International Thermonuclear Experimental Reactor: A Thermonuclear Future

M. Goswami$^1$ and S. Sahoo$^2$

$^1$Department of Physics, Regional Institute of Education (NCERT), Bhubaneswar - 751022, Orissa, India. E-mail: manasigoswami1@yahoo.com

$^2$Department of Physics, National Institute of Technology, Durgapur – 713209, West Bengal, India. E-mail: sukaadevsahoo@yahoo.com

1. Introduction

Nuclear power or energy is an outcome of basic research in the field of reactor physics. The discovery of the neutron in 1932, the discovery of nuclear fission in 1939 followed by the first critical fission reactor in 1942 marked the beginning of a new source of energy for the mankind that is nuclear energy. The nuclear process other than fission which yields nuclear energy is called fusion. As we know fission and fusion are two diametrically opposed nuclear processes. In fission a heavy nucleus is split by a neutron into two (or more) parts of almost always different masses. Energy is released in the process along with some neutrons. In fusion two very light nuclei combine, if they have sufficient energy and create various reaction products that have much more kinetic energy than the reactant. During last four decades, electricity from nuclear fission process has become a reality and many countries contribute a significant proportion. However, the demands for energy are increasing so rapidly and stock of conventional energy resources are depleting so fast, that it is now realised that nuclear energy derived from nuclear fission process can not alone cater to the present day need. Hence the world is now determined to explore the unlimited source of energy which is possible only from the nuclear fusion. As a resurgence of interest in fusion energy programme, a collaborative agreement had been signed in July, 1992, by Euratom and the Government of Japan, Russia, and the United States of America with the objective of demonstrating the scientific and technical uses of fusion for peaceful purpose$^1$. Such a high level multinational effort aimed at designing the world’s first fusion reactor, the international thermonuclear experimental reactor (ITER).

2. Background of ITER

It all began in early 1988, when an agreement was reached by European Community, Japan, the United States of America and Soviet Union to jointly conduct conceptual design activities for the ITER until 1990, under the auspices of the international atomic energy agency (IAEA). Since May 1988, participants from the four countries meet regularly in Garching Federal Republic of Germany to carry out the design work.
However, the project took a new turn in 1992, when Russia, the USA, the European Community, and Japan decided to develop jointly the ITER on the basis of TOKAMAK (Toroidal Chamber in Magnetic Coils) technology. Soviet physicist developed the first Tokamak units in line with Sakharov’s idea. Sakharov had suggested the tokamak concept in 1960’s. The world physicists offered 114 thermonuclear concepts; however, TOKAMAK alone has survived. Sakharov’s concept was eventually tested at the Kurchatov Research and Development Institute and it proved to be a success. Russian achievements in the field of superconductors as well as unique electron plasma heating methods were instrumental in implementing ITER project. In 2003 China and South Korea joined in this project. Canada had joined in this programme in 2001 but walked out in 2003. It is a matter of pride that India was invited into the multimillion dollar project as a full partner at the ITER negotiations meeting in Jeju, South Korea in December 2005. At present the members of ITER project are the USA, Russia, the European Union, India, China, Japan and Republic of Korea. This ITER project will cost an estimated $5 billion and now involves seven countries. A country whose territory has been chosen to build the reactor shall contribute 50 percent of the total. The members of the project debated this issue rather hotly with France and Japan offering to accommodate the reactor. It was eventually decided to construct the ITER reactor in France’s Cardarache (Province). ITER project is set to remove the last obstacle to the creation of the world’s first thermonuclear power plant that promises to solve global energy and environmental problems. A thermonuclear power plant may materialize by 2030. The current architectural plan of ITER’s fusion power plant is given in Fig. 1.

![Fig. 1: architectural plan of ITER’s fusion power plant](image-url)
**Schematic Diagram of a Fusion Reactor**

**PLASMA**
In the plasma 80% of the energy produced by the D-T reactions is in the form of energetic NEUTRONS which escape across the magnetic field and are then trapped in a surrounding blanket which contains LITHIUM. Plasma in the reactor core is self-heated by the α-particles which contain the remaining 20% of the energy produced.

**BLANKET**
In the blanket the NEUTRONS
- react with the LITHIUM to produce TRITIUM and HELIUM
- as they slow down their kinetic energy is converted into heat.

**Shielding Structure**

**HEAT EXCHANGER**
Heat is extracted to raise steam to drive conventional turbines for electricity generation.

**Superconducting Magnets**

**Deuterium Fuel**

**Tritium**
TRITIUM is recycled to the vacuum vessel to refuel the plasma.

**Tritium and Helium**
TRITIUM and HELIUM formed in the lithium blanket and plasma exhaust gases are separated by cryodistillation.

**He**
HELIUM is exhausted as waste.

**Electric Tower**

**Steam Boiler**

**Turbine and Generator**

Fig. 2: Schematic diagram of a Fusion reactor."
3. Objectives of ITER

The specific technical objectives of ITER science and technology programme includes:

(a) achievement of self-sustained burn
(b) extended burn time
(c) safe operation of a reactor like device
(d) testing of components under reactor like conditions.

To reach these objectives, ITER will be operated in two phases

- A physics phase, focused mainly on the achievement of the plasma objectives.
- A technology phase, devoted mainly to engineering objectives and the testing programmes.

The plasma physics objectives i.e. to demonstrate controlled ignition and extended burn of deuterium-tritium plasma, with steady state as an ultimate goal will be reached through

- Inductive plasma operation under condition of controlled burn
- Extension of the burn pulse towards the steady state using non-inductive current drive.
- Provision of current carrying capability as required for ignition experiments in which the predictions regarding plasma confinement are studied.

The engineering and testing objectives are

- to validate design concept and to qualify engineering components.
- to demonstrate the reliability and maintain ability of the reactor system and
to test the main nuclear technologies (blanket modules, tritium production, extraction of high-grade heat appropriate for generation of electricity)

4. Reactor Structure and Materials

In brief the reactor is composed of a support structure, a cryostat with superconducting magnet, a vacuum vessel and the first wall being an integrated blanket (Fig. 2). The blanket includes structural materials, a neutron absorber, and high heat flux components i.e. plasma facing armour and heat sink.

Energy leaves the plasma in form of electromagnetic radiation and kinetic energy of particles. Plasma surrounding wall is irradiated by ions, charge exchange neutral electrons, photons and neutrons. All of them modify material properties, from very surface to the bulk. Therefore blanket materials must be compatible with ultra high vacuum, cryogenics (cryo pumps), magnetohydrodynamics, neutron irradiation and handling of high heat loads. As a consequence, there are stringent requirements regarding the properties of plasma facing materials (PFMs). These are materials of high thermal conductivity, good thermomechanical properties, high resilience to thermal socks, non-magnetic, low activation by neutrons, high resistance to radiation damage, low accumulation of hydrogen isotopes accompanied by low chemical affinity to hydrogen in
order to avoid chemical erosion leading to the formation of volatile compounds. High affinity to oxygen towards formation of stable and non-volatile oxides is also important for guttering oxygen impurity species in a reactor. However, properties of no single element, compound or alloy can satisfy all points of the above list. Only a few candidate materials for the plasma facing wall are seriously considered: carbon fibre composites (CFC), beryllium (Be) and tungsten (W). Behaviour of these elements under plasma condition i.e. particle bombardment and high heat flux deposition is very different. Therefore, their distribution on the reactor wall is different. Beryllium is used for the main chamber wall port limitors, and baffle, whereas the diverter dome and upper vertical target are covered with tungsten tiles. CFC covers the lower vertical target where the greatest power is deposited. A detailed distribution of tungsten and carbon is shown in Fig. 3.

The list of candidate structural materials for the blanket comprises mainly austenitic steel (e.g. 316L), vanadium-titanium alloys (V – Ti, V – Ti – Si , V – Ti – Cr) and silicon carbide composites. Major requirements emphasize mechanical strength and low activation by neutrons. Low activation and increased resistance to radiation damage are also crucial for ceramic insulators and components of invessel diagnostics such as optical fibres, cables, mirrors, and windows. The most important processes affecting material properties are discussed in more detail in section 5.

5. Fuel Cycle

In order to start a fusion reaction the two nuclei must have enough energy to overcome the repulsive Coulomb force acting between them. When both nuclei approach each other, at sufficiently close distance the short range attractive nuclear force becomes dominant. For the above purpose the fusion fuel must be heated to high temperature. D–T reaction (which usually involve in a fuel cycle) the gas temperature must exceeds $5 \times 10^8$ K before a significant fusion rate is feasible. At such temperature the gas exists as a macroscopically neutral collection of ions and unbound electrons which is called plasma. In the vacuum chamber the fusion fuel is generally irradiated by a multiplicity of focused laser beams, causing the micro-explosions necessary for initiating thermonuclear reaction i.e. the plasma state. This fuel is compressed towards the centre of the chamber by means of an inhomogeneous magnetic field. The fuel cycles involving deuterium and tritium are given by the reactions:

**D – D fuel cycle**

$$D + D \rightarrow ^3H \ (1.01 \text{ MeV}) + p \ (3.03 \text{ MeV})$$
$$D + D \rightarrow ^3He \ (0.92 \text{ MeV}) + n \ (2.45 \text{ MeV})$$

The above two equations have about equal probability.

**D – T fuel cycle**

$$D + T \rightarrow ^4He \ (3.52 \text{ MeV}) + n \ (14.07 \text{ MeV})$$

All the above reactions are exothermic in nature.
In December 1993, for the first time in history a reactor fuel mix of 50% deuterium and 50% tritium was used in a Tokamak Fusion Test Reactor (TFTR) operated at the Princeton Plasma Physics Laboratory. Initial TFTR experiments yielded 6.0 million watts. By November 1994, TFTR achieved 10.7 million watts of power.

Fig. 3: A schematic view of the ITER structure and distribution of plasma facing materials\textsuperscript{5}.

For fusion reaction to take place for a D–T or D–D plasma it should obey the Lawson’s Criterion. At a given temperature ‘T\textsubscript{0}’ of a plasma the chances of fusion reaction to take place, depend on the minimum particle density N in it and the time τ for which they are confined together. According to Lawson’s Criterion with a particular composition of the plasma having a volume V, there is a minimal value of the confinement parameter Nτ for which the power component resulting from the deceleration of charged particles in the plasma compensates the power required for
maintaining the plasma at a suitable temperature. For a D–T plasma whose ignition temperature $\sim 8\times 10^8$ K, $N_\tau$ has a value of $3\times 10^{14}$ s/cm$^3$ where as for a D–D plasma whose ignition temperature is $\sim 5\times 10^8$ K, $N_\tau$ is $10^{16}$ s/cm$^3$. The above data shows that a D–T mixture react about $10^2$ times faster than a D–D mixture even at lower ignition temperature. So D–T reaction is more preferred. Moreover a D–D reaction has smaller cross section and lower Q-value than that of a D–T reaction. Therefore the D–T cycle is taken as the fuel of a first generation fusion reactor.

D–3He fuel cycle

A third possible fusion reaction is: $D + ^3He \rightarrow ^4He + H$ ($Q = 18.2$ MeV).

$^3$He isotope has been identified as a potential fuel to use in nuclear fusion since the 1950s. But there is an estimated of only 15 tonnes of $^3$He on earth’s surface. However, lunar surface is rich in $^3$He, which can be used to produce energy through fusion.

The fusion reactivity is shown in Fig. 4 for the three reactions cited above. The D–T fusion reactivity is much greater than that for other potential fusion reactants, which is the reason why achieving the necessary condition for D–T fusion is the principal goal of the present phase of fusion research. The charged particles produced in the D–T reaction are slowed down in the plasma itself with an yield of 20% of power (from He). Another 80% of power ($\sim 14$ MeV) is carried off by the neutrons. Let us now discuss the procedure to heat the fuel/plasma for possible fusion reaction.

6. Magnetic Confinement

A fusion plasma can not be maintained at thermonuclear temperature if it is allowed to come in contact with the walls of the confinement chamber because, material evoked from the walls would quickly cool the plasma. Fortunately, magnetic fields can be used to confine a plasma within a chamber without apparent contact with the wall.

The basis for the magnetic confinement of plasma is the fact that charged particles spiral about the magnetic field lines. The radius of the spiral, or gyroradius, is inversely proportional to the strength of the magnetic field, so that in a strong field charged particles move along magnetic field lines. Basing on the above principle there are two ways of confining the plasma: (i) closed toroidal confinement systems, (ii) open (mirror) confinement systems.

(i) Closed Toroidal Confinement System

The magnetic field lines may be configured to remain completely within a confinement chamber by the proper choice of position and current in a set of magnetic coils. The simplest such configuration is the torus, shown in Figs. 5 (a,b). A set of coils can be placed to produce a toroidal field $B_\phi$. Particles following along the closed toroidal field lines would remain within the toroidal confinement chamber.
The curvature and non uniformity of the toroidal field produce forces which act upon the charged particles to produce ‘drift’ motions that are already outwards, which would, if uncompensated, cause the particles to hit the wall. A poloidal magnetic field must be superimposed upon the toroidal magnetic field in order to compensate these drifts, resulting a helical magnetic field which is entirely contained within the toroidal confinement chamber. This poloidal field may be produced by a toroidal current flowing in the plasma (tokamak) or by external coils.

The tokamak concept⁷ (Fig. 6) which was invented in USSR in the mid-1960s, has been the most extensively investigated worldwide and is the most advanced. The toroidal field is produced by a set of toroidal field coils which encircle the plasma. The
poloidal field is produced by an axial, or toroidal, current which is induced by the transformer action of a set of primary poloidal field, or ohmic heating.

(ii) Open (Mirror) Confinement System

It is possible to confine a plasma magnetically within a fixed confinement chamber, even when the magnetic field lines themselves do not remain within the confinement chamber, by trapping the charged particles in a magnetic well. The principle is illustrated in Fig. 7, where the particle’s speeds along parallel ($v_\parallel$) and perpendicular ($v_\perp$) to the magnetic field $B(s)$ are denoted.

Because the force exerted on a moving charge by a magnetic field $B(s)$ is orthogonal to the direction of particle motion, no work is done on the particle and its kinetic energy (KE) remains constant:

$$\frac{1}{2} m \left[ v_\parallel^2(s) + v_\perp^2(s) \right] \equiv KE = \text{constant}$$

The angular momentum is also conserved, so that

$$\frac{1}{2} \frac{m v_\perp^2(s)}{B(s)} \equiv \mu = \text{constant}.$$  

Thus, the velocity along the field line can be expressed as:

$$v_\parallel(s) = \frac{2}{m} \left[ KE - \mu B(s) \right].$$

For particles with sufficiently large values of $\mu$, the right-hand side (RHS) can vanish as the particle moves along the field line into a region of stronger magnetic field. When the field is strong enough that the RHS vanishes, the particle is reflected and travels back along the field line through a region of reducing, then increasing magnetic field strength until it comes once again to its reflection point, at which the field strength is again large enough to make the RHS vanish.

![Fig. 5a](image-url)
Fig. 5b

Fig. 5 (a, b): Closed toroidal confinement.

Fig. 6: Tokamak Schematic\textsuperscript{7}.
7. Heating of Plasma

Heating a plasma to thermonuclear temperature for many years has been one of the principal goals of the fusion research. In 1978, this goal was first achieved in a tokamak. The following processes are usually adopted for plasma heating.

(a) Neutral Beam Injection (NBI)

Plasmas have been heated to the thermonuclear temperature regime by neutral beam injection (NBI) system, which is the most advanced and successful plasma heating technique to the date. Injection of highly energetic hydrogen or deuterium atoms into plasma, where they become ions and give up their energy to the plasma ions and electrons via Coulomb scattering, has proven to be a highly successful means of plasma heating. The injected particles must be neutral in order to pass through the magnetic field surrounding the plasma. While neutral beam injection is the principal auxiliary heating technique used on the present generation of tokamak experiments and will continue to be so for the next few generation of tokamaks, there are some technological and economical factors that provide an incentive to develop other heating techniques.

(b) RF Heating

Foremost among the other alternate heating techniques are electromagnetic waves which are commonly known as rf heating techniques because in most instances, the energy absorption occurs within the radiofrequency range. The three most promising rf methods are electron cyclotron heating, lower hybrid heating and ion cyclotron heating, all of which are based upon the absorption of a wave in the vicinity of a plasma resonance.

(c) Adiabatic Compression

Compression of the plasma in major or minor radius by means of a change in the magnetic field results in an increase in internal plasma energy. This technique has been confirmed experimentally on the small tokamak. Because of the rather large pulsed power and the oversized plasma chamber required for the compression, adiabatic compression presently is not considered as a viable, primary heating method.
8. Fueling

Fueling with a controllable mixture of deuterium and tritium will be required during the startup phase and possibly for control purposes during the burn phase. Refueling during the burn will require roughly equal amount of deuterium and tritium. Two methods of refueling are used in present tokomaks, and both can be extended to meet the needs of future reactors.

Gas puffing is the simplest method. In this mechanism gas injected into the chamber reaches the centre of the plasma.

At times it is desirable in order to control density profiles to refuel in the plasma interior. Frozen pellets of a few mm in diameter are injected into plasmas at velocities up to $10^3$ m/s, using light gas guns and centrifugal accelerators. These pellets velocities are adequate for penetrating the outer regions of a reactor level plasma before complete pellet ablation occurs, but velocities of $5–10 \times 10^3$ m/s would be necessary to fuel preferentially the center of such plasmas.

9 (a) Tritium Breeding Blanket

The D–T fuel cycle involves the release of 14 MeV neutrons which cannot be retained in the vacuum chamber by the confining magnetic field. This neutron energy can be made available only as heat. All considerations of energy conversion require that the heat be removed at a useful high temperature. Therefore, to stop escaping neutrons and put them to good use, the vacuum chamber in a CTR running on this fuel cycle is surrounded by a blanket where neutrons collide with the nuclei of blanket material and produce heat. In fact, 80% of the energy carried by escaping neutrons is released in this region. The blanket region also serves another important purpose of breeding tritium. Deuterium (D), the naturally occurring isotope necessary for the D–T fuel cycle can be obtained from oceans in large quantities (0.0153 atom % of hydrogen). However, tritium occurs only in minute traces in nature, since it decays (into $^3$He) with a half life of 12 years. Thus if deuterium-tritium (D-T) reaction is envisaged the tritium must be somehow regenerated in a side reaction.

Lithium has so far been considered to be the most suitable element for tritium production due to its relatively high cross-section and Q-values for (n,t) reactions. Natural lithium has two isotopes $^6$Li (7.42 atom %) and $^7$Li (92.58 atom %), the two nuclear reactions leading to tritium production are:

\[
^6\text{Li} + n \rightarrow ^3\text{H} + ^4\text{He} + 4.78 \text{ MeV},
\]

\[
^7\text{Li} + n \rightarrow ^3\text{H} + ^4\text{He} + n - 2.465 \text{ MeV}.
\]

The first reaction is exothermic, while the second one is a threshold reaction and can occur only for neutron energies more than 2.9 MeV. However, the (n,t) cross section of $^6$Li for capture of thermal neutron is 945 barn (at 2200 m/s), which is a large value and is approximately proportional to the reciprocal of the neutron speed. Hence, tritium can conveniently be bred as part of the fusion reactor cycle if lithium is incorporated in some
form as a breeding blanket round the plasma: this has the additional advantage that neutron capture in $^6\text{Li}$ (7.42 % abundant, the remainder being $^7\text{Li}$) results in the liberation of 4.78 MeV of energy, thus increasing the net energy output per fusion reaction. The effective total nuclear heat per fusion reaction thus becomes,

$$U'_{\text{DT}} = U_{\text{DT}} + 4.78 \text{ MeV} = 22.37 \text{ MeV}.$$  

Therefore, although the fusion reactions occurring in a reactor take place between deuterium and tritium, the consumable are deuterium and lithium.

Besides lithium metal, its many ceramics and compounds / alloys like Li$_2$O, LiF, LiAlO$_2$, Li$_2$SiO$_3$, Li$_7$Pb$_2$ have also been studied in some detail. A detailed study of the blanket assumes great importance because the success of futuristic D–T fuel cycle based fusion reactors critically depends on our ability to breed tritium necessary for fuel self sufficiency.

In addition to the tritium breeding material the blanket also contains a certain amount of structural material such as stainless steel, titanium, nickel alloys etc. that absorbs neutrons. To compensate the loss of neutrons, an arrangement for multiplication of neutrons by some nuclear reaction is also made inside a blanket. Materials like Be and Pb have been used extensively for neutron multiplication through the (n, 2n) reaction. To stop escaping the energetic neutrons from the blanket a reflector is necessary to minimize the breeder thickness. Such a multilayer blanket assembly is usually termed as a blanket system.

An important parameter in fusion reactor blanket design consideration is the tritium breeding ratio (TBR). It is defined as the number of tritium atoms produced per fusion neutron released. In recent years, considerable efforts have been made to obtain vital information needed for selecting suitable breeding material and feasible configuration that would yield a TBR value over unity by a margin necessary to compensate the tritium losses and radioactive decay.

(b) Characteristics of Tritium Breeding Material

The tritium breeding material may be solid or liquid. The liquid breeders, as a class, have the advantage that they can be circulated through the blanket, which allows the bred tritium to be extracted by processing the breeder external to the blanket. However, difficulty with the liquid breeders is that their melting point is above room temperature, which requires that internal heaters must be provided to maintain the breeder molten during the reactor shutdown or to melt it before start up. Liquid metals, particularly those containing lead, tend to be rather corrosive, which is another problem associated with the liquid breeders$^8$.

Lithium undergoes exoteric reactions with air and water, which constitutes a potential safety problem. These reactions are suppressed when lithium is in the form of binary or ternary ceramic compound (the oxygen-containing compounds like Li$_2$O, LiAlO$_2$, Li$_2$SiO$_3$ etc.). The vigor of the reaction can also be reduced substantially by reducing the lithium density (e.g. Li$_{17}$Pb$_{83}$, which is in ratio of 17 lithium atoms to 83 lead atoms). The tritium breeding yield is generally improved with increasing lithium
atom densities; with the exception of those breeding materials which themselves contain a good neutron multiplier such as Pb or Bi. The breeding materials must be capable of achieving a tritium breeding ratio in excess of unity, which provides for the possibility of a self-sustained D–T fuel cycle. A neutron multiplier must be included in the blanket in order to achieve TBR > 1. The achievable TBR depends on the thickness of the blanket.

Now let us discuss various concepts for the blanket design which have been proposed, experimented and believed to have the potential to meet the optimal value of TBR (∼1.05).

(i) Aqueous Salt Concept:
In this case the lithium bearing salts (i.e. LiOH, LiNO₃) are put in the water coolant to achieve tritium breeding. This concept requires a comparatively simpler design and tritium is extracted from the blanket using the same technology as applied for recovering the tritium from other water cooled blankets. The disadvantages here are the loss of neutrons as well as the possible radiolysis and electrolysis of water. For such a blanket a significant amount of Be is required for neutron multiplication so as to compensate the neutron loss.

(ii) Solid Breeder Concept:
The most promising breeding blanket concept is the solid breeder. From safety point of view the solid breeders are preferred because they have structural and safety advantages. The solid breeder in which there has been lot of interest is Li₂O, which has the highest atom density and best breeding capability than any other oxides. The other favorable solid breeders are the ternary ceramics LiAlO₂, Li₂ZrO₂ and Li₂SiO₃. In this conceptual blanket system pressurized ³He gas is used as a coolant for convenient recovery of the bred tritium. Be is also used here as neutron multiplier.

(iii) Lead–Lithium Concept:
In Lead–Lithium concept, the solid compound Li₇Pb₂ or an eutectic Li₁₇Pb₈₃ is used as a tritium breeding material. The eutectic may be in a solid or in a liquid state. During the off-time the eutectic material is directed out of the reactor to recover the bred tritium. When eutectic is in the liquid state it serves the purpose of a multiplier, breeder and coolant. In the case of solid eutectic or Li₇Pb₂ the coolant is either ³He or pressurized water. The liquid Li₁₇Pb₈₃ eutectic blankets present the potential advantage of permitting an overall less complex design because neutron multiplication, tritium breeding and tritium transport are ensured by the same material.

(iv) ³He – Gas Loop Concept:
This concept is based on breeding through ³He (n,p)⁴t reaction. In this configuration ³He gas is homogeneously mixed with some moderating materials like C, Be, BeO, H₂O etc. ³He has a very large cross-section for tritium production at low energies (σ_{T,p} = 5328 b at 2200 m/s) and also serves the purpose of a coolant. Here, Pb or Be are used as multiplier.
(v) **Thermal breeding Concept:**

A new concept has recently been proposed\(^{10,11}\) in which the 14 MeV neutrons produced in the plasma are slowed down to very low energies before interacting with the breeding material. This concept is applicable to those breeding materials which have large tritium production cross-section at low energies. The blanket system is provided with an additional arrangement, for slowing down the neutrons, in between the multiplier and breeding region. One may use Be which will serve the dual purpose of neutron multiplication \((n, 2n)\) and slowing down of neutrons. Alternatively, the arrangement can comprise of a moderating medium in which breeding material has been intercalated.

10. **Blanket Coolants Structural material**

Three classes of coolants are being considered for fusion reactor blankets – gases, water and liquid metals. Several types of structural materials are being considered – austenitic and ferritic stainless steel, high–nickel based alloys, and refractory alloys based on vanadium, niobium, molybdenum, and titanium. The principal features of these coolants and structural materials are briefly reviewed below.

Helium is the gas coolant, with which there has been the most experience, primarily in the gas – cooled fission reactor field. Helium is an inert gas, which precludes many adverse chemical reactions that are a problem with other coolants. However, impurities in the helium present problems. It also has the potential for operating at very high temperature to achieve high thermal to electrical conversion efficiencies. Helium, along with the other gases, has rather than poor heat removal properties, which leads to a requirement for a large heat transfer area, large gas flow rates, and corresponding large pumping power. Helium cooled reactors would ideally operate at coolant pressures of 5–10 MPa, maximum gas temperatures of 500 – 1000°C, and gas flow velocities of 30–100 m/s. Pressurized water (H\(_2\)O, or D\(_2\)O) is an attractive coolant because of its good heat removal properties which has been experienced from the fission reactor field\(^{12}\). The principal limitations of water as a coolant relate to the requirement for high pressure and relatively low outlet temperature, so its chemical reactions with some of the potential tritium breeding materials, and to the difficulty of extracting any tritium which leaks in to the cooling system.

Lithium and other alkali liquid metals have excellent heat removal properties at low coolant pressure. With lithium or lithium eutectic there exists the possibility that the coolant can also serve as the tritium breeding material. There is a significant base of liquid metal technology from the fast fission reactor program. The principal limitations relative to the use of lithium or a lithium-lead eutectic as a coolant are related to (i) corrosion of the structural material; (ii) magnetohydrodynamics (MHD) effects on pressure drop and heat transfer; (iii) high melting point; and (iv) chemical reactivity with water and air.

11. **Blanket Tritium Recovery System**

For the recovery of tritium from a breeding blanket the appropriate techniques is employed. A proper choice of structural material and coolant can enhance the tritium
production rate. If a liquid breeding material (e.g. lithium metal or lead–lithium eutectic) is used, then the breeding material can be circulated outside the blanket and continuously processed to remove the tritium.

Tritium recovery from solid breeding material (e.g. Li₂O, LiAlO₂, Li₄SiO₄, Li₂SiO₃, LiAl, Li₇Pb₂) can be achieved by slowly circulating low-pressure helium through the breeding material and subsequently removing the tritium from the helium stream. Approximate temperatures required for effective tritium removal from some solid breeding materials are 400°C – 600°C for Li₂O, 700°C – 800°C for LiAlO₂, greater than 700°C for Li₄SiO₄ and Li₂SiO₃, and 400°C for Li₇Pb₂. The most efficient tritium recovery process, at least for Li₂O, involves the addition of oxygen to the helium to oxidize the tritium and then the trapping of T₂O. However, with Li₇Pb₂, the presence of oxygen leads to the formation of an oxide film that acts as a barrier to tritium recovery and lowers the melting point which already is quite close to the operating temperature required for tritium release.

12. Conclusion

The ultimate goal of fusion research is to construct and operate an energy generating system. Hence, the above mentioned aspects of nuclear as well as of material science are necessary and important to accomplish the success of ITER. In ITER the temperature gradients between the plasma and the surrounding wall will probably be the greatest in the universe and the operation will be associated with intense nuclear radiation. Therefore, the technology for ITER presents challenges not encountered in present day machine. This includes development and construction of the components capable of reliable performance in highly radioactive environment. Fusion if realized under terrestrial conditions, offers inexhaustible and CO₂ emission free energy supply. However, a reactor is a nuclear device and radio active aspects of fusion will undergo intense public scrutiny. International cooperation established around ITER is a very important step on the track towards commercial fusion. In this sense fusion related material research is an attractive field for a young generation of scientists and nuclear generation.

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