combine good strength at elevated to high temperatures and radiation resistance. Nuclear power plants consist in priciple of a vessel which contains the reactor core, core internals, and the core support structures, piping, coolant circulation equipment, heat exchange equipment or direct cycle electricity generation plant. A list of the main components of the different types of nuclear power plants is shown in Table 3.1. Accelerator driven systems are, dependent on the cooling used, comparable with the respective fast reactor requirements. The main structural components of fusion plants are first wall,

W. Hoffelner, *Materials for Nuclear Plants*,
DOI: 10.1007/978-1-4471-2915-8_3,

Table 3.1 Main components for different nuclear fission plants. The brackets for the MSR indicate that also fast reactor concepts are considered

Component	LWR	SFR	GFR	SCWR	VHTR	LFR	MSR
Reactor pressure vessel	X		X	X	X		
Reactor vessel	X[a]	X				X	X
Fuel pins	X	X	X	X		X	
Special fuel compounds			X		X		X
Core internals	X	X	X	X	X	X	X
Graphite core					X		(X)
Core support	X	X	X	X	X	X	X
Piping	X	X	X	X	X	X	X
Coolant pumps or blowers	X	X	X	X	X	X	X
Steam generator	PWR	X	X	X	X	X	X
Intermediate heat exchanger		X	X		X	X	X
Direct cycle electricity	BWR		X	X	X		

a ...RBMK

divertor and blanket. These components represent a variety of service exposures, a variety of geometries (from huge pressure vessel forgings to claddings with a wall thicknesses of a few hundred micrometer) and a variety of production routes (from cast/forged to powder metallurgy). There are different life-time expectations for the different components. The questions of maximum useful life become increasingly important for current LWRs where life extensions from 40 years to 60 and even more years are in discussion. For generation IV reactors the anticipated design life is already 60 years. The challenge for these future plants is the missing long-time experience and also missing long-time data. In contrast to components which are changed on a regular basis (e.g. fuel pins) there are central elements which cannot easily be exchanged (e.g. the reactor pressure vessel). Sound and reliable manufacturing of components is a necessary requirement for save long-term operation. In this chapter primarily the manufacturing aspects will be dealt with.

3.1.1 Vessel

Several nuclear power plants in which the primary cooling medium must be kept under pressure (LWR, SCWR, VHTR, GFR) need a pressure vessel. The vessel is a barrier to the outside and it therefore must fulfil most stringent safety measures. The main demands for the vessel materials are:

- High strength for temperatures up to maximum operating temperature (including accident conditions)
- High fracture toughness and low ductile to brittle transition temperature (DBTT)

Table 3.2 Characteristics of a RPV for a 1300 MW PWR

Reactor pressure vessel	
Diameter of cylindrical shell	5,000 mm
Wall thickness of cylindrical shell	250 mm
Total height	12,362 mm
Weight without internals	507,000 kg
Material: ferritic steel	20 MnMoNi 55
With austenitic cladding (about 5 mm)	X6 CrNiNb 1810

- High creep rupture strength and excellent creep properties whenever thermal creep has to be considered in design (e.g. VHTR hot vessel option)
- High resistance to coolant exposure (corrosion resistance)
- Homogeneous microstructure and homogeneous mechanical properties
- High thermal stability (thermal and radiation embrittlement)
- Very good weldability and very good non-destructive inspectability

RPV's for LWRs are fabricated from low alloy carbon steels (typically A302, A533B and A508-type). Characteristic numbers for a PWR-RPV are given in Table 3.2. A cut through a pressure vessel of a pressurized water reactor is shown in Fig. 3.1. This figure does not only show the complex shape of the vessel with its flanges and penetrations. It gives also an impression from the core internals. The RPV is a welded construction fabricated from quenched and tempered, low alloy Mn–Mo–Ni steel with primarily tempered bainitic microstructure. Welding a bent plate longitudinally would be the most simple way to get a cylindric geometry. Therefore early RPV-technology was based on plates rolled into appropriate shapes and welded together into an integral vessel. One major disadvantage of such a procedure is, that the weldment is highly loaded by the internal pressure and therefore welded constructions with longitudinal weldments were replaced by welding forged rings together which needed suitable forging procedures and forging tools to be developed. The distinct advantage realized through the use of forged ring components is reduction of the welding line, as compared to plate construction type vessels. This reduction helps to significantly reduce the manufacturing cost of a RPV, as well as shorten the construction time. It also increases its reliability and reduces the time required for in-service inspection (ISI). Realization of the shell with no longitudinal welding line in the core region is quite an improvement in the safety and reliability of a RPV.

Most advanced RPV designs avoid weldments in the belt line which shows highest radiation exposure and tendency for radiation embrittlement. Figures 3.2a [1] and b [2] show a late production step of a reactor head with several penetrations and the ring forging process. The head is bolted to the vessel. It can be removed periodically (e.g. for re-fueling). Flanges and penetrations are welded

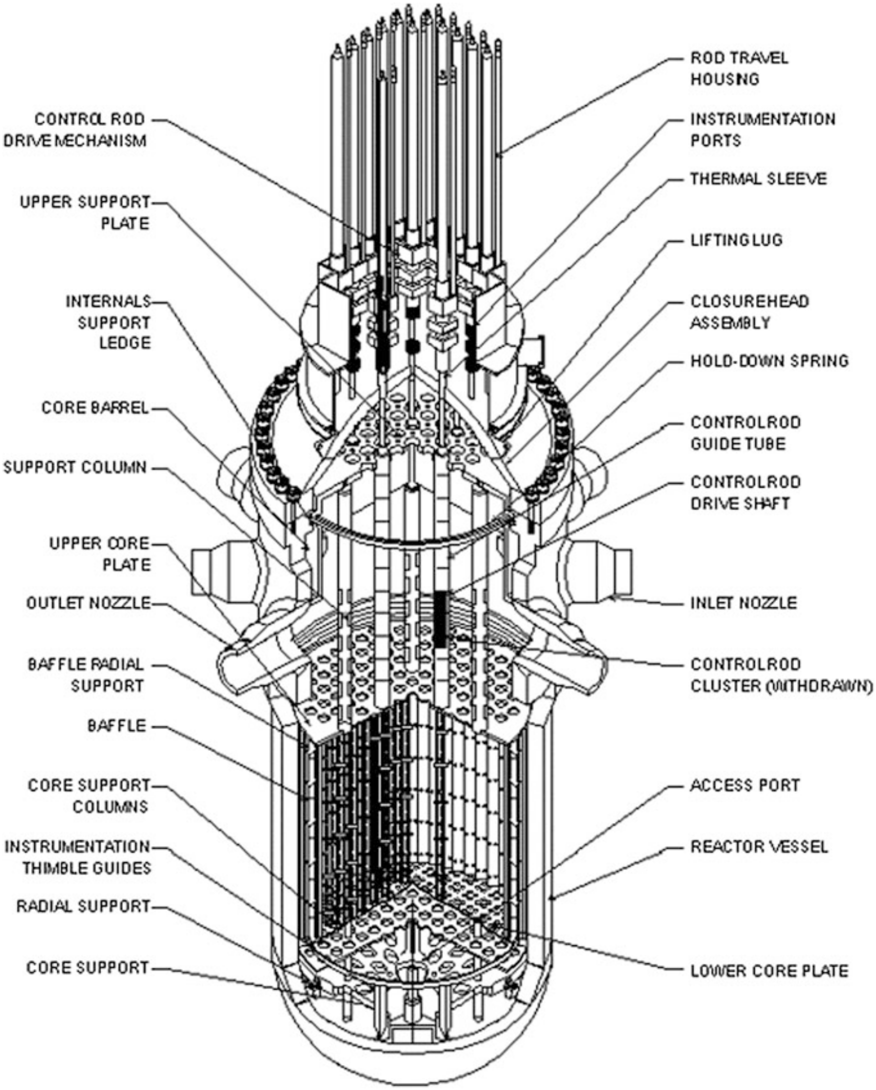


Fig. 3.1 Cut through a pressure vessel of a pressurized light water reactor (*Source* [47])

into the vessel. The inside of the vessel is clad with an approximately 10 mm thick stainless steel corrosion barrier. Weld compositions are usually not the same as the base metal and may vary significantly between different welds or even within a particular weld in the same vessel. Final heat treatments are in the range of 600–650 °C for periods of 10–50 h followed by a slow cool. The resulting microstructures vary from tempered bainite to mixtures of bainite and ferrite. Complex finer scale microstructural features include a range of matrix and

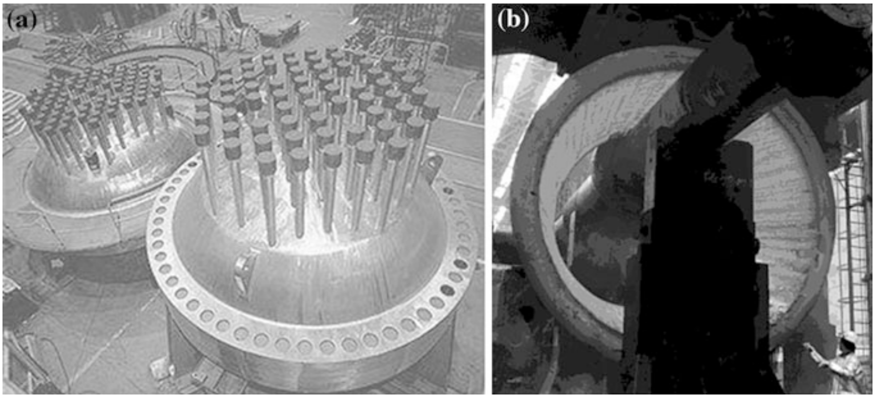


Fig. 3.2 Reactor pressure vessel head and ring forging procedure.**a** Rector Pressure vessel head (*Source* [1], US-NRC) **b** Ring forging process (*Source* [2], Doosan Heavy)

boundary carbide phases, inclusions, and dislocations. While welds are often the most sensitive material, embrittlement of base metal regions can also make a significant contribution to the RPV failure probabilities due to their greater volumes in the pressure vessel. Although LWR-vessels are already huge forgings the dimensions of RPVs for advanced nuclear plants are even larger. For the pressure vessel of a SCWR much thicker forgings would be required to accomodate the much higher internal pressure. There were even discussions going on to move to higher strength steels. Two potential materials are A508 Grade 4 N Class 1 and a developmental steel, 3Cr-3WV [3].

For both gas cooled reactors (VHTR, GFR) the basic demands remain the same. According to the proposal of the GENIV Roadmap a vessel temperature of about 600 °C was originally envisaged as the RPV material temperature. Such a temperature can not be reached with low alloy steels as shown in Chap. 2, Fig. 2.26, where the allowable stresses applicable for 300,000 h are shown as a function of temperature. As the low alloy steel has almost no creep resistance, alternatives have to be found. One possible candidate is the low alloy ferritic–bainitic steel of type 2.25Cr-1Mo and the other alternative is a ferritic–martensitic steel of type mod 9Cr 1Mo. The 2.25Cr-1Mo option was chosen for the Japanese HTTR [4].

For further advanced VHTRs the mod 9Cr 1 Mo steel with much better creep behaviour was envisaged. However, uncertainties concerning large forgings, welding procedures, tendency to cyclic softening and other expected problems did not allow considering this material as a real option today. Therefore, international research is going on to provide materials knowledge necessary for reliable safety assessments and technological improvements to promote this steel for the next generation of gas cooled reactors.

The modified 9 Cr-1Mo steel is also considered as RPV-material for the GFR. For currently planned HTRs in the US, in China or in Korea intense cooling will reduce the temperature into a regime where SA 508 (i.e. currently used low alloy

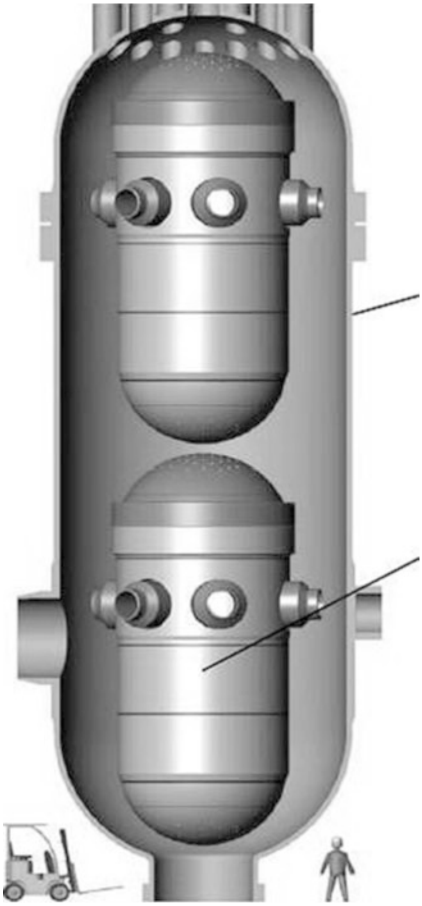


Fig. 3.3 Dimensions of a reactor pressure vessel for a gas cooled reactor compared with a PWR-vessel [48]

steel) type material can be used as a cold vessel option. However, more advanced, future concepts will operate at higher metal temperatures in the creep regime and consequently a reconsideration of 9Cr-1Mo needs to be done. One of the major problems of the HTR vessel is its huge dimension which is shown in Fig. 3.3.

For the liquid metal reactors (SFR, LMR) und for the molten salt reactor (MSR) no pressure vessels but only vessels are needed. Although they don't have to be designed for high stresses they are heavy metallic constructions (Fig. 3.4) which have to house all internals and also cooling media (Fig. 3.5). Figure 3.4 shows the installation of the main vessel of India's Prototype Fast Breeder Reactor into the safety vessel. The main vessel (stainless steel), which is 12.5 m in diameter and 12.5 m tall, will form the heart of the fast reactor, which will contain more than 1000 tons of sodium. The internals of the SFR vessel as planned for the JSFR ar shown in Fig. 3.5. Main challenge for the vessel is the long-time behaviour in the coolant environment (sodium, lead–bismuth, lead, molten salt). Preferred vessel material for the SFR are austenitic steels. In case of the LMR austenitic as well as

Fig. 3.4 Installation of the main vessel of India's prototype fast breeder reactor, PFBR (copyright 2011 Nuclear Engineering International magazine, www.neimagazine.com)

ferritic–martensitic steels are considered for temperatures up to 550 °C. Structural components for the MSR are nickelbase alloys of type Hastelloy N. The inner surface of all vessels is exposed to the cooling medium which needs precautions to prevent corrosion damage. The inside of LWR vessels is cladded with a corrosion resistant austenitc material which is applied by welding. The pressure vessels of helium cooled reactors don't have any cladding. For LMRs and for temperatures exceeding 550 °C FeAl-type coatings (sprayed) are under consideration. For the vessel of the MSR besides Hastelloy N bulk material also the iron-nickel based alloy IN-800H with a Hastelloy N or other claddings were proposed. Cladding procedures are described separately, later in this chapter.

3.1.2 Fuel Elements

Fuel elements are the central part of a nuclear reactor. Fuel claddings contain the fuel and they have to fulfill the following important requirements:

- remain gas tight against fission gases
- sustain the load built up by the fission gas
- accomodate high irradiation doses (high burnup)

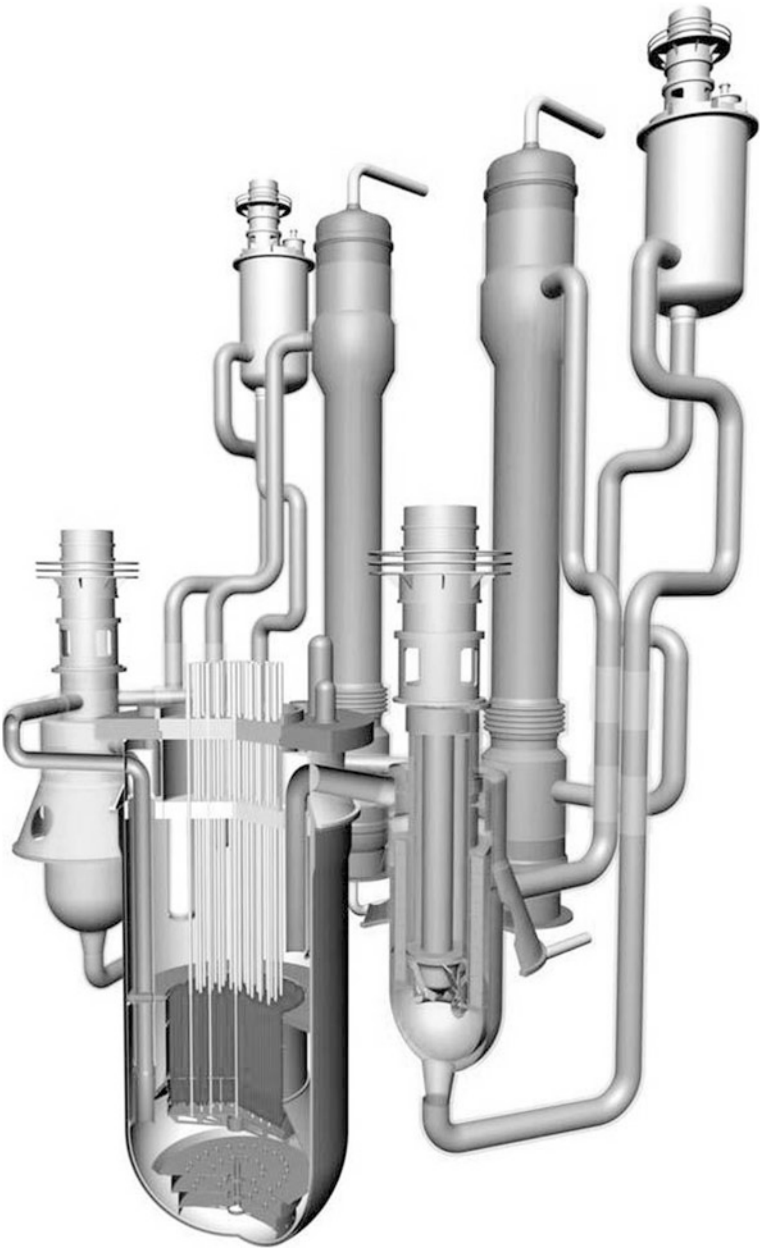


Fig. 3.5 Japanese sodium fast reactor (JSFR) reactor vessel and internal structures (*Source* [49], copyright: JAEA-Research 2006-042)

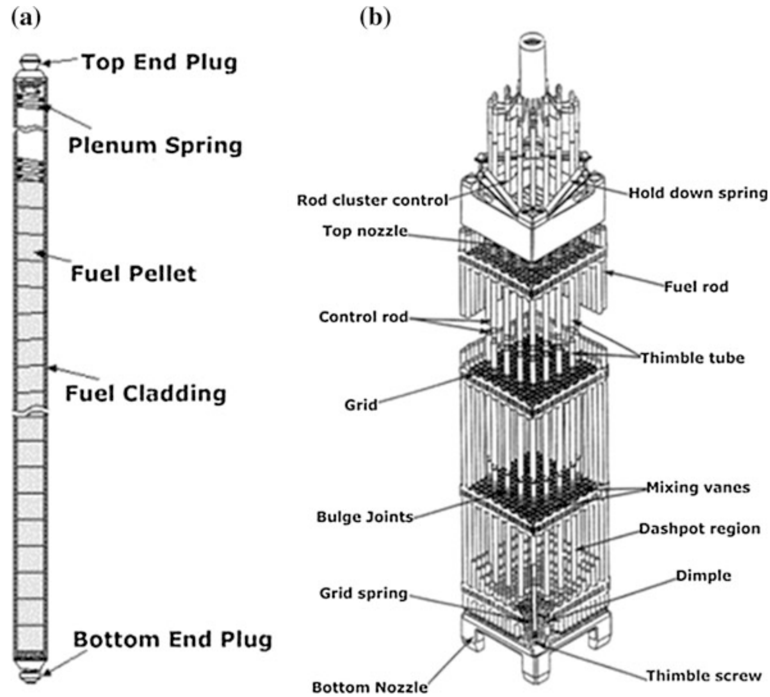


Fig. 3.6 Fuel pin and and fuel assembly of a light water reactor. **a** Fuel pin of a light water reactor. **b** Fuel assembly of a light water reactor (*Source* [70])

- resistivity against corrosion from coolant
- accomodate fuel–clad interactions
- cheap production of fuel assembly

Our considerations will remain limited to the structural aspects of cladding materials. Claddings for light water reactors are made of zirconium alloys (Zircaloy). Advanced nuclear reactors need other types of fuel containing elements because of the higher operation temperature and the fast neutron spectrum.

A thin walled pipe containing fuel tablets is the current fuel pin of reactors using water as moderator/coolant (Fig. 3.6a). These pins are assembled to fuel elements as shown for an LWR in Fig. 3.6b. Fuel claddings for LWRs are manufactured out of Zirconium alloys (type Zircaloy). Also the CANDU-reactors have fuel pins. Figure 3.7 shows a sketch of the pin-calandria design [5] and Fig. 3.8a closeup of a typical CANDU fuel- element end plate with the bundle inside a Zr–Nb 2.5 % pressure tube [5]. The rather thin walled pressure tube replaces the RPV in this case. As the hoop stress on pressure vessels is directly proportional to diameter; the small diameter pressure tube walls can be much thinner than the thick walls required for a PWR pressure vessel. Thin Zircaloy walls do not absorb many neutrons; hence the moderator can be placed outside the fuel area in a low

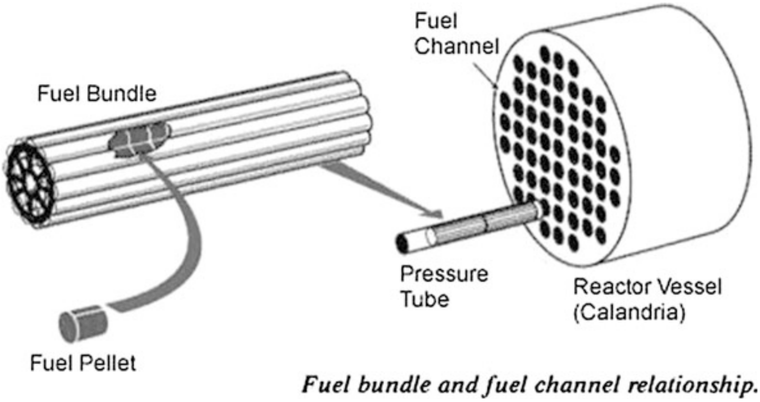
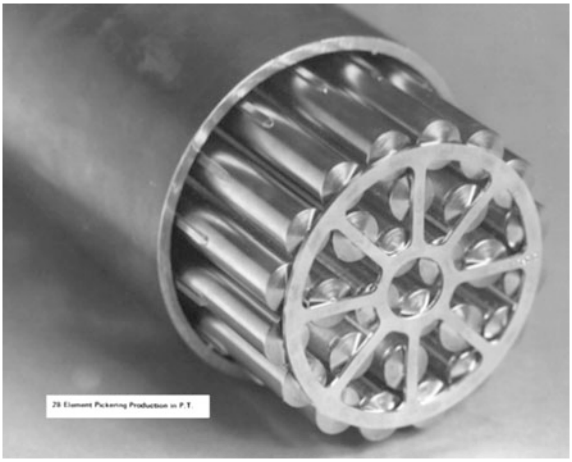


Fig. 3.7 Sketch of the fuel arrangement in a CANDU-reactor. The pressure tubes fit into the channels of the Calandria which acts as pressure vessel [5]

Fig. 3.8 Typical CANDU fuel- element end plate with the bundle inside a Zr–Nb 2.5 % pressure tube [50]



pressure calandria. This is the difference between pressure tube reactor design and pressure vessel reactor design.

The SCWR is the advanced reactor closest related to the water cooled current designs. The design of the fuel elements is therefore similar to the water reactors (either LWR or CANDU-type). However, The higher temperatures require cladding materials other than Zircaloy. Ferritic martensitic steels, low swelling austenites or even oxide dispersion strengthened steels are considered as options (see Table 3.3).

The fuel for liquid metal cooled ractors (SFR, LFR) is also contained in clads. Figure 3.9 shows the fuel assembly of a Korean SFR design as an example. Figure 3.10 shows the core assembly where the fuel rods and wrapper wires are assembled in the duct. The high flux and the operation conditions of these reactors

Table 3.3 Exposure conditions and structural materials proposed for a SCWR (*Source* partly [67])

Component	Temperature °C	Dose dpa	Material
Fuel cladding	280–620	15	Ferritic martensitic, low swelling austenitic, ODS
Spacer grids/wire wrap	280–620	15	Ferritic martensitic, low swelling austenitic
Fuel assembly duct	280–500	15	Ferritic martensitic, low swelling austenitic, SiC/SiC
Upper guide support	280–500	0.021	Ferritic martensitic, advanced austenitic
Upper core support plate	500	0.021	Ferritic martensitic, advanced austenitic
Lower core plate	280–300	0.3	Ferritic martensitic, advanced austenitic, 304L
Core barrel or shroud	280–500	3.9	Ferritic martensitic, low swelling austenitic
Threaded fasteners	280–500	<4	IN-718, 625, 690, advanced stainless steel

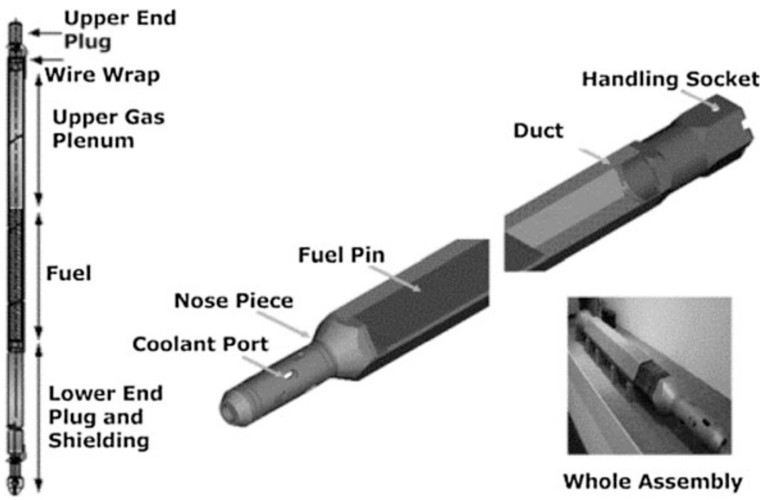


Fig. 3.9 Typical SFR fuel pin (*Source* KAERI, [51] http://ehome.kaeri.re.kr/snsd/eng/organization/organization1-1.htm)

provide a real challenge for the materials for cladding, wrapper and duct. Currently, austenitic steels modified for better swelling resistance, ferritic martensitic steels and oxide dispersion strengthened materials are considered as candidates. Liquid lead–bismuth may cause cladding corrosion and therefore surface protection with MCrAlY-coatings (M stands for metal) is considered as a possible option. Coating pocedures will be describe later.

The gas cooled reactors follow (at least partly) a different fuel concept. The fuel element of the high temperature gas cooled reactor which uses a thermal neutron spectrum is shown in Fig. 3.11. The so called TRISO (tristructural isotropic) pellet

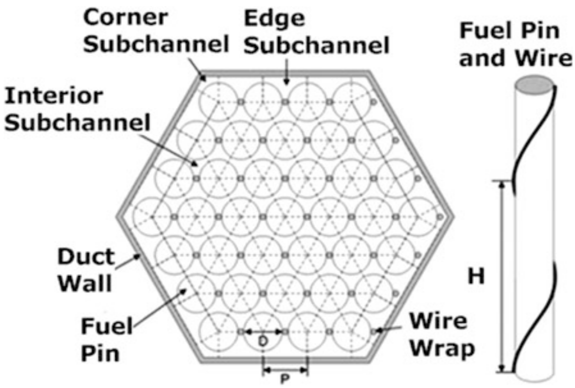


Fig. 3.10 Arrangement of fuel pins with wire wrap in a duct of an SFR (*Source* [52])

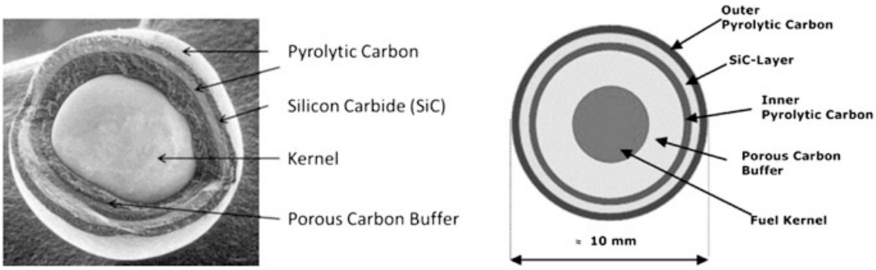


Fig. 3.11 TRISO coated fuel particle. **a** (V)HTR fuel [71]. **b** Schematic drawing of a TRISO coated particle

consists of a small particle of reactor fuel (there are several different combinations of uranium, plutonium and thorium that are suitable and have been tested) which is coated with four layers of carbon based materials. The innermost layer is porous pyrolytic graphite which is designed to provide an expansion volume for the gases that are released as the heavy metals are fissioned. The next layer is dense pyrolytic graphite whose purpose is to seal in the gases. The third layer is silicon carbide (SiC) whose purpose is to seal in certain fission products that are capable of diffusing through the pyrolytic graphite. Finally there is an outer coating of pyrolytic graphite. In this case carbon and SiC fulfill the structural role of the cladding. These particles can be embedded in a graphite sphere (pebble) for the so called pebble bed reactor or the can be embedded in fuel compacts stacked in prismatic blocks (prismatic design). Both options are shown in Fig. 3.12.

Different fuel concepts are considered for the gas cooled fast reactor (Fig. 3.13). They could be either advanced particle options (similar to the VHTR), platelets with compartments which are filled with fuel and cladded pellets. Which

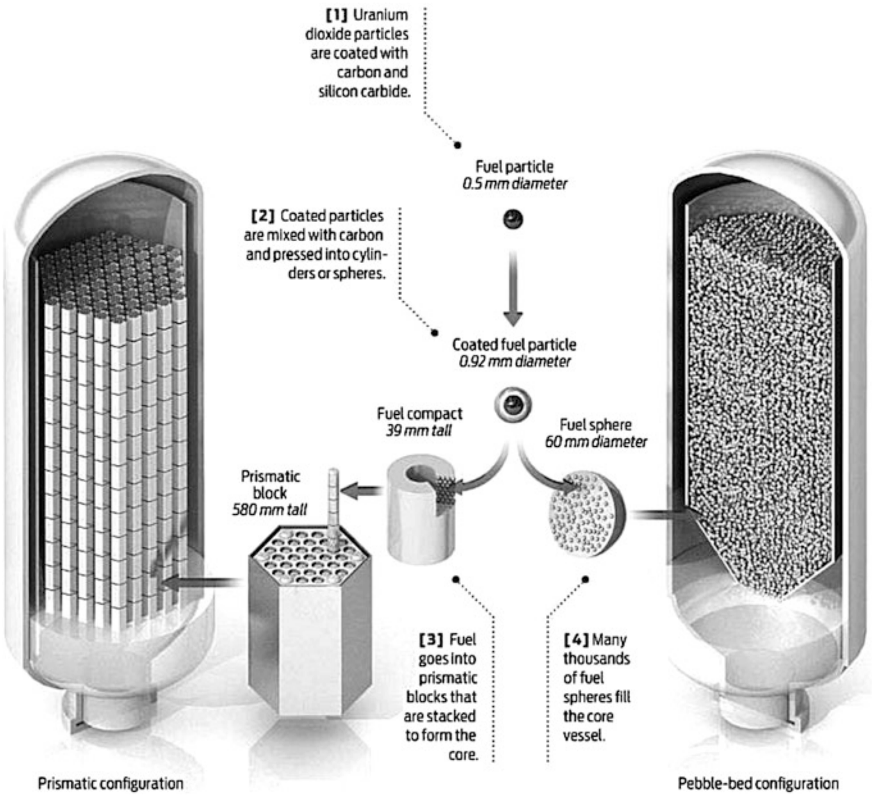


Fig. 3.12 The two options for a high temperature gas reactor (HTR). (Copyright: Artwork by Bryan Christie)

of these options will finally be chosen is not quite clear [6]. The cladded pellet would be a straightforward development from existing concepts. The problem is, however, that the core-temperature of a GFR can raise very quickly up to 1600 °C in case of a loss of coolant (LOCA) accident. This is due to the low thermal inertia of the system. Metallic core-elements should therefore be avoided (except refractory alloys). Fibre reinforced SiC would be basically an option as cladding material. But the danger remains that the ceramic material would not stay gas-tight which would lead to fission gas release. Therfore, refractory metal liners on SiC compound claddings or even refractory claddings are considered as possibilities. The pebble type fuel is also considered as a thermal option for a molten salt reactor like the advanced high temperature reactor (AHTR) [7]. In MSR fast reactors the fuel is also a molten salt which is carried by a carrier salt (see introduction) which makes structural elements like claddings un-necessary.

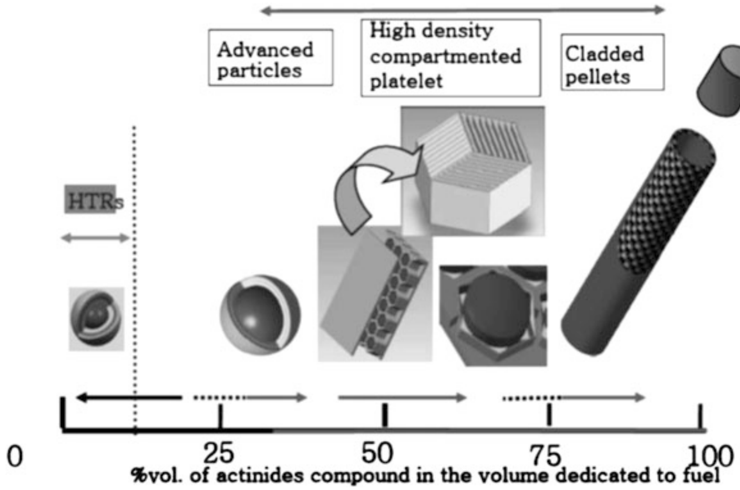


Fig. 3.13 Currently discussed types of GFR-fuel (*Source* [6])

3.1.3 Control Rods

Control rods are necessary to control the neutron flux in a reactor. They can be removed from or inserted into the central core thereby increasing or decreasing the number of neutrons which will split further uranium atoms. This affects the thermal power of the reactor, the amount of steam produced, and hence the electricity generated. They are moved by the control rod drive mechanisms being mounted on the reactor pressure vessel. The mechanical function of the control rods is exclusively the positioning of chemical elements with a sufficiently high capture cross section for neutrons. Movement and positioning of the drives is most important and the control rod elements have to maintain its mechanical integrity in the reactor environment. For the GFR and the very high temperature reactors control rods most probably need ceramic parts. Fiber reinforced ceramics (C/C, SiC/SiC) are potential candidates. A design for a control rod segment made of fiber reinforced ceramic has been given in [8] (see Fig. 3.14).

3.1.4 Other Reactor Internals

Reactor internals (except fuel elements) serve primarily as guiding or supporting elements necessary to support the fuel bundles or to guide the stream of the coolant. Most important internals of a PWR are shown in Fig. 3.1. For BWRs the elements are similar though system specific differences exist. The materials used

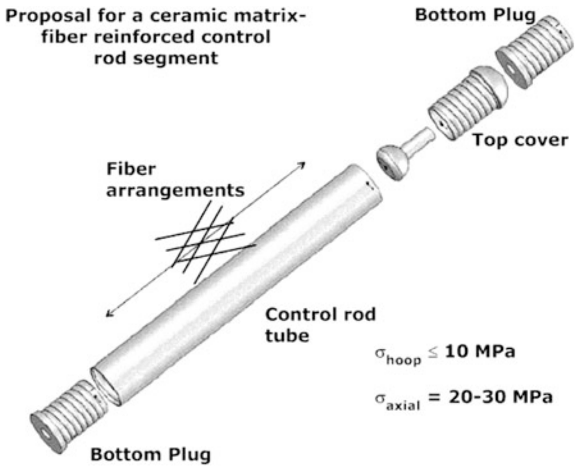


Fig. 3.14 Proposal for VHTR SiC/SiC control rod element (*Source* [53])

are primarily austenitic steels. In light water reactor environments reactor internals suffer stress corrosion cracking which is a very important field for safety considerations and life-extension programs. Such problems will be discussed in more detail in Chaps. 6 and 8.

For some thermal reactors graphite is used as a moderator instead of water. The Russian RBMK reactor has a huge graphite block structure as the moderator. Water is circulated in more than 1000 vertical tubes with about 9 cm diameter passing through the reactor core to remove the heat produced by 2 sets of long fuel assemblies, which are also mounted in the vertical tubes. The graphite structure is contained in a steel vessel (approximate diameter of 13 m). A helium–nitrogen mixture is used to improve heat transfer from the graphite to the coolant channels and reduce likelihood of graphite oxidation [9]. It is important to notice at this stage that graphite (though carbon) cannot really burn. This becomes also evident by a description of the fire occurring at the Chernobyl accident [10]: "*The reactor power of Chernobyl Unit* 4 *increased to* 100 *times its normal value in* 4 s. *That is a huge burst of energy and it made everything exceedingly hot. Thus, red hot graphite was ejected from the core. Upon meeting air, some oxidation of the geometrical surfaces of the fractured blocks immediately occurred, and a mix of carbon dioxide and carbon monoxide was produced. Where the CO was dominant, it immediately reacted with the oxygen of the air, producing a flame—genuine 'burning', but not of the graphite. Large red-hot graphite projectiles also landed on flammable material, such as asphalt roofing, and the heat provided from the graphite was sufficient to start fires—again, genuine 'burning' but not the graphite itself. The real test is to use a propane/oxygen flame to heat a block of graphite—say* 1 kg—*to white heat. If you then turn off the propane and allow the pure oxygen to impinge on the white-hot graphite, it cools it down, rather than 'fuelling'*

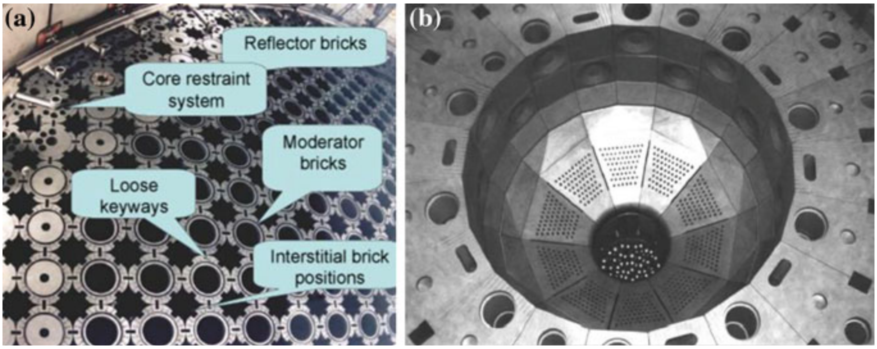


Fig. 3.15 Graphite cores of gas cooled reactors, advanced gas reactor (AGR) [54] and high temperature reactor [55]. **a** with the permission of B. Marsden (Manchester) **b** with the permission of Y. Sun, INET

a fire. Folks should also remember that bags of graphite powder were also used as fire extinguishers—I myself used them to put out a Magnox fire in the hot-cell line at Berkeley Nuclear Laboratories. Please also think about arc lamps, use as electrodes at high temperature in (for example) aluminium smelting, etc. etc."

Also the AGR uses graphite as moderator. The mean temperature of the hot coolant leaving the reactor core was designed to be 650 °C. In order to obtain these high temperatures, and to ensure useful graphite core life (graphite oxidises readily in CO_2 at high temperature) a re-entrant flow of coolant at the lower boiler outlet temperature of 278 °C is utilised to cool the graphite, ensuring that the graphite core temperatures do not vary too much from those seen in the earlier designed Magnox stations [11]. Figure 3.15a show the core of an Advanced Gas Reactor before fuel insertion.

Although the Generation IV concepts are primarily fast reactors needing no moderators, two of the three thermal concepts have graphite cores. These are the very high temperature reactor (VHTR) and the thermal version of the molten salt reactor. The core of the Chinese pebble bed reactor HTR-10 is shown in Fig. 3.15b. The core is a full graphite construction with openings and channels for control rod drives and gas flow. The pebbles are visible at the bottom. Dimensional changes and internal stress induced by neutron irradiation are real challenges for quality of graphite and core-design. The molten salt reactor is a very versatile concept. In its thermal version a liquid molten fluoride salt is circulating through the graphite reactor core. However, the long-term compatibility of graphite and molten salt would need further attention. As currently fast molten salt concepts (without graphite moderator) are considered graphite limitations for MSR are no longer important.

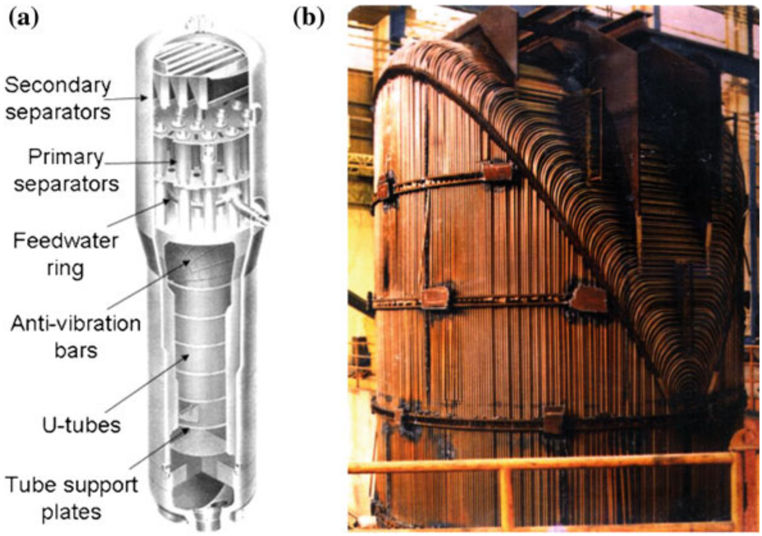


Fig. 3.16 Steam generator top view of an old steam generator from a nuclear power plant (*Source* [56]). The *left figure* shows a new modern steam generator for comparison (*Source* [57]). **a** Advanced steam generator. Copyright: Mitsubishi Heavy Industries **b** Old steam generator (with permission of US-NRC)

3.1.5 Piping and Steam Generator

Steam Generators are required whenever heat is converted into steam to drive a steam turbine or to use it for process purpose. A steam generator for nuclear applications is a cylindrical reservoir containing approximately 5.000 inverted U-shaped pipes. The hot medium in the primary circuit, coming from the reactor (or the primary heat source), circulates through the steam generator pipes. The heat carrier (usually water) in the secondary circuit flows along the outside of the tube bundle. When it comes into contact with the heated pipes, the secondary circuit water starts boiling and is converted in steam. Therefore, the water and steam in the secondary circuit do not come into contact with the coolant of the reactor. In this manner the steam generator acts as an additional safety barrier between the nuclear reactor and the outside world. Moist separator and steam dryer condition the steam before it enters the steam turbine. The steam generator vessel is similar to the reactor pressure vessel a huge welded construction. Several pipes must be corrosion resistant. The materials used will be discussed later. Figure 3.16a shows a cut through modern steam generator for PWR plant. Figure 3.16b shows the upper portion of the pipes after long-term sevice exposure.