Advanced Materials and Fabrication Challenges of Fusion Reactors

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Abstract

With the achievement of scientific feasibility of controlled fusion of isotopes of hydrogen in a 'hot' plasma under magnetic as well as under inertial confinement, a distinct possibility has emerged to employ thermonuclear fusion for the production of electricity. To make this dream come true, international effortsin a coordinated and co-operative manner are presently being made to build ITER – the International Thermonuclear Experimental Reactor – to test, in this first step, the concept of 'Tokamak' for net fusion energy production. To exploit this new developing option of making energy available through the route of fusion, India too embarked on a robust fusion program under which we now have a working tokamak - the Aditya and a steady state tokamak (SST-1), which is on the verge of functioning. The program envisages further development in terms of making SST-2 followed by a DEMO and finally the fusion power reactor. Further, with the participation of India in the ITER program beginning in 2005, and recent allocation of half – a – port in ITER for placing our Lead – Lithium cooled Ceramic Breeder (LLCB) Test Blanket Module (TBM), meant basically for breeding tritium and extracting high grade heat, the need to understand and address issues related to materials for these complex systems has become all the more necessary. Also, with increasing power from the SST stages to DEMO and further to PROTOTYPE, the increasing demands on performance of materials would necessitate discovery and development of new materials and processes for fabrication of complex components from these.

Due to the 14.1 MeV neutrons that are generated in the D+T reaction exploited in a tokamak, the materials, especially those employed for the construction of the first wall, the diverter and the blanket segments, suffer crippling damage due to the high He/dpa ratios. To meet this challenge, the materials that need to be developed for the tokamaks are steels for the first wall and other structural components, copper alloys for the heat sink, and beryllium for facing the plasma. For the TBMs, the materials that need to be developed include beryllium and/or beryllium-titanium alloys for neutron multiplication, lithium-bearing compounds for tritium generation, and the liquid metal coolants *like lead-lithium eutectic that combine the functions of both the neutron multiplier and the tritium breeder. The other materials that need attention include superconducting materials like NbTi, Nb_3Sn and $Nb_3Al$for the tokamaks, ceramic coatings on structural materials (or, inserts) to offset the effect of corrosion and the MHD drag in liquid metal cooled TBMs, and a host of other materials, alloys and compounds.*

This paper deals with the issues related to development, characterization and qualification of both the structural as well as the functional materials required to carry forward the challenging task of harnessing fusion energy in engineered systems. The challenges being faced to manufacture complex components from these materials also form the subject matter of this paper.

1. Introduction

The global requirement of energy is ever increasing and more critically so in developing economies like India. Out of the various options available today to exploit fossil fuels for producing electricity, we are able to harness only a small fraction of our requirement because of the constraints on the quality and quantity of the fossil fuels and other required materials. Furthermore, the need to control the damage to climate due to carbon dioxide emissions, inevitable in fossil fuel use, has prompted serious and urgent searches for alternative potentially cleaner and sustainable sources of energy. The renewable sources, at the existing level of technology, are difficult to exploit to meet the ever increasing demand. Nuclear fission, a huge source of energy, is already being exploited increasingly even though it is saddled with apparently intractable radio-active waste disposal and proliferation concerns. Making energy available through the route of fusion of light elements is the option that is now attracting increasing attention. This is because the thermonuclear fusion of light elements - the source of light and heat of *the Sun* and *the Stars* - if exploited on earth, will not only be a cleaner but also, a potentially infinite source of energy. It was discovered that, formidable though this task is, it can be accomplished in a number of ways. Each of the methods involves, in the first step, creation of plasma, a fully ionized state of matter. The plasma is then heated to temperatures of hundreds of millions of degrees and '*confined* ' in this hot condition, either magnetically or inertially, for times long enough for 'thermonuclear fusion' to occur. Extremely intense lasers are employed to cause a micro-explosion, similar to that in a hydrogen bomb, in a tiny pellet of frozen mixture of deuterium and tritium to achieve thermonuclear fusion in the concept based on the inertial confinement of the plasma. In the concept based on the magnetic confinement of the plasma, fusion of deuterium and tritium, the hydrogen isotopes, can be achieved by using a device called *theTokamak*, an acronym for the Russian name 'toroidalkamera (chamber) magnet and katuschaka (coil)'. It has taken over half a century of effort to achieve the scientific break-even in thermonuclear fusion by either of the methods of confinement of the hot plasma. The engineering break-even is yet to be achieved. The current scenario indicates that it may only be a few years away but once it is achieved, potentially infinite source of clean energy will become available. Recognizing this in time, India has launched [1] a fusion research program by establishing the Institute for Plasma Research at Gandhinagar, Ahmedabad. The Institute is now a constituent unit of the Department of Atomic Energy. Under this program, India has already a working tokamak, *Aditya*, a machine for '*Plasma Physics*' research. The program has further progressed and a steady-state Tokamak-1 (SST-1) is on the verge of functioning. The program envisages further development in terms of making a SST-2 by 2022, followed by a DEMO by 2037, and finally a fusion power reactor by 2050. Further, in the recent past, India joined as an equal partner in the International Thermonuclear Experimental Reactor (ITER) program at Cadarache, France, in which the commitment of the country lies in delivering a large inventory of sophisticated components. India has also beenallotted [2] half-a-port for placing a Lead – Lithium cooled Ceramic Breeder (LLCB) Test Blanket Module (TBM) in ITER.The primary objectives of placing this TBM in ITER are to test its capability to (i) breed tritium with Tritium Breeding Ratio (TBR) >1, and to (ii) extract the high grade heat from the tokamak with acceptable thermal efficiency.

2. Materials for Tokamak

For the construction of the '*Tokamak*' and its '*Blanket*',which, presently, is in the form of Test Blanket Modules (TBMs),a number of materials – both structural and functional – are required to be developed. As they have to face the 14.1 MeV neutrons generated due to the fusion of deuterium and tritium in the *Tokamak*, structural materials have to have the radiation stability apart from other relevant properties and the functional materials the required level of integrity under the operating conditions. Finalizing the specifications for materials, their development, characterization, production and suitable fabrication into components is a major challenge. With increasing power and expected effective life of the tokamaks from the experimental stage to the demonstration stage and then further to the prototype stage, this challenge increases and calls for continued development of superior materials and processes.

Before listing the demands that these devices - the tokamaks and their blankets - would impose on materials for their safe and reliable operation, it may be worthwhile to look at the '*Tokamak*', the D + T reaction that is exploited in it to get the *'net'* energy, and the subject of radiation damage of materials due to the 14.1 MeV neutrons.

As stated already, tokamak is a device that is based on the concept of magnetic confinement of plasma in which a mixture of D and T is burned to obtain the energy. The fusion reaction that takes place may be written as:

$$D+T = {}^{4}He\ (3.5\ MeV) + n\ (14.1\ MeV).$$

The world has been getting closer and closer at achieving its goal of obtaining net power from a tokamak so much so that the D-T plasma could be held for about 5 seconds in the Joint European Torus (JET) at Culham in UK resulting in fusion power of over 15 MW. Similar results were obtained at the Princeton Physics Laboratory in USA. These achievements gave impetus for setting up of a major international magnetic fusion project known as the International Thermonuclear Experimental Reactor (ITER) at Cadarache, France to obtain fusion power of 500 MW (from a 50 MW input) for up to 500 seconds. Seven countries, including India, are participating in it. It will be the first fusion experiment to produce net power – ten times the amount used to heat the plasma andwill allow scientists to explore the physics of burning plasma at energy densities close to that of a commercial power plant. An artist's view of the ITER, is shown below in Fig.1. A schematic view of the materials layout around the plasma appears in Figure 2.

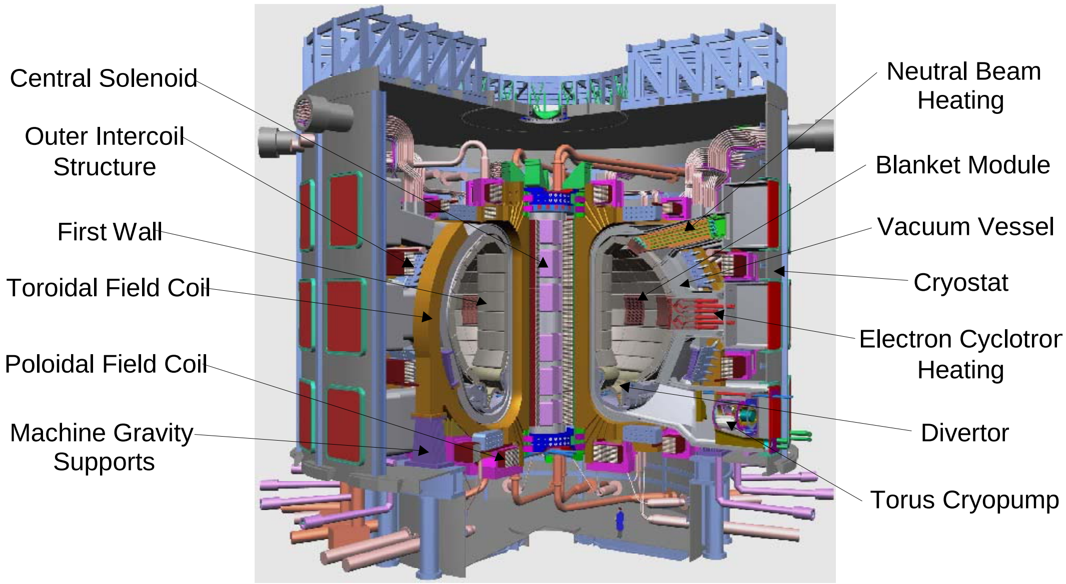

Figure 1. Artist's view of ITER, a 'Tokamak'

The plasma is confined by toroidal and poloidal magnetic fields in the form of a ring in a vacuum vessel that has the shape of a toroid and the heat generated due to the fusion in the plasma is extracted by an appropriate coolant, the He gas and/or a liquid eutectic alloy flowing in the blanket modules that are sitting close to the plasma. The heat is transported to the coolant through the walls of the TBMs by both electromagnetic radiation from plasma and the electrically neutral 14.1 MeV neutrons that escape from the plasma into the walls of the TBMs and the functional materials in TBMs. Also, since the blanket consists of Li^6 either in the form of a ceramic compound or liquid metal (pure lithium or lead-lithium eutectic alloy), it transmutes to tritium by the (n,α) reaction giving rise to additional heat to the coolant. Further, when the 14.1 MeV neutrons escaping from the plasma enter the walls of the TBM, complications arise [2, 3] both due to the radiation damage (displacements and transmutations) of lattice atoms caused by them. Because of the high cross section of these high energy neutrons to cause the (n,α) and the (n,p) reactions with almost all elements, atoms constituting the walls of the TBMs undergo these reactions leading to the formation of both helium and hydrogen in them at high rates causing serious damage to the structural material.

The material behavior at the high He/dpa ratios (dpa, displacements per atom is the unit in which the displacement damage of the lattice is expressed), likely to be encountered by the materials of the first wall of the tokamak as well as the materials in the TBMs, is yet to be completely understood. The challenge to put appropriate structural and functional materials in the tokamak as well as in the blanket module in a configuration to enable these devices to function for the intended time is, indeed, a challenge for the materials scientists.

When the design and construction of the TBMs for even the experimental ITER is considered, the relevance of the points put forward until now becomes further evident.

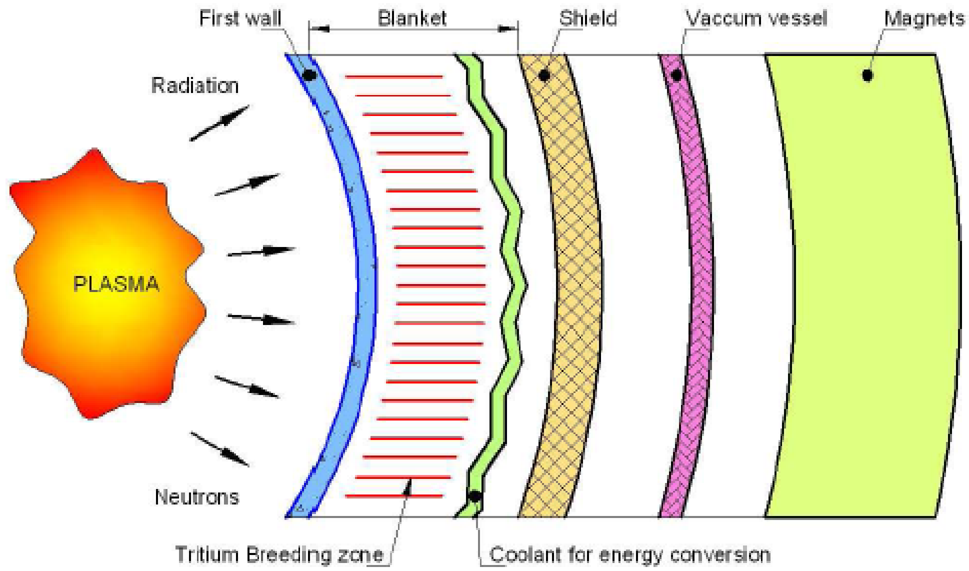

Figure 2.Schematic view of the arrangement of materials in a tokamak.

The first wall of the tokamak is the wall that is nearest to the plasma and, therefore, experiences, the high He/dpa ratios due to the damage caused by the high energy neutrons apart from the high heat flux. The diverter and the limiter also fall in the same category. If material sputters into the plasma, the plasma may get quenched. To avoid this from happening, an element that either does not sputter due to the neutrons (and, occasionally, electrons and other ions from the plasma) hitting it or, else, it does not quench the plasma despite the fact that it sputters, is selected. High Z (atomic number) elements fall in the first category in that they sputter less and the low Z elements, even though they may sputter into the plasma, they are not strong enough to quench it. The selection of the material that has to face the plasma is based on this consideration. Once selected, this element has to be an integral part of the first wall. Next to it in the first wall, especially in the diverter, has to be a material that can act as a heat sink and carrier of heat away from the first wall to avoid its excessive heating. This generally is OFHC (oxygen free high conductivity copper) alloyed with a little bit of Cr (< 1 wt%) to give the Cu the required tensile strength and even a lesser content of Zr (<0.1 wt%) to impart the required fatigue strength. Alternatives available are also listed in table 1. Next to the plasma facing material or to the heat sink (as the situation may demand), and bonded to it, is the structural material, generally a steel. This is the one that actually takes the entire load. Initially, austenitic stainless steel 316 was selected for use as the first wall structural and continues to be material of construction for the first wall of ITER in the form of low activation 316 LN (IG), IG meaning the ITER grade. However, because of its tendency to swell more under irradiation as compared to the ferritic steels and unacceptable fatigue life above 600 ^{0}C, especially with He (generated due to (n,α) reactions) in it, the material of choice for the first wall now for the DEMO reactors is the low activation Ferritic/Martensitic (F/M) steel (FMS), F82H, or, its equivalents.

Table 1. Materials for the First Wall of a Tokamak

First Wall Plasma Facing [5-8]

- Low Z – Be, C-C composites – high sputtering but less quenching
- High Z – W, Mo-based alloys – low sputtering but high quenching

First Wall Heat Sink [9-12]

- Cu-Cr-Zr alloy
- Copper alloys - dispersion strengthened by alumina

First Wall Structural [13-22]

- Steels [13-18]
- Low activation austenitic steels [SS 316L(N) IG] for the first wall of ITER
- Ferritic/martensitic steels (F82H, EUROFER) for the DEMO and the TBMs
- Nanostructured ferritic/martensitic ODS steels or nanostructured high nitrogen carbide dispersion strengthened (CDS) F/M steels for the PROTOTYPE

- Vanadium alloys [19,20]

- SiC-fiber/SiC composites [21,22]

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3. First Wall Structurals

3.1. Steels

Low activation is achieved by selection of appropriate alloying elements and control of impurities, both substitutional and interstitial. The alloying elements and the impurities in the steel have to be such that, due to the neutron irradiation, they do not transform to their respective long half-life radio-active isotopes. Typical compositions of the alloy F82H and its equivalents given in table 2, are actually derivatives of the commercially available modified 9Cr-1Mo steel. The changes in composition to arrive at F82H or, its equivalents, have been made to ensure the desired low activation due to irradiation as well as to increase the high temperature capability of this steel. The limits to which the various elements in this steel need to be controlled to achieve the low activation are given in table 3.

Table 2. Typical compositions of the various F/M steels for the first wall of the tokamak

Steel/Composition	Cr	W	Mn	V	Si	C	Ta	N	Fe
F82H	7.46	1.96	0.21	0.15	0.10	0.09	0.023	0.006	balance
JLF-1	9.0	2.0	0.45	0.25	0.2	0.10	0.07	0.05	balance
Eurofer 97	8.9	1.1	0.47	0.2	----	0.11	0.14	----	balance
CLAM	8.98	1.55	0.40	0.21	----	0.11	0.15	----	balance

Table 3. Limits on the contents of impurities and interstitials in FMS for the desired low activation

Element	Wt ppm desired	Wt ppm achieved
N	<300	600
P	<50	20
S	<50	20
B	<10	2
O	<100	100
Nb	<0.1	1
Mo	<1	30
Ni	<10	200
Cu	<10	100
Al	<1	30
Ti	<200	100
Si	<400	110
Co	<10	500

However, even F82H, or its equivalents, in their wrought form are not acceptable for the prototype reactors because of the envisaged life of 30 years for these reactors and the unacceptably large quantities of He that would accumulate in these steels in this period. The alternative has been found in the form of a nano-structured F82H capable of distributing the He into small bubbles by nucleating them on the surfaces of Ti-Y-O complexes introduced in the steel in extremely large numbers through the route of attrition of powder of the steel with nanoyttria and hot extrusion or HIPing of the milled mixture. This steel is known as the 3rd generation oxide dispersion strengthened (ODS) F/M steel [23, 24]. Research continues [25] the world over for other easy-to-produce materials that might fit the requirements of the first wall of a commercial tokamak. The list of many such materials is given in table 1.

3.2. Vanadium Alloys

Vanadium-based alloys, mainly V-4Cr-4Ti and V- 5Cr-5Ti, are very attractive for the structural part of the first wall of a tokamak due to their low activation, good high temperature strength and surface heat capability and, also because, they exhibit low DBTT in the unirradiated state, negligible irradiation-induced embrittlement at temperatures more than 400 ^{0}C, and good swelling resistance. However, they have strong affinity for solutes such as oxygen, carbon and nitrogen, which leads to matrix embrittlement and reduced compatibility with liquid Li. Also, they have high solubility, diffusivity and permeability of tritium, which can lead to embrittlement at low temperatures. Despite the fact that strong MHD effects are observed when these alloys are used in conjunction with liquid lithium as a coolant and the fact that industry is yet not mature in the large-scale production and fabrication of these alloys, Russia has decided to use vanadium alloys for the first wall of their tokamaks.

3.3. SiC_f/SiC Composites

SiC_f/SiCceramic composites have attractive properties for both structural as well as functional applications. They have the ability to operate with acceptable mechanical properties up to almost 1100 ^{0}C. This offers potential increase in fusion reactor efficiency. The SiC_f/SiC composites also have very good tolerance against neutron irradiation up to very high temperatures and an inherent low level of long-lived radioisotopes. Until recently, SiC_f/SiCceramic composites exhibited significant degradation in mechanical properties upon irradiation due to non-SiC impurities causing easy interfacial debonding. Improved performance was obtained as a result of development of stoichiometric, crystalline SiC fibers and advanced fiber/matrix interfaces such as multilayered interfaces. Whereas the recent progress was driven by the availability of almost stoichiometric fibers with higher thermal conductivity and higher thermal stability, the next step will be to tailor the properties of the composite to the specific application by choosing the appropriate fiber architecture, fiber to matrix interface and densification processes. Despite the all-out efforts to improve upon the properties of the SiC_f/SiC composites, they still have low surface heat capability and are quite brittle. Also, the production rates of hydrogen and helium in these composites due to the (n,α) reactions are extremely high. There is also a possibility of leakage of helium gas coolant into the fusion plasma, as most of the available SiC_f/SiC ceramic composites are still porous and anticipated to be vulnerable to widespread microcracking. There is also the issue of fabrication and joining of large sub-components of SiC_f/SiC composites and of the methodology of their structural design. Despite their potential to be the structurals of the first wall of the tokamaks, it appears that there is still a long way to go before they completely qualify.

A comparison of the properties of the three types of first wall structural materials is made in table 4. At the moment, there is little choice but to go for F/M steels as the industrial experience of fabrication and joining of vanadium and its alloys is not as much developed and because of inherent brittleness, the bulk SiC and SiC_f-SiC composites still do not qualify for use. Further, because of poor compatibility of vanadium with the Pb-Li alloy, only pure Li can be used in combination with it necessitating the need of Be or beryllide as multiplier. Russia has designed [26] its liquid TBM that has vanadium alloy as its structural material.

Table 4. Comparison between the properties of various structural materials short-listed for the FW

Property/Material	FMS	V-4Cr-4Ti	SiC_f/SiC
Temperature Window, ^{0}C	300-600	400-700	700-1000
Surface Heat Capability, kW/K.m	4.32-2.74	4.61-4.63	1.05
Thermal Expansion, 10^{-6}/K	11.1-12.3	10.3-11.4	2.5
Thermal Conductivity, W/K.m	33.4-32.3	31.3-33.8	12.5
DBTT*, ^{0}C	<20	250-300	Brittle
RIS** and He effects	✓	✓	✓

* Ductile to Brittle Transition Temperature
** Radiation Induced Segregation

3.4. The Critical Issues

The critical issues related to the first wall struturals include their transmutation and displacement damage due to the high-energy neutrons, manufacturing the large sized intricate shapes and their joining and codes for qualification of the materials for use in fusion environments. So far as the damage due to neutrons is concerned, all the effects that occur in the core of fast reactors occur in the fusion environment also, but with greater intensity.

Helium produced because of the (n,α) reactions of the neutrons with the atoms constituting the first wall is an issue that is difficult to deal with. The rate of production of He in the material due to its irradiation particularly by the 14.1 MeV neutrons in a tokamak is very high (in the range of 200-600 appm/year for steel) and, therefore, in its lifetime of 30 years, the material is likely to accumulate huge amounts of He. Since the solubility of He in any metallic matrix is known to be zero, the high temperature helium embrittlement is an issue of major concern. Furthermore, this He, under thermal fatigue likely to be experienced by the first wall of a tokamak, limits the life of the first wall austenitic steel severely. To overcome this challenge, the F/M steel has been substituted for the stainless steel 316 as this has a much better thermal conductivity. The issue of high temperature helium embrittlement is being further addressed by distributing He into nano-sized bubbles by developing ODS F/M steel of 3rd generation [23,24] in which yttria particles having sizes less than 3nm diameter are distributed in large numbers (10^{23} particles/m^3). Since the nano-sized (18-20 nm dia) yttria gets refined to less than 3nm dia during attrition of its mixture with steel powder only in the presence of Ti, this is to be added to the mixture before attrition. Ti-Y-O complexes form due to attrition. Interestingly, Ti is the only element that can effectively achieve this. The reason is yet to be established. Besides, the Ti-Y-O complexes act as sites for the nucleation of He bubbles [27, 28].

The other issue relates to manufacturing of components, particularly joining of materials. Friction stir welding, electro-discharge welding, and diffusion bonding by HIP are the technologies that are currently being developed to advanced levels for meeting this challenge [29-34].

3.5. Materials for Other Components of Tokamak

Structural materials for the other components of a tokamak [35] are listed in table 5. Methods of manufacturing these materials and the components of the tokamak out of these are well understood.

Table 5. Materials for the other components of the Tokamak

Materials	Thermal Shield	Vacuum Vessel & Ports	VV Support	Blanket Support	Diverter
SS 304 (plates)		✓	✓		
SS 304 L plates	✓				
Ti-6Al-4V (plates)	✓			✓	
Steel 660 (bolts)	✓	✓	✓		✓
Alloy 718 (bolts and plates)	✓	✓	✓		✓
NiAl bronze (rod and plates)			✓	✓	✓
Steel 430 Borated steel plates, SS 316		✓			
SS 316 L(N)-IG (plates & pipes)		✓			
Cu-Ni-Be (collar)				✓	

4. The Blanket

Before the tokamak or, for that matter, even a device based on the concept of inertial confinement, can qualify to be an effective device for producing electricity, it is obvious that, apart from achieving its engineering breakeven, methods have to be established for breeding the tritium (unavailable on earth) at rates more than the rates at which it shall be burned in the core of the tokamak and, also, methods have to be found to extract the *heat* produced due to the fusion of D and T, majority of which is carried by the neutral 14.1 MeV neutrons. It is the blanket that has to perform these two functions [2]. It has to breed tritium with a Tritium Breeding Ratio (TBR) of more than one and, also, has to extract nuclear heat efficiently. Also, it is obvious that the design of the blanket would depend on the concept employed to confine the plasma.

Keeping these functions of the blanket in view, a number of concepts for the tokamak Test Blanket Modules (TBMs), for the purposes of testing alone, have been proposed for ITER. It is, however, mandatory that their designs be DEMO-relevant as ITER is only a test reactor. The DEMO-relevancy of the design has been imposed to ensure that something that works now in ITER will also work for the commercial reactor for which it is being tested in ITER.

Table 6. Functional Materials in the TBM

For neutron multiplication

- Beryllium, Be-8at%Ti (beryllide), BeO in solid form
- Liquid lead

For Tritium breeding

- Li^6 enriched liquid lithium or eutectic Pb-17at%Li
- Li^6 enriched ceramics like lithium titanate and lithium silicate

For Tritium extraction

- He (purge gas through the ceramic breeder)
- Liquid lead lithium eutectic

For self-healing coatings (required to reduce the MHD drag and prevent tritium permeation)

- Aluminaon FMS
- AlN, CaO, Er_2O_3 or Y_2O_3

Some of the TBMs that have been designed for ITER are termed as solid test blanket modules and some as liquid test blanket modules, the difference being the physical state of the breeder material in the TBM. If the breeder (basically, Li^6) is in the form of a solid ceramic compound, it is solid breeder TBM and, if the breeder is in liquid state (as pure Li liquid or eutectic Pb-Li alloy liquid), it is called a liquid breeder TBM. In the case of a solid TBM, the coolant, more often than not, is He. In one such concept proposed by Japan, it is water. To have enough neutrons for the breeding reaction, Be or beryllide is to be inserted in the solid TBM as a neutron multiplier. The solid TBM thus consists of the structural material (low activation F/M steel), the ceramic breeder (lithium titanate or lithium silicate), the neutron multiplier (Be or beryllide) and the coolant, He. The material of construction of TBM has been chosen to be F/M steel to gain experience with this material as this is a candidate for the first wall of a DEMO. When Pb-Li is used, Li works as the breeder and Pb as the neutron multiplier. The liquid itself sometimes is made to act as the coolant as well. As a coolant, it creates the extra issue of Magneto-Hydro-Dynamic (MHD) drag on its own flow in the TBM, which raises further requirements in terms of electrically insulating coatings on steel to reduce the drag, powerful pumps to push the liquid through the TBM and, of course, the integrity of the material under forced flow at high temperature of liquid metal. However, obviously, there is no need to insert Be or beryllide for neutron multiplication in this case. Various concepts of TBMs, based on either solid or else liquid type breeder, have been proposed by various countries participating in the ITER program [36-38]. India has, however, proposed a new '*hybrid*'blanket that contains the solid breeder as well as the liquid breeder. The parameters of the Indian hybrid blanket known now as Lead Lithium cooled Ceramic Breeder Test Blanket Module (LLCB-TBM) are shown in table 7.

Table 7. Parameters of the Indian LLCB-TBM

Breeder and Coolant	Li^6 being a tritium breeder, either lithium titanate or lithium ortho-silicate enriched in Li^6 are used as solid breeders and pure liquid Li or Pb-Li eutectic liquid enriched in Li^6 are used as liquid breeders. Because the Indian TBM has both the solid and the liquid breeders, the concept is known as an '*hybrid*' concept. The Pb-Li eutectic liquid also acts as a coolant in this TBM.
Neutron Multiplier	Pb in Pb-Li
Structural Material	Indian Reduced Activation Ferritic/Martensitic Steel (IN-RAFMS)
Electro-insulation	Al_2O_3 coating on the inner walls of the TBM box that would be in contact with the Pb-Li eutectic liquid. This helps to reduce the drag on the flow of the eutectic liquid due to MHD and, also, prevents permeation of tritium into the structural steel and surrounding environment
Coolant for the Structural	Helium Gas at 80 bar with inlet temperature of 300 ^{0}C and outlet temperature of 550 ^{0}C
Purge Gas to scavenge Tritium from Ceramic Breeder Pebbles	Helium gas at low pressure

The hybrid concept proposed by India has been accepted by ITER for its implementation and is elaborated in the following paragraphs. Be it a solid, a liquid or a hybrid TBM, its design must ensure that (i) the net tritium breeding ratio >1 (to be met through neutron multiplier and enrichment of the breeder material in Li^6), (ii) efficient extraction of heat (from heterogeneous volumetric nuclear heat generated in the TBM) while maintaining the temperatures of the structurals and the functionals within their allowed windows, (iii) liquid metal coolant circulates despite the MHD drag if it is the coolant, and (iv) safety of the TBM, the tokamak, the environment and peoplein and around the tokamak and, above all, (iv) the design of the TBM and the materials that go into it have to be compatible with the DEMO design. The design of our Lead-Lithium Ceramic Breeder (LLCB) TBM is also been done keeping these points in view. From an artist's point of view, the location of a TBM, be it solid or liquid, in the tokamak [39] is shown in figure 3 and the schematic of Indian LLCB TBM is shown in figure 4.

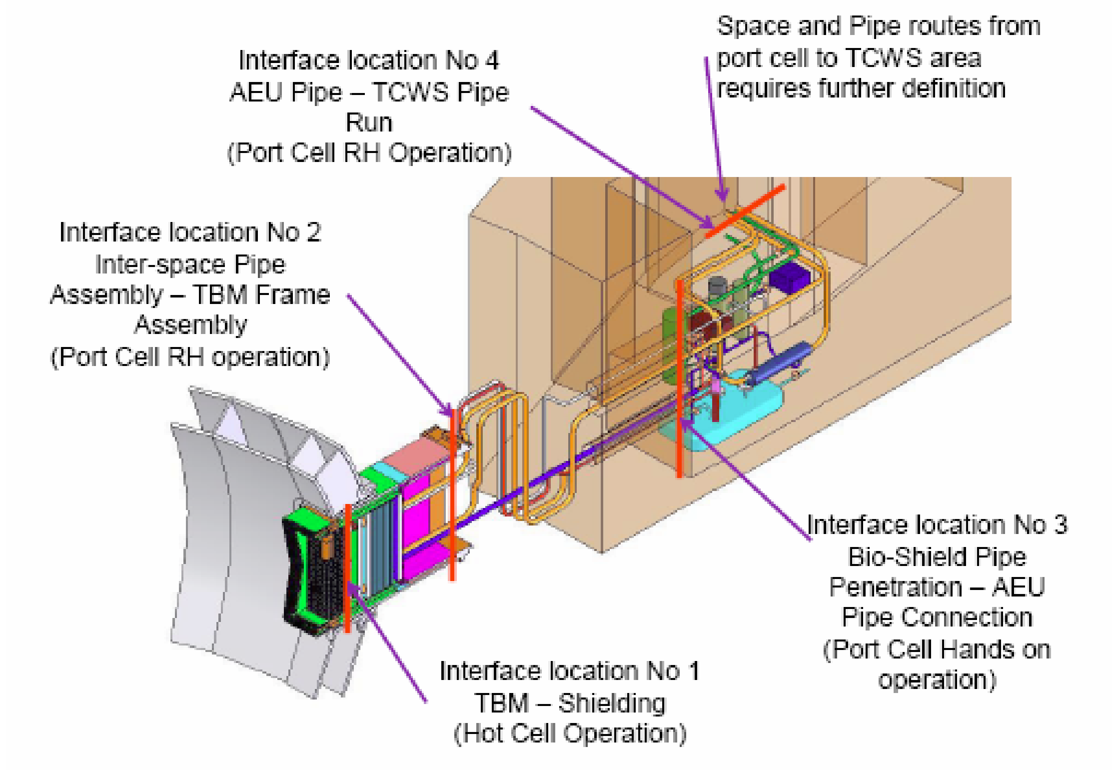


Figure 3 An artist's view of the TBM and its proximity to plasma

• Ceramic Breeder: Li Titanate
• Coolant, Multiplier and breeder: Pb-Li
• FW coolant : Helium, 80 bar 300-525 C
•Pb-Li Mass flow: 42 Kg/s
• Velocity (ref): 0.2 m/sec

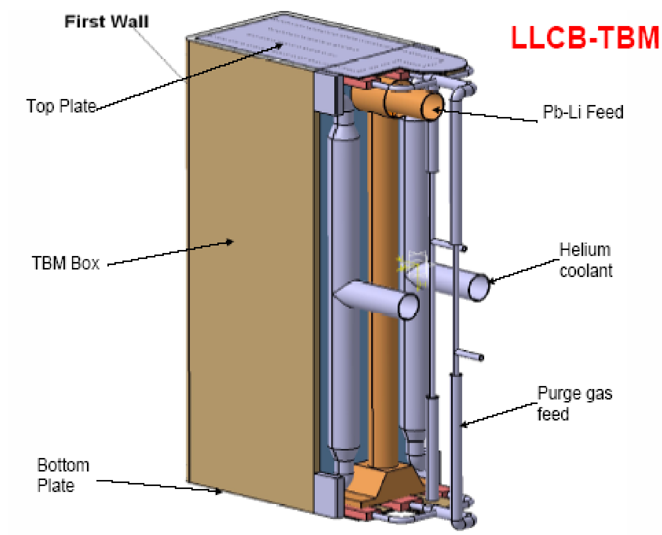


Figure 4. The Indian Lead-Lithium Ceramic Breeder (LLCB) TBM and its schematic

5. The Superconducting Magnets

The low temperature superconductors required for the tokamak, are at an advanced stage of development [40-41], while the high temperature superconductors needed for advancing into commercial fusion power are making good progress. The required operating conditions are listed in table 8, and the two accepted LTS materials and their characteristics in table 9. For the Nb_3Sn superconductors, internal tin strand fabrication process is adopted.

Table 8. Operating conditions of the superconducting magnets for experimental tokamaks

- High field variations (dB/dt ~ 2T/s)
- Very high structural & operational loads (~1000 MPa)
- High vacuum (10^-6 mbar)
- High inductive loads (~ 100 H or more)
- High fields (12-14 T: central solenoid & TF, 6-8 T: PF)
- Very high stored energy (> 100 GJ)

Table 9. Characteristics of superconducting magnets

NbTi	Nb_3Sn
Solid solution T_c of 9.8 K H_{c2} of 11 Tesla at 4.2 K The alloy is produced by multiple EB melting-fabrication through a thermo-mechanical route	Intermetallic compound T_c of 18 K H_{c2} of 22.5 Tesla at 4.2 K Nb – Sn reaction is carried out only after magnet fabrication

High temperature superconductors (HTS) discovered since 1986 have taken the T_c beyond the liquid nitrogen temperatures. They have found application in the current lead between the low temperature section and the room temperature section. Table 10 gives the properties of the popular HTS materials.

Table 10. Properties of Popular HTS Materials

Material	T_c (K)	H_{c2} (T)
MgB_2		

	39	17
YBCO ($YBa_2Cu_3O_7$)	93	150
BSCCO ($Bi_2Sr_2Ca_2Cu_3O_{10}$)	110	108

YBCO and BSCCO appear to be the prominent candidates for producing high current carrying cables suitable for future fusion reactor magnets. Being brittle materials, they need to be produced in the form of composites containing other materials which enable adequate mechanical properties for handling.

YBCO can only be made as multilayer tapes wherein the superconductor is deposited as a thin film on a nickel-base alloy with an oxide buffer layer in between, and this is further sandwiched between two copper layers. This is because of a highly anisotropic dependence of critical current density on the orientation of the magnetic field with respect to the crystal axes. The oxide buffer layer promotes deposition of the high temperature superconducting (HTS) film with the beneficial texture. The dependence of the critical current density on the texture is reported to be weaker in a Zr-doped YBCO tape being developed currently. A schematic of the layer sequence in a typical YBCO based tape is shown in figure 5.

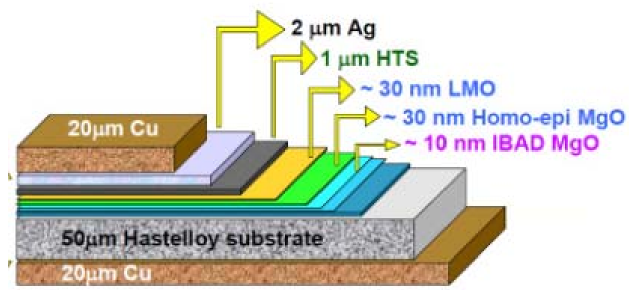


Figure 5. Typical layer sequence in a YBCO based tape

BSCCO is produced both as tape in a silver matrix, and as a round wire. The tape is anisotropic, while the round wire is isotropic. But due to its critical magnetic field dropping to very low values above 40 K, it cannot be used at temperatures above 40 K. Currently for high field applications above 77 K, YBCO based conductors are the only choice [42-43].

There is a tremendous amount of research activity in this area, but it may take some time before these materials are used in applications with current densities and other requirements typical of fusion reactor magnet systems.

In the ITER HTS current leads are used to provide the transition between room temperature and 4 K. A total of 60 HTS current leads are required, with a total nominal current capacity of 2.64 MA. Trials at low temperatures with HTS current leads of 52 kA and 68 kA capacity were successful [44].

6. Developments on the Materials Front for the Tokamak and its Blanket in India

The issue of materials is the second most important issue (the first being ignition of the plasma and its sustenance) to be resolved for commercial exploitation of the fusion power through tokamaks. What needs to be done, to begin with, is to develop the above listed materials with characteristics and life that are desired for their application in the environment of tokamaks. Li needs to be enriched in Li^6 followed by manufacture of lithium ceramic pebbles with desired characteristics both as individual pebbles and collectively. The same applies to the beryllium and beryllide pebbles required for the helium cooled solid ceramic breeder modules. The desired low activation of the F/M steel needs to be achieved by further refining the steel. The issue of coatings that would reduce the MHD drag without getting corroded or eroded by the flow of high temperature liquid coolant needs to be addressed. Apart from these, the unknown domain of behavior of all these materials and their joints in the fusion environment of 14.1 MeV neutrons needs to be explored both theoretically as well as experimentally by using the current level of knowledge in this area and the available sources of irradiation till the International Fusion Materials Irradiation Facility (the IFMIF) becomes functional [45]. New Materials Test Reactors (MTRs) and innovatively designed dual beam irradiation facilities need to be developed. The task before the world scientific community to develop and qualify materials for this high-tech application is thus indeed formidable.

Undeterred by the above challenges, once the concept of a hybrid TBM, the Lead-Lithium cooled Ceramic Breeder (LLCB) for ITER was conceived by India, a Design Description Document (DDD) of the LLCB incorporating the thermal-hydraulic design, the neutronic design and the safety design was prepared and submitted to 'ITER Organization' for obtaining a port for testing of the TBM. The concept was accepted and half-a-port in ITER was allowed to India to test this concept. Since then, the work of detailed design of the LLCB, the fabrication of the TBM box using indigenously developed RAFMS, development, characterization, qualification and production of the other materials like the ceramic breeder and the liquid Pb-Li coolant, development of coatings on steel to prevent the permeation of tritium and to offset the MHD drag, neutron irradiation and testing of materials, evaluation of various configurations of the materials in the TBM to achieve the highest possible TBR and efficient extraction of heat, the development of the auxiliary systems of the TBM, testing and inspection of all the materials and components, the assembly of the TBM and its integration to ITER at Cadarache, its remote installation and dismantling at the ITER site, instrumentation for the diagnostics of the TBM, enrichment of lithium, extraction of tritium from the purge gas and the liquid Pb-Li coolant all are being worked out actively [46]. The low activation FMS has been indigenously manufactured and characterized [47]. It qualifies for the intended use. Processes for manufacturing the lithium-titanate pebbles have been worked out [48]. Method has been set to produce the difficult-to-make Pb-Li alloy. Liquid lead-lithium loops to study the dissolution, corrosion and erosion of steel in flowing liquid metal have been successfully run using the indigenously developed magnetic pumps. Innovative methods are being attempted to fabricate the 'First Wall' of the TBM. Enrichment of lithium to 90% for use in eutectic Pb-17 at% Li and 30-50% for use in ceramic breeders is required. Enrichment can be carried out by various processes such as chemical exchange, exchange distillation, thermal diffusion, gaseous diffusion, electrolysis, and centrifuge. One of these has been found to be promising.

Apart from all these, beryllium pebbles, likely to be used as neutron multiplier if an opportunity arises to pair up with an ITER partner who is going to test the solid breeder, can be prepared by rotating electrode process (REP) in which an arc is struck between a rotating beryllium anode (consumable) and tungsten cathode (plasma dissoluble electrode) in an inert gas atmosphere. The leading edge of the beryllium electrode melts due to the intense heat generated by the arc plasma. Molten beryllium droplets are splashed out due to the centrifugal force and solidify in the inert gas atmosphere forming beryllium pebbles. Along with production of pure beryllium, the beryllide, REP is being actively developed. Also, the low temperature superconducting magnets of both NbTi and Nb_3Sn type are being actively developed for the field coils of the tokamak. Also, even before participation of India in the ITER program, the heat sink, namely, the Cu-1wt%Cr-0.1wt%Zr alloy, was successfully developed and characterized [11,12,31,49].

7. Challenges in fabrication and joining of materials for components of the tokamaks

In section 6 above, it has been briefly pointed out the efforts that India has begun making on various fronts to implement the Indian program on making, installing, commissioning and testing the IN-LLCB-TBM in ITER. It has also been repeatedly pointed out in the text above that the challenges lie not only in developing new materials for tokamaks but also in manufacturing the intricate components of tokamaks and their blankets from these materials. The following few paragraphs deal with the challenges that the world is facing in manufacturing some components like the vacuum vessel, diverters, superconductors and test blanket modules for ITER.

7.1. The Vacuum Vessel

The Vacuum Vessel is a dough-nut shaped and hermetically sealed double walled stainless steel container in which the fusion plasma exists. Its internal surfaces are lined with the blanket modules to protect from the high energy neutrons produced in the fusion reaction, and it has the diverter located at the bottom of its internal surface. It measures about 19 m across, with an internal diameter of 6 m and a height of 11 m. It has 44 ports for remote handling, diagnostics, heating and vacuum systems. It weighs more than 5000 tons. The Vacuum Vessel is shown in figure 6, along with a cutaway drawing and one of the nine sectors making it up.

It is planned to fabricate the vacuum vessel in the form of nine toroidal sectors, which will be joined by field welding on site during assembly [50]. Each of these 40° sector will consist of two 20° sectors. An inner and an outer shell, of 40 mm thickness, separated by poloidal ribs keeping the shells apart by 0.45 to 0.83 m, define each sector. The material of construction is SS 316L(N)-IG with controlled nitrogen content and cobalt content at less than 0.05 %. The tolerances for the vessel are specified to be +/- 20 mm in height and width, and +/- 5 mm in wall thickness. The necessity of replacing toroidal field coils or a vessel sector when they malfunction will require an on-site remote refabrication capability for parts of the vacuum vessel. The requirements on the vacuum vessel are: tight manufacturing tolerances, ability to withstand large electromagnetic loads, to contain the shielding for the magnets, to provide for natural convection cooling, and to provide for remote maintenance operations.

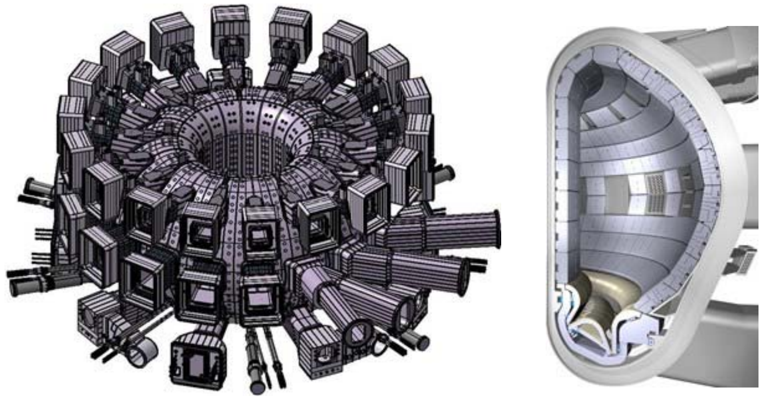


Figure 6. ITER Vacuum Vessel (left), a cutaway drawing (center), and one of the nine sectors making up the vacuum vessel (right).

The fabrication issues involved were not easy to resolve without the fabrication and testing of large scale models. After some small scale models, a full scale sector model was constructed and tested. The issues that could be addressed only with the full scale model were the magnitude of welding distortions and achievable tolerances.

Narrow gap TIG, EB welding and metal active gas (MAG) welding were selected for sector fabrication to achieve low distortion after some trials. The required tolerances were achieved by the development of new types of joints and/or by varying the welding sequence. The initial dimensions of components were chosen to achieve the final dimensions with shrinkage wherever applicable [51].

For NDT of the field joints, ultrasonic testing (UT) with two-mode piezoelectric probes was selected. The probes emit and receive both longitudinal and shear waves with an angle of 45 or 60°, enabling a high resolution image of a flaw. Flaw sizes of about 2 mm in 40 mm thick plates were detected.

Nd:YAG laser cutting and welding, reduced pressure electron beam welding and a phased array UT are being pursued for development of VV sector assembly and on-site maintenance procedures. With a reduced pressure electron beam welding technique (using nearly 0.1 mbar, 150 kV and up to 19 kW), 60 mm deep welds with minimum welding distortion were achieved. A 6 kW Nd:YAG laser could demonstrate both welding with good results, and 60 mm thick cutting with an edge width of 1.5 mm at a speed of 0.15 m/min. Partial mockups of the vacuum vessel can be seen in fig. 7 [52, 53].

7.2. The Test Blanket Module

TBMs, as elaborated already, have a complex design and norms for their acceptance for installation in ITER are yet not fully defined. As it is, the fabrication of this component is very challenging. This paper highlights some of the challenges associated with fabrication of Indian LLCB-TBM. Figure 8 shows schematic drawing of the LLCB-TBM and different subcomponents of the same. It consists of following important subcomponents

- First Wall

- Solid Breeding Modules
- Inner and Outer Back Plate Assemblies
- Top and Bottom Plate Assemblies

All these individual components are complex in design and further need to be assembled through weld joints which are again very complex and difficult to access leading to challenges in fabrication. Another challenge is the material out of which these components have to be fabricated. The material for TBM fabrication (IN-RAFMS) is a new material and still under development. Therefore, there is not much experience about its fabricability, including weldability. However, since this material is being developed on the lines of ASTM A387 Grade 91 steel or Modified 9Cr-1Mo; on the fabricability of which exist [54] a great deal of experience. In the paragraphs that follow, we discuss the challenges associated with fabrication of TBM from the perspective of TBM design, material and manufacturing processes by considering individual components of the TBM.

Figure 7. Partial mockups of the VV sector.

7.2.1. Challenges in fabricating the First Wall (FW)

First wall of the TBM faces plasma in the core of ITER and acts as containment of the lead-lithium coolant cum liquid breeder and the solid breeding boxes. This is made of Indian RAFMS (IN-RAFMS)of composition Fe-9Cr-1W-0.24V-0.08Ta-0.1C, which is still evolving [47].

First wall has a U-shape with each arm being ~0.5 m. Radius of curvature at the bends is equal to thickness of the plate, i.e., 28 mm. It is made of 28 mm thick plate of IN-RAFMS. It has 64 square channels of 20x20 mm cross-section which are running all along through the U-shaped wall. These channels are parallel to each-other with a 5mm thick rib separating the channels. These channels are meant forcirculation of He coolant at 80 bar pressure at

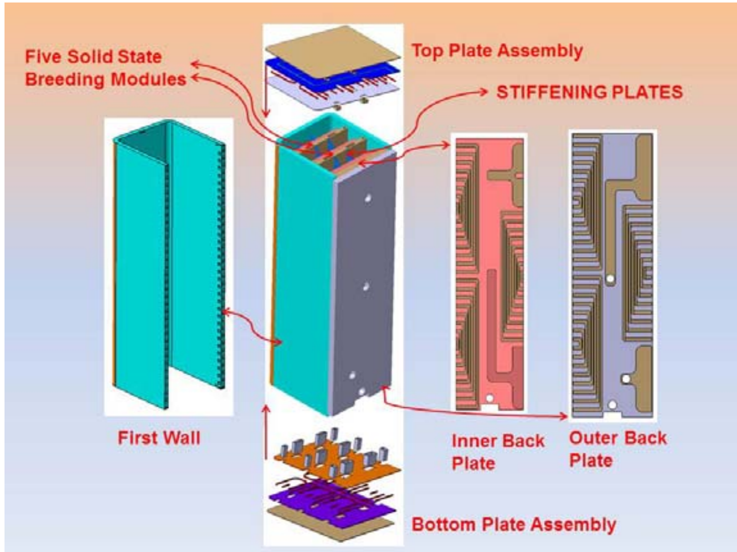


Figure 8. Schematic Diagram Showing Indian TBM and its Different Sub-Components operating temperatures in the range of 400-450 ^{o}C.

Fabrication of this component is difficult because of the following reasons:

- Complex design
- No weld seam is allowed at plasma facing surface
- Bonding of Be-tiles on the plasma facing wall

HIPing is being actively explored world over to fabricate this component. Different aspects of the HIPing process like finish (roughness and cleanliness) of the mating surfaces, temperature, pressure, etc. have been extensively studied [55]. A significant effort in this direction was made by the Japanese group [34], which fabricated a nearly 1/3rd width prototype of Japanese water cooled TBM (top and bottom cover unlike ours) by HIPing of U-shaped tubes of square cross-section sandwiched between U-shaped plates. However, microscopic examination revealed lack of bonding at the mating surfaces. Besides, the charpy energy in of the HIPed joints is merely 20% of that of the parent metal. Though acceptance criteria for FW are still not frozen, it is not difficult to assume that such a lowcharpy energy can never be acceptable as it is a critical component and it has to operate at high pressure. This shows that relying on HIPing alone is not a good idea as the mating surface is very large (it will be ~16 m^2 for LLCB-TBM if Japanese approach is followed). There is always some finite probability of having lack of bonding as we are trying to annihilate such a huge parting surface in one go. Therefore, following two alternative approaches are being explored by Indian TBM manufacturing team to manufacture the U-shaped first wall of the TBM.

7.2.1.1. Cutting of square channels by wire EDM followed by hot bending

This approach involves cutting of straight square channel by wire EDM followed by hot bending. The advantage of this approach is that there will be no mating surfaces and, therefore, no lack of bonding. However, challenges remain in cutting such a long square channel by wire EDM. Some of the challenges are ensuring geometry of the square channels post bending within tolerance, bending the plate to the desired radius of curvature equal to the thickness of the plate, and the possibility of formation of defects like cracks during the bending process. To overcome these challenges, simulation of the bending process is required to design and optimize the bending process.

Some preliminary work has been done at BARC to explore this route. Straight channels with square cross-section (20x20 mm) and 300 mm length in 30 mm thick plates were produced in ASTM A387 Grade 91 plates. Hot bending process was computationally modeled, simulated and optimized. Hot bending was performed to produce L-bends in the plate with square channels (fig.9). The radius of curvature is 100 mm which is much more than that in the current design of the FW. This means, still there is a lot of gap between the design end and manufacturing process end in terms of radius of curvature at the bend and modifications in the design and / or innovations in the process are needed to bridge this gap.

Destructive and non-destructive (ultrasonic testing) evaluation of the channel cross-section and wall thinning at the bend locations were carried out. It was found that there is not much deviation in the channel cross-section at the bend location. However, there was considerable thinning on the outer wall near the bend location. The wall thickness on the outer surface had thinned from 5 mm before bending to 4 mm after bending near the bend location. This implies either design should provide for this tolerance or suitable allowance will have to be provided in the manufacturing process to take care of wall thinning during the bending process. No cracks were observed in the square channels post bending. Fig. 10 shows the UT scan probe orientation, a typical scan image showing possible defects, and the actual machining marks which caused the signal.

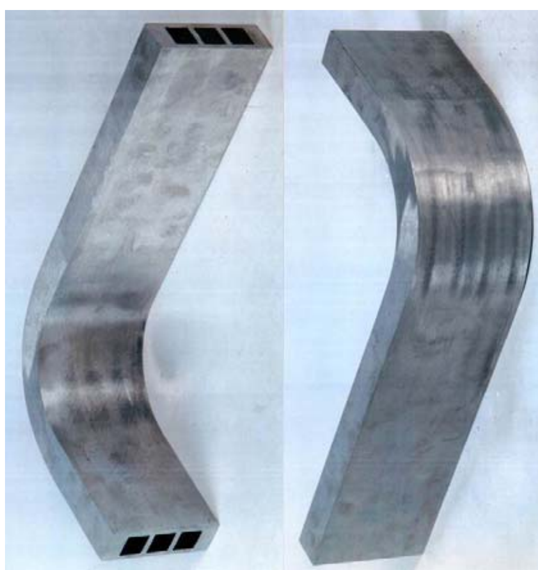


Figure 9. Showing different view of L-bends with three square channels (20x20 mm cross-section, 300 mm length)

7.2.1.2. Machining of open channels, bending followed by closure of individual channels with strips using laser welding

This is another approach, which relies on capability of laser welding process to produce sound and defect free weld joints with negligible distortion along complex seam geometry. This approach looks upon FW as a shell structure and tries to produce it by high power laser welding. The weld joints are designed such that all the weld seems are on the inner surface of the FW to comply with the requirement that no weld seem should be on the plasma facing surface.

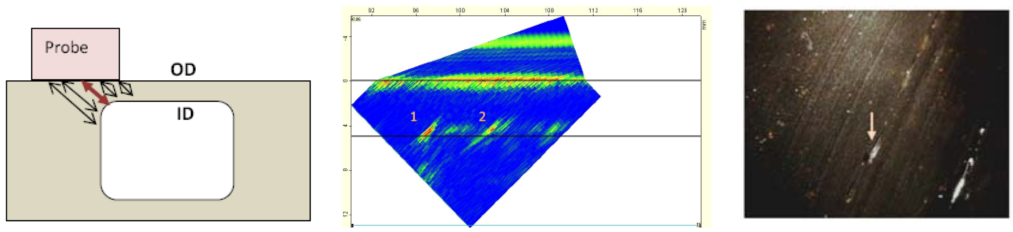


Figure 10. S scan probe configuration (left), S scan image showing defect during angle beam examination (center), machining marks on the channel inner surface which probably caused the indication (right).

In this approach the advantage over HIPing lies in huge reduction in the parting area (from 16 m^2 in case of HIPing to just 2 m^2 in case of welding), which needs to be annihilated to fabricate the LLCB-TBM. In boding, less is the parting area, less is the problem. Besides in case of HIPing, the entire parting area is annihilated in one go while, in this current approach, the parting area is annihilated in incremental manner giving an opportunity of midcourse correction. There is no opportunity to rectify defects like lack of bonding in case of HIPped FW while local weld repair is possible in case some defect is detected in post weld NDE.

The concern in this approach, however, was that weld seams being so close to each other (separated by just 5 mm), fusion zone and HAZ of the adjacent welds may overlap. To see if this really happens, straight square channels with 20x20 mm cross-section and 230 mm length were produced in ASTM A387 Grade 91 steel using this approach.

Ultrasonic Tandem Technique (UTT) is employed for detection of flaws that are oriented perpendicular to the scanning surface. In the present case, flaws like lack of fusion and lack of penetration are expected to be oriented perpendicular to the scanning surface. Hence, tandem technique was developed. Unlike conventional pulse-echo technique that uses a single probe as both transmitter and receiver, tandem technique uses two separate probes, one as transmitter and other as receiver. Both are positioned on the same surface with a separation distance decided by the depth at which the flaw of concern is expected. Ultrasonic phased array examination using tandem technique was found to be effective for NDE of laser weld joints in these specimens. Simulation studies were carried out to arrive at the most suitable linear phased array probe for the given configuration of the weld joint. The experimental work carried out on four weld joints using the available phased array probe detected defects of varying levels in all the weld joints. The presence of these defects was confirmed by radiography. These results indicated that the phased array tandem technique is very sensitive to detect the defects of concern in the laser weld joint of this configuration.

These channels were destructively examined to see weld joint cross-section. There was no evidence of interaction between the fusion and or HAZ of the adjacent weld joints.

Besides, the weld joint was sound and free of defects. Weld joints produced in the same material under identical welding conditions were tested for tensile behavior and the weld joint was stronger than the parent metal. Further work is now in progress to fabricate a FW prototype with 7 square channels through this approach.

7.2.2 FEM simulation of bending

The analysis of high temperature bending followed by cooling to room temperature to predict resultant stress pattern and distortion in the square channel is essentially an elasto-plastic structural and thermal steady state nonlinear analysis. The overall process is a complex process and requires optimization to get the reasonable result. This analysis was, therefore, carried out in two parts. First, analysis of hot bending was performed at 700 ^{o}C with suitable modeling of die and punch (fig. 11a) with realistic clamping method using material property at that temperature. Thereafter, cooling of hot bent channel to room temperature was analyzed.

The behavior of modeling was verified by doing a simple elastic bending with actual dimensions of the channel. This provided better insight for selection of contact elements to simulate ‘Punch-Channel’ contact pair and ‘Channel-Die’ contact pair. The behavior was found satisfactory for further elasto-plastic analysis.

The trial for the elastic-plastic analysis was carried out with actual model. The selection of material parameters for simulation is very important which is quite non-linear. In the initial trials the elements in the model failed under excessive plastic strain. This was verified with trial runs in other software packages. The size and number of the elements were modified for convergence of the solution.

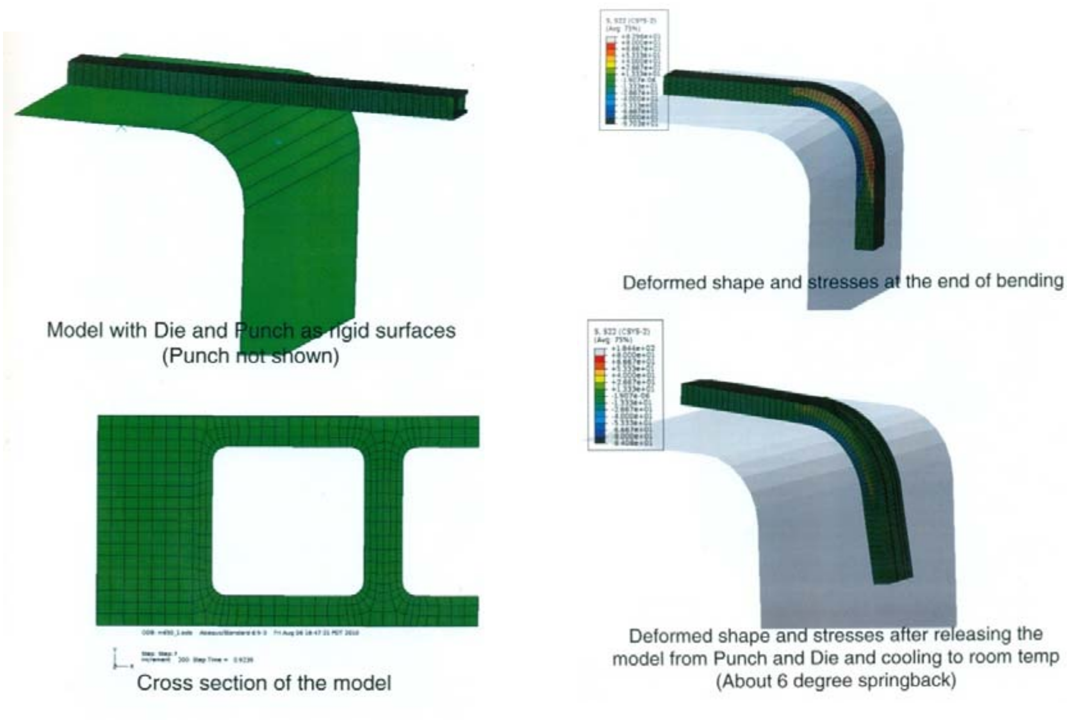
The deformation of the channel cross-section is low at the start (0^o) and end (90^o) of the bending. However, maximum deformation occurs at the 45^o location of the bend. It has been observed that the model springs back by 6^o after removal of the punch and the die (fig.11 b). Majority of the cross-section in the bend region is plastic state during bending.

7.3 Challenges in Development/Fabrication of Superconductors for Coils of the Tokamak

The ITER Magnet system consists of 48 elements generating a magnetic field nearly 200,000 times than that of the Earth. These include 18 superconducting Toroidal Field and 6 Poloidal Field coils, a Central Solenoid, and a set of Correction coils whose purpose is to magnetically confine, shape and control the plasma inside the Vacuum Vessel [56]. Fig. 12 shows a schematic of the ITER magnet system and one of the toroidal field coils.

Both the Central Solenoid and the Toroidal Field coils use Nb_3Sn as the superconductor material operating at high magnetic field (12 T). For the Poloidal Field coils and the Correction coils the field is somewhat lower, and they use NbTi. All the coils are liquid He cooled at about 4 K.

The total weight of the toroidal coils is about 6500 tons, making them one of the biggest components of ITER machine, along with the Vacuum Vessel. The coils use Cable-In-Conduit superconductors, wherein a lot of superconducting strands are cabled together in a structural jacket providing for liquid helium cooling.



(a) (b)

Figure 11: (a) Model for hot bending process, (b) deformation and stress profile after hot bending.

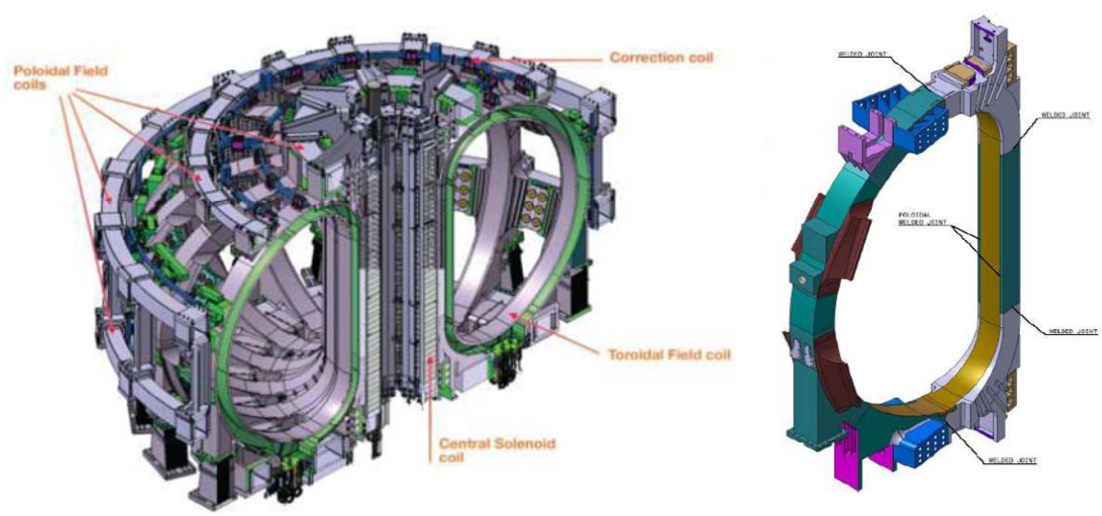


Figure 12. The ITER Magnet System (left) and a Toroidal Field Coil (right)

The six independent poloidal field coils maintain the shape and stability of the plasma. They are placed outside of the toroidal magnet system. The smallest of these coils will be manufactured offsite, while the other five will be wound onsite in a dedicated building due to their size. These coils also use Cable-in-Conduit conductors, although of two different kinds of strands [57].

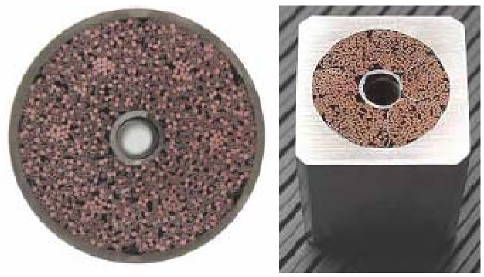


Figure 13. The typical cable crosssection for TF coils (left) and CS / PF coils (right).

The two different kinds of Cable-in-Conduit Conductors used in ITER magnetic system are shown in cross section in fig. 13. The fabrication of magnetic coils from these cables involves winding and assembly, some of which is done on site.

The central solenoid induces the current which drives the plasma, acting like a transformer. Six vertically stacked independent coils made of Cable-in-Conduit conductor of Nb_3Sn essentially make up the central solenoid. Each of the coils consists of multiple pancake winding units minimizing joints. A high voltage capability up to 29 kV is ascertained by a resin-impregnated glass-polyimide electrical insulation. The large electromagnetic forces expected during operation require the conductor jacket material to possess good fatigue behavior.

8. Summary

The world is attracted to exploit the nuclear fusion of light elements - the source of heat and light of *the Sun* and *the Stars* - mainly because it is potentially an infinite source and relatively clean. However, the task is formidable. In the last half-a-century, notable advances in technology have been made to achieve thermonuclear fusion through the routes of magnetic confinement as well as inertial confinement of the *hot* plasma and the scientific break even has been achieved. Thus, though the scientific challenge involved in this task has been largely overcome, the engineering challenge persists. Coupled to this challenge exists the necessity to design and develop new materials with desired multiple attributes, and breeding of tritium, and its handling. The attributes that the new materials must possess are the resistance to14.1 MeV neutron irradiation, capability for long term operation at high temperatures and low propensity to induced activation. Designing and developing the right materials will solve but one of the problems. The design of a fusion device and its blanket is very complex by nature and further innovations will have to be used to overcome the material limitations. Fabrication might be the key to open up more degrees of freedom for design.

For the success, the world must pool its resources – both scientific and financial – to overcome the challenges of venturing into inadequately chartered domains of activities and the prohibitive costs of research and development on the complex systems involved. India, both as a partner in ITER and in its own capacity, is fully committed and deeply engaged to contribute.

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