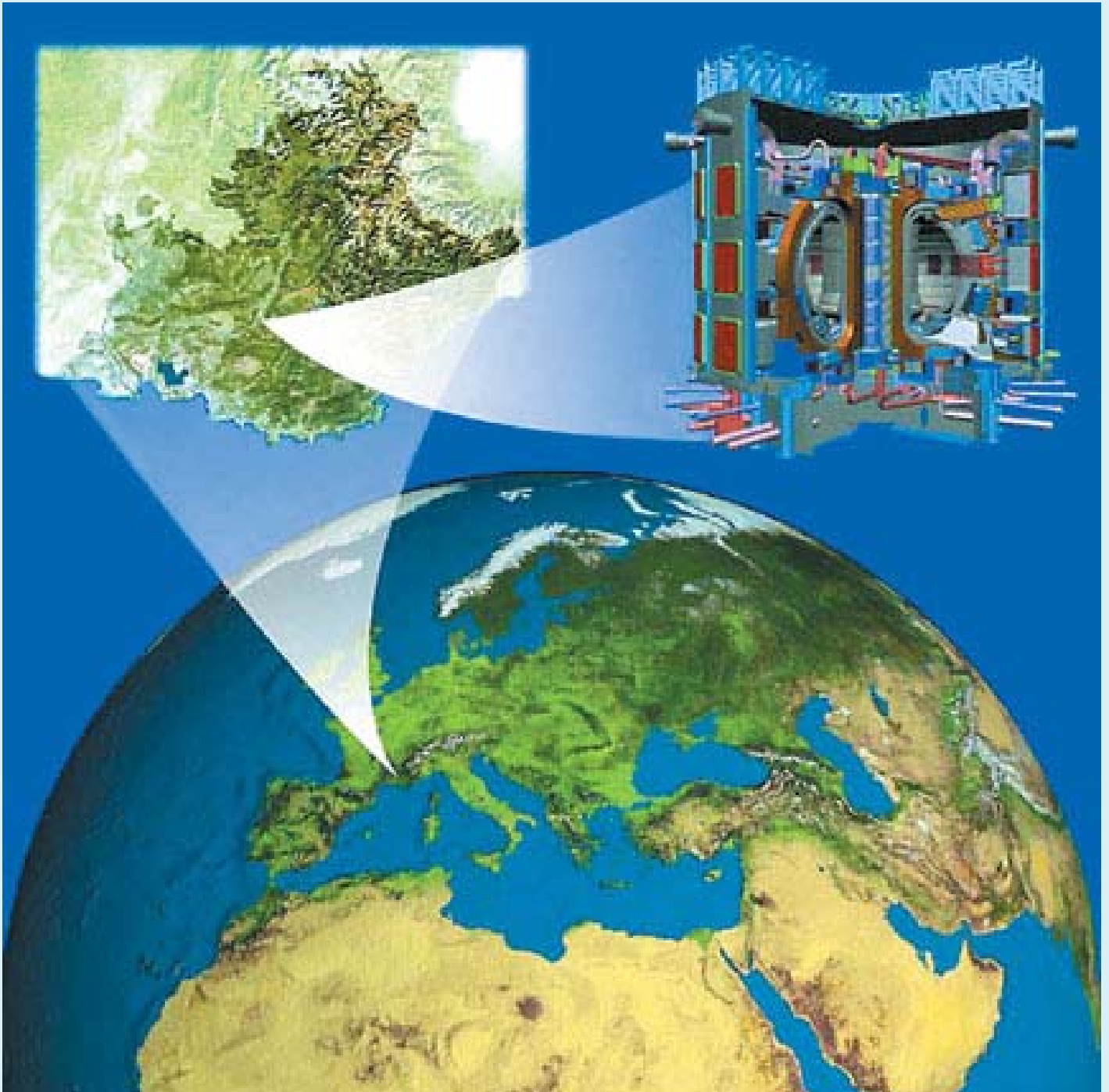


Nuclear India

May - June 2006/ VOL. 39/NO.11-12

International Thermonuclear Experimental Reactor



India's accession to ITER

Fusion, as explained in the articles in this issue, is the energy source of the sun and stars and for the past 50 years scientists are working to use fusion to generate energy on earth. A fusion power plant, like a fission power plant would not produce any greenhouse gases. Fusion technology has no proliferation concerns and is a potential source of safe and environmentally benign source of energy.

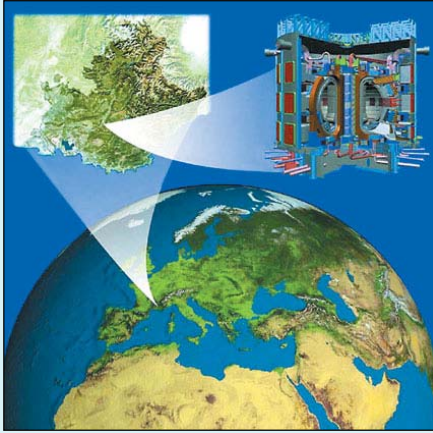
Experiments to harness fusion are being conducted in various laboratories around the world. India also has a modest programme in fusion and has plans to harness this technology to provide long-term energy security. Considering the complexity of this technology, Japan, European Union, the then Soviet Union and the United States established a collaborative project in 1985 to harness fusion energy. It is for the first time that several countries were thinking of collaborating on a large project of great scientific complexity and therefore, the entire process of planning went through long negotiations on several issues particularly the location of the project and relative contribution of the participants. In between the process of negotiations, the United States withdrew and then joined again. Canada joined and then withdrew. In addition to the original 4 partners, China and South Korea also joined the consortium of parties to the ITER venture. It is gratifying that all issues have been resolved and the text of the implementation agreement has been finalized to the satisfaction of all.

Two years ago, European Union encouraged India to join ITER and a team from EU visited India for initial discussions in October 2004. This was followed by more such interactions at the international level and approvals at home and India

was finally admitted to ITER negotiations on December 5, 2005 at a meeting held at Jeju, South Korea. The meeting of the negotiating teams at Jeju was a landmark meeting as it signified not only India's admission to the ITER venture, but also marked the end of negotiations. With the joining of India, more than half of the world's population is party to the ITER venture.

Agreement for implementing ITER, called Joint Implementation Agreement (JIA), is to be initialed and signed in the next few months. Initialing of the agreement is planned for May 24, 2006 at Brussels and signing towards the end of the year followed by ratification a few months later. With the ratification of the agreement, it will become possible to launch ITER International Organization and set up an ITER Council for decision making. In the meanwhile, ITER preparatory committee (IPC) has already started working to iron out all issues before the formal launch of the international organization. The last meeting of the IPC, the eighth in the series, was held in Goa during April 25-28, 2006 and discussions took place on all administrative, scientific and technical issues.

Some decisions related to top management structure have also been taken. Kanema Ikeda from Japan has been nominated to be the Director General of the prospective ITER organization and Norbert Holtkamp of EU has been nominated to be the Principal Deputy Director General. Both Ikeda and Holtkamp were present at the Goa meeting of the IPC. A few more Deputy Director Generals are expected to be nominated in the months ahead. Discussions on preparatory work will continue in the next meeting of the IPC scheduled to be held in early July at Cadarache.



The ITER INDIA Project

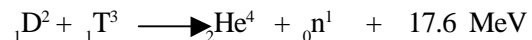
*Prof. P.K. Kaw, Director,
Institute for Plasma Research*

ITER, which is the Latin for “the way”, is an acronym for the International Thermonuclear Experimental Reactor. It is a prestigious international project which will nearly complete the scientific and technological investigations required to build a prototype demonstration reactor DEMO, based on the magnetic confinement scheme of controlled thermonuclear fusion. India has recently joined ITER as one of the seven full partners, the others being China, European Union, Japan, Korea, Russia and USA. India will contribute equipment worth nearly 500 million US dollars to the experiment and will also participate in its subsequent operation and experiments. This equipment will largely be made by Indian industries.

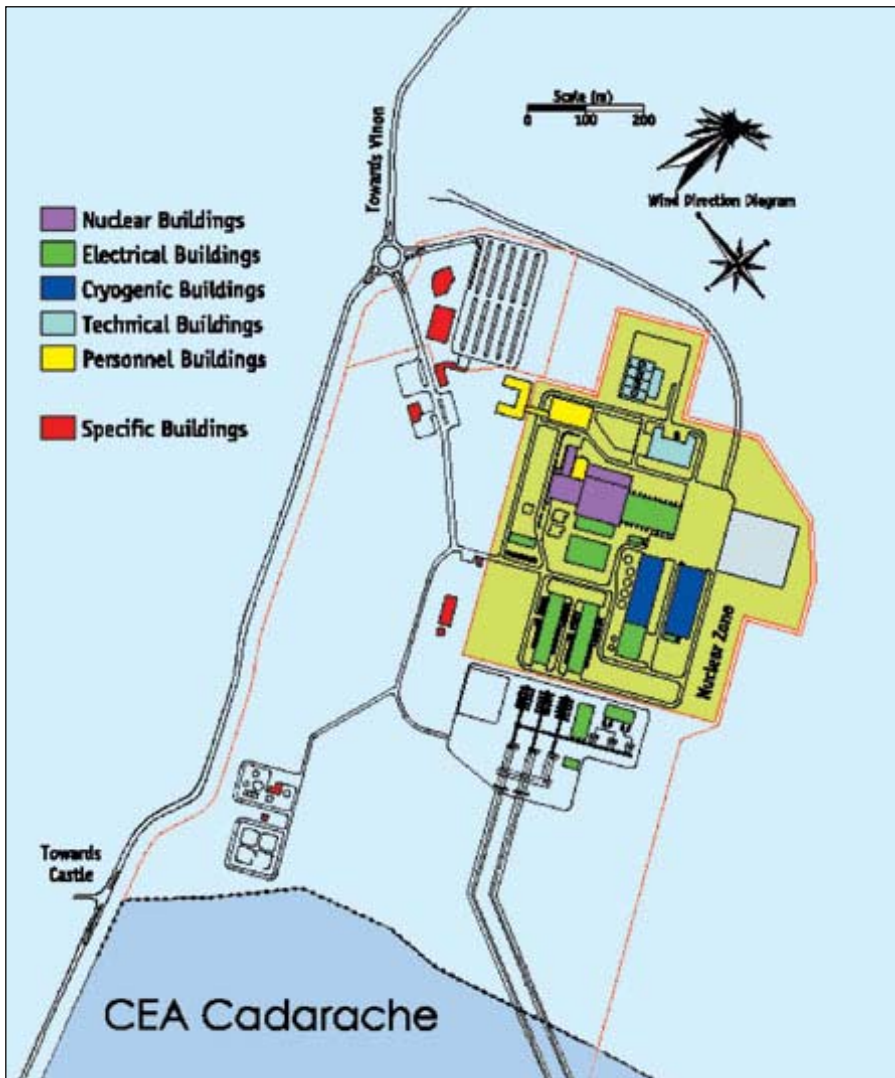
India’s energy needs are enormous. With a rapidly growing economy and the rising expectations of its citizens to enjoy a decent standard of living, the energy requirements of India are simply staggering. We have one fifth of the world’s population but our per capita electricity consumption is still only a quarter of the average of the world, 1/13th of that of Western Europe and 1/30th of that of the United States. Today we are consuming about 130 GW of power, 95% of which comes from thermal or hydro sources. This number is likely to go above 1000 GW by the middle of this century. If this power continues to be produced by the mix we have today, the consequences for our environment are ominous. Therefore, we have to change the energy mix and go to a more aggressive pursuit of nuclear energy and renewable sources. We have an ambitious nuclear energy programme. Right now about 3% of our electricity generation is based on nuclear power. This power generation is based on reactors using natural uranium as fuel and heavy water as moderator and coolant — a technology we have mastered. We are now in the process of building the first 500 MW prototype fast breeder reactor at Kalpakkam, Tamilnadu. This is a follow up on the successful experiments with the 40 MW Fast Breeder Test Reactor. We would like to bring the share of nuclear power to about 10 per cent by the year 2020. Nuclear fusion is viewed as an advanced

successor technology to nuclear fission, and is likely to play a commercially important role sometime in the second half of this century.

Nuclear fusion is the process which has kept the stars burning brilliantly for billions of years. On the earth its devastating power has been seen through the hydrogen bombs. The most convenient fusion reaction is that of heavy isotopes of hydrogen, which are either readily available or may be readily bred from available material in earth’s crust and the oceans :



For the past fifty years or more, controlled thermonuclear fusion experiments have been investigating how to confine a low density fusion grade plasma of deuterium and tritium at temperatures approaching hundred million degrees by magnetic fields, so that slow and controlled release of fusion energy may become possible. This search has led to a successful magnetic bottle concept, viz. the tokamak concept, in which the magnetic confinement geometry is created by a combination of fields produced by external coils, and fields produced by plasma currents. The plasma is heated by the plasma currents and by injection of radio frequency waves and energetic neutral particle beams into the plasma. Once the fusion reaction is ignited, the fusion plasma can be kept hot by stopping of energetic helium nuclei in the plasma. The electrically neutral neutrons carry their energy out of the plasma where it is collected in a blanket, used to generate steam and utilized for electricity generation by the use of standard steam turbine cycles. Large experiments with millions of amperes of plasma current and tens of Megawatts of injected power (like JET in Europe and JT-60U in Japan) have produced fusion reactor grade plasmas with breakeven conditions. Empirical scaling laws have been established which indicate that an experiment of the size of ITER will produce an energy amplification by a factor of 10 and will thus be able to generate about



Aerial view of ITER Project site and its layout

500 mega watts of fusion power. This is why it is important to do an experiment of the size of ITER before designs for prototype commercial fusion reactors can be finalized.

India has had a fusion research programme of its own, since the early eighties. Two tokamaks have been indigenously built at the Institute for Plasma Research (IPR) near Ahmedabad, and a small tokamak has been imported from Toshiba, Japan at the Saha Institute for Nuclear Physics, Kolkata (SINP). The SINP tokamak has been used for an intensive study of low rotational transform tokamak discharges. ADITYA, the first indigenously built Indian tokamak, has been extensively used for the study of plasma turbulence in the edge and scrape-off layer regions. This novel work led to the discovery of intermittency in tokamak turbulence, which is related to the presence of coherent structures in the turbulence and leads to bursty transport effects. The second IPR tokamak, SST-1, is a steady state superconducting tokamak and is currently undergoing commissioning tests. It will have megawatts of ion cyclotron and neutral beam based auxiliary heating. These two tokamaks and associated auxiliary equipment have been built by Indian industries with designs and integration responsibilities taken up by IPR. Many sophisticated diagnostic tools have also been developed at IPR and SINP.

India's contributions to ITER are largely based on the indigenous experience and the expertise available in Indian industry. India will be fabricating the 28 m dia, 26 m tall SS cryostat, which forms the outer vacuum envelope for ITER. The vacuum vessel shields made of 2% boron steel and occupying space between the two walls will also be designed and fabricated by India. It will also take up the design and fabrication of eight 2.5 mega watt ion cyclotron heating sources, complete

with power systems and controls. It will also take up the fabrication of a diagnostic neutral beam system which will give crucial information about the physics of burning plasmas in ITER. India will also be responsible for a number of other diagnostic subsystems. Finally, India will contribute to cryo-distribution and water cooling subsystems. All this equipment will have to be built with ITER quality standards and in a time

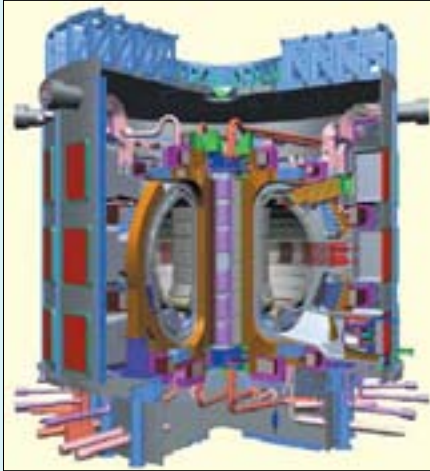
frame (approximately ten years) determined by the International Team at the host site in Cadarache, France. This is the challenge.

The opportunity that participation in ITER offers us, is enormous. This is the first time that we shall be full partners in a prestigious international experiment. We shall have to come to international standards of quality, safety, time schedule maintenance etc immediately. Indian scientists and

engineers will get direct hands-on experience in design, fabrication, operation etc of the latest fusion technologies. India will get access to a number of fusion technologies on the scale relevant to fusion reactors for the first time. If we backup the ITER INDIA effort with an aggressive well focused national programme, it will allow us to leapfrog by at least a couple of decades.



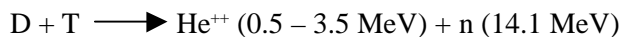
Aerial view of ITER Project site



ITER and India's contribution

Prof. D. Bora
Institute for Plasma Research

In the quest for new energy sources, the world is pinning its hope on controlled thermonuclear fusion as one of the promising futuristic alternate sources of energy. The simplest fusion reaction is when two nuclei of heavy forms (isotopes) of hydrogen (Deuterium and Tritium) fuse together to produce Helium and liberate energy in the form of fast neutrons:



Fast neutrons can be trapped in a blanket, producing heat, which may be used to generate steam and produce electricity using conventional turbines. Once realized, it would have endless source of fuel to continue and very limited controlled radioactive waste. Thus an environment friendly energy source is in the horizon.

It is well known to the Plasma Physics community that at present magnetically confined fusion research has come a long way to start building a test experimental reactor that would pave the way to harness fusion energy commercially. It is heartening to note that finally the site for ITER (International Thermonuclear Experimental Reactor) project is selected, as Cadarache in France and the legal entity will be formed in very near future.

The ITER project originated when the need for a next-step experiment aimed at demonstrating the scientific and technological feasibility of fusion energy for peaceful purposes was recognized among the leading fusion programmes world-wide. A conceptual design was initiated by the European Union, Japan, the Russian Federation, and the United States of America in 1987 and was completed successfully in 1990. It was pursued under IAEA auspices. However towards the end of the engineering design phase, it was recognized by the Parties that due to financial constraints, it was difficult to procure a financial commitment towards the construction of ITER. Therefore, new technical guidelines for minimizing costs by reducing goals, but still retaining the overall programme

objectives of the ITER/EDA agreement were established.

A Special Working Group (SWG) was constituted under the terms of the ITER EDA Agreement, which reviewed the design and recommended a unanimous opinion that the design meets the programmatic objective of demonstrating the scientific and technological feasibility of fusion.

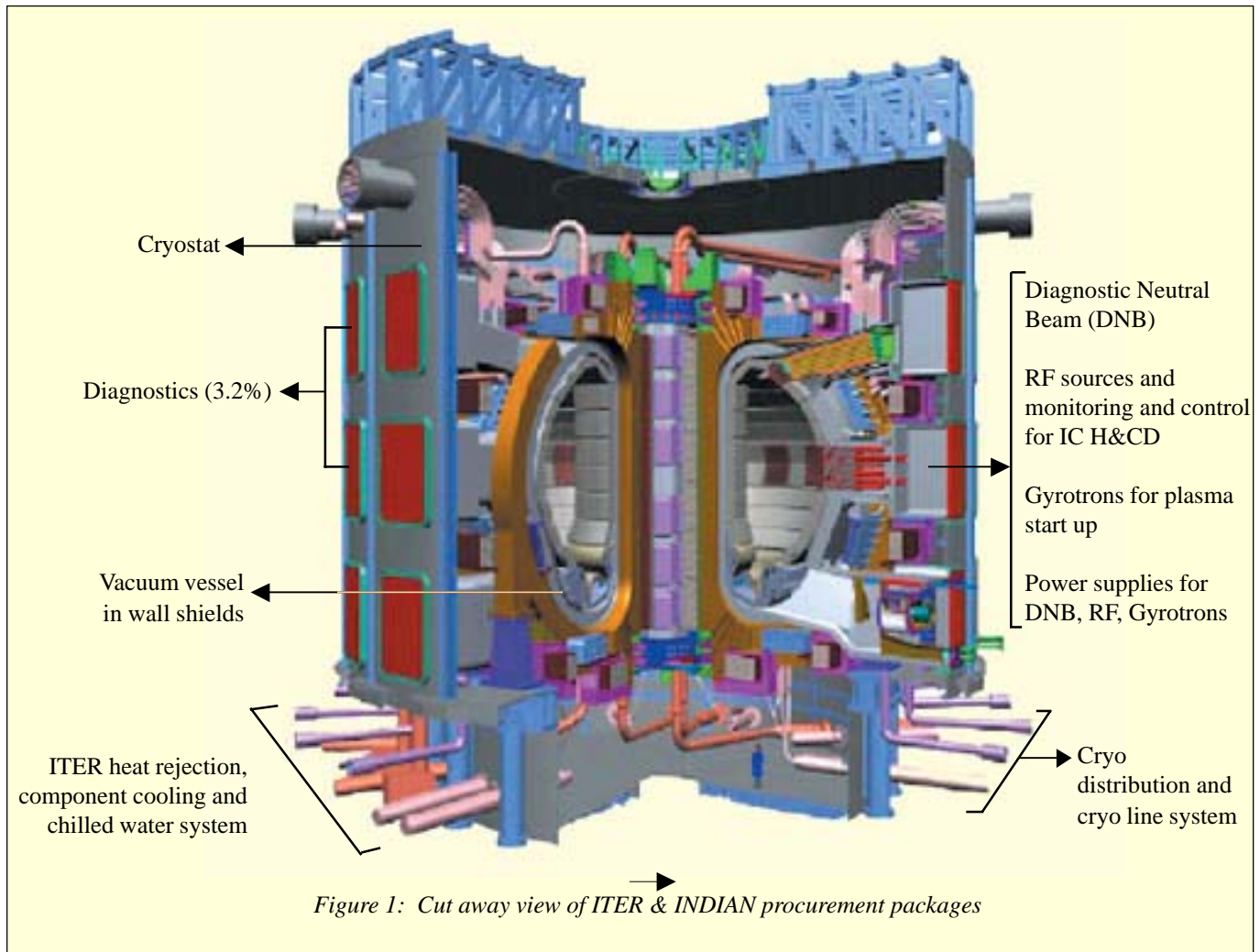
ITER means "the way" in Latin. It is a step between today's studies of plasma physics and tomorrow's electricity-producing fusion power plants. It is an international collaboration to build the first FUSION SCIENCE EXPERIMENT capable of producing a self-sustaining fusion reaction, called "burning plasma".

Unique features will be ITER's ability to operate for long durations and at power levels ~500 MW, sufficient to demonstrate the physics of the burning plasma in a power plant like environment. It will also serve as a test-bed for additional fusion power plant technologies. Once such system becomes a reality, fission-fusion hybrid system also becomes very attractive.

ITER is a long pulse tokamak with an elongated plasma and a single null poloidal divertor (Fig.1). Table 1 gives the main ITER design features and parameters.

Aim is to have nominal inductive operation that would produce a fusion power of 500 MW for a burn length of 400 seconds. The major components of the tokamak are the superconducting toroidal and poloidal field coils which magnetically confine, shape and control the plasma inside a toroidal vacuum vessel. The centering force on toroidal magnets is reacted by the central solenoid. The TF (toroidal field) coil cases are used to support the external PF (poloidal field) coils.

The vacuum vessel is a double-walled structure supported on the toroidal field coils. The magnet system together with the vacuum vessel and plasma facing components inside the vessel are supported by gravity supports. They are located one beneath each sector. Removable components like blanket modules, divertor



ITER Parameters

● Total fusion power	500 MW (700MW)
● Q = fusion power/auxiliary heating power	≥10
● Average neutron wall loading (0.8 MW/m ²)	0.57 MW/m ²
● Plasma inductive burn time	≥300 s
● Plasma major radius	6.2 m
● Plasma minor radius	2.0 m
● Plasma current (I _p)	15 MA (17.4 MA)
● Vertical elongation @95% flux surface/separatrix	1.70/1.85
● Triangularity @95% flux surface/separatrix	0.33/0.49
● Safety factor @95% flux surface	3.0
● Toroidal field @ 6.2 m radius	5.3 T
● Plasma volume	837 m ³
● Plasma surface	678 m ²
● Installed auxiliary heating/ current drive power	73 MW (100 MW)

cassettes, port plugs such as the limiter, heating antennae, test blanket modules, and diagnostics sensors, absorb most of the radiated heat from the plasma and protect the vessel and magnet coils from excessive nuclear radiation. The divertor exhausts the helium ash and limits the concentration of impurities in the plasma. Other in-vessel components are chosen so that they do not contribute unacceptably to the concentration of impurities in the plasma. The shielding blanket design has a possibility of replacement on the outboard side by a breeding blanket at a later time.

The heat deposited in the internal components and the vessel is rejected to the environment via the tokamak cooling water system (comprising of individual heat transfer systems). The system is designed to preclude

releases of tritium and activated corrosion products to the environment. Some parts of these heat transfer systems are also used to bake the plasma facing surfaces inside the vessel by releasing impurities. The tokamak is housed in a cryostat, with thermal shields between the hot parts and the magnets, and support structures which are at cryogenic temperature.

Gas and solid hydrogen pellet will be injected into the vacuum vessel with the help of the tokamak fuelling system. Low density gaseous fuel will be introduced into the vacuum vessel chamber by a gas injection system. The plasma will be initiated with the help of electron-cyclotron-resonance discharge in a circular configuration, which will lean on the limiter. The plasma will be shaped to an elongated divertor configuration as the plasma current is ramped up. After the current flat top (nominally 15 MA for inductive operation) is reached, subsequent plasma fuelling (gas or pellet) together with additional heating for ~100 second leads to a high Q DT burn at 500 MW. With the help of non-inductive current drive from the heating systems, the burn duration will have capability to extend to ~3600 s, or longer. Plasma control is provided by the poloidal field system, and the pumping, fuelling (D,T and impurities such as N₂, Ar) and heating systems, based on feedback from diagnostic sensors.

Safety and licensing issues have been incorporated into the operation scenario. The current design focuses on confinement as the overriding safety function of equipment, other functions being recognized as being required to protect confinement. A “lines-of-defense” methodology is used to obtain the required level of safety while balancing the functional requirements of systems and components. Successive barriers are provided for tritium (and activated dust). These include the vacuum

vessel, the cryostat, active air conditioning systems, with detritiation and filtering capability in the building. Confinement and effluents, normal as well as accidental, are filtered and detritiated, in such a way that their release to the environment is As Low As Reasonably Achievable (ALARA).

Realizing that ITER is an important step on the path to develop fusion energy, India initiated the process of joining ITER as equal partner by showing its desire to the already existing six partners. After a series of steps and negotiations India has become a partner in the ITER project. That makes seven equal partners namely, China, the EU, India, Japan, South Korea, the Russian Federation and the United States of America.

A common understanding on procurement packages for each partner was reached in the Jeju (South Korea) meeting and finalized. The Procurement Allocation amongst the seven Parties has been developed to enable the successful realization of ITER construction, according to the available resources and overall project schedules. The allocation has been made aiming at reduction of project risks and definition of clear responsibilities. The sharing ratio of in-kind procurements by the seven Parties is about 4:2:1:1:1:1:1 respectively.

Procurement packages that require strong design integration and/or the ITER Organization will be procured on-site installation. Corresponding resources are assigned to Fund. India, like the other partners, is making contribution termed as in-cash contribution.

Detailed adjustments on Procurement Allocation and on costs of specific items may arise, while preserving the overall cost sharing by the Parties. In particular, detailed adjustments might take place as a result of the design review conducted

by the ITER Organization and of the pre-qualification, which will be needed for the critical procurement packages shared by multi-Parties.

The committed scope of the Indian contribution towards ITER in kind includes :

- 1) ITER cryostat,
- 2) Vacuum vessel in-wall shields,
- 3) Cryodistribution lines and Cryo-lines,
- 4) ITER water cooling system (Tokamak Cooling Water System, Heat Rejection System, Component Cooling Water System and Chilled Water System),
- 5) 9 numbers of RF power sources each of 2.5 MW in the frequency range from 35 to 60 MHz (Total power of 20 MW) and associated power supplies, monitoring and control system,
- 6) ECH Assisted Plasma Start-Up System (rf sources),
- 7) 8 numbers of Regulated High Voltage Power Supplies (26 kV, 175 A) for Ion Cyclotron Heating and Current Drive Systems, 3 numbers of Regulated High Voltage Power Supply Systems (80 kV,30A) for Gyrotrons of Plasma Start-Up System, 1 number of Regulated High Voltage Power Supply (100 kV,70A) System for Diagnostic Neutral Beam (DNB) and Plasma Arc Power Supplies (1 MW) for the Beam Source,
- 8) 3.2% of the ITER Diagnostics consisting of Optical Emission Plasma Diagnostics and
- 9) Diagnostic Neutral Beam (100 kV, 24 A).

India is also making in-cash contribution of 2.1% for Procurement Packages that require strong design integration and/or the ITER Organization and will be procured on-site installation.

The construction period of ITER is ten years, which will be followed by operation in two phases. Third is the decommissioning phase.

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Fusion Devices in India

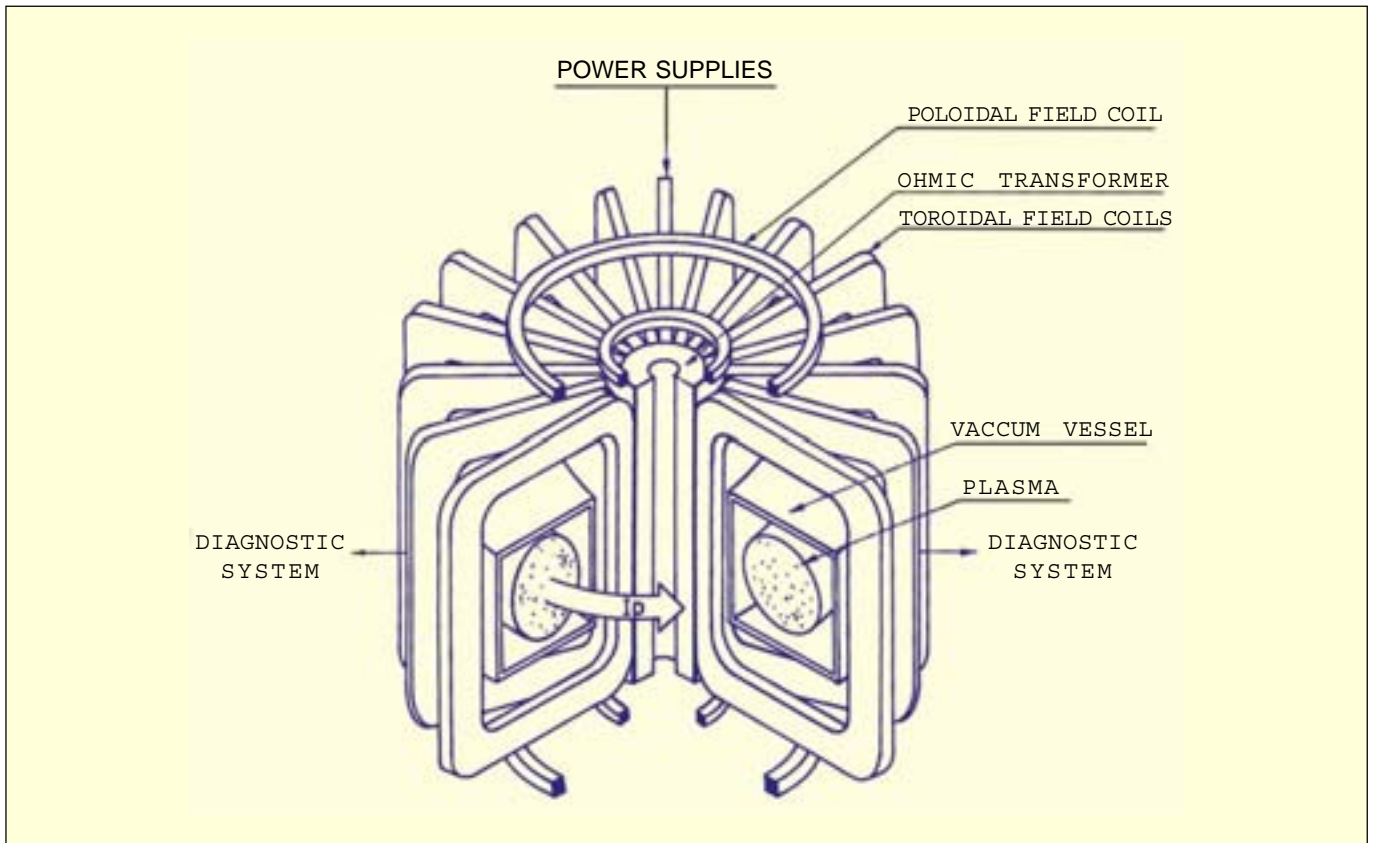
Prof. Y.C. Saxena
Institute for Plasma Research

Fusion research aims at achieving combining the nuclei of light elements to form a heavier element. This nuclear reaction results in the release of large amounts of energy – typically a million times more energy than can be obtained by combining atoms chemically. In a fusion reaction, the total mass of the resultant nuclei is slightly less than the total mass of the original particles. This difference is converted to energy as described by Einstein’s famous equation, $E=mc^2$.

For fusion reactions to occur, the particles must be hot enough (temperature), in sufficient number (density) and well contained (confinement time). These simultaneous

conditions are represented by a fourth state of matter known as plasma. There are three principle mechanisms for confining these hot plasmas - magnetic, inertial and gravity. Magnetic confinement utilizes strong magnetic fields, typically 100,000 times the earth’s magnetic field, arranged in a configuration to prevent the charged particles from leaking out (essentially a “magnetic bottle”). Inertial confinement uses powerful lasers or high energy particle beams to compress the fusion fuel. The enormous force of gravity confines the fuel in the sun and st0ars.

Fusion Programme in India concentrates on Tokamaks, which are special magnetic bottles and are the most



successful fusion devices today. Tokamak is the acronym for Russian word “*Toroidalnya Kamera Magnetnaya Katushka*” meaning the toroidal chamber and magnetic coils. The plasma is confined in the vacuum chamber using toroidal magnetic field produced by external toroidal field (TF) magnets, placed around the vacuum chamber, and the poloidal magnetic field produced by internal plasma current (Fig.1). The self magnetic field of the plasma current and the external toroidal magnetic field generate the closed magnetic field surfaces, which confine the plasma in the vessel. There are also vertical and poloidal field (PF) coils to control the radial movement of the plasma and to shape the plasma. The plasma is produced by gas breakdown using Ohmic transformer which is also used to drive current in the plasma and thus heat the plasma for a period fixed by the flux stored in the Ohmic transformer. The Ohmic transformer consists of a solenoid placed in the central bore of the vacuum vessel and some compensating coils. It acts as a primary and the gas column inside vacuum vessel acts as a secondary of the transformer system. By changing the current in the primary, a loop voltage is induced in the secondary, which breaks down the gas and runs current through the plasma.

The plasma discharge duration can be a fraction of a second to few seconds (depending upon the size of the Tokamak) if only the ohmic transformer is running the current in plasma. This duration can be increased with additional current drive and heating system, e.g. using Radio Frequency (RF) waves and/or energetic Neutral beam injection. Long duration steady state discharges can thus be sustained. For such steady state plasma discharges, it is desirable to replace the magnetic field coils made of copper by superconducting

Table I : Parameters of Tokamaks in India

Parameters		SINP Tokamak	ADITYA Tokamak	SST-1 Tokamak
Major Radius	R(m)	0.30	0.75	1.1
Minor Radius	a(m)	0.045-0.075	0.25	0.20
Toroidal Field	B_T (T)	2.00	1.50	3.0
Plasma Current	I_p (kA)	75	250	220(330)
Plasma Duration	(s)	0.02-0.03	0.25	1000
Plasma Cross-section		Circular	Circular	Elongated
Elongation		—	—	1.7 - 2.0
Triangularity		—	—	0.4 - 0.7
Configuration		Poloidal Limiter	Poloidal Limiter	Double/single Null Poloidal divertor
Type of Magnetic Field Coils (TF and PF)		Copper Air cooled	Copper Water cooled	Superconducting At 4.5K
Current Drive		Ohmic Transformer (Iron Core)	Ohmic Transformer (Air Core)	Ohmic Transformer (Air Core) + LHCD
Heating		Ohmic	Ohmic	Ohmic + ICRH
Vacuum Vessel		Conducting Shell (Al)	Vessel with electrical break	Vessel without break
Location		SINP Kolkata	IPR, Gandhinagar	IPR, Gandhinagar
Commissioning		1988	1989	2005
Design and Fabrication		Toshiba, Japan	Indigenous	Indigenous

Important investigations in each of these regimes are listed in the following.

Experiments in normal q_a regime:

- Drift wave like instability in Tokamak core region.
- Anomalous current penetration.
- Temperature fluctuation induced anomalous transport.
- Origin of inversion of up-down potential asymmetry.
- Observation of Runaway Electrons by ECE.
- Observations of Current holes & negative currents.

Experiments in low q_a regime:

- Accessibility condition for very low q_a (VLQ) and ultra low q_a regimes.
- Anomalous Ion heating in VLQ discharges.
- Edge Biasing experiments in VLQ discharges.
- Runaway Electrons in startup phase of VLQ discharges.
- Variation in Up-down asymmetry with q_a .



Figure 2: A view of SINP tokamak

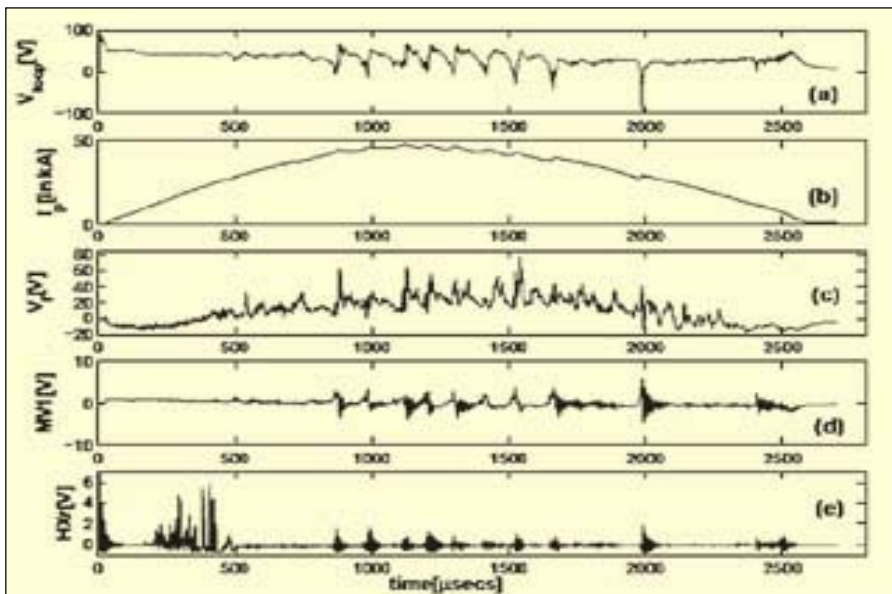


Figure 3: A typical Tokamak discharge in SINP Tokamak: (a) Loop voltage signal (V_{loop}); (b) Plasma Current (I_p); (c) Langmuire Probe voltage (V_p); (d) Mirnov Signal (MVI); (e) Hard X-ray burst (HXr)

coils in order to reduce power consumption.

Fusion research in India is currently being carried out in two Tokamak devices, the ‘‘SINP Tokamak’’ at the Saha Institute of Nuclear Physics, Kolkata, and the ‘‘ADITYA Tokamak’’ at Institute for Plasma Research (IPR), Gandhinagar. The assembly of a new Steady State Superconducting Tokamak ‘‘SST-1’’ has been completed at IPR and this device is being commissioned. The parameters of these devices are listed in Table 1. This article gives a brief

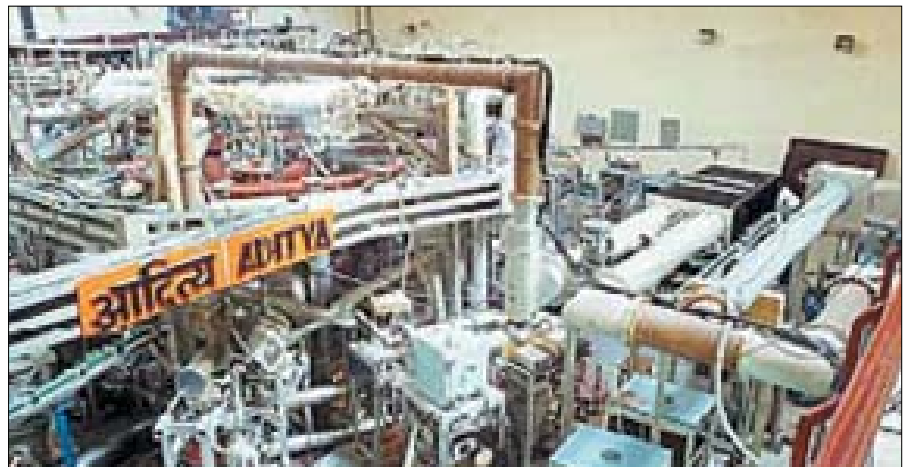


Figure 4: A view of tokamak ADITYA with associated diagnostics and ICRH heating system

introduction to these devices and summaries some major results.

SINP Tokamak

The SINP Tokamak is a small iron core Tokamak with a pulse duration about 20 ms. Designed and fabricated by M/s Toshiba, Japan the device was installed and commissioned in 1987. A picture of the device is shown in Fig. 2. The Tokamak operates in both low edge safety factor, q_a ($2 > q_a > 1$) and normal q_a (above -2.5) modes. A large number of diagnostics, including the special diagnostics like, Spectroscopy, Microwave interferometry, Internal magnetic coils, Rogowskii coils, Time of flight analyzer, and Hard X-ray systems for Runaway electrons which has been developed indigenously and added to the SINP Tokamak.

A set of plasma parameters observed in typical discharge in SINP Tokamak is shown Fig.3. Experiments have been carried out both in the normal q_a and low q regimes.

Future programme on the SINP Tokamak include investigations on the role of runaways in H modes, turbulence studies using new data analysis techniques like wavelet analysis, fractal analysis and higher order spectral analysis.

ADITYA Tokamak

ADITYA, the first Indian Tokamak, conceived, designed, largely

indigenously fabricated and erected at the IPR, is a medium size low field Tokamak. Commissioned in 1989, it has a plasma of circular cross-section with major radius 0.75 m and minor radius of 0.25 m. A toroidal field of up to 1.5 T can be produced at plasma

center with the help of 20 toroidal field coils spaced symmetrically around a toroidal vacuum vessel. The chief scientific objectives of ADITYA are (i) investigation and control of edge phenomena; (ii) investigation of density and current limits of a

Tokamak with special emphasis on interesting phenomena like MARFES, detached plasma, disruptive instabilities and their control; (iii) study of novel regimes of operation such as those with currents dominantly in energetic current carriers etc. Fig. 4 shows a view of ADITYA.

The TF coils, each having 6 turns with a current of 50 kA through each of the turns, are picture frame type, having a bore of 0.78 m x 0.90m and outer dimensions of 1.03 m x 1.26 m. The Ohmic transformer for ADITYA consists of a central solenoid, which stores the required flux and three additional pairs of compensating coils in order to minimize the field within the plasma region. The solenoid produces a peak field of 3.2 Tesla within the bore to store a flux of 0.6 Volt-sec at peak current of 20 kA. The vertical field for ADITYA is generated by means of two pairs of the vertical field coils designed to carry 10 kA peak current.

The vacuum vessel of ADITYA is a torus of major radius 75 cm with a square cross-section of side 60 cm, a volume of ~ 16 cu m and total surface area of ~ 75 sq.m. The vessel, designed for ultra high vacuum (UHV), is made out of SS 304 L material and assembled in four quadrants. The vacuum vessel has been subjected to various wall treatment procedures in order to be compatible for UHV $\sim 10^{-8}$ torr. A combination of Glow Discharge Cleaning and low temperature Pulsed Discharge Cleaning is used for keeping the surface clean. The vacuum vessel is pumped by two turbo-molecular pumps each having a pumping capacity of 2000 litres per second for air. In addition during discharge cleaning, two cryopumps, each having a pumping capacity of 10,000 litres/sec for water vapour and condensible hydrocarbon, are also used.

Pulsed power (~ 50 MW per pulse) is required to energize the magnetic

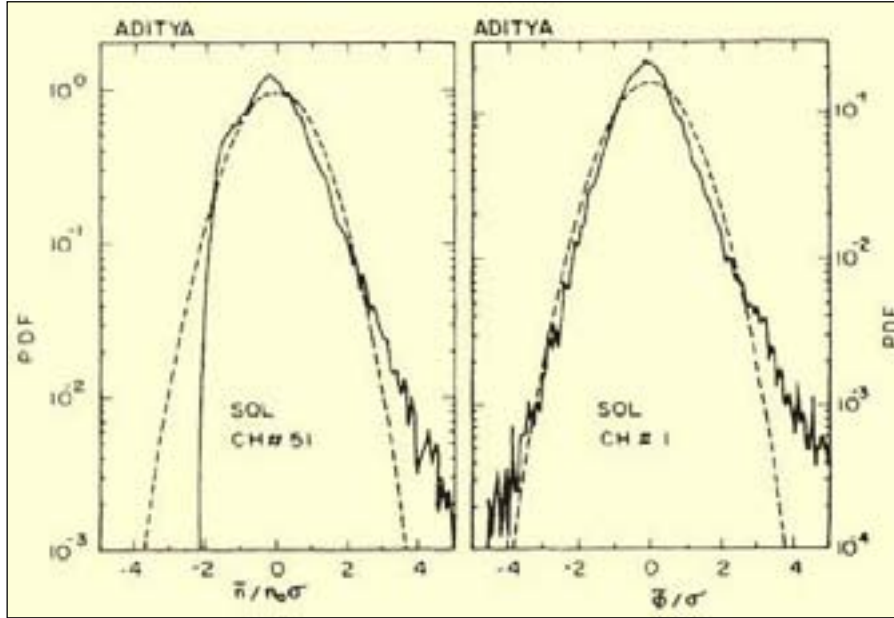


Figure 5: Probability Distribution Functions for density (left) and potential (right) fluctuations in edge plasma of ADITYA.

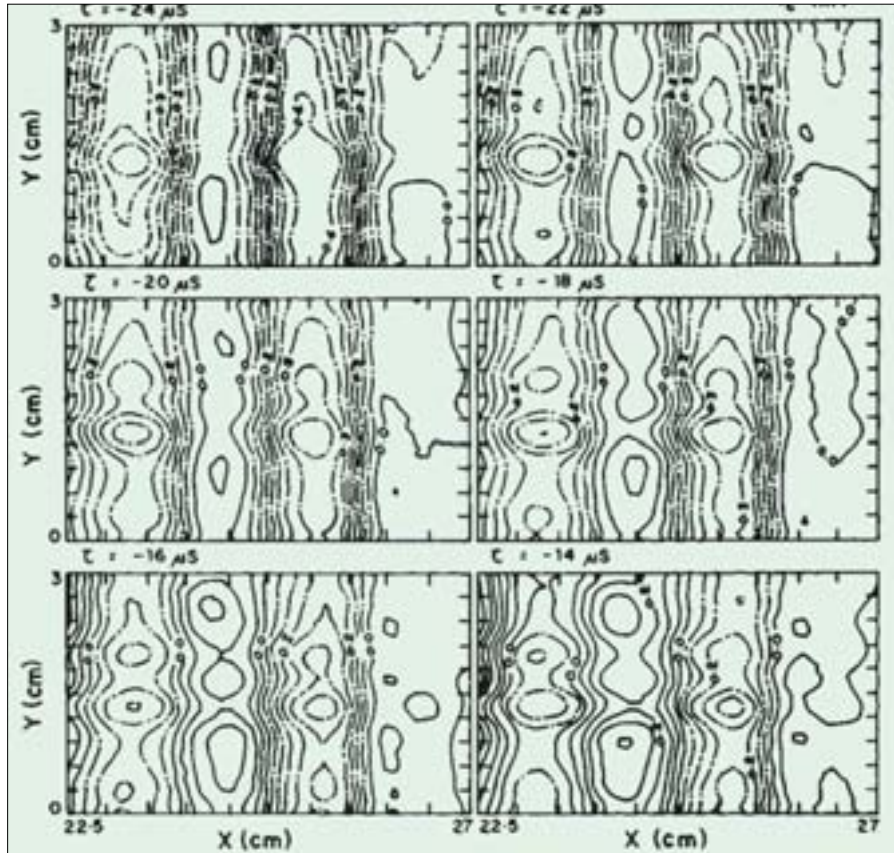


Figure 6: Coherent structures in edge turbulence in Tokamak ADITYA

field coils of ADITYA. Two types of pulsed power systems are provided (1) a triggered D.C. power supply cum capacitor based power system for low parameter regime operations; (2) a controlled rectifiers cum wave shaping circuits system drawing energy from the mains grid for full operational parameters. A large number of key diagnostics e.g. microwave interferometer, Laser Thomson scattering, Visible and UV spectroscopy, soft X Ray imaging, ECE radiometer, Langmuir and magnetic probes have been developed and deployed on ADITYA. The device operates under full control of automated data acquisition and control system.

In Tokamaks, edge and core plasma are integrated and phenomena in edge influence core transport profoundly. Experiments in ADITYA have been dedicated to edge turbulence studies. The intermittency in Tokamak edge turbulence was discovered in ADITYA. The probability distribution functions of the density and potential fluctuation were found to be non-Gaussian (Fig.5). This was followed up by first observation of coherent structures in conditional statistical analysis of the fluctuations (Fig. 6). The results indicated the particle transport in Tokamaks is not a steady ooze but comes out in bursts. Other important experimental results include scale dependence of the fluctuation dependent particle transport, and Levy's statistics of edge turbulence.

ADITYA has been upgraded recently to obtain better parameters and performance. The vacuum system has been improved by improving the surface cleaning facilities. To increase the plasma energy content during the discharge, auxiliary heating systems have been integrated. A 20 – 40 MHz, 200 kW Ion Cyclotron Resonance Heating (ICRH) system has been integrated to ADITYA vacuum vessel. A 28 GHz, 200 kW gyrotron

based electron cyclotron resonance heating (ECRH) system has been successfully commissioned on ADITYA Tokamak. ADITYA is regularly being operated with the transformer-converter power system with ~100 msec 80 - 100 kA plasma discharges at toroidal field of 8.0 kG.

SST-1 Tokamak

A steady state superconducting Tokamak SST-1 (Fig. 7) has been

designed, fabricated and assembled indigenously and at IPR. The objectives of SST-1 include studying the physics of the plasma processes in Tokamak under steady state conditions and learning technologies related to the steady state operation of the Tokamak. These studies are expected to contribute to the Tokamak physics database for very long pulse operations. The SST-1 Tokamak is a large aspect ratio Tokamak,



A view of SST-1 tokamak

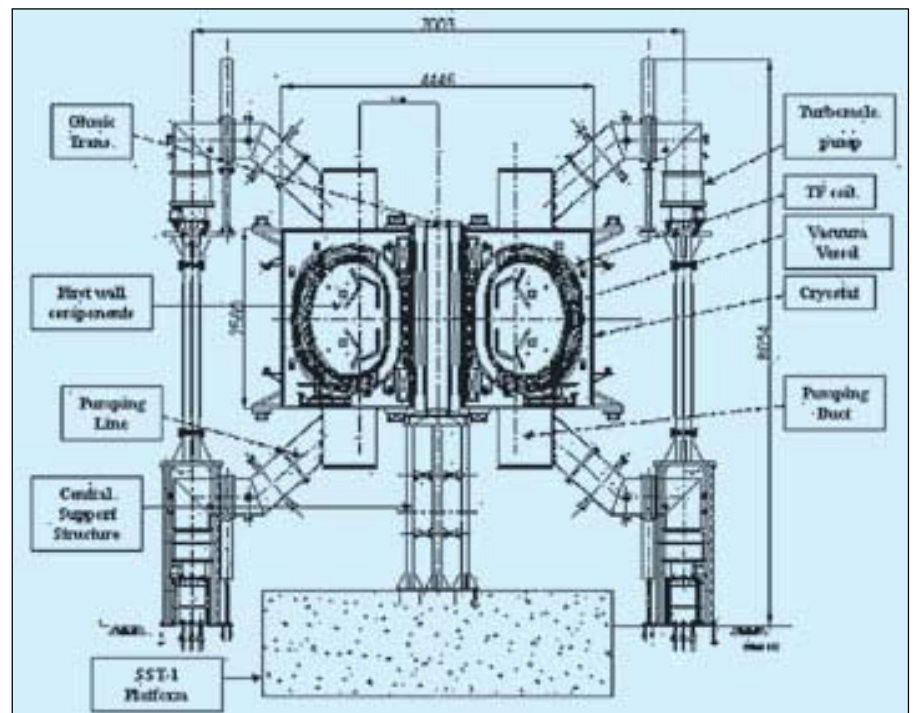


Fig. 7: A schematic cross-section of SST-1 Tokamak.



Figure 8: TF Coil of SST-1



Figure 9: A view of vacuum vessel

confinement and (ii) a high vacuum compatible cryostat to provide operational environment to all superconducting magnets. Vacuum vessel is made of sixteen modules to maintain modularity with sixteen TF coils. Poloidal cross-section of the vessel is close to 'D' shape with height of 1.62 m, a mid-plane width of 1.07 m. Vacuum vessel, with a total volume of 16 cu m and surface area of 75 sq.m, is fabricated from SS304L material. A view of the interior of the vacuum vessel is given in Fig. 9. Cryostat is toroidally continuous sixteen-sided polygonal vacuum chamber, made of SS304L with a volume of 35 cu m and surface area of 59 sq.m, and encloses vacuum vessel and all superconducting magnets. Superconducting coils operating at 4.5 K are protected against thermal radiation from hot surfaces using liquid nitrogen cooled panels. These panels are maintained by passing liquid nitrogen at a rate of 1200 litres/hour.

Sixteen numbers of turbo molecular pumps, each with a pumping speed of 5000 l/s at 10^{-3} T for hydrogen, are to be connected to sixteen pumping lines on the vacuum vessel. Two closed cycle cryo pumps will be used during wall conditioning of vacuum vessel. The net pumping speed provided for cryostat using two turbo molecular pumps is 10,000 l/s. Vacuum vessel and cryostat will be pumped down from atmospheric pressure to 10^{-3} torr using two separate root pumps of 2000 cu m/hr capacity.

Plasma Facing Components (PFC) of SST-1, consisting of divertors, passive stabilizers, baffles and limiters are designed for long pulse operation. A design based on the mechanical attachment of the isostatically pressed graphite tiles, having 100 w/mK thermal conductivity, to the copper alloy (Cu-Cr-Zr & Cu-Zr) heat sink back-plates is adopted for the PFC. The heat sinks

configured to run double null diverted plasmas with significant elongation (κ) and triangularity (δ). It has a major radius of 1.1 m and minor radius of 0.20 m. A plasma current of 200 kA will be produced in a toroidal field of 3 T. Radio frequency waves in different frequency bands will be used to drive the current and heat the plasma for 1000s. Superconducting magnetic field coils are deployed to produce the toroidal and poloidal fields in SST-1.

The Magnet System of SST-1, consists of sixteen superconducting (SC) TF magnets, nine SC PF magnets, a pair of resistive poloidal field magnets, a pair of single turn Feedback Coils inside the SST-1 vacuum vessel and a system of five ohmic coils apart from a pair of resistive vertical field coils.

The TF coils are D-shaped (Fig.8). For a maximum current of 10 kA per turn the magnetic flux density at plasma center is 3.0T with a stored energy of 56 MJ. The bore of each TF coil is ~ 1.2 m x 1.8 m. All coils are connected in series and are protected against the quench by suitable dump resistance, switching and sensing system. The PF coils are cylindrical in shape, have diameters

ranging from 0.9m to 2.7m and allow for a variety of Plasma equilibria with wide range of elongation and triangularity. The Ohmic transformer of SST-1 has a central solenoid of diameter 0.9 m and height of 2.6 m, and stores a flux of 1.2 volt-sec at peak current of 20 kA with maximum field of 8T in the bore.

A unique feature of the SC Magnet System of SST-1 is that the same basic conductor is used for both the SC TF and PF magnets. A CICC, based on NbTi with high copper to SC ratio in the strands has been designed and manufactured, for the first time, for use in Tokamak magnets. The CICC consists of 135 strands of 0.86 mm ϕ each, twisted at a cabling pattern of $3 \times 3 \times 5$. Each strand consists of $10\mu\text{m}$ ϕ , 1224 NbTi/Cu filaments and has an overall Cu/NbTi ratio of 4.9:1. The final stage cable is jacketed with a 1.5 mm thick SS304L conduit through roll forming and welding process giving a final dimension of 14.8 mm \times 14.8 mm. The operational current of CICC is 10 kA at 4.5K and 5.0T field with a critical current of 36kA.

SST-1 has two vacuum chambers, (i) a UHV compatible vacuum vessel for plasma production and

are cooled with water flowing through SS tubes brazed on the copper alloy back-plates so as to keep the surface temperature below 1000 °C at peak heat flux.

The SC magnets of SST-1 are cooled using forced flow of supercritical helium (SHe) through the void space in the CICC. Further the magnets have to be energized from power supplies at room temperature using vapor cooled current leads, which evaporate liquid He at cold end to gas He at ≈ 300 K at the warm end of the lead. A LHe plant with a capacity of 400 W refrigeration at 4.5 K and 200 l/h liquefaction at the pressure 1.2 bar (a) has been designed and fabricated by M/S Air Liquide DTA, France and installed at IPR. The plant also provides the refrigeration capacity of around 250W for the heat dissipation in the cold circulation pump. A cold circulation system, for the flow of SHe through the SC coils in a closed cycle, forms part of the LHe plant. An external dual bed, full flow, on-line purifier with automatic regeneration is provided to remove impurities. A buffer main control dewar (MCD) is provided in the plant, to absorb the heat loads generated within the SC coils and return cold helium vapors to the cold box at a constant temperature and pressure. The SHe coming from the SC coils delivers heat to the MCD through a suitably designed heat exchanger before returning to the cold pump. The MCD also serves as a reservoir for supplying LHe to the current leads. The capacity of the MCD is 2500 l. A total He gas inventory of ~ 3100 Ncu m is required for the refrigerator/liqefier and SCMS. Two high-pressure tanks, each of 25 cu m capacity at 150 bar provide storage equivalent to twice of the required inventory. A medium pressure storage system, at 14 bar with a total capacity of 272 cu m, provides capacity to store total inventory as pure gas.

Thermal radiation shields, cooled by liquid nitrogen (LN_2) are present between the SC coils and vacuum vessel as well as between the cryostat and SC coils. The LN_2 consumption, during different phases of operation, varies between ≈ 500 l/h to ≈ 1500 l/h. Three storage tanks, of 35000 l capacity each, have been installed for storing LN_2 . The LN_2 is purchased commercially and filled in these tanks to replenish the consumed liquid. The gas/vapors from the applications are released to atmosphere. A LN_2 management system, consisting of the main LN_2 transfer line, the phase separator/sub-cooler Dewar containing suitably designed heat exchangers, the LN_2 distribution & return lines and N_2 gas vent line, has been designed and installed. The system is controlled by fully automated instrumentation and control system based on a PLC.

SST-1 will have three different high power radio frequency systems, the Ion Cyclotron Resonance Heating (ICRH) system, the Lower Hybrid Current Drive (LHCD) system, and Electron Cyclotron Resonance Heating (ECRH) system, to additionally heat and non-inductively drive plasma current to sustain the plasma in steady state for duration of up to 1000 sec. ICRH system, developed indigenously at IPR, operates between 20-100 MHz range at different fixed frequencies and is modular in out put power. The LHCD system, at 3.7 GHz, is based on two 500 kW, CW Klystrons with four outputs. The ECRH system is based on a 200 kW, CW Gyrotron at 83.6 GHz. Neutral beam injection system capable of delivering 2 MW of neutral beam power in 30 – 80 keV of H^0/D^0 beams, in an operational duty cycle of 1000 S On/ 5000 S Off, is being developed for the purpose of heating the SST-1 Tokamak plasma.

A distributed control and data acquisition system has been installed

for the operation of SST-1. Commissioning tests are presently being carried out to prepare the machine for first plasma.

Summary

SINP Tokamak and ADITYA Tokamak are presently operation in India and are being used in fusion related science and technology development. A superconducting Tokamak SST-1 is being commissioned and is expected to contribute to the physics of steady operations of the Tokamaks.

From page 8

India has committed in this long term programme with the following objectives, namely,

- 1.To fulfill the commitment of delivering Procurement Packages accepted by India.

2. To contribute to the research in burning fusion plasma in the ITER.

3. To acquire self-sufficiency in the critical area of fusion reactor technologies by actively participating in construction and operation of ITER.

At the successful completion of this project India will be ready to build its fusion reactor

For the present, it is a matter of pride that the Indian fusion research activities have been recognized by the International community and as result of which we have become the seventh partner in the ITER project. It is the second largest international scientific venture ever undertaken by mankind. It is now for us to prove our competence by supplying the packages, which are in our share.



Fusion Technology Development in India

Prof. S.K. Mattoo
Institute for Plasma Research

The world wide development of fusion technology as an energy source has a unique continuity from its history to the contemporary present and linkages to the future are contemplated through a vision of road map of making available commercially usable power. An unusually long period of many decades of years of intensive R&D is still required before fusion technology matures into an energy technology. Very substantial scientific and technical challenges must be mastered for building even fusion experimental devices, which act as test beds for the development of fusion technology. These fusion experiments have to be necessarily large in size and built in several steps of increased complexity. Substantial financial investments are needed besides mobilizing a large scale of resources over a very long period. These conditions make it imperative that we have a national

vision on development of fusion technology in the country and are an active participating party in any international cooperation.

To generate competence in the design, construction and operation of fusion reactors, we have built a series of plasma devices, core and auxiliary systems, with increased technological complexity over last 20 years. These devices have varied from simple linear systems to toroidal systems. While Aditya is a Copper pulsed system, Steadystate Superconducting Tokamak (SST-1) uses super conducting coils, auxiliary heating systems of RF and neutral beams, heat-load bearing first wall, divertor for control of plasma-wall interaction, an extensive web of sophisticated diagnostics for measuring different plasma quality parameters and a high speed and a large volume data acquisition and control system. In parallel, this framework

Vacuum vessels



of programme has ultimately led to a trained manpower both in the institutions and in the industry. This has led to a competence for participation in the international cooperation programme. In this report we catalogue some of the important fusion technologies we have developed while building these devices. However, some important elements of fusion technology remain to be developed. Areas where initiatives need to be taken are pointed out.

International cooperation programme is oriented towards the reactor research. It consists of two intermediate phases of ITER and DEMO (Demonstration Fusion Power plant) before construction of the first commercial fusion reactor in around 2050. ITER is a seven-sided partnership between the European Union, Japan, Russian Federation, the United States of America, China, South Korea and India. DEMO, not an international cooperation as yet, is intended to demonstrate the technical feasibility of a fusion power plant and generate electricity in continuous operation for the first time. In parallel to ITER, construction of a special high-intensity fusion neutron source is needed to develop and test low activation material.

Vacuum technology of large vessels

Large vacuum vessels are required for SST-1. In an ultra high vacuum large toroidal vessel (16m^3) plasma is formed, heated and manipulated. A cryostat of 39m^3 maintains the vacuum surrounding the superconducting magnets. The vacuum vessel has close to D-shape cross section, weld joints that can allow re-use of vessel and cryostat components, hot nitrogen gas baking and is electro polished and ultrasonically cleaned. In comparison our first tokamak, Aditya, is a vacuum vessel of 2m^3 with a square cross-section of $0.6\text{m} \times 0.6\text{m}$.



First wall with actively cooled CFC tiles

Beam line for 1.7 MW neutral beam injector consists of 2 large rectangular chambers (40m^3) and a double walled chamber for Large Volume Plasma Device (LVPD) is used to source a 9m^3 quiescent, uniform plasma of density 10^{12} per cm^3 over a diameter of 2m.

This experience has created manufacturing and design capability of vacuum vessels suitable for fusion devices in the industries in India.

First wall

First wall is the solid material surface which comes in contact with burning fusion plasma edge and receives all the radiated power of neutral and charged particles, neutrons and electromagnetic radiation in the frequency range of visible to x-rays and gamma rays. It should not contribute high-Z impurities to cool the plasma and extinguish the fusion fire and it should have the capability of removing the received heat at $\sim 6\text{ MW/m}^2$ in a fusion reactor.

For SST-1, we have developed actively cooled CFC tiles bolted to Cu-Cr-Zr with a capability of heat removal at power density of $\sim 1\text{ MW/m}^2$. Next step is to develop braze joints between these two dis-similar



Superconducting TF coils under test

materials to raise capacity to 10 MW/m^2 .

Superconducting magnets

Very strong magnetic fields (\sim few Tesla) are required to confine the plasma in the vacuum vessel and prevent it from touching the walls. If conventional resistive electromagnets are used, too much energy is wasted in the form of heat. To limit the energy needed to produce the magnetic field, super-conducting magnets are used. These magnets have to operate at liquid helium temperature (4.5 K).

Liquid helium at 4.5 K is continuously passed around the magnet strands of to keep it at low temperature and ensure it remains in a super-conducting state. Once energized, these magnets can operate continuously with very high efficiency and therefore are perfect for a steady state fusion reactor. As these magnets run at liquid helium temperature it is necessary to operate them in vacuum to prevent the heat in the atmosphere from boiling off the helium.

Super-conducting magnets are made from NbTi/Cu Cable-in-Conduit-Conductors (CICC's) of 24 pieces with a piece length of 600m each. The CICC consists of

135 strands with 0.86 mm diameter each twisted at a cabling pattern of 3×3×3×5. Each strand consists of 10µm-dia 1,224 NbTi/Cu filaments and has an overall Cu/NbTi ratio of 4.9:1. The final stage cable was jacketed with a 1.5 mm thick SS304L conduit through roll forming and welding process giving a final dimension of 14.8 mm×14.8 mm. These CICC's are then wound into the final shape of the magnet. One of the major engineering challenges for fusion has been to be able to manufacture very large superconducting magnets that operate reliably and safely.

The superconducting magnets are the most expensive components of a fusion reactor.

Cryogenic Systems

A 1.3 kW, 4.5K refrigerator/liquefier with a cold circulator provides forced flow cooling of superconducting magnets and an 110W refrigerator uses a vacuum screw compressor to cool million l/s speed cryopumps to 3.8K. Both the systems are supported by a management of warm helium gas and LN₂, integrated flow schemes, robust control systems, an indigenously developed current and helium feeds, transfer lines, large size cryostat test facility and electric isolators at 5kV.

RF Power Systems

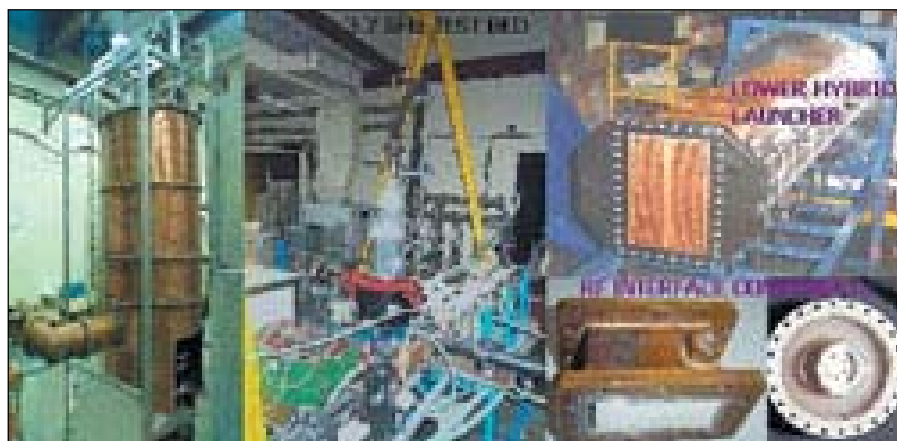
Full RF spectrum, meter to millimeter waves, has been exploited to provide MW steady state power, high power and UHV compatible transmission line components, dummy loads, vacuum windows, high power (>1.5MW) launchers, transmission line components, high power passive components and a test bed at 3.7 GHz. Our experience, expertise and integrated resources have all worked together through a range of indigenous development programmes that enable us not only support fusion research in India but also other power intensive steady state applications in space and industry.

Neutral Beam Injector

To date a positive ion based neutral beam technology has been developed to extract 31 MJ for 14 S at the rate of 1.7 MW at 41 kV. The injector has the capability to provide hydrogen and helium beams at beam energies up to 80 kV. Construction of injector has led to development of cryopumps with million l/s speed, heat exchangers with capability of removing heat at power density of 15MW/m², steady state ion extractor cum accelerator, multi-pole bucket plasma boxes with plasma density of 10¹² per cm³, integrated control system for operations and special alloys of Cu-Cr-Zr.



Neutral beam components



High power RF components



Regulated high voltage power system

Regulated High Voltage Power Supplies

IPR have built highly reliable 80 kV, 130 A regulated high voltage power supplies. These supplies are fully solid state, modular, use synchronized pulse width modulation, have programmable turn on/off time (3 ms to 300 ms), low fault energy of < 5J and clearing time of < 3 ms.

The supplies are used to power triodes, tetrodes, klystrons, gyrotrons and neutral beam injectors for fusion applications, communication and nuclear engineering.

Engineering services

Making of a fusion reactor is a complex activity of design, modeling, prototyping of concepts, development of engineering processes, manufacturing, metrology, quality assurance and management of resources, schedules and budget. Design of systems, right from their conceptualization to the emergence of the final engineering drawings; have used softwares like AUTOCAD, MECHANICAL DESKTOP, ANSYS, ANSOFT, KORBA, and POISSON etc. have been used at various stages. Analyses of the design outputs using these softwares for various components of SST-1 have helped in finalizing their basic frame ware through studies of response of mechanical structures to thermal loads, electromagnetic fields and the interactions of coupled fields and, wherever applicable, to flow of cryogenics/water.

BRACC, SWHAP, SPARK 1.0, ACCOME, LSC, BRAMBILLA AND ECRCYL have assisted in deployment of auxiliary heating and current drive systems on SST-1.

SST-1 and its sub-systems are integrated, operated, controlled and monitored through a network of computers and specially written application soft wares. Inputs for these tasks were obtained from modeling various scenarios of operation of SST-1 plasma through

use of TSC, IPREQ, PEST2, ERATO AND DEGAS.

A large amount of data pertaining to housekeeping and experiments on SST-1, is acquired on dedicated data acquisition systems connected to SST-1 and its sub-systems. A single discharge of 1000 seconds long yields a data volume of 10 GB distributed over 5000 data channels. Such a large volume of data is managed and manipulated through a network of devices consisting of specially developed software tools of analysis, feedback and integrated control loops, interpretation codes and archival systems.

A clear need exists for development of a whole range of engineering and material production processes and manufacturing techniques for constructing DEMO like reactor. Alloys and jointing processes for dissimilar materials (alloys, metals and ceramics) have to bear large stresses and forces, fatigue, erosion, and nuclear transmutation. Facilities for novel jointing processes such as high-pressure electron beam welding, hybrid metal inert gas and laser welding, impulse magnetic field welding and large area brazing form primary requirements for fabrication of fusion reactor grade components.

New Technology Initiatives

We are contributing cryostat, vacuum vessel suppressor system, RF and diagnostic neutral beam power, cryo- and cooling water distribution system and some diagnostics towards construction of ITER as a consequence of fusion technologies we have developed while building various plasma devices and two tokamaks of Aditya and SST-1. Participation in ITER will give us access to operation of systems contributed by other participants and operation of the fusion reactor, but may not give to our industry manufacturing and production capability of those systems. Thus

ITER is not a solution to the shortfalls in the fusion technology of the country. ITER is a window of opportunity for laying a plan for infrastructure in fusion.

The energetic neutrons released from fusion reactions do not interact with the plasma. The role of the blanket, which will surround a commercial reactor, is to slow down the neutrons, recover their energy and use them to transform lithium into tritium. The tritium can then be extracted, processed and added to deuterium for refueling the reactor. We need to develop concept for breeder blankets and probably use one of them on ITER.

Fusion reactors will require advanced low activation materials to ensure that the fusion wastes will not be a long-term burden to the future generations. These materials must be resistant not only to neutrons but also to high surface heat loads and thermal cycling. Thus a concentrated effort needs to be directed towards the development of special materials that can withstand the hostile environment existing in such reactors. A facility is currently being considered under an international collaboration called the International Fusion Material Irradiation Facility (IFMIF) to test such these new materials. We should participate in this international cooperation.

In making of superconducting coils, we have generated a competence for winding of coils in BHEL. This needs to be augmented. We have to establish an infrastructure for production of superconducting material, forming of strands and a cable suitable for winding.

The internal structure of a fusion reactor will become radioactive during operation due to neutron radiation and the use of Tritium. It is necessary, therefore, to be able to replace components inside the

Continued on page 29

Shimomura: A Conversation

Dr. Yasuo Shimomura is the Interim Project Leader of the International Thermonuclear Experimental Reactor (ITER) project. Prof. P. I. John from the Institute for Plasma Research talked to him in the background of the 12th ITER Negotiation meeting, which was held in Jeju, South Korea from 1-8 December 2005. During this meeting, India's accession to ITER as a full member was announced



John: This is a historic moment for ITER since the negotiations are coming to a successful conclusion and its transition to an international entity will take place soon. For India, this is equally a memorable occasion since we have become a full member of the ITER project. Thank you for finding time for this conversation amidst your pre-occupations with the negotiations. Let me start by asking about the historical origin of the ITER concept. Is it true that Gorbachev first took the initiative?

Shimomura: Yes. Academician Velikhov sold the idea of a joint development of fusion energy with US and other countries to First Secretary Gorbachev. In 1985 in Geneva, he proposed this collaboration to President Reagan who agreed immediately. Politically, the time was such that some visible collaboration between Russia and US was needed. Initial collaboration was of course for a study.

John: What was your background when you joined ITER?

Shimomura: I was born in Osaka in Japan in 1944. I did my Ph. D. in plasma physics in Osaka University in 1971. I moved to the Japan Atomic Energy Research Institute (JAERI), which was constructing the first tokamak JFT 2. This was a conventional tokamak. Japan wanted to start advanced tokamak research. So along with Dr. Masaji Yoshikawa, we proposed diverter tokamak experiments and I became the leader of the project from 1973. This, JFT-2a/DIVA was the first tokamak with a poloidal diverter in the world. Our team was very young and very active. We were able to do all fundamental diverter experiments except helium removal. Major results were understandings of mechanism of metallic impurity contamination, its control, the behavior of heat and particles in the scrape off layer plasma, and requirements of the diverter in a reactor.

Based on the DIVA experiment, I developed the poloidal

diverter concept for reactor applications, which was proposed to INTOR in 1979. There was also this novel idea of keeping all the poloidal coils outside the toroidal coils. The diverter has relatively short with wide throat. All these ideas were employed in the INTOR concept. This scheme is now common to many tokamaks including ITER. Based on our work in Japan, I was invited to Germany by Dr. Martin Keilhacker to work on the ASDEX machine. I worked with him on cold and dense diverter plasmas, which we now know as the semi-detachable plasma. This was the first experiment on this topic. Around this time I became the head of the JT 60 experiment in 1983.

John: What was the strong motivation to join ITER?

Shimomura: There were two major reasons. While working with these machines, I had an intense desire to see and operate burning plasmas. I also had a good experience of working with scientists from different parts of the world. I thought that ITER would give me an opportunity to fulfill both these aspirations.

Around that time, in 1986 I was asked by Dr. Ken Tomabechi to participate in a group, which was defining the scope of the next machine. This work was initiated to follow up the meeting between President Reagan and First Secretary Gorbachev on starting collaboration on the development of fusion energy. The group consisted of Academician Kadomtsev, Paul Rutherford, Ken Fowler, Romano Tosci, Folker Engelmann and Ken Tomabechi in addition to me. This group defined the objectives of a machine, which would achieve thermonuclear fusion and the collaboration possible around this concept. The name ITER did not exist at that time. I was the youngest and Romano said "Yasuo is the only person in our group who will operate ITER."

John: What was the work like in the initial phase?

Shimomura: In 1988, we, EU (European Union), JA (Japan), RF (Russian Federation) and US (United States), started the conceptual design activity for ITER. Scientists and engineers worked at Garching for several months per each year for three years and developed the conceptual design of ITER as well as the necessary research and development programme. We chose the major parameters such as major radius 5.2 m, minor radius 2 m etc. The plasma current was 22 MA with very highly elongated plasma. However, each participating team developed a different machine under the same envelope. A very important result of this activity was the recognition that an international team could work together very well by collecting and developing different ideas. This was also a lesson in developing good working relationship among the working scientists and engineers.

At this time, participating governments did not anticipate that a machine would be built based on the international collaboration among the Parties. The major objective was to understand the next device and identify the technologies that would be needed for it. Through this design study, we generated an understanding about the concept of the machine necessary for a burning plasma with associated technology necessary to be developed. In parallel with this activity, there was progress in tokamak experiments like JET, JT 60, TFTR and other machines in producing data on high temperature plasmas from which we could extrapolate to necessary reactor plasmas relatively accurately.

John: How large was the group in the beginning?

Shimomura: During CDA there was a coordination committee with Dr. Tomabechi as the chairman. There were about 40 permanent members and about 200 part time contributors. Key members of the conceptual design group started to think about the possibility of constructing a machine with collaboration among EU, Japan, US and Russia and therefore the Parties started to investigate possible future activities, which could continue to the construction of the device. This led to the ITER Engineering Design Activity (EDA).

Governments reacted positively to this idea and allocated about \$ 1 billion, which included the cost of research and development. The inter-governmental agreement was established in 1992. The objective of this activity was to generate sufficient information that would lead to the construction of a machine. This collaboration was based on equal contribution from the four Parties. During the latter half of this activity, the Parties started to discuss the possibility of building a single machine together by all countries. This EDA started in 1992 and finished in 2001. The maximum staff number was 150 with supporting staff

of 100. The budget for technology R&D in the four Parties was \$ 750 million. The Director coordinated all technical activity. For the first two years Rebut was the Director who was followed by Robert Aymar. I was the Principal Deputy Director from the beginning. I became the Interim Project Leader in July 2003.

Due to the economic difficulties in the Soviet Union and the withdrawal of US from the project, we faced difficulties. Following a Japanese proposal, the machine size was reduced and correspondingly the cost reduced by a factor of 2. This modification was done during 1998 to 2001 and then the final design report was completed in July 2001. The Parties tried to prepare the construction agreement; however due to lack of resources, the negotiations did not smoothly proceed. In early 2003 China joined the ITER programme followed by Korea and around this time, the US also decided to rejoin. Sufficient resources were available now. Japan offered Rokkasha site to build ITER and EU made a counter-offer of Cadarache site. Site selection became a contentious issue and with the difficulties in selecting the site, the negotiations almost came to a stop by June 2005. However this was finally resolved and the site at Cadarache was selected.

John: Can you recall how you grew professionally during your career?

Shimomura: I was in my early forty when I joined ITER. I joined ITER to do experiments on burning plasma with a scientific team drawn from all over the world. I have always enjoyed working with many excellent people from different backgrounds capable of very creative hard work. At the same time I got very tired with complex international political negotiations in which people often forget their overall objective, which should be the interest of the project. I believe that mainly young persons must do ITER construction so that they can complete the machine and operate the machine for many years. This means you need to work at least for about 15 years from now. I planned to retire myself about two years ago. Dr. Aymar left the project and the project was in difficulties as mentioned above and I could not leave it. Now, the ITER Project is stable and the Director General Nominee was selected and it is a time for me to leave ITER.

John: What were the problems in working with an international group?

Shimomura: Work was never a serious problem; but family was. When you are to live in a foreign country for long time, education of the children and their future tend to suffer. If your wife wants to work, a suitable job may be difficult to find. My wife taught at college in Japan and

enjoyed it, but she could not find a similar job in US or in Germany. I have three sons. Two are educated in foreign country and one went back to Japan after finishing high school. Now two are in Japan and one in Australia. A long term work can be enjoyed much better in your own country due to a variety of reasons.

John: What were the major breakthrough ideas in the ITER project?

Shimomura: The original ITER design was optimized only for ignition. Mainly Drs. Rebut and Aymar designed this. In the original EDA, which was done in 1992-1998, we had very serious difficulty to promote ITER because of lack of support mainly due to the cost, its reduced scope and big uncertainties of fusion energy. Economic problems in Russia and US withdrawal also added to these difficulties. Therefore, I tried to change the design completely as originally suggested by Dr. Hiroshi Kishimoto based on the recent developments both in technology and physics and also consideration of future reactors. Then the ITER council asked me to organize a task force including experts in the Parties to develop a possible new concept. At the time, Dr. Aymar was the director. This was to conceptualize a more attractive device at reduced cost. The present device is based on the results from this task force, which include also the results of technical research and development during the original EDA. After establishing this concept we could recover support from especially Japan and Europe. I also put my effort to involve China and Korea by visiting these countries in 2000 and meeting leading scientists in both countries. to explain the ITER project and its benefits to these countries. I also explained to them that in the near future young scientists and engineers in these countries may become leaders of fusion community in the world because many of fusion scientists in US, EU and Japan are getting too old. They do not have new projects under construction and hence they are losing their knowledge and experience.

John: ITER has taken such a long time to come to realization. Could we have made faster progress?

Shimomura: I think one of the main reasons is that it takes long to convince governments in these matters. For instance in Japan, we had very hard discussion about three years in the fusion community, followed by those with the science community and also the wider community. I believe the same process was also needed in other countries. In addition to this, each party has its own political and financial systems. To accommodate all these takes time. This delay is in fact the weakest point of ITER project. The other

reason was that the original design was not attractive and we, the ITER Team, were not active to find a better solution. A serious problem arising out of long times is the loss of people and knowledge. For example during EDA we spent \$ 750 million in the industry to build up technology and experience necessary for ITER construction which also resulted in the creation of many experts in Home Teams and industries as well as in the International Team. We lose this knowledge and knowhow since people are leaving. Because of this, it is critical to start ITER construction quickly especially to collect young engineers so that they can catch up with design and technologies quickly.

John: Did you bring a special quality in your job being a Japanese

Shimomura: The differences between nationalities are not as big as the difference between personalities. So in that sense I had no significant difficulty in the way to work. However one general difference may be that the Japanese try to understand what other people are saying and thinking. If one finds a better way, one tries to include it into his idea and try to develop a better one before fighting with each other. Some time people fight with each other by sticking to their own ideas forgetting the final goal. Final goal means reaching a better solution. In this sense I believe that I made an impact on the way of work in the team especially during the development of the new concept.

John: How did international politics affect the ITER project?

Shimomura: Positively. Survival of ITER over these years despite differences is a proof of this. International competition stimulates good work. The negative side is the delay in order to accommodate different requirements and other complexities. Sometime the national wish is not consistent with the requirements of the project.

John: Where you disappointed when the US left ITER in between?

Shimomura: Yes. My understanding as to the reason for that was that the US fusion community consisted mainly of only physicists. When the 1996 fusion budget was reduced they cut major engineering aspects. In US, there was no home center for ITER unlike in Japan or EU. This means no center to promote ITER. We were disappointed with the US departure, but JA and EU pushed the project strongly during this hard time. Why they came back is of course clear.

John: Do you think other countries, like Brazil would also join ITER?

Shimomura: Other countries may be able to join ITER through existing Parties or directly as associate members. There is no campaign to have other countries as full partners.

John: What will your role be in ITER from now?

Shimomura: I expect that by early March next year Mr. Ikeda will start to work for ITER as Director General Nominee on a full time basis. Then I intend to resign as the Interim Project Leader. If requested by the preparatory committee and the DG Nominee, I will work for the ITER Project as a senior advisor for a few months.

John: What are the long term plans?

Shimomura: I will retire. I have no plan except to lead my private life.

John: What are your hobbies

Shimomura: I enjoy mountain climbing, skiing, snorkel, nature, and jungles, as well as reading various kinds of books. I like mountains in all seasons especially in Tyrol, Bavaria and south Italia, Japan and Himalaya. I have spent 15 years in foreign countries and my family has been scattered for a long period. For example I am with my wife in Germany and my eldest son just came back from Thailand to Tokyo where he worked as an art teacher, my second son, a vet, is near Tokyo but will move to Okinawa soon, and my third son is in Canberra studying art. I need to recover a comfortable and stable life, and better communication with my family, not only with my sons but also brothers, sisters, and relatives as well as Japanese friends.

John: Where will you live in Japan?

Shimomura: In Mito city, 20 km from Naka. I have a

house there. It's a typical Japanese house with some rooms in western style.

John: Are you on the whole satisfied with your life?

Shimomura: In the ITER project I have enjoyed work with challenging goals. Work with different nationalities with different cultures, ideas and way of work was satisfying. In this sense I have been very happy. But I was not so happy about the delays and slow decision-making. And sometimes these delayed decisions came from lack of support from communities or government, but many times these delayed decision came from conflict among the Parties. For my private life, I need more time.

John: Do you believe in the kind of stories that you find in science fiction as in star trek where there are fusion reactors driving rockets.

Shimomura: I have never linked fusion with science fiction. Fusion is a very difficult system. I think that it would be possible for us to decide whether fusion will be a significant source of energy only after we operate at least one experimental reactor. It is too early to say. It is very obvious that we can get energy from fusion. The question is how the system can be simplified, made reliable and with reasonable cost as a system commonly used in the world. This depends on development of plasma physics, material, and fusion technologies. These conditions are not clear now. So I believe we have to be more careful about development programme of fusion. ITER is certainly the main device for the next step. It may not be able to solve all problems. If we are lucky it will be possible to construct DEMO immediately after ITER operation. Results may suggest necessary improvements. Development of fusion energy might not be very straight one. It is also the reason why we need to involve major countries in the world and also many scientists, engineers, industries and laboratories. So in this sense ITER is not the final step.

DAE Websites	
Atomic Minerals Directorate for Exploration and Research	http://www.amd.gov.in/
Bhabha Atomic Research Centre	http://www.barc.ernet.in/
Indira Gandhi Centre for Atomic Research	http://www.igcar.ernet.in/
Raja Ramanna Centre for Advanced Technology	http://www.cat.gov.in/
Variable Energy Cyclotron Centre	http://www.veccal.ernet.in/
Board of Radiation and Isotope Technology	http://www.britatom.com/
Heavy Water Board	http://www.heavywaterboard.org/
Nuclear Fuel Complex	http://www.nucfuel.gov.in/
Electronics Corporation of India Limited	http://www.ecil.co.in/
Indian Rare Earths Limited	http://www.indianrareearths.com/
Uranium Corporation of India Limited	http://www.ucil.gov.in/
Bharatiya Nabhikiya Vidyut Nigam Ltd	http://www.bhavini.nic.in/
Nuclear Power Corporation Of India Limited	http://www.npcil.nic.in/
Board of Research in Nuclear Sciences (BRNS)	http://www.barc.ernet.in/webpages/brns/
National Board for higher Mathematics (NBHM)	http://www.nbhm.dae.gov.in/



Commercial Spin-off of Fusion Technologies

Prof. P. I. John

Institute for Plasma Research

Plasma medium offers unique opportunities for high energy density, non-equilibrium and atomic scale processing. The technology is environmentally benign and economical in material and energy inputs. High temperatures, high chemical reactivity, microscopic electric fields, sheaths, radiation and particle flux are the tools for plasma-assisted processing. Plasma processing has transcended conventional material processing applications and has emerged as an enabling tool with a wide spectrum of applications relevant to the modern industrial society.

In 1990, IPR took an initiative in establishing links with industry for development and commercialisation of plasma-based industrial technologies. The Facilitation Centre for Industrial Plasma Technologies (FCIPT) was set up in 1997 to consolidate all activities in technology development, demonstration, incubation and commercialisation. FCIPT has established a multi-disciplinary group of scientists and engineers and infrastructure for process and instrumentation development including prototypes and pilot plants for industrial scale job working to generate database on instrument performance, process reliability and economics. Industry, research establishments and universities extensively use advanced material characterisation facility set up to support the in-house activities. Manufacture and supply of complete reactors to industries and research institutions is an integral part of the technology transfer process.

Pulsed Plasma Nitriding

Incorporation of atomic nitrogen in metals by thermochemical diffusion from radical-rich nitrogen-hydrogen plasma produces a hard, wear resistant case. Plasma nitriding is medium temperature low distortion surface engineering process capable of producing tailored microstructure and hardness profile.

State-of-the-art systems use actively heated hot wall furnaces. Elevated temperatures of the vacuum vessel

wall increase thermal efficiency and temperature uniformity. When the work piece is heated by radiation in a hot wall chamber, the heat transfer will be a function of the surface area and the temperature difference between the radiating hot wall and the work surfaces. Instrumentation developments include IGBT pulsers upto 100 A capacity, arc interruption capability, computer based process control etc. In-house facilities include large industrial nitriding systems for technology demonstration and job work. The plasma nitriding technology was recently transferred to M/s Milman Thinfilm Systems Pvt. Ltd., Pune.

The cavity and the core inserts for plastic moulds made of P20 steel are subjected to high injection pressures and to erosive/corrosive plastic material during processing and the lifetime of the moulds is limited by the material wear properties. Formation of pits on the surface in contact with hot plastic reduces service life due to fatigue and erosion wear. In collaboration with Indo-German Tool Room, Ahmedabad, cavities and core inserts used in manufacturing toothpaste caps were nitrided to a surface hardness of ~850 HV and a case depth of ~300 microns with excellent surface finish and minimal distortion. In field operations, the plasma nitrided components survived more than one million shots as against half a million shot of the untreated components. Investigation of the parting surface after extended usage showed no significant dimensional changes but revealed partial erosion with 100-m pits of irregular shape distributed non-uniformly over the entire surface.

A campaign of nitriding of hydropower components made of AISI 13CrNi4 steel to mitigate wear due to silt was taken up in collaboration with NHPC. 13CrNi4 annealed at 790-815 °C. and austenized at 955-980 °C and tempered at 595-620 °C has a typical hardness of 350-400 HV. Since silt has a hardness of 800 HV, it was believed that hardening these components beyond 800 HV, would increase the life of these components. FCIPT was

successful in developing nitriding process for AISI 13CrNi4 martensitic stainless steel to produce a surface hardness of 1200HV and a case depth of 250 microns.

Other strategic applications include indigenisation of the nitriding of solar panel drive gear of ISRO satellites, which was previously being done by a British company. Plasma nitriding is being promoted by the Department of Science and Technology as a replacement of polluting technologies by setting up a plasma nitriding system at the Indo-German Tool Room at Ahmedabad. A series of awareness campaigns and workshops are also being planned for promotion of plasma nitriding as an eco-friendly technology.

Plasma Ion Implantation

Large area directed ion flux generation and implantation in metals using ion sheath as the extraction optics makes PII inexpensive and less complex compared to beam ion implantation. The surface modification improves wear and corrosion resistance and beam impingement aids film deposition. PII has high efficiency for nitrogen and carbon

incorporation. Processes developed include nitriding of metals, modification of plastics and doping of electronic devices. PII formed the focus of a collaboration programme with the Technical University, Clausthal and the Institute of Ion Beam Physics, Dresden under the Indo-German Surface Engineering Initiative of the Department of Science and Technology.

FCIPT has developed a 1-metre diameter 2-metre long ion implanter using 50 kV hard tube pulser and dc and inductively coupled radio frequency plasma source. Plasma density can be enhanced by multicusp magnetic trap. Operation at elevated work piece temperature is possible. Innovations include nitrogen incorporation using a combination of shallow ion implantation and diffusion at elevated temperature. A boron source is available for shallow implantation in semiconductor devices.

Nitrogen implantation on AISI 316 was done to understand the change in surface properties as a function of implantation voltage upto 20 kV at constant substrate temperature. The microhardness and SEM measure-

ments show 8 m depth of penetration, independent of substrate bias voltage. XRD results indicate presence of expanded austenite phase with nitrogen incorporation and without CrN formation. The PII facilities have promoted research collaboration with a number of institutions; for example IIT, Kharagpur and National Metallurgical Laboratory. Basic studies on ion implantation of AISI 52100 ball bearing steel are an example. PSII was investigated on its effect in countering hydrogen embrittlement of Cu-strengthened HSLA-100 steel. Low energy nitrogen ion implantation produced an enhancement in the linear strain to failure under embrittling conditions. This has been attributed to the introduction of residual compressive stresses and the reduction of hydrogen flux. A PIII setup supplied to IIT, Kharagpur is being used by TATA Steel for studies of nitrogen implantation on 52100 steel.

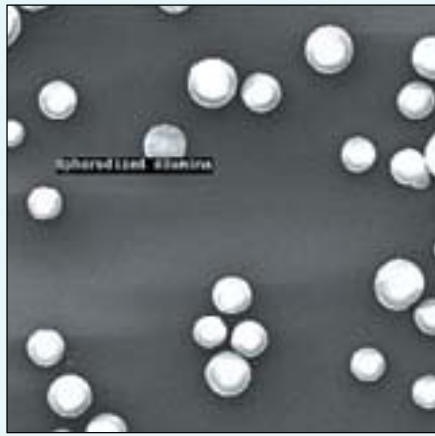
Plasma-enhanced CVD and Polymerisation

The plasma environment can enhance the deposition and growth of thin films on metals and polymers by physical and chemical methods. These films can be designed for improving surface hardness, wettability, optical properties and corrosion protection. Plasma enhanced chemical vapour deposition employs metal organic monomers, which are decomposed in the plasma and made to react with hot surfaces for the deposition of films. 1 cubic metre box type demonstration reactor has multi-kilowatt impedance matched RF generator, auxiliary temperature control and vapour-phase monomer injection systems. PECVD technology is being augmented with the addition of impulse, microwave and variable frequency techniques.

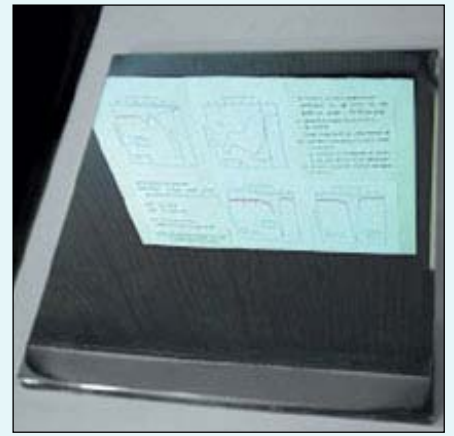
Glass-like polymerised silicon based thin film coating involves



The Facilitation Centre for Industrial Plasma Technologies, which links the Institute for Plasma Research with industry



*Alumina particles
spheroidised
in a thermal
plasma jet*



*Large area silver
coated aluminium
mirror with a
protective silicon
oxynitride coating*

dissociation of HMDSO leading to formation of Si-O chains, which deposit as glass-like coating on substrates. The coating consists of highly coherent, cross-linked, pinhole free Si-O-Si bonds and has good adherence to the substrates. Micron thick SiO₂ diffusion barrier films on decorative brassware and aluminised automobile reflectors inhibit corrosion and provide high quality surface finish. Coatings are immune to solvents like acetone, petroleum ether, water, dilute acid and base and have undergone extensive industry acceptance tests. A prototype system is being built for MHSC, Moradabad to coat SiO_x on brass articles, which forms a large export market.

SiO_xN_y films have been grown on single crystal p-type Si (100) wafers and on textured Silicon solar cells, using Hexamethyl-disilazane (HMDSN). The average bond coordination is N_{av} ~3, representing device quality film. Process parameters are optimised to produce carbon free (<1%) films of thickness ranging from 30-3000 Å, useful for electronic and optical applications. In a joint project with BHEL, Electronics Division, Bangalore, FCIPT have grown 1/4 'blue-violet' anti-reflection coatings on batches of solar cells with uniform chemical composition (SiO_{1.35}N_{0.5}) and thickness (870 ± 15 angstroms), and demonstrated a photovoltaic efficiency increase of ~1% (AM 1).

Metal mirrors are employed in large area optics, such as telescope mirrors. To protect silver coated metallic mirrors from drop in reflectance due to tarnishing, a thin dielectric film of Silicon Oxy-Nitride is deposited, which acts as a barrier layer for silver surface when exposed to air and does not allow atmospheric oxygen to tarnish the silver substrate. Large area coating with dimensions upto 35 cm x 35 cm has been productionized.

Surfaces can be made lubricious by depositing fluorocarbon coatings with low coefficient of friction. Precursors obtained from pyrolysing Teflon scrap yield organic precursors, which are fragmented into various radicals (CF, CF₂, CF₃) and grow the film on the surface of the substrate. The coating is smooth, defect free and has good adhesion and corrosion resistance. The XPS study shown in Figure 8 revealed the presence of C, F and small quantity of N and O and the presence of -C-CF, -CF, CF₂ and CF₃ functional groups on the surface of the coating. The concentration of -C-CF, -CF, -CF₂ and -CF₃ functional groups change when the substrate temperature was varied from room temperature to 150°C. The low substrate temperature technique is now being adapted for coating elastomer seals for application in the fast breeder reactor.

Plasma modification of Surfaces

Plasma interaction with polymer surface leads to removal of organic materials, cross-linking via activated species of inert gases, ablation and surface chemical restructuring by addition of polar functional groups. These processes increase the surface energy and improve adhesion to other materials. Plasma treatment can produce hydrophobic or hydrophilic surfaces on metals, plastics, glasses and polymers.

Environmentally friendly non-aqueous techniques in the field of natural and synthetic textile finishing are in increasing demand. Plasma interaction with fabrics can lead to changes in the wettability and the surface texture. This leads, for example to an increase of printing quality, dye-uptake, dye adhesion etc. The etching technology is being developed for applications to angora wool fibres for improved spinnability in collaboration with the Central Wool Research Board and National Institute of Design.

Conventional metal reinforced rubber products require both an adhesive to bond the metal to the rubber and a separate curing system to increase the mechanical properties of the rubber. In an associated work for Triton Valves Ltd., Mysore, brass valves were treated in the air plasma generated by the 10 kHz pulsed dc plasma source. Plasma treated brass

valves were moulded with rubber and showed improved adhesion. The present process use chemicals thereby leading to production of polluting liquid effluents.

Minerals and Ceramic Processing

The high temperature and enthalpy of thermal plasma jets with large temperature gradients enable novel routes of material processing. FCIPT has developed a number of medium power capacity non-transferred plasma torches for ceramic and mineral processing.

Large volume thermal plasma jet formed with sublimating multi-electrode graphite anode is used for in-flight dissociation of zircon and spheroidisation of ceramic powders. Successful in-flight dissociation of zircon sands was demonstrated in collaboration with C.Z. Zircon Ltd., Himachal Pradesh under a DSIR project. Treatment yielded 95% mono-clinic Zirconia with a specific energy consumption of 4-6 kWh/kg. The amorphous and highly reactive silica phase can be leached out to yield ~95% monoclinic ZrO_2 phase. In spheroidization of alumina and chromia ceramic powders to produce free flowing particles, the yield is more than 98% with a specific energy consumption of 6 to 7 kWh/kg. The particle size varies from 35-90 m. From the XRD peaks it was found that particles have corundum (trigonal) structure.

The steep temperature gradient and high quenching rates of thermal plasma systems are utilized for generation of sub-micron aerosols of metals, ceramics and composites. Plasma-based aerosol generators can provide very high throughputs with control on generation rate and size. High purity sub-micron Aluminum Nitride has been synthesized by vapour phase nitridation of Aluminum. Evaporating magnesium metal in nitrogen plasma and reacting it with oxygen added as sheath gas

generated magnesium oxide aerosols. Powder sampling was carried out using Andersen sampler at the flow rate of 1.7 cu.m/hr. The particle size distribution peaks at 1 mm. The particles are found to be highly agglomerated clusters of very fine primary particles.

Plasma Pyrolysis of Medical Waste

Safe disposal of medical waste is a topical problem with the growth in health care services and facilities. Burning the waste in open air can be never complete, with small quantities of many organic and chlorinated organic compounds as well as pathogens surviving and dispersing dangerous diseases. The essential problem of conventional incineration

is that the generation of heat is closely coupled to the reaction chemistry. Oil fired incineration produces harmful products due to insufficient temperature in the process chamber. This can cause air pollution or the toxic pollutants can remain at the bottom ash, eventually finding way into landfills.

In plasma pyrolysis, electrical energy is converted to a plasma stream with temperatures more than 10,000 degrees. The intense heat causes the waste molecules to disintegrate forming fragments of compounds and the highly reactive plasma environment can catalyse homogeneous and heterogeneous chemical reactions. The most likely compounds to form from



Plasma Pyrolysis System

Table 1. Plasma pyrolysis gases: CPCB and measured emission limits

Gas	CPCB Limit Concentration (ppm)	Measured Concentration (ppm)
CO	100	40 – 85
NO _x	450	3 – 5
SO ₂	50	1 – 20
HCl	50	3 – 10

carbonaceous matter are methane, carbon monoxide, hydrogen, carbon dioxide and water. The recombination of the plasma produces intense ultraviolet radiation, which can destroy pathogens completely. The volume reduction is 95%.

In collaboration with TIFAC, FCIPT has developed a pyrolysis system for the safe destruction of medical waste. The system is compact with a footprint of 2.5 metres by 1.5 metres. An automated load lock valve allows waste charging without air entry during the pyrolysis process. Hollow anode graphite plasma torch without forced nitrogen injection comprises of tubular graphite anode and rod shaped graphite cathodes. Cathodes are rotated around their axis for uniform erosion and linearly moved to control the torch impedance using programmable logic circuit. The pyrolysed gases are burned in the combustion chamber reaching the statutory 1100°C temperature and specially designed baffles keep the residence time at 2 seconds. The technology of plasma pyrolysis has been transferred to M/s Bhagwati Pyrotech Pvt. Ltd., Ahemdabad

Pyrolysis technology is being adapted for safe disposal of plastic carrybags for ecologically sensitive locations like Andamans and Nicobar Islands, Sikkim, Goa and Himachal Pradesh under a DST sponsored project. Additionally more systems are being installed at locations such as Haryana, Uttar Pradesh, Andhra Pradesh, and Tripura. BARC and IPR are collaborating to adapt pyrolysis technology low level nuclear active waste destruction.

Pyrolysis converts waste into CO, methane and Hydrogen with a composition, which is superior to gas produced in biomass gasification. Super thermal pyrolysis, possible with high temperature plasma jet; result in efficient production of CO and H.

Table 2. Comparison of by-products in plasma and biomass gasification

Gases	Biomass	Plasma Pyrolysis
CO + H ₂	38-44%; H ₂ : 18%	45-60%; H ₂ : 33%
CO ₂	4-5%	4-8%
N ₂	50%	30-50%
CH ₄	1-3%	3-6%
Other HC	<0.5%	2 -7%
HCl; NO _x	Negligible	< 20 ppm; < 85 ppm
Total Combustible Gases	39-47%	50-65%

The controlled quantity of the steam is introduced which reduces the soot particulate by 20-50% and increases the concentration of CO and H₂, which is clean fuel having high calorific value.

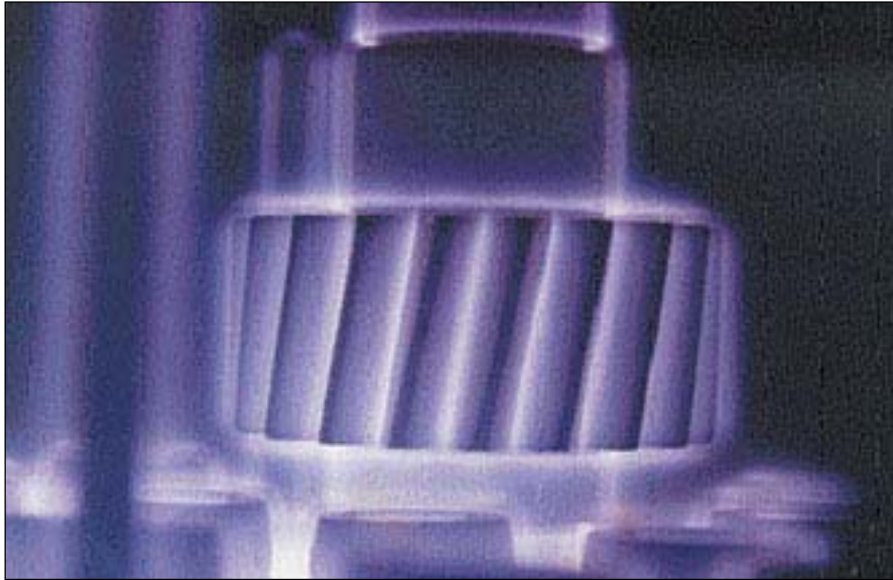
Destruction of Volatile Organic Contaminants

Emission of volatile organic compounds from vehicular and industrial sources is a major source of atmospheric pollution. The dilute concentration of these compounds often poses problems in development of economic and efficient remediation techniques. Non-thermal plasma environments exploit the high plasma chemical reactivity of radicals and energetic electrons to dissociate toxic molecules. The highly mobile free electrons permeate the gas column and fragment the pollutant gas molecules into non-toxic compounds.

Atmospheric pressure non-equilibrium plasma sources are essential for economically viable processing of dilute streams of toxic molecules in an ambient gas. A number of types of plasma sources have been developed for VOC treatment. Microdischarges called streamers produce non-thermal

electron population through the formation of plasma filaments produced by highly localized space charge waves, which enhance the applied electric field in the wave front and propagate by the electron avalanche in this field. Streamers yield good power efficiency since the ions are stationary and hence do not consume power. Streamers are generated using sub-microsecond high voltage pulsed corona discharges or with dielectric barrier discharges.

Benzene or n-Hexane molecules are dissociated into hydrocarbons, hydrogen, carbon monoxide and lower hydrocarbons like methane, ethane, ethylene, acetylene, propane propylene etc. Free radicals combine with the fragments to form non-toxic molecules and compounds such as CO₂, CO, light hydrocarbons and hydrogen. Equivalent chemical destruction reaction rate associated with the temperature of 1000°K can be realized with non-thermal plasma operating at room temperature. Further fragmentation of these molecules does not occur and the concentration of ethylene and acetylene hydrocarbons decreases with increasing benzene concentration at constant power.



Abnormal glow discharge in nitrogen-hydrogen plasma provides radicals, which contribute nitrogen for diffusion into the steel surface.

Plasma Sources, Power Systems and Diagnostics

A spin-off from basic research at IPR is the development of a variety of plasma sources, specialized power drivers and diagnostic instruments. The growing research activity in plasma physics and plasma processing in universities and the prohibitive cost of imported systems makes a good case for indigenous development and supply of these systems.

ECR-plasma sources have been extensively applied to numerous low-pressure plasma processing applications as etching, deposition and ion implantation. To meet the indigenous demand for this system for research applications, FCIPT have developed an ECR microwave plasma system using a 2.45 GHz 800 Watt commercial magnetron. Typical parameters achieved using argon gas at operating pressure of 5×10^{-5} mbar and magnetron power of 620 W were 2.5×10^{10} /cu.cm plasma density and 5.5 eV electron temperature. Low-pressure glow discharge configuration with constricted anode geometry (CAPS), which improves ionization efficiency by electrostatic trapping, has been developed and characterized.

Plasma loads are inherently complex due to non-linearity in the current-voltage relationship, large range of time-dependent variation of impedance due to discharge mode transitions, reactive impedance due to complex conductivity, sheaths and current filamentation, tendency to arc due to current localisation at electrodes etc. The power supplies for driving plasma loads have to be designed to meet these demanding functional requirements and should have provisions for safe and reliable operation over a large range of voltage and current. FCIPT has designed and developed a variety of power sources based on solid state and hard tube devices for such plasma loads as abnormal glow discharges, magnetron sputter sources, RF capacitive and inductive sources, vacuum and atmospheric arcs, pulsed corona discharges etc. A 3 kW, 150 Hz trapezoidal waveform power supply has been developed to improve the performance of low pressure high power UVC lamps. Regulated High Voltage Power supply for klystron is being developed for BARC.

Electronically swept Langmuir probes with radiofrequency

compensation and vacuum drives for automated positioning has been developed. Simpler probe systems for basic plasma research are now being supplied to universities and research institutions. Presently a sophisticated plasma diagnostics setup for characterization of Hall effect thruster for LPSC and ISAC is being developed.

From page 19

machine remotely through use of mascot and delicate maneuvers. Remote handling equipment must be capable of manipulating components weighing up to 50,000 kg. This essential technology for ITER and future reactors is being developed in Europe. Components of R&D project in this area has to demonstrate the basic feasibility of the remote maintenance scenario for the divertor of a reactor which includes the in-vessel cassette removal/replacement and hot cell refurbishment, consumable components in the auxiliary heating systems, in-situ repairs etc.

Summary

Today assets of fusion technology in the country are clearly visible. We now have a network of institutions and industries practicing one or the other facet related to building of fusion devices and a trained manpower of engineers, technologists and scientists. This has enabled us to participate in the international cooperation programme of ITER. Experience of ITER will hone our skills. We will learn to create and handle burning fusion plasma. However, ITER experience needs to be complemented. New initiatives need to be taken in the areas of blanket, superconducting and advanced low activation materials, remote handling and maintenance etc.

International Collaboration: IGCAR experience

Dr. Baldev Raj

Director, Indira Gandhi Centre for Atomic Research

The mission of IGCAR is to render comprehensive Research, Development and Design support to the Indian Fast Reactor Programme. Over the years, comprehensive expertise has been developed in all the relevant fields, including reactor engineering, safety, materials, and reprocessing. In the process, IGCAR has established itself as a nationally and internationally reputed research centre not only in the primary areas of its activities but also in many associated areas. This has resulted in a gradual increase in the number of international interactions and collaborations. In deed, in several cases, the proposals for collaboration have originated from the foreign laboratories, a testimony to the overwhelming and clear strength of IGCAR in the relevant areas.

IGCAR has been an active participant in the International Working Group on Fast Reactors, and other International Atomic Energy Agency (IAEA) activities of interest to fast breeder reactor (FBR) programmes. IGCAR hosted during 13-17 January 2003, the IAEA Technical Meeting on "Primary Coolant Pipe Rupture Event in Liquid Metal Cooled Fast Reactors", in which international experts from several interested member countries participated. IGCAR has also been actively participating in international cooperative research programmes of IAEA in the areas of reactor engineering, reprocessing, and safety. IGCAR has actively participated, along with France, Japan, Korea and Russia on the investigation of thermal stripping damage of the expansion tank of Phoenix reactor. The thermal

hydraulic as well as structural damage predictions made by IGCAR scientists matched very well with in-plant data. In another cooperative research programmes in which IGCAR contribution was greatly appreciated was on intercomparison of computer codes to predict seismic behaviour of liquid metal fast breeder reactor cores. In the cooperative research programmes on core mechanics of FBRs, the predictions by IGCAR scientists were found to be in good agreement with the experimental observations. IGCAR is actively participating in several current cooperative research programmes of IAEA. One example in the area of reactor physics is on updated codes and methods to reduce calculated uncertainties in reactivity coefficients in liquid metal fast breeder reactors; China, France, Japan, Korea, Russia and USA are collaborating in this project. Another example in the area of reprocessing is on the separation efficiencies for La, Ce and Nd in the oxide electro-winning process using $MgCl_2$ based electrolyte, in collaboration with China, Czech Republic, Japan, Korea and Russia. IAEA has initiated an International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) with a view to ensuring sustainable nuclear energy towards fulfilling energy needs for the twenty-first century. India is a member of this project. As part of this project, IGCAR is actively participating in a joint case study on assessment, using the INPRO Methodology, for an Innovative Nuclear Energy System based on a Closed Nuclear Fuel Cycle with Fast Reactors (CNFC-

FR). The objectives are to assess the long term viability of technology options and innovations, and identify areas where research and development is required. The participating countries are China, France, India, Korea and Russia, and Japan as observer. IGCAR is an active member in the Technical Working Groups on nuclear fuel cycle options and nuclear data evaluation, and Standing Advisory Group on nuclear energy.

As a part of the protocol between DAE and the 'Organisation Européenne pour la Recherche Nucléaire (CERN)', Geneva, so far five scientists have been deputed from IGCAR to CERN, each for a period of one year, for carrying out testing, evaluating, and training the 1200 huge super-conducting di-pole magnets for the Large Hadron Collider (LHC); this is a giant underground accelerator designed for accelerating protons to 14 TeV and lead nuclei to 1150 TeV for basic studies in particle physics, the next major project of CERN.

IGCAR has also been actively participating in the collaborative research programmes with the premier R&D institutes in Germany under the aegis of Indo-German Bilateral Agreement. In the early years, the areas of collaboration included sodium chemistry, high temperature mass spectrometry, high temperature calorimetry, and solid state physics including radiation damage and low temperature physics. Subsequently, the range of collaboration has considerably expanded to include many other front line areas of common research interest in science and technology, including vibration noise monitoring and analysis for diagnostic purposes in nuclear power plants, and micro-meteorological studies using acoustic Sonic Detection And Ranging (SODAR) system for prediction of dispersal of radionuclides. These



European Organisation for Nuclear Research, Geneva, Switzerland

projects also fostered close collaboration between Indian and German scientists, including mutual visits. The experience and expertise thus generated have been valuable not only to IGCAR, but also to the host institutions in Germany. An important area of direct interest is that of high temperature mechanical properties, namely, creep, fatigue and their interaction, and also fatigue and creep-fatigue crack growth behavior in liquid sodium. Over the years several projects have focussed on high temperature components: creep-fatigue interactions and fracture mechanics properties and their correlation for high temperature materials, including steels and their weldments, and also Ni-base superalloys; methodology for high temperature component life assessment; and monitoring & repair welding of high temperature components. The results from these studies have clarified many basic issues, and enhanced capability in life prediction of engineering components. An extensive project addressed the issue of environmental degradation of stainless steels and their welds and the effect of nitrogen content thereon, and corrosion properties of spun-melt

ribbon and bulk metallic glasses of Zr-based systems with superior corrosion resistance. Another such extensive study has been on the science of oxidation of high temperature materials, a topic of perennial interest to power plants. Indo-German Collaborative Programme on Non Destructive Evaluation (NDE) is an extensive collaboration that charted several new paths - including development of ultrasonic and micromagnetic techniques with innovative approaches for characterization of microstructures and deformation process, and development of eddy current impedance imaging for quantitative characterization of defects and also knowledge based systems for NDE applications. Two IGCAR scientists obtained their doctoral degrees from a German University based on the investigations carried out under this programme. This interaction has considerably enhanced IGCAR impact and recognition as one of the premier NDE research centre in the international NDE community, and is a fine example of the importance and impact of such collaboration with premier international institutions.

The early collaboration of IGCAR with the French Atomic Energy Commission (CEA) dates back to 1969, when the fast breeder test reactor (FBTR) was conceived to be built by adaptation from the French fast reactor Rapsodie, with several design modifications. IGCAR reestablished collaboration with the French Atomic Energy Commission (CEA) in 1989 to exchange computer codes in the field of thermal hydraulics and structural mechanics. Under this collaboration, from CEA IGCAR received CASTEM 2000, PLEXUS and TEDEL codes for structural mechanics analysis. Recently, DAE has established collaboration with CEA in wider spectrum of areas related to safety of fast reactors.

IGCAR is a partner in some of the international collaborative projects under the aegis of Department of Science and Technology. An ambitious DST-DAAD collaborative project is aiming at comprehensive understanding and quantitative characterization of thermomechanical fatigue in power plant materials, and IGCAR and Universität Siegen, Germany, have been identified as the principal partners. Under the above project so far 4 Indian scientists

visited Germany and 4 German scientists visited India. University of Science and Technology, Beijing and IGCAR are collaborating on a project on remnant creep life prediction model for power plant components. Another project is on yield criterion for superplastic metals and alloys, with Moscow State University, University of Hyderabad, and IGCAR as the collaborators; recently a Russian scientist visited IGCAR for carrying out experimental work for this project.

Mention must be made here of collaboration between IGCAR and some internationally reputed laboratories, another reflection of the prestige of IGCAR as a research centre. In an Indo-French Collaborative Programme on Ferrofluids, a device has been developed that can measure forces of the order of 10^{-13} N between colloidal particles, and it has been effectively

utilized for synthesizing ferrofluids and their emulsions with long term stability. The programme led to filing of four patents that include force apparatus, ferrofluid based magnetic flux leakage measurements for NDE and optical filters. The technology is now being adapted to develop dynamic seals for sodium pumps used in fast reactors. In collaboration with Centre for NDE, Iowa State University, a three dimensional boundary element model has been developed and validated for detection and quantitative evaluation of surface and subsurface cracks in structural components. In collaboration with Michigan State University, East Lansing, modelling of response of SQUID sensors developed in IGCAR for detection of defects through flux leakage measurements has been taken up for optimization of the sensor design.

The contribution to expertise pool

of IGCAR from personal initiatives deserves a mention here. A few of our scientist have been awarded the prestigious Alexander von Humboldt Fellowship, which gave them the opportunity to work in premier laboratories in Germany. Many of our scientists have spent between one to two years as Post-Doctoral fellows, or shorter periods, in many leading laboratories in Japan, USA, Europe, and Australia. The areas of research for such visits have invariably been in the frontiers of current or near-future interest to IGCAR. Similarly, several scientists from foreign laboratories have visited IGCAR for periods ranging up to 1 year, to work on projects of common interest. These interactions promote inter-laboratory co-operation and cross-fertilization of ideas, and benefit the programmes of both the IGCAR and the concerned foreign institution.



IGCAR coordinated in sculpturing of the largest bronze Nataraja idol ever cast; this was presented by the Department of Atomic Energy to CERN, Geneva.

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