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Review Article

Materials Research and Development Opportunities in Fusion Reactors

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Next generation fusion reactors would demand new materials and technologies that can sustain extreme nuclear environment. Materials development activities related to structural materials, shields, superconducting magnets, breeder materials, plasma facing materials, function materials and coatings have been accelerated in the last two decades. Out of these, recent research activities are focused mainly on structural materials, plasma facing materials and breeder materials. Generation of appropriate nuclear testing and materials characterization facilities is also required to complement such materials research & development activities. In this context, an overview of the scenario of materials R&D activities in India and abroad has been discussed in this chapter.

Keywords: Fusion reactor; materials; ceramics; irradiation; blanket; divertor.

1. Introduction

Thermonuclear fusion of two hydrogen isotopes results in release of energy and energetic neutrons which can be utilized for electricity generation. Out of various hydrogen isotope reactions viz. D-D, D-T & T-T (D: Deuterium, T: Tritium), the D-T reaction has the largest cross-section at the lowest energy. D-T Fusion reaction can be conducted by different confinement methods viz., gravitational confinement, inertial confinement, and magnetic confinement. While different fusion confinement techniques are being explored, the most widely pursued technique is magnetic confinement fusion. A tokamak is a device which uses magnetic field to confine plasma to the shape of a torus. Such magnetic confinement is required since no solid material could withstand extremely high temperatures of plasma. Tokamak is one of the several types of magnetic confinement devices, and is one of the most researched devices for producing controlled thermonuclear fusion power. Tokamaks such as DIII-D in San Diego, USA; Joint European Torus (JET), Culham UK; Tore Supra at CEA, Cadarache, France, etc. are being used for thermonuclear fusion. In November, 2006, seven partner countries viz. European Union, Japan, USA, Russia, South Korea, China and India joined hands to set up a nuclear fusion reactor named International Thermonuclear Experimental Reactor (ITER) at Cadarache, France and the first plasma shot is scheduled in 2020. ‘‘Aditya’’ is India’s first tokamak developed by the Institute for Plasma Research, Gandhinagar. Subsequently, as the next step, a steady state tokamak (SST-1) reactor has been commissioned and India has been working on its domestic fusion programme as well.

The fusion of deuterium (D) and tritium (T) under magnetic confinement would lead to generation of energetic 14.1 MeV neutrons by the following reaction (Naujoks, 2010):

$$D + T \rightarrow {}^{4}He\ (3.517\ MeV) + n^{0}\ (14.069\ MeV) \quad (i)$$

The D-T reaction (eq. i) yields highly energetic 14.1 MeV neutrons which can be utilized to generate fuel (T) from Li as well as extract energy from the kinetic energy of neutrons. This is expressed by the following equation:

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$$^{6}Li_3 + n^0 \rightarrow {}^{4}He_2 + {}^{3}T_1 + 4.78 \text{ MeV} \qquad \text{(ii)}$$

Thus, Li utilizes the fast moving neutrons to generate T fuel and heat, which is later on converted to electricity. This conversion of energy of neutrons to heat and electricity is done by the blanket module of the reactor as discussed in section 2.2. A cutaway view of the ITER model is indicated in Fig. 1 (Suri *et al*., 2010). This high energy of neutrons also leads to severe damage of structural and functional materials in the reactor, i.e. radiation damage, generation of transmutation products and consequences in the mechanical and metallurgical properties thereof. It is therefore necessary to understand the type of environments prevailing in different sub-systems of the fusion reactor and their operational requirements. Fig. 2 indicates a schematic of various subsystems (Suri *et al*., 2010).

As shown in Fig. 2, the magnetically confined plasma core has a temperature of ~5 million °C. The energy from 14.1 MeV neutrons has to be utilized for T production and electricity generation. The sub-assemblies and their functional requirements are spelt out in Table 1 (Baluc *et al*., 2007).

2. Materials-related Challenges in Fusion Environment

As discussed in the previous section, many materials research and development needs have been generated for building a suitable fusion reactor plant. Most of the R&D activities on materials for fusion reactors is mainly focused on plasma facing (first wall and divertor), breeder materials and structural materials. This includes materials development, fabrication technologies, characterization and functional validation of the developed materials. A brief overview of the materials R&D needs have been summarized in the following sections.

2.1 Plasma-facing Components and Divertor

The hot plasma confinement by magnetic confinement in a tokamak reactor involves plasma-materials interaction. The first solid surface of the reactor facing

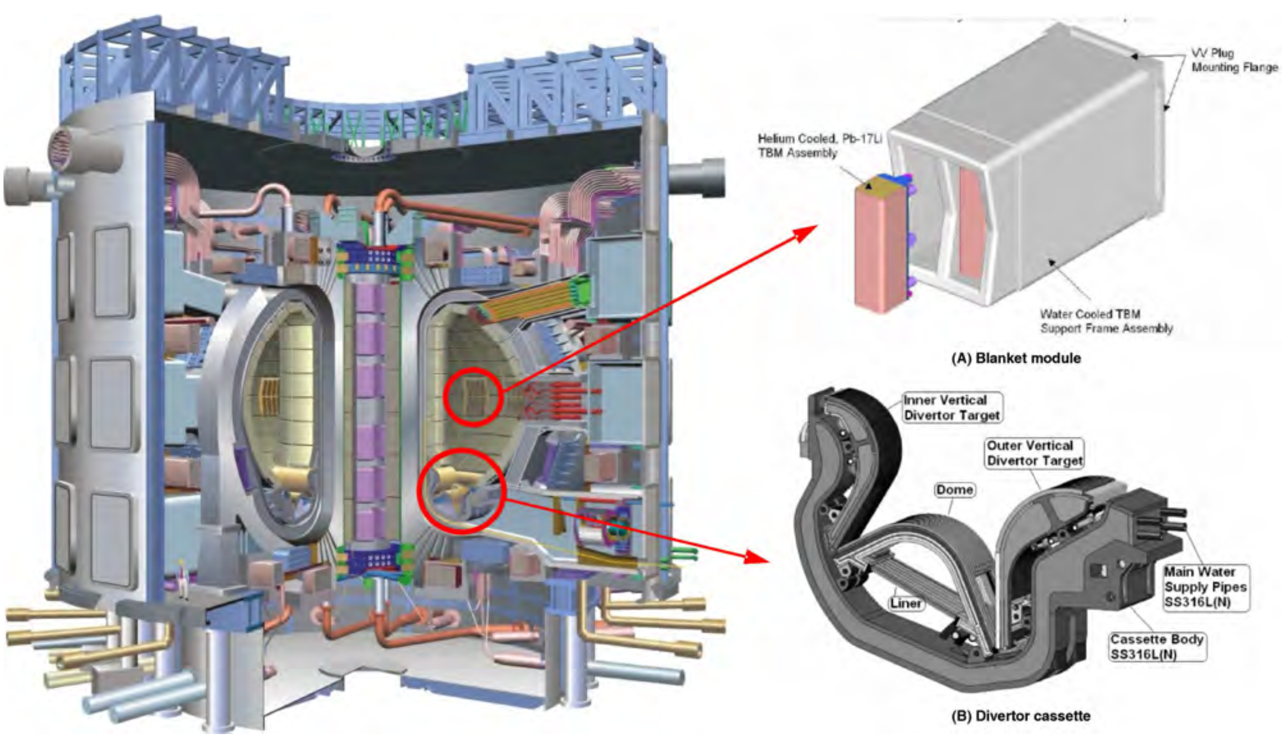

Fig. 1: (A) Cutaway view of ITER model showing the glimpse of various sub-systems such as first wall, divertor, breeder blanket module, superconducting magnetic coils, vacuum vessel etc. (B) gives a magnified view of the test blanket module (C) magnified view of the divertor cassette assembly [Ref: Suri *et al.* (2010)]

Table 1: Generic information about subassemblies and their functional objectives [Ref: Baluc *et al.* (2007)]

	First wall	Divertor	Breeder Blanket
Functional objectives	To shield subassemblies from thermal loading and plasma exposure	To extract particles / He dust, survive high heat flux	To breed T fuel, utilize 14.1 MeV neutrons, shield sub-assemblies from neutrons, extract heat for electricity generation
Plasma facing materials	W, W-based alloy, W coated SiC, Be, W coated ODS/ RAFM steels, flowing liquid Li	W-based alloy (ODS-W etc.), W-coated SiC_f/SiC; flowing liquid metal: Li, Ga, Sn, Sn-Li	As in the first wall
Neutron multiplier	—	—	Be, $Be_{12}Ti$, $Be_{12}V$, Pb
Tritium breeding material	—	—	Liq. Li, Eutectic Pb-Li, Li based ceramic pebbles (Li_2O, Li_4SiO_4 + 2.5%SiO_2, Li_2TiO_3, Li_2ZrO_3, $LiAlO_2$)
Structural material	RAFM steel, ODS steel, V-based alloy, SiCf/SiC	ODS steel, W-based alloy	RAFM steel, ODS steel, V-based alloy, SiCf/SiC
Coolant	—	Water, He	Water, He, Eutectic Pb-Li, Li

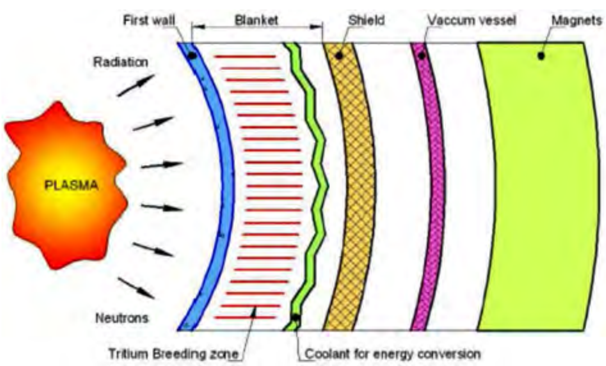

Fig. 2: A schematic diagram of arrangement of sub-systems in a tokamak reactor. [Ref: Suri, Krishnamurthy *et al.* (2010)]

the hot plasma is known as the ''First wall'' and the assembly is often referred to as plasma-facing components (PFC), whereas divertor is a device (or assembly) that allows the online removal of material and He ash from plasma. The first wall surface faces impact of energetic particles leading to erosion of the surface and irradiation damage, and leads to trapping of D or T in re-deposited layers of eroded species leading to a radioactive inventory buildup in the reactor. The impinging energetic particle spectrum varies across the poloidal circumference of the tokamak reactor, and hence different materials can be used depending on the particle energy and flux at a given location in the reactor. Fig. 3 as reported by Reith and co-workers (Rieth *et al*. 2013) provides an idea about the first wall and blanket assembly. As indicated in Fig. 3, area (1) indicates the first wall typically associated with the plasma facing side of the blanket assembly, area (2) indicates the plasma

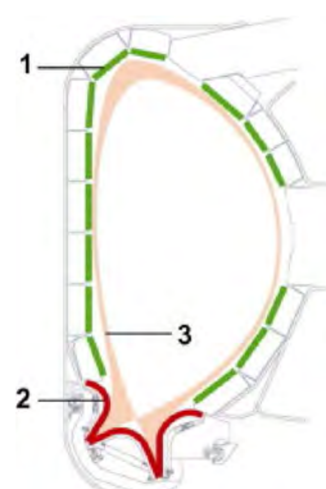

Fig. 3: The illustration shows the cross-section of a tokamak. (1) The plasma facing part of the blanket boxes – so called first wall. (2) The high heat flux cooling layout of the divertor. (3) The magnetic field lines which direct exhaust particles (mainly He) to divertor target plates [Ref: Rieth *et al.*(2013)]

facing side of the divertor and area (3) indicates the magnetic field lines which exhaust the exhaust particles to divertor target plates. The PFCs have to face high heat flux (<1 MW/m^2 for blanket first wall and up to 20 MW/m^2 for divertor target plates), suffer from erosion of PFCs due to particle impingement. Reaction of D or T with eroded particles also poses a threat of radioactive dust.

The selection of material for the first wall is critical. On erosion, low Z materials are fully ionized in the central plasma and only radiate bremsstrahlung, the high Z elements still have bound electrons that emit line radiation which leads to strong plasma cooling (Schmid and Roth, 2010). On the other hand, low Z materials (viz., C, Be, etc.) have higher erosion rates than high Z elements (viz. W, Mo, etc.). The ITER first wall with a heat flux <1 MW/m^2 considers Be or W as plasma-facing material. One of the reasons attributed to replacing carbon-based PFCs to W-based PFCs is the reduction of hydrogen or tritium retention (Philipps 2011). However, under high hydrogen fluence and low temperatures (<600 K), blistering of W has been observed (Neu, 2010). Be poses health hazards and hence is still under evaluation stage. This W can either be coated on the structural material of first wall (RAFM steel) by thermal spray technique (e.g. vacuum plasma spray) or can be fixed as a solid tile. The fabrication process (thermal spray coating or fabrication of solid tile block) for affixing on the structural component of the first wall is being investigated globally.

The divertor assembly (Fig. 4) (Griffith, 2008) on the other hand considers both low Z and high Z materials. The main function of divertor assembly is to remove the scrape off layer (SOL) (mainly He dust), shield subsequent sub-assemblies from very high heat fluxes (10-20 MW/m^2) and extract heat out of the divertor. W and Carbon Fibre Composites (CFCs) have been the preferred choice of materials for such divertor target plates in ITER. In order to improve the erosion resistance, oxide dispersion strengthening concept is being explored in target plates or first wall components and powder metallurgical processing routes are reported to be promising. (Rieth *et al.* 2013) has reported development of ODS-tungsten alloys

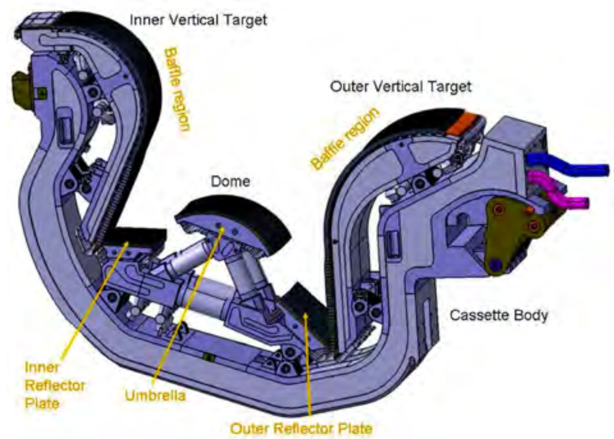

Fig. 4: Illustrative diagram of ITER divertor cassette assembly. Image emphasizes the arrangement of plasma facing components and the cooling arrangement provided to extract the heat. Multiple casettes together will be assembled in the torus to form a divertor of reactor [Ref: Griffith (2008)]

such as W-2Y_2O_3, W-2La_2O_3 etc. in Europe, while W-La_2O_3, W-TiC, W-TaC and W-Y_2O_3 are being reported to be developed in China (Yan *et al*., 2013). Fabrication processes such as wet chemical method, mechanical alloying, sintering of pure and doped W by spark plasma sintering, resistance sintering under ultra-high pressure are being pursued. Properties of such W-based PFCs are also being studied through embrittlement studies, blistering, hydrogen retention, radiation damage and high heat flux testing. Another challenge to divertor design is the fabrication of an appropriate heat sink along with W armour.

The heat from the armour tile has to be extracted, for which a heat sink made of CuCrZr – a precipitation hardened copper alloy has been reported as candidate material (You *et al*., 2013; Rotti *et al*., 2014). An SS 316 (LN) grade pipe brazed through CuCrZr alloy shall circulate water for cooling and heat extraction purpose. The joining of this CuCrZr alloy with refractory armour material such as W is a challenge owing to the large coefficient of thermal expansion (CTE) difference. Vacuum brazing of W/CuCrZr and C/CuCrZr has been found successful as a joining technique. W/Cu tile fabrication using oxide free high conductivity (OFHC) Cu casting in vacuum followed by brazing of W/Cu with CuCrZr has been found promising. Studies on functionally graded coatings of

W-Cu on CuCrZr alloys are being pursued as a promising solution. One of the limitations of W/Cu functionally graded materials (FGMs) is loss of strength at elevated temperatures due to presence of Cu. As a solution to this, recent studies (You *et al*., 2013) have been reported on W/CuCrZr FGMs. Porous W skeletons infiltrated with CuCrZr alloy melt was prepared in the form of functionally graded composite as promising materials for divertor applications. Graded composite samples of W/CuCrZr in 1:1 ratio were reported promising for further development. Apart from this, novel concepts such as liquid metal (Li) plasma-facing components are being evaluated at Princeton Plasma Physics Laboratory. Other concepts involve vaporization of Li such as tungsten alloy with Li, wherein lithium would get evaporated and would thus be able to sustain the heat fluxes (Wong *et al*. 2001). Areas such as tritiated water corrosion of SS316 (LN) tubes are also important as the radiolytic and decomposition products enhance the corrosion rates (Bellanger, 2008). Further, presence of tritium would add to possible stress corrosion cracking issues, which need to be studied and mitigated.

2.2 Test Blanket Module

The fusion reactor programme is driven by the ultimate goal of developing large-scale power plants for production of electricity. The success of a fusion power plant is dependent on the high-grade heat extraction capability and efficiency of the tritium breeding blankets (Kumar *et al.*, 2008). A blanket module as the name suggests is an assembly which blankets the core of the fusion reactor and utilizes the energy of neutrons to generate T fuel and heat for electricity production. Thus, the main function of a breeder blanket module is to generate T fuel from Li by utilizing the 14.1 MeV neutrons generated from reactor core; to extract the heat for electricity generation purpose generated from the n-Li reaction and shield the other sub-systems from the radiation damage by the energetic neutrons. This can be achieved by appropriate design concepts, selection and or development of appropriate breeder materials and their processing techniques and the choice of appropriate structural material with relevant compatibility with breeder material. Since the T can be generated from reacting the neutrons with ^{6}Li (see eq. ii), all the breeder materials are Li or Li-based compounds or alloys. Based on the form of Li, the blanket design can either be categorized as liquid breeder blanket, solid breeder blanket or mixed type. Different blanket module designs with solid or liquid or dual breeder breeder concepts have been proposed by different countries which are summarized in Table 2. The ITER reactor will be useful for testing and validating different blanket concepts as proposed by partner countries and the performance information will be helpful for development of a DEMO relevant blanket module design.

Both the solid and liquid breeder concepts have their pros & cons. The main advantage of the solid breeder is that it offers good compatibility between breeder, structural material and the coolant. However, one of the major drawbacks of solid breeder concept is the costly fabrication and re-processing of the ceramic breeder material. Against this, the liquid breeder concept offers efficient heat & fuel extraction, and easy maintenance. A general comparison between the two breeder blanket concepts is provided in Table 3 (Bornschein *et al*., 2013). The Indian Lead Lithium Ceramic Breeder (LLCB) blanket involves use of both solid and liquid breeder concepts. The conceptual sketch of the Indian TBM (LLCB) as indicated in Fig. 5 (Rajendra Kumar *et al*., 2012), illustrates the arrangements of the ceramic breeder columns and liquid breeder (Pb-17Li) flow channels. The flow of Pb-17Li in the RAFM steel channels results in severe corrosion of RAFM steel. (Krauss *et al*., 2012) reported a dissolution rate of RAFMS at the rate of ~400 µm/ year which is approximately equivalent to 4 kg/m^2 per year of TBM-dissolved corrosion products. Such dissolution of structural material would not only lead to possibility of section thinning and leak out, but also poses a threat of chocking of the flow channels due to re-deposition of dissolved corrosion products at colder zones. Apart from corrosion, another major concern is the T permeation into steel. The T generated from Li during the Pb-17Li flow path or from the ceramic pebble bed channels would permeate into the steel and thereby increase the radioactive T

Table 2: Overview of different breeding blanket concepts

TBM design concepts	Country	Brief outline of the blanket design
Helium Cooled Lead Lithium (HCLL)	EU	He & PbLi as coolant and breeder; RAFMS (Eurofer '97) as structural material
Dual Function Lithium Lead (DFLL) – He cooled quasi-static lithium lead (SLL)	China	He as coolant; self-cooled PbLi in quasi-static condition as breeder; RAFMS (CLAM steel) as structural material.
Dual Function Lithium Lead (DFLL) – Dual cooled Lithium Lead (DLL)	China	He/LiPb as coolant & breeder; RAFMS (CLAM steel) as structural material;
Lead Lithium Ceramic Breeder (LLCB)	India	He as coolant; self-cooled PbLi as breeder; Li_2TiO_3 as solid breeder; RAFMS as structural material
Helium Cooled Liquid Lithium	USA	He as coolant; self-cooled liquid Li as breeder; RAFMS with SiC_f-SiC inserts as structural material
Lithium Vanadium (Li-V)	Russia	Self-cooled lithium as coolant cum breeder; Vanadium alloys as structural material
Helium Cooled Molten Lithium (HCML)	Korea	He as coolant; molten Li as breeder; RAFMS (Eurofer) as structural material
Water Cooled Solid Breeder (WCSB)	Japan	Li_2TiO_3 as solid T breeder; Be pebbles as neutron multiplier; He and water as coolants; SiCf/SiC inserts; F82H (RAFMS) as structural material

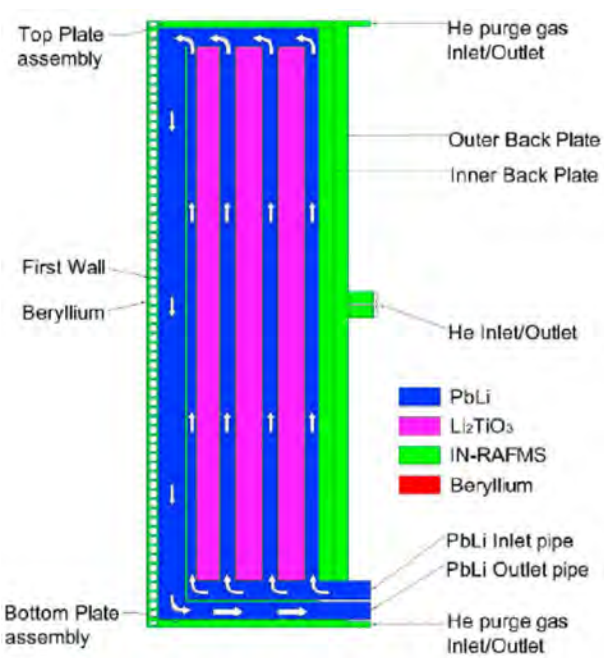

Fig. 5: Schematic diagram of Indian TBM (Lead Lithium Ceramic Breeder) concept. The arrows eutectic breeder while the columns in between (pink colour) represent solid Li_2TiO_3 breeder columns. [Ref: Rajendra Kumar *et al.* (2012)]

inventory buildup in the reactor, which is undesirable. The interaction of magnetic field B with the flowing liquid metal also induces a magneto hydrodynamic drag (MHD), which increases the corrosion rates of steel as well as increases the required Pb-17Li pumping pressures. Coatings have been reported to be an inherent solution to all the three major issues of blanket modules. Such coatings will need to be compatible with Pb-17Li; resist permeation of T into steel with permeation reduction factor >100; provide electrical insulation for mitigating MHD issues and can also be coated on complex geometries (Smith *et al.*, 2002). The coatings should have low radiation induced conductivity after neutron irradiation. A variety of coatings such as AlN, Al_2O_3, Er_2O_3, Y_2O_3, CaO, AlN, $(Cr_2O_3+SiO_2)$ + $CrPO_4$, ZrO_2, Al_2O_3+FeAl, etc. have been explored for different blanket concepts as reported in literature survey (Jamnapara 2013). Alumina and erbia coatings have been widely reported as candidates for the ITER reactor, wherein RAFM steel is considered as structural material. The structural materials are an integral part of the blanket module and its development and testing are inevitable. Studies on structural materials in blankets have been briefly spelt out in section 2.3.

Table 3: Comparison of solid and liquid breeder design concepts [Ref: Bornschein *et al.* (2013)]

	Solid breeder	Liquid breeder
Breeder material	Ceramics: LiO_2, $LiAlO_2$, Li_2SiO_3, Li_4SiO_4, Li_8ZrO_6, Li_2TiO_3	$Li_{17}Pb_{83}$, Flibe (LiF, BeF2)
Neutron multiplier	Be, $Be_{12}Ti$	Pb, Be
Coolant	He cooled, Water cooled	He cooled, water cooled, self cooled, dual cooled
Structural material	RAFM steel	RAFM steel
Advantages	Tritium extraction less challenging	No breeder damage or swelling; adjustable breeder composition
Difficulties	Blanket replacement, tritium permeation into coolant	MHD drag, corrosion, tritium permeation into coolant

2.3 Structural Materials

Structural materials for power plants are derived from existing high strength materials used for extreme environments. Materials and alloys having reduced activation elements (to reduce radioactive inventory), high resistance to creep, fatigue, low ductile to brittle transition temperatures, and capable of being used at higher temperatures under neutron irradiation are desired for fusion reactors (Baluc, 2006). After more than two decades of research, only three candidate material systems appear to have the potential to meet the low activation, high performance goals: 8-9 Cr ferritic/martensitic steels (including reduced activation variants), SiC_f/SiC composites and V-Cr-Ti alloys (Bloom *et al*., 2007).

2.3.1 Effect of Neutron Irradiation

Irradiation damage occurs due to impingement of high energy particles viz. electrons, ions, neutrons, protons on the atoms of structural or functional materials, wherein the atoms are displaced from their regular lattice positions yielding Frenkel defects viz. vacancies and interstitials (Ronald and Klueh, 2001). While a few tens of eV are required to displace an atom, the neutrons in fusion reactors with 14.1 MeV energy will create significant radiation damage in the irradiated material. The extent of displacement damage is expressed in terms of how often an atom is displaced from its normal lattice position during irradiation as displacement per atom or dpa. In addition to the displacement damage, neutrons also cause transmutation reactions with atoms of the irradiated material. Such transmutation products may either be another metal atom or gas atoms viz. helium and hydrogen by (n, α) and (n, p) reactions, respectively. It is well-known that the production of small amounts of He within lattice may have pronounced effects on material properties. Vacancies become more mobile for irradiation above $0.3T_m$, and result in formation of dislocation and cavity. Swelling of irradiated components occurs due to cavities formed by dissolved gases. As a result, the combination of gaseous transmutant products and radiation damage has to be monitored closely in the form of He/dpa ratio. An overview of the defect production in steels for different irradiation facilities has been reported by Baluc *et al*. (2007) and listed in Table 4.

Overall, it can be said that radiation damage leads to issues such as radiation-induced segregation, radiation hardening and embrittlement, phase instabilities due to radiation-induced precipitation, irradiation creep, volumetric swelling due to void formation, and high temperature He embrittlement (Tavassoli, 2002; Baluc *et al*., 2011).

2.3.2 Reduced Activation Steels

The safety and environmental concerns in fusion reactors involve radioactivity in blanket and first wall structures. The radioactivity of the exposed fusion reactor components should be short-lived and hence, the structural material selection should involve low activation or reduced activation elements, specifically

Table 4: Defect production in steels for various irradiation facilities [Ref: Baluc *et al.* (2007)]

Defect production (in steels)	Fusion neutrons (3-4 GW reactor, 1st wall conditions)	Fission neutrons (BOR 60 reactor)	High energy protons (590 MeV proton accelerator)	IFMIF (high flux test module)
Damage rate (dpa year^{-1})	20-30	~20	~10	20-55
Helium (appm dpa^{-1})	10-15	=1	~130	10-12
Hydrogen (appm dpa^{-1})	40-50	=10	~800	40-50

considering the high energy of fusion neutrons. The term ''low activation material'' is often used for those which allow hands-on maintenance and minimize waste disposal. In general, fusion relevant alloy development involves resistance to irradiation effects and higher temperature performance ability including structural and functional requirements.

Many studies were conducted in 1980 on type 316 austenitic stainless steels, including effects of fusion relevant He production and displacement damage on properties and microstructural stability (Zinkle, 2005). Austenitic stainless steels are reported to have higher swelling due to He bubbles generated from Ni transmutation reaction than ferritic martensitic steels. The alloy development thus shifted to high strength Fe-Cr-based steels with 2¼Cr to 12Cr steels with different alloying elements for purpose of microstructure strengthening (Klueh and Harries, 2001). The strengthening mechanism was the formation of carbides of V, Cr, Nb, etc. at the prior austenite grain boundaries and within the grains (on lath boundaries). Typical composition for ferritic martensitic alloys being developed in India for fusion reactor applications is listed in Table 5. Elements with long half-life transmutants such as Ni, Mo, Nb, Cu, Co, Al, N, etc. have been replaced by low activation counterparts viz. Mn, W, Ta and C so that the steels can be safely handled after a shorter cooling period as against cooling time of 1000 years required for conventional Cr-Mo-Ni-Nb containing ferritic martensitic steels (Saroja *et al.*, 2011). Such steels are produced by vacuum induction melting (VIM) followed by vacuum arc refining (VAR) method. Vacuum techniques are used since undesirable elements having high vapour pressure are readily volatilized during vacuum melting. A typical RAFMS microstructure as indicated in Fig. 6 involves lath martensite boundaries packed in a prior austenite grain. The $M_{23}C_6$ type carbides are observed to be segregated on prior austenite grain boundaries while the MX type precipitates are present on lath boundaries within prior austenite grains, and such precipitates render strengthening effect to the steel structure. An advancement of this alloy is the oxide dispersion strengthened (ODS) ferritic martensitic steels (Paúl *et al.*, 2005). Particle reinforcement is one of the reliable strengthening mechanisms of FM steels. In such steels, particle reinforcement is provided by addition of minor quantities of nano-sized yttria (Y_2O_3) powders. Such ODS steels can be manufactured by

Table 5: Composition of Indian RAFM steel [Ref: Saroja *et al.* (2011)]

Element/ steel	Indian RAFM	Eurofer 97	Element/ steel	Indian RAFM	Eurofer 97
Cr	9.04	8.5-9.5	B	0.0005	<0.001
C	0.08	0.09-0.12	Ti	<0.005	<0.01
Mn	0.55	0.2-0.6	Nb	0.001	<0.001
V	0.22	0.15-0.25	Mo	0.001	<0.005
W	1.00	1.00-1.20	Ni	0.005	<0.005
Ta	0.06	0.05-0.09	Cu	0.001	<0.005
N	0.0226	0.015-0.045	Al	0.004	<0.01
O	0.0057	<0.01	Si	0.09	<0.05
P	0.002	<0.005	Co	0.004	<0.005
S	0.002	<0.005	As+Sb+ Sn+Zr	<0.03	<0.05

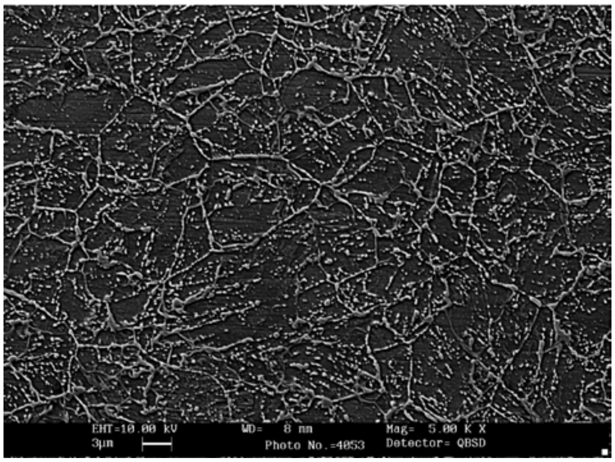

Fig. 6: Microstructure of RAFM steel observed under SEM. Note the carbide precipitation around prior austenite grain boundaries. Minor precipitates within prior austenite grains are indicative of lath martensite boundary

powder metallurgical route followed by hot isostatic pressing to compact shapes.

2.3.3 Vanadium Alloys

In a blanket module, liquid metal flow through structural material and presence of ceramic breeder and neutron multiplier need to be handled through the right choice of candidate material. Vanadium alloys are such that they not only have excellent compatibility with Li, but also provide room for high operating temperatures and eliminate the need of ceramic breeders or neutron multipliers (Chen *et al*., 2011). Vanadium alloys are one of the candidate materials for structural components of blanket modules in the fusion reactor owing to their high temperature strength, high thermal stress factor, low long-term activation, excellent compatibility against Li, etc. The vanadium alloy with V-4Cr-4Ti type composition has been developed for fusion blanket applications by US, Japan, Russia and China (Muroga *et al*., 2002; Chen *et al*., 2011). There are two ways to improve the strength of V-alloys; one of the methods involves increase in amount of alloying elements such as Cr, etc. Addition of Al and W as alloying elements in V-4Cr-4Ti alloy has also been known to enhance the mechanical properties of the alloy. The other strengthening mechanism is to produce fine-grained and particle-dispersed V-alloys using mechanical alloying.

It is important to note that the V-Cr-Ti alloy will endure severe hardening and loss of strain hardening capability by neutron irradiation below 400°C. Also the mechanical strength of reference V-4Cr-4Ti alloy is not as high as FM steels at <~600°C, although it has higher strength at higher temperature up to ~750°C. Efforts are being made to expand the operating temperature window and improve the mechanical properties as well. Such V alloys are specifically considered candidates for Li-V blanket concepts where liquid Li is flowing up to 700°C in V-alloy channels with corrosion rates as low as 7.5 μm/year. V-alloys are thus an attractive candidate for advanced self-cooled fusion blanket concepts such as Li/V. However, issues such as irradiation creep and He embrittlement behaviour still remain unsolved.

3. Status of Materials Development in India

Institute for Plasma Research (IPR), Gandhinagar initiated the plasma physics programme in 1982 and developed ‘‘*Aditya*’’ which is India’s first tokamak system. Subsequently, IPR started developing a steady state tokamak (SST-1) which is under final stages of completion. With passage of time, as India participated in ITER programme (November 2006), the activities towards technological development for fusion reactor were accelerated. Many activities pertaining to development of fusion reactor materials have been initiated at Indira Gandhi Centre for Atomic Research (IGCAR), Bhabha Atomic Research Centre (BARC) and Institute for Plasma Research (IPR).

Research on first wall and divertor development activities at IPR involves materials development and processing studies of W, W-based alloys, CFC, CuCrZr alloy, their joining studies and the performance tests such as high heat flux testing (Khirwadkar *et al*., 2011; Singh *et al*., 2011; Patil *et al*., 2013). Reduced activation ferritic martensitic steels have been developed for the Indian TBM programme as per the composition specified in Table 5 (Saroja *et al*., 2011). Physical metallurgy of the developed Indian RAFMS has been reported by a few studies (Raju *et al*., 2009; Raj *et al*., 2010). Studies related to the development of blanket relevant materials such as RAFM steels, ceramic breeders (Li_2TiO_3), compatibility studies of liquid breeder with structural materials under magnetic field, neutronics studies and development of simulation codes, welding and joining techniques, etc. have been reported (Rajendra Kumar *et al.*, 2012). Activities related to development of alumina-based coatings (Al_2O_3 + FeAl) for blanket applications have been extensively studied and developed by the authors (Jamnapara *et al*., 2012a; 2014a). Attempts have been made to explore the insulation properties of alumina films grown by thermal and plasma processing. Plasma grown alumina films on FeAl surfaces had been found to yield improved dielectric properties as compared to thermally grown alumina films (Jamnapara, 2015a). In continuation to this, α-Al_2O_3 + FeAl-based coatings on P91 steels have been successfully generated using hot dip aluminizing followed by normalizing and tempering treatments using plasma oxidation process. A novel patented concept of using plasma as an oxidation tool has been developed so as to accelerate the transformation of metastable θ-Al_2O_3 to α-Al_2O_3 (Jamnapara *et al*., 2015b). The compatibility of such alumina coatings generated by thermal processing and plasma oxidation treatment against Pb-17Li have been conducted at 550°C for 1000 hours under static mode (Jamnapara *et al*., 2014b). The plasma processed stable α-Al_2O_3 has been found to be immune against Pb-17Li for 1000 hours duration with no weight loss, while the bare P91 substrate revealed 7.23 mg/cm^2 weight loss which was ~7 µm of substrate degradation as confirmed by SEM-EDS. Further tests under dynamic Pb-17Li conditions are being planned in near future. FeAl-based coatings for welded areas of blanket structures using electrospark deposition technique have also been explored on a preliminary level by the authors (Jamnapara *et al*., 2012a). Interesting features of a quasi-amorphous interface without any grain boundary has been observed which could be a potential barrier for hydrogen permeation. The electrospark deposition technique can also be explored as a possible coating process for first wall applications needing W coatings.

Materials development activities for extreme environments such as nuclear fusion would not be possible without nuclear grade materials testing, characterization and validation facilities. Neutron irradiation sources equivalent to 14.1 MeV are not available and need to be set up. An International Fusion Materials Irradiation Facility (IFMIF) is being conceptualized (Garin and Sugimoto, 2008) and planned for validation trials of materials to be qualified for fusion environment. The IFMIF is being planned by EU, Japan, Russia and USA; and being managed by the International Energy Agency. The primary mission of IFMIF is to generate a materials database to be used for design and construction of various components of DEMO-type reactors. IFMIF will be an accelerator-based, high-energy neutron source mainly composed of two 125 mA deuteron accelerators and a flowing liquid Li target (Baluc *et al.*, 2011). In the high flux test module of the IFMIF, the irradiation conditions will be very close to the ones expected in a DEMO type reactor at the level of first wall, at least in terms of damage rate (22-55 dpa/year) and rates of He production (10-12 appm dpa^{-1}) and hydrogen (40-50 appm dpa^{-1}). In addition to static material interaction in the high flux test module, more sophisticated *in situ* creep-fatigue tests on structural materials and *in situ* tritium release experiments on different tritium breeding materials are foreseen in medium flux test module, where a damage rate of 1-20 dpa/year will be reached (Baluc *et al.*, 2011). India will need to actively participate in such international facility to build a domestic fusion reactor.

4. Indian Fusion Reactor Programme

The energy requirement of India is expected to grow by almost 10 times of the present requirement in the next 50 years. Out of the installed capacity of 120 GWe power, 95.5% is produced through thermal and hydro while just 2% is produced through nuclear energy. With the introduction of the new concepts of fission technology in India viz. prototype FBR and thorium reactors for near future, fusion is viewed as natural advanced successor technology to fission for producing large amounts of electricity (Srinivasan and Deshpande, 2008). The Indian plasma physics programme started in 1982 resulting in the inception of the Institute for Plasma Research (IPR) at Gandhinagar, Gujarat, India. India's first tokamak device "Aditya" was demonstrated in 1989. Later on, the Steady State Tokamak-1 (SST-1) programme was initiated at IPR. The SST-1 machine (designed for hydrogen plasma) is mainly focused on studies related to plasma physics and supporting instrumentation and subsystems. With India's participation in the ITER project, the D-T fusion reactor technology will be enhanced and would be useful for India's domestic fusion reactor programme. The Indian TBM programme has aggressively progressed with the joint efforts of IPR, IGCAR, BARC and other organizations in India. The Indian TBM concept is to be tested in one half port of ITER reactor at Cadarache, France.

As a roadmap towards the Indian fusion programme (Rajendra Kumar, 2012), India has proposed a next stage of SST-1 machine: a D-T machine named Stead State Tokamak-2 ('SST-2'). SST-2 will be a medium-sized tokamak reactor with D-T operation. SST-2 operation will enable the testing of various indigenously built subsystems of the fusion reactor, specifically addressing tritium breeding and handling issues, robotics and remote handling, alpha particle issues, fusion materials development, etc. Further, the SST-2 will address the availability of a machine for breeding blankets and testing and validation of novel materials developed for DEMO applications. The SST-2 will become the stepping stone towards realization of a DEMO reactor.

5. ITER Project and India's Role

The first conceptual design of ITER device was started in April 1988. A revised design was finalized in 2001 by the ITER Council. Owing to the domestic steady state tokamak programme, India joined ITER in December 2005 and the ITER agreement between seven partners was signed in November 2006. Thereafter, the ITER design and construction work has been in progress. It was projected (Holtkamp, 2009) that ITER would initiate its first plasma shot in 2018. The divertor which was initially a carbon-based (CFC) divertor has now been confirmed as full tungsten-based divertor (Merola *et al*., 2014). This is because carbon is not permitted during nuclear use owing to the potential risk of rapid generation of tritium dust. Thus, the replacement of a CFC divertor by a full W divertor is due to safety reasons as well as cost reduction and maintenance. In 2009, the ITER Council created a high-level advisory body, ''The TBM Programme Committee'', was appointed to ensure the on-time delivery of the test blanket systems at the ITER site (Giancarli *et al.*, 2012). The test blanket module system concepts proposed by the seven partners of ITER underwent a final design review in 2013 and is now entering the engineering and procurement phase (Merola *et al.*, 2014).

India is one of the seven partners and shares some cost of components of ITER. A schematic view of the in-kind contributions of the seven partners is shown in Fig. 7, wherein India's contribution has also been indicated (Kaname, 2010). 'ITER India', is a project governed by an empowered board under the Department of Atomic Energy, Government of India, and is committed to the delivery of the components to be provided by India as ''in-kind contribution''. The contribution to TBM for ITER excludes the contributions shown in Fig. 7.

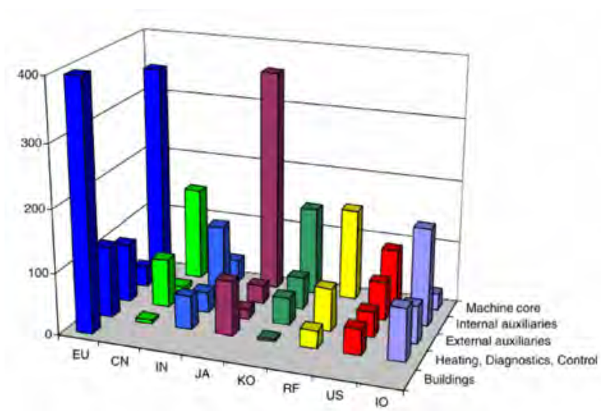

Fig. 7: Contribution of ITER partners to ITER [Ref: Kaname (2010)]

Conclusions

The fusion reactor programme has taken a large leap owing to the increasing energy demands and the limited reliability of energy sources. The extreme environments in fusion reactors pose a challenge to the materials research and development community, which is being worked upon aggressively. Without appropriate materials, we cannot expect a fusion reactor. While most of the materials research is being focused on first wall, divertor, breeder and structural materials, appropriate thrust is also being made on setting up of fusion grade testing and validation facilities. With India's domestic power demand, it would need to strengthen its domestic fusion programme as next generation energy source. Materials research and development activities are thus inevitable for a better tomorrow.

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