Research Opportunities In National Fusion Science And Technology Programme

Report NFP – 01

Prepared by

Programme Committee, PSSI Workshop on National Fusion Programme: ITER and Beyond

Institute for Plasma Research, Bhat, Gandhinagar 382 028, India

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About this Document

Title: Research Opportunities in National Fusion Science and Technology Programme

Purpose: the NFP document and future versions will serve as the master database for all research and development activities proposed and performed under this programme

Accessibility: It will be distributed to potential project investigators (PI) as printed reports and shall be put on the ITER-India website.

Numbering: The present version is NFP-1. Subsequent modifications will have progressive numbering (NFP-2, NFP-3 etc)

Hierarchies: Broad area, topic, research problem and research proposal. The content will get modified with each version

Research Proposals: Research Proposals nucleated during the present and future workshops will get included in NFP-2 and subsequent versions along with relevant project details

Project details: Principal Investigator, collaborator, funding details, review data, progress reports etc.

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Annexure – I Format of proposal

India and ITER

ITER (*International Tokamak Experimental Reactor)* is an important step on the path to develop nuclear fusion as a viable, long-term energy option. Realizing the importance of nuclear fusion to the future national energy security, India has joined ITER as an equal partner along with China, the EU, Japan, Korea, the Russian Federation and the United States of America. Information on ITER can be found on the ITER website http://www.iter.org

The Institute for Plasma Research (IPR, Gandhinagar, http://www.ipr.res.in) is the domestic agency empowered to design, build and deliver advanced systems and sub-systems forming part of ITER project, which have been assigned to India under this agreement. These include the ITER cryostat, Vacuum vessel in-wall shields, Cryo-distribution lines and Cryo-lines, ITER water cooling system, RF power sources and associated power supplies, monitoring and control system, ECH Assisted Plasma Start-Up System, Regulated High Voltage Power Supplies, Diagnostics consisting of Optical Emission Plasma Diagnostics and Diagnostic Neutral Beam.

Some of the areas of work that India has been assigned or has committed to ITER project includes (1) ITER cryostat, (2) Vacuum vessel in-wall shields, (3) Cryo-distribution components and Cryo-lines, (4) ITER water cooling system (Heat Rejection and Component Cooling Water: Material and Transportation), (5) Nine 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 one test stand), associated power supplies and monitoring controls, (6) ECH Assisted Plasma Start-Up System (3 Gyrotrons at 120 GHz/1 MW/10s and one spare), power supplies and controls, (7) Eight numbers of Regulated High Voltage Power Supplies (26 kV, 175 A) for Ion Cyclotron Heating and Current Drive Systems, Three numbers of Regulated High Voltage Power Supply Systems (80 kV,30A) for Gyrotrons of Plasma Stat-Up System , One 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 Microwave Diagnostics (9) Diagnostic Neutral Beam (100 kV, 24 A).

Beyond ITER: The Necessity for a National Fusion Programme

While IPR prepares for meeting the Indian commitment to ITER, it has been realized that it is imperative to start planning a long-term programme aimed at developing competence in all aspects of fusion science and technology within the country. This is to ensure that the country shall be ready to take up designing and building a demo reactor after the successful operation of ITER.

The programme clearly is of a multidisciplinary, multi-institutional character, requiring expertise from a variety of fields, ranging from the frontiers of fundamental science to sophisticated technologies.

Adequate funds have been earmarked in the ITER-India programme to fund topical research projects in all these areas.

IPR-PSSI Workshops

The major objective of the proposed series of workshops is to bring experts from other areas of physics and technology to a common platform, expose them to the challenges involved in the programme mentioned above and to discuss the strategies for realizing the programme. It is hoped that this and the future workshops would create a community of fusion scientists and technologists from academic institutions, and national laboratories who will to contribute towards the success of this activity.

The first workshop will be held in IPR on **8-10th November 2006**. The workshop will consist of theme presentations on the topics mentioned above and also will have talks by invited specialists on the topics of their competence. Working groups will be formed for identifying and discussing specific areas of collaborative research. It is expected that at the end of the workshop, definite ideas on collaborative, multi-disciplinary research projects would be nucleated, which will then be taken up for funding.

Major Research and Development Areas

To start with, ten major areas have been identified which are as follows;

Plasma Theory & Simulations: MHD activity, plasma turbulence, physics of energetic particles.

Advanced Material Technologies: Beryllium, Borated steels, Low activation steel and Vanadium, Ceramic and Graphite composites, High strength Copper alloys, Electro-deposition, Special brazing and welding techniques, Advanced fabrication and heat treatment technologies, Surface coatings, Corrosion engineering etc.

First wall component Engineering: Design of high heat and nuclear radiation facing Limiters, diverters, blankets etc.

Cryogenic and Magnet Technologies: Cryopumps and Cryogenic systems, Cryo-transfer lines, Superconducting cavities and Superconducting material development, Nb:Sn cables, HTSC Current lead development, development of high field, large volume Superconducting magnetic systems etc.

RF and Microwave Technologies: development of Klystrons, Gyrotrons, High power Tetrodes, Microwave components, RF and microwave transition technologies etc.

Beam Technologies: Ion sources; High and Low energy neutral and ion beam developments etc., Multi Megawatt neutral beam transport

Power Engineering Technologies: Regulated HV Power supplies, Insulators.

Advanced Data Acquisition and Control Engineering:

Software and hardware developments

Nuclear Technologies: Studies on radiation effects on materials, detectors, mirrors, windows etc., Tritium recovery and handling, radiation monitor developments etc.

Plasma Diagnostics: neutron, gamma ray, proton and alpha particles diagnostics, Laser based diagnostics, Microwave and Infrared diagnostics, UV-VIS-X-ray spectroscopy, Imaging technologies, Tomography, Neutral beam based diagnostics, Analysis codes and software developments etc

In the rest of this document, after a brief introduction to each of above areas, the topics of possible research collaborations have been given. While some of them are oriented towards fundamental research, others pertain to technology development. It is hoped that this document would assist collaborators to identify and discuss research projects of their competence and interest. If more information is needed in any of the topics/proposals below, the scientific programme committee of the workshop will be very happy to provide them.

1. Plasma Theory & Simulations:

Theory and simulation studies play an integral part in guiding the design and operation of the magnetic confinement fusion (MCF) devices. They have helped in the understanding of better modes of operation of these devices with improved particle and energy confinement times. A proper understanding of plasma transport in these devices is associated with some of the frontier problems of significance in fundamental science e.g. that of turbulence in magnetized fluids. The theoretical and computational challenges involved in the study of plasma behaviour in MCF devices in the specific context of the upcoming machines viz., ITER and future DEMO reactors have been enlisted below. The ITER and subsequent DEMO reactors will be machines for carrying out feasibility studies for the possible commercialization of the fusion energy. These devices, therefore, will operate in longer pulse and/or continuous mode and give the first opportunity to investigate issues related to a self-sustained 'burning plasma'.

1.1 MHD Activity in Plasmas

Plasma is a magnetohydrodynamic (MHD) fluid. A proper confinement of plasma therefore demands that considerations for MHD stability be met in design as well as operation of fusion machines. For devices operating with confinement time less than 10 secs, the elimination of fast growing ideal MHD instabilities, suffices. However, for longer duration discharges in ITER and DEMO like machines the non ideal effects associated with resistivity, electron inertia etc., manifesting over slower time scales would become relevant for the stability and transport of the plasma in such devices. The non ideal effects free the plasma from the topological constraint of magnetic flux conservation and the resultant reconnection of magnetic field lines drives several new varieties of instabilities. The neoclassical tearing and resistive wall modes are some such modes, which are crucial in this regard.

MHD 1: stabilization of sawtooth oscillations by fast particles:

Sawtooth oscillations at q=1 surface – can trigger Neoclassical tearing modes, Implications for ITER where fast particles can stabilize sawtooth instability

MHD 2: Physics of small magnetic islands

Excitation threshold – the small island physics is still not well understood

MHD 3: Neoclassical tearing modes

Identification and avoidance of Neoclassical tearing modes which - can cause deterioration of energy confinement, limit beta and sometimes lead to disruptions

<u>MHD 4:</u> Modeling Resistive Wall Modes

Beta limiting instability – associated with external kink modes

<u>MHD 5:</u> Implications of error field induced instabilities

<u>MHD 6:</u> Internal Transport Barriers

Understanding localized internal instabilities associated with sharp pressure gradients of ITBs

<u>MHD 7:</u> Modeling

Modeling of ELMs, energetic particle modes, disruption physics, runaways, VDEs, etc. Theoretical modelling with MRE has many weaknesses e.g. polarization current term; needs better validation through use of 3D nonlinear MHD codes

1.2 Issues related to burning plasma

Experiments on burning plasma would be carried out on ITER and DEMO like reactors. Each fusion reaction between the DT nuclei produces energetic neutrons and the alpha particles. The neutrons being neutral immediately leave the machine. The alpha particles, which are charged, stay in the device for a longer time keeping the DT fuel hot for a self-sustained fusion reaction to proceed. However, once the alpha particle gets thermalized it should be removed from the machine to avoid dilution of the DT fuel. There are thus two major challenges involved in the successful operation of burning plasma experiments. Energetic alpha particles must be confined for a duration in which it can appreciably transfer its energy to the fusion fuel. Secondly, the confinement time for the thermalized alpha particles should be extremely small. In addition to these the impact of energetic alpha particles on the MHD stability and transport also needs proper investigation.

<u>ASH 1:</u> Models to optimize the helium ion transport.

Fast helium ion must be confinement for longer time for efficient plasma heating and the slowed helium ions should be deconfined as early as possible as they degrade the whole plasma confinement

<u>ASH 2:</u> Understanding neoclassical diffusion of helium ions from the core to the edge.

<u>ASH 3:</u> Effect of residual thermalized helium ions on the dilution of the D-T fuel

For ITER like machines, the effect of residual thermalized helium ions on the dilution of the D-T fuel, and on overall machine operation needs to be modeled

1.3 Integrated Control for ITER-like Machines

The fusion plasmas are very far from thermal equilibrium and many different reservoirs of free energy can drive instabilities, which grow to large amplitudes. Therefore, plasma is often in highly nonlinear state. The plasma phenomena vary in length scale from sub-millimeters (plasma instability, turbulence) to meters (radius of the device) and in time scale from tens of nano-seconds (wave-plasma interaction) to few thousands of seconds (duration of the discharge). The performance prediction of ITER-like machines should include these multi scale phenomena. Present physics and numerical techniques address these issues very well individually or few scales together. The challenge is to couple all the multi-scale phenomena together so that one can predict the performance of plasma in a reactor. If a simulation is to accurately resolve rapid-time scale and shorter wavelength phenomena, then dealing with slow-time scale and longer, device-scale length phenomena will require special mathematical and computational methods. Such type of simulation will not only help to understand basic theories but also provide support for fusion experiments.

Modeling of fusion-relevant plasmas require qualitative improvements and innovations to enable cross-coupling of a wider variety of physical processes and to include the presence of multiple time scales, ranging over fourteen orders of magnitude, and multiple spatial scale, ranging over eight order of magnitude. The computational domains are geometrically complex and the solutions are extremely anisotropic. In many cases, the physics approximations are not completely understood and hence the model equations are unclear. The underlying physics is coupled with nonlinearities. Taken in isolation, models have been developed or are under investigation for each of these challenges.

Integrated model development is envisioned to be a research tool that contains comprehensive coupled self-consistent physics problems, which would then be used to guide experiments and be updated with ongoing results. In other words, it would serve as an intellectual integrator of physics phenomena in fusion plasma experiments.

This development will require advances in fusion physics (to develop the underlying physics models and elucidate their mathematic basis), applied mathematics (to further develop suitable algorithms for solving models on the appropriate computer architecture, and to define frameworks within which these algorithms may be easily assembled and tested), and computer science (to provide an architecture for integrated code development and use, and to provide analysis and communication tools appropriate for remote collaboration). Strong collaborations, forged across these disciplines and among researchers working in different topical areas, will be essential to the program.

MOD 1: Multiple length and time scales and development of hybrid tools

Extension of similar experiences from other walks of sciences such as model for weather prediction; modeling of biological organism, etc. can be utilized for the coupling of plasma phenomena with scales of various length and time.

MOD 2: Development of non-equilibrium statistical physics formalisms

Modification of transport processes triggered by instabilities could lead to quasi-plasma turbulence (weak) or full scale turbulence in a reactor. Driven, dissipative systems such as tokamaks need extensive non-equilibrium statistical physics formalisms, tools and measures.

MOD 3: Upgrading the existing tokamak simulation codes

The Tokamak Simulation Code (TSC) is one of the tools available to predict the plasma behaviour in tokamak. This has to be improved by adding various phenomena seen in experiments recently as well as other relevant phenomena of reactor plasmas. In reactor, one would expect modification of sawtooth oscillation due to the presence of high energetic particles and their coupling with unstable modes in the plasma edge known as Edge Localized Modes (ELMs). There exists Neo-Classical Tearing modes, which tear the plasma current column. These are slow growing modes seen in long pulse experiments. Resistive Wall Modes (RWM) are instabilities which grow on resistive time scale of the wall and limits the attainable plasma pressure in the device. There are mechanisms to control/suppress these activities externally by applying either RF, using external coils or through other input power. Modules related to these activities and their suppression have to be added so that one can predict plasma behaviour accurately.

MOD 4: Modeling effect of ELMs on ITER plasmas

Even though ELMs are edge phenomena, they affect the energy confinement of core plasma, which is a crucial parameter of the fusion triple product. Edge plasmas, contaminated by impurities due to the interaction of plasma with the components facing plasma are rich in atomic processes. It is important to understand the influence of these phenomena on ELMs activity and is necessary for controlling the ELMs to improve the design of tokamak.

MOD 5: Integrated code development

The development of integrated simulation code is an important activity and may extend over a decade to complete. This tool will integrate all the physics, electromagnetic, auxiliary power and so on. The resistive MHD equations (such as continuity, momentum, energy along with Maxwell's equations) with the actual geometry of the vessel has to be solved using Finite Element Method. This tool will be used to predict the long pulse operation of a typical experimental shot and will suggest the way to decide the next shot. This can be used to avoid plasma disruptions, generation of energetic runaway electrons so that the device does not get damaged.

MOD 6: Development of integrated modeling

Integrate existing models that work in isolation, and the integrated model must have capabilities to predict likely MHD activities, turbulence, transport and other mechanisms related with the reactor operation, and should interact with the real time machine control.

1.4 RF Wave-Plasma Interaction

In an ITER like machine, ICRH and LHCD will play a dominant role in heating and current drive schemes. For ITER like machines LHCD will operate at 5 GHz and deposit steady state power ~ 50 MW. To understand and then to optimize the LHCD operation, fundamental physics underlying in the interaction of such waves with the plasma needs to modeled. Recent experiments have indicated their predominant role in MHD activities including sawtooth effects. A number of simulation codes exist, however ITER like machines have extended parameters and there is a need to develop codes that are described below.

WPI 1: Development of computer codes for wave plasma interaction

For this, a flexible and comprehensive computer code has to be developed with the following modules: wave spectrum module, ray-tracing module, power absorption module, plasma equilibrium module, transport module, Fokker-Planck module etc.

WPI 2: Integrated model for the edge plasma

ICRH uses an antenna along with a Faraday Shield in front of the antenna. Interaction of the same with the edge plasma happens hence it is desired to develop an integrated model for the edge plasma with realistic modeling of RF sheath, modeling of Plasma rotation due to ICRF power and wave coupling in ELMy plasma.

WPI 3: Model to study the synergistic effects

In big machines neutral beams and ICRH are frequently used simultaneously and as both will deposit a large amount of power in the plasma it is required to develop a model to study the synergistic effects

1.5 Plasma Surface Interaction

In ITER like machines where the plasma durations are long and its interaction with the surface is also for a long period, there are serious issues related with plasma sputtering of carbon based surfaces, contamination of plasma and subsequent entrapment of tritium. Disruptions generate runway electrons, which interact with Plasma facing Components (PFCs). Some of the open areas related with modeling are:

PSI 1: Edge plasma - SOL coupling codes

Edge plasma – Scrape Off Layer (SOL) coupling codes have to be developed and have to be integrated with plasma turbulence codes.

PSI 2: Plasma interaction with graphite surfaces

Plasma interaction with graphite surfaces, carbon sputtering and re-deposition, sometimes results in the formation of hydrogenated thin films. These films can remove significant amount of hydrogen (deuterium and tritium) from the reactor and will affect fuel inventory inside the machine

PSI 3: Plasma interaction with ITER like first wall

Plasma interaction with ITER like first wall and blistering / flaking / creep studies of W, Be, B – Experiments, involves development of codes with inputs from Material Science

PSI 4: Interaction of runway electrons with PFC

Characterization of interaction of runway electrons with PFC and estimation of heat flux and material damage, plasma contamination will be needed.

1.6 Neutronics-related Modeling

In a reactor, significant amount of helium ions, neutrons and gamma rays are generated which will not only interact with the plasma facing components but will interact with the structural and other materials which does not see the plasma directly. A lot of these materials, which are stable under normal plasma environment, might lose their stability in such an environment. The modeling aspects involved with this are listed below;

NEU 1: Neutron induced Damage of materials

Damage of materials in the presence of high energy, high flux neutrons and simultaneously the material being at high temperature; look for synergy effects of radiation and high temperature; post irradiation effects, ductile to brittle transition, fracture, micro-structural evolution, etc.

NEU 2: Estimation of Neutron cross sections

Estimation of Neutron cross sections at all neutron energies relevant to ITER like machines and development of direct nuclear reaction models (spallation, pick-up & stripping) are of importance.

NEU 3: Neutron Shielding

Development of materials and process codes that will be used to design Neutron Shielding

<u>NEU 4:</u> Helium induced material damage

The role of fusion-relevant helium transmutation on deformation and fracture of structural material, enhanced cavity swelling

<u>NEU 5:</u> Development of MD simulation of radiation damage

Development of MD simulation of radiation damage is necessary to understand, displacement cascades and interstitial motion, KMC simulation for microstructure evolution as a function of time and temperature.

2. Advanced Material Technologies

In ITER scenario, a lot of advanced engineering skills and materials have to be developed, however their stability in a neutron environment is of prime importance. This is an area where a lot of activities are envisaged, and may serve as a benchmark for material evaluation. This aspect (stability in a neutron environment) is not mentioned below, though being the most important aspect. Under advanced material technologies the following R&D activities are identified;

2.1 Development of Special Processes and Skills

In construction of big tokamak machines there are many different materials that are to be joined where each material has very large areas. To achieve such deliverables special techniques have to be developed. The following are some of the research topics.

PRC 1: Special brazing and welding techniques

Establishment of special brazing and welding techniques leading to development of large area electron beam welding facilities, friction welding and other forms of welding.

PRC 2: Dissimilar material joints

Development of techniques and facilities for dissimilar material joints, especially ceramic to metal joining for application in high power tubes, NBI components, feed-throughs, etc.

PRC 3: Advanced fabrication

Advanced fabrication would include large area machining having strict dimensional tolerances, involving robot-assisted fabrication, in-situ characterization of fabricated surfaces

PRC 4: Heat treatment technologies

Heat treatment technologies are to be developed that will be used make large volume ceramics, heat treat special steels and joints, etc.

2.2 Development of Facilities for Special Material Fabrication

Special materials, which have been proposed for use in ITER requires very complicated sequence of processes for their machining and handling. To develop and manufacture those special materials, with the know how of those special processes for the future, the following topics can be identified;

MAT 1: Development of Borated steels, Low activation steel and Vanadium alloys

Low activation Steel is the material suitable for the structural material for the tritium breeding Blanket material in the fusion reactor. Low activation steel can go well with the Pb-Li eutectic coolant. If liquid Li is used as a coolant material then it is better to use Vanadium alloy as the structural material. Development and fabrication of low activation ferritic steel need to be undertaken.

MAT 2: Fabrication and characterization of SiC composites

SiC based composite materials are good because these materials can work at high temperatures thereby increasing the thermal efficiency of the system. However, their thermo-mechanical properties have to be tested after neutron irradiation. Fabrication of these materials has to be done in such a way that post-irradiation property degradation is minimized.

Fabrication of SiCf/ SiC composite as a structural material for higher thermo-mechanical characteristics and better irradiation resistance is a crucial area. Testing of out-of-pile Mechanical and thermo-mechanical properties before and after irradiation with neutrons/ions of high (MeV) energies. The test on the small samples will be done using charpy and tensile tests. Similarly thermal conductivity experiments will be carried out using suitable heat source and detector arrangements.

MAT 3: Microstructural studies of SiC composite before and after irradiation

Microstructural studies will include the details of TEM and SEM morphological studies, voids, defects, phases and interphases formed between the fiber and the matrix. The structure after irradiation is expected to show more damage, however it will be important to see the collapse or the phase transformation such as amorphization occurring at high doses or longer irradiation time. This will give idea about the resistance of the composite material to the irradiation.

MAT 4: Fabrication and characterization of ODS steel

Fabrication / Synthesis of SiC_f/SiC composites to satisfy the challenges such as thermal conductivity improvement, radiation stability – Differential swelling between SiC fibres, that are not fully dense or crystalline, carbon interphase and SiC matrices, gaseous transmutation rates, hermetic behaviour. Stochiometric and highly crystalline SiC fibres such as Hi-Nicalon S type and Tyranno SA will be used for composite fabrication. These fibres will be coated with carbon for controlling mechanical behavior of the composite through the fibre-matrix interphase control. Structural optimization and characterization of thin carbon interphase will be taken up. Out -of-pile Mechanical properties testing of ODS (Oxide dispersion strengthened) F/M steel before and after irradiation with neutrons/ions of high (MeV) energy. The test on the small samples will be done using charpy and tensile tests.

MAT 5: Study of Irradiation of F/M steel using KeV and MeV ion beams at different temperatures and He/dpa ratios.

Studies on Ferritic Martensitic alloys will be focused on the understanding of the mechanisms by which the improvement/ degradation in the mechanical properties can be coupled with the optimum temperature and He/ dpa ratio. Such coupling is possible through the microstructural investigations. Interesting studies could be void swelling, defect structure, bubbles formation etc. after irradiation. Diffusion and segregation of constituent elements along with the vacancy migration has to be understood for the behavior of the material in the irradiation environment.

MAT 6: Microstructural studies of ODS F/M steel before and after irradiation

Microstructural studies will include the details of TEM and SEM morphological studies, voids, defects and phases formed. The structure after irradiation is expected to show more damage , however it will be important to see the temperature at which the resistance of the composite material to the irradiation is maximum.

MAT 7: Development of Holey fibers for optical diagnostics

In the Holey fibers, the light travels through the 7-micron-diameter hollow core of a 50-micron-wide photonic crystal fiber made from glass tubes fused together lengthwise in a honeycomb pattern. The holey pattern acts as a "corral" that keeps a narrow range of wavelengths from escaping the core, while letting other wavelengths out. These fibers can help control the spreading of an optical pulse and even guide a single mode over all wavelengths, both important features for high data rate transmission. Holey fibers have been shown to almost entirely eliminate optical non-linearities and dispersion, and to provide exotic optical characteristics. Recently it has also been demonstrated that by filling the core of this fiber with hydrogen, it can be used as an efficient, low-power Raman converter. For the optical diagnostics, it is being proposed to use these fibers to improve upon the signal level. Development of these fibers in India needs to be taken up as it has potential for industrial use.

MAT 8: High strength Copper alloys

2.3 Development of Special Coatings

Development of specialized functional coatings and characterization of unwanted coatings by the tokamak plasma are the areas to be addresses under this sub-topic. While some of the components need to be coated with specialized coating before integrating to the tokamak, some of the fresh components get coated by the plasma itself that are facing the plasma environment. Procedures have to be developed to remove them in-situ. The activities, which have been identified in these areas, are as follows;

COT 1: development of thick electro-deposition of Cu

Development of thick electro-deposition of Cu on large areas for NBI applications

COT 2: PVD Coatings

Development of physical vapor deposited Cu on insulators/ceramics for critical RF applications

COT 3: Plasma spray coatings

Development of plasma sprayed Aluminum Nitride and other high-end insulators for NBI and blanket

COT 4: Coatings for blankets

Development of arc deposited and sputter deposited metal nitrides and metal oxides for blanket applications

COT 5: development of special reflecting coatings

Development of special reflecting coatings for optical mirrors mounted on the reactor wall; these mirrors have self-cleaning abilities or can be cleaned in between the shots by some in-situ techniques

COT 6: hydrogenated amorphous carbon films

Development of processes by which hydrogenated amorphous carbon films formed on reactor walls and shadows can be removed and trapped hydrogen liberated

2.4 Database for Material Properties

At the moment a comprehensive database of material properties due to combined effect of plasma heat and particles, neutrons, helium ions, and other aspects (like thermal and mechanical loading cycles, fatigue, creep, etc.) is not available. Though some scattered databases exist, it is felt needed to generate a complete and comprehensive database with all required properties. Out of those properties, the most important requirement seems to be the neutron irradiation environment, which needs a lot of small experiments to be done.

MDB 1: Database for LHCD, NBI, ICRF components

LHCD, NBI, ICRF components which are in contact with the plasma

MDB 2: Database for optical properties

Optical and ports through which information exchange with plasma happens

2.5 Development of Characterization Facilities

Materials characterization forms an integral part of the ITER like activity, as a large number of materials have to be developed along with the understanding of their performance in very harsh environments. Some of the facilities that have to be developed and trained to be used are.

MCF 1: Characterization facilities

Dynamic SIMS, X-ray photoelectron spectroscopy, Electron Microscopy (TEM, SEM), Atomic force microscopy, X-ray diffraction

MCF 3: Characterization of corrosion

Study of corrosion - potentiostat

MCF 4: Mechanical Properties Testing

Mechanical Properties Testing includes, micro and nano indenter, wear, fatigue, creep testing, adhesion, thickness and other measurement facilities

3. First Wall Engineering

The design and development of first wall of the tokamak is a challenging task, because of the fact that the components of this first wall will bear the brunt of the high heat and particle flux from the plasma. So for the components to withstand such harsh conditions, design and fabrication of those components should have included a comprehensive performance evaluation of these materials in ITER like conditions, which need to be developed. Some of the activities foreseen under this head are as follows;

FWM 1: Beryllium tiles

Beryllium is a First Wall Material used in ITER. It is needed to develop panels with Beryllium tiles cladded with actively cooled heat sink material such as copper alloy or stainless steel.

FWM 2: cladding/attaching of Beryllium tiles with heat sink material

Technological challenges include cladding/attaching of Beryllium tiles with heat sink material that can withstand head flux of the order of 1MW/m2.

FWM 3: special handling techniques for Beryllium

Beryllium is toxic in nature, so development of special handling and machining tools and special precautionary measures.

4. Cryogenic and Magnet Technologies

ITER like machines have a very high demand of cryogenic and magnet technologies. The work elements involve indigenous development of Cryo-pumps and Cryogenic systems, Cryo-transfer lines, Superconducting cavities and Superconducting material development, NbTi & Nb_3Sn cables, HTSC Current lead development, development of high field, large volume Superconducting magnetic systems etc. Some of the activities under this head are;

MAG 1: Fusion Relevant Cable-in-Conduit-Conductor Design

Superconducting Magnet System provides the basic equilibrium and confinement to plasmas in magnetically confined Fusion Devices. NbTi & Nb_3Sn based high current carrying high field compatible Cable-in-Conduit-Conductors (CICC) are employed as the base conductor in the magnet winding packs. Design of such CICC involves optimization of micron sized superconducting filaments, stabilizer to superconductor ratios, twisting pitches and stages, thermo-mechanical compatibility of the jacket materials, current sharing and redistribution characteristics in response to normal and off-normal disturbances conceived in fusion plasma scenarios etc. Sophisticated codes are required to interactively design the CICC, model their applied superconductivity related characteristics, electrical and normal zone characteristics that serve as vital input to the cable design and manufacturing technologies.

MAG 2: Quench Simulations in Fusion Relevant Superconducting Cable-in-Conduit-Conductors and Magnets

Irreversible superconducting to normal transitions under the influence of spatially and temporally varying disturbances are known as quench in superconductors. The quench dynamics and normal zone propagation characteristics are largely dependent on the helium flow conditions and plasma scenarios as well as on the intrinsic cable characteristics like stability margin, energy margin, current redistribution, `n' value and ramp rate characteristics etc. Codes are required to study the quench phenomena in fusion conductors and magnets incorporating the cryogenics aspects that are usually used as design input to both magnet design as well as cryogenic flow distributions.

MAG 3: Ramp Rate Limitation & Current Redistribution Investigations in

Fusion Relevant Magnets

It is often observed that cable-in-conduit-conductor wound superconducting magnets quench at much lower currents under rapidly varying transport currents and/or fields compared to nominal quench current values under DC conditions. Experiments focused at understanding these phenomena and their modeling are presently underway in global scenarios. Innovative experiments and modeling investigations would enable design tools for fusion relevant magnets design.

MAG 4: Fusion Magnet System Design

Design of Magnet System for Fusion Devices are very demanding and highly inter-disciplinary in nature. Shape and size of the plasma, plasma initiation, plasma equilibrium evolution characteristics, shape control issues, electro-magnetic forces and loading, structural compatibility, irradiation issues, cryogenics issues, protection and magnet energy dump issues etc are made input into a reliable magnet system design. Further the design is optimized to space requirements and cost. Integrated designs incorporating the above aspects are presently conceived as a possible design approach to magnet system design.

MAG 5: Development of Cryo-Compatible Insulation System for High Field High Current Carrying Superconducting Magnets

Insulation system forms an integral part of superconducting magnet winding packs. Thermo-mechanical compatibility inside the winding pack, stiffness and rigidity as a part of the load bearing structure and electrical integrity (break-down voltage characteristics) are the goodness parameters of the insulation system together with its appropriateness for use in cryogenic environment. Insulation systems suitable for vacuum pressure impregnation as well as for those for the winding pack are planned to be developed for fusion relevant magnets.

MAG 6: Manufacturing Technology Development for Large Superconducting Magnets

Winding of very large magnets appropriate for fusion applications involving large size cable-in-conduit conductor is required to be developed indigenously. The winding line requires precisely designed large take-up spool, driving mechanisms, bending and forming rollers, on-line insulation stations etc. The manufacturing also involves consolidation fixtures and tools, handling fixtures and vacuum pressure impregnation moulds etc that requires heavy and precise engineering. Design and winding pack manufacturing tools, detailed analysis of each of the components as well as that of the integrated assembly is done prior to the winding.

MAG 7: Indigenous Development of Superconducting Strands and Cable-in-Conduit Conductor

NbTi and Nb_3Sn based strand are required to be developed for making superconducting magnets for ITER like machines. These strands are required to be twisted and jacketed inside a conduit with minimal residual stress. The metallurgical process ability, extrusion parameters and the mechanical aspects related to the fabrication of strands of kilometers long single length as well twisted and jacketed cable with reduced degradation etc. would be investigated.

ITER like machines have a very high demand of cryogenic and magnet technologies. The work elements involve indigenous development of Cryo-pumps and Cryogenic systems, Cryo-transfer lines, Superconducting cavities and Superconducting material development, Nb:Sn cables, HTSC Current lead development, development of high field, large volume Superconducting magnetic systems etc. Some of the activities under this head are;

5. RF and Microwave Technologies

In ITER like machines RF and microwave based techniques play a very big role in plasma heating, current drive and in keeping the plasma in fusion conditions. These technologies rely on development of high power Klystrons, Tetrodes, and require development of RF & Microwave components, transition technologies etc. Some of the work elements involved are;

RFM 1: Development of High power vacuum tubes –Tetrode/Triode

High power vacuum tubes like Triode; Tetrode etc. are widely used for RF generators in MHz frequency range. The work includes RF, electrical, mechanical, material science, technology development in the area of metal to ceramic joint, fabrication, assembly and low-high power testing

RFM 2: Dummy load

High power dummy loads are used for testing of RF generators. Such loads can be used as matched as well as mismatched loads. RF power is absorbed in the load and changes the temperature indicating RF power level

RFM 3: Setting HV testing facility for CW operation

Design, procurement, assembly and testing on dummy load (cooling facility should also be included) with fully automated DAC system.

RFM 4: Filament, beam focusing, cavity design for beam bunching

Filament, beam focusing, cavity design for beam bunching in GHz frequency range; Design, fabrication and characterization. Final assembly with developed tubes along with full rating testing

RFM 5: Assembly, characterization, conditioning, fabrication & testing of klystron

The tube should be developed, tested and delivered: finally a facility should be developed to deliver future requirements for national programs.

RFM 6: Window/vacuum feed-through in GHz/MHz frequency range

The component would be used to isolate the vacuum and air - side of the transmission line system. The proposal needs designing, fabrication and testing of proto-type window and delivery of final window. The fabrication involves high temperature vacuum brazing techniques between ceramics to metals

RFM 7: Microwave transmission line components

Development of wave guides, coaxial transmission line, bends, directional couplers, phase shifters, Power splitter, combiner

RFM 8: RF measurements

RF detection system, amplitude & phase measurement system near antenna mouth.

RFM 9: Impedance transformers

Transformers, transforming modules; The components would be used for delivering high power RF to the antenna system. Several input lines would feed the RF power to this component and would deliver the RF power as per the geometry of the antenna design with minimum of insertion loss, proper phase balance, good isolation between various arms and low VSWR. The proposal needs designing, fabrication and testing of proto-type components. The fabrication involves precision machining and good brazing/welding techniques

RFM 10: High Power antenna system

The components would be used for coupling high power RF into the plasma with good coupling property and low reflection. The proposal needs designing, fabrication and testing of proto-type components. The fabrication involves precision machining, UHV compatibility and good brazing/welding techniques.

RFM 11: Wide Band Circular Corrugated Wave guide for microwave diagnostics

Wide band circular corrugated wave guide is to designed in different sizes (dia.=90 mm and 63.5 mm). This wave guide will be used as over-sized wave guide to transmit the emissions (~ 100 GHz to 1000 Ghz) from Plasma to the radiometer and Michelson interferometer. Simulations are to be carried out by using standard software like Microwave studio for the best performance. The technical specification to be generated and fabrication has to be done. After fabrication testing has to be done for the required performance.

RFM 12: Development of Mitre Bend, transitions, UHV Brewster angle

quartz window

Mitre bend which will be used in the path to bend the Wide band corrugated wave guide is to be designed. Also the 'Transition' from rectangular wave guide (D-band) to circular corrugated wave guide is to be designed and developed. Ultra high vacuum compatible Brewster angle quartz window with CF flange (sizes 63 CF, 100 CF, 150CF) also need to be developed.

6. Beam Technologies

The neutral beam system for ITER consists of two heating and current drive (H&CD) NB injector and a diagnostic neutral beam (DNB) injector.

In each H&CD NB injector has beam energy of 1 MeV of negative deuterium (D^-) ion beams of 40A is neutralized to form D^o neutral beam, which delivers 16.7 MW power to ITER plasma to contribute heating the plasma and driving current for steady state operation. The H&CD NB system will be able to operate for 1 hr. The required current density for D^- ions is 200 A/m^2.

The DNB system injects 200 keV, 60A of D^- ions to ITER plasma to probe beam for active charge exchange recombination spectroscopy to measure the He ash density in the core plasma region. It also monitor the light emitted from the excited He atoms following charge exchange with the beams. The required current density for DNB is 300 A/m^2

6.1 Neutral Beam Injection

Neutral Beam Injection forms the major heating mechanism in tokamaks. Though a lot of experimental results have been published in the literature, still more needs to be done in the light of long pulse operation. The identified issues in the long pulse operation are as follows:

NBI 1: **scaling laws in low densities:**

An increase in the L-H transition power threshold in the low density operation have been observed in some experiments. It has also been observed that the L-H transitions at lower densities are slower than at normal densities. The mechanism and the physics reasons for such observations are not yet identified. Experiments in SST-1 with Neutral Beam and RF heating would address to these issues.

NBI 2: power thresholds dependence on heating mode.

The dependence of the L-H threshold on the heating method has been observed in some machines. While for an ECRH plasma the threshold is high, in the case of dominant ion heating NBI, it is near normal. A lower edge temperature for such plasmas indicates the possibility of a different role of the electron and ion channels in the L-H transition. Measurement of edge T_i needs to be carried out to address to these issues.

NBI 3: wall recycling effects over longer periods (1000 S +).

To realize a steady state operation in a tokamak, high performance plasma needs to be sustained for long duration. Important characteristic time scales are the time for current profile to become stationary and the time for the change in particle recycling due to saturation in wall recycling. It is important to study these effects over long time scales, in order to understand the dynamics of long pulse operation. Such experiments can be carried out in SST-1.

NBI 4: **long term Z_{eff}, effects on wall conditioning, effects on boronization**.

Recent experiments on wall pumping in tokamaks show a saturation behavior of the wall. This changes the operation behavior. Increase in carbon generation has been observed under saturated wall conditions. Understanding the process of carbon generation in long pulse plasmas is important for controlling divertor plasma, which ultimately controls the Zeff in the plasma. Such experiments are planned in the SST- 1 tokamak under Neutral Beam Injection.

NBI 5: Experiments on profile of beam driven current profile and effect of fluctuations on fast ion orbits and a good experimental validation of NBCD calculations

The profile of neutral beam driven current in the tokamaks is a matter of current interest, as it forms a significant contributor to the total plasma current. Experimental observations and code predictions do not have perfect agreement on the beam driven current profile. In view of this it is important to study neutral beam current driven further, including effects, which have not yet been considered.

NBI 6: duct conditioning and issues with NB power transport through duct.

Power transport to the plasma takes place through the Neutral Beam-Torus interconnecting duct. For short pulse discharges, the duct once conditioned, remains so for shots lengths equivalent of few S. Wall recycling effects dominate the de-conditioning processes. The long pulse behavior of duct is therefore an interesting area of study. It involves, conditioning techniques, material selection, application of sputter resistant coating, etc.

NBI 7: Synergy effects & Database on Synergy effects using NB & RF heating systems in shaped plasmas

Synergy effects – i.e. heating & current drive using multiple schemes in the same machine have led to interesting observations on achieving higher confinement modes in tokamak plasmas. To date experiments have been rarely carried out on shaped plasmas. With multiple heating schemes available in SST-1, it is envisaged that experiments would be carried out to study synergy effects and important database for future devices is a likely outcome.

NBI 8: Steady State NB heating in SST-1

Objectives: Establish operational parameter space – initial phase begins with short pulse discharges - Heating scenario under different magnetic configuration – effective NB power utilization - Validate L-H threshold power (~ 0.6 – 2.1 MW) for SST-1 – assess inputs to scaling- Develop time dependent heating profile - Develop particle influx model and comparison with LP database

NBI 9: Study of B field compensation & NB-Torus duct dynamics

Objectives: Establish and effective compensation scheme – (Soft iron shield vs active compensation) - Assess duct conditioning – material effects, behavior over short and long pulse - Re-assess thermal design – based on conditioning inputs - Assess long pulse duct behavior – material erosion, effects on plasma, diagnostics etc.

NBI 10: Study the synergy effects under steady state heating scenarios in SST-1

Objectives: Working out a synergy map for SST-1 (matrix of most efficient use of non inductive drives in different parameter regimes) - Short pulse injection of RF + NB including phasing studies - Study long term effects of synergy heating – plasma performance, impurity influx etc

6.2 Physics of Energetic Particle Beams

The study of beam optics it is necessary to develop the computer code, which will give the input to the design of ion extractor system.. As some commercial codes are available which are very costly and vendor can give only EXE file so we are handicapped to change anything or want to know some information we want. However many physical phenomenon e.g., multi-aperture extraction, beam-mean interaction, plasma meniscus near the aperture etc are not included in the code.

In case of negative ion extraction, multi-species e.g., H^-/D^- ions, electrons are present in the plasma meniscus near the extraction grid aperture. The electrons are co-extracted along with the negative (H^-/D^-) ions, these electrons have to be separated from the negative ions other wise production of negative ions shall be reduce due to collision with the electrons, this is called stripping loss. Dipole magnets embedded in the acceleration grid do the separation of these co-extracted electrons. The dipole magnetic field bends the trajectory of the electrons, which dump on acceleration grid. The presence of this magnetic field make the asymmetry to the beam path, hence 3D computer code is necessary to study the negative ion beam optics.

6.3 Positive Ion extraction from plasma source

In case of positive ion extraction, single species (e.g. H^+ ion) is present in the plasma meniscus formed near the acceleration grid (which is connected to plasma box) aperture, and then positive ions are extracted, accelerated to the desired energy, beam optics is well understood. This will help to understand the physics involved in positive ion extraction from plasma source. Then in the next step we have to study the physics of negative ion extraction and identify the critical issues to be solved computationally.

ION 1: Beam optics

Understanding the role of magnetic field on the beam divergence, mechanism of space-charge compensation and reduction mechanism of beamlet-beamlet divergence, effect of filtering out the co-extracted electrons and corresponding 2ndary electron emission, effect of stripping loss

ION 2: 2D Modeling of ion sources

Development of 2D code for positive ion extraction from plasma source for ion beam energy of 20keV to 100 keV taking into account the space charge effect of the beam.is important. This is to be extended to 3D code for multi-aperture extraction. This work will give beam-beam interaction. The knowledge of positive ion beam optics study by using 2D and 3D code will generate confidence to jump into the development of negative ion extraction code.

ION 3: 3D Modeling of positive ion sources

Development of 3D code to study negative ion beam (200 keV of D^-) optics study for ITER like machine. As mentioned above 2D code will not give correct beam divergence due to the presence of dipole magnetic field for separation of co-extracted electrons. This 3D code will include multi-aperture negative ion beam extraction and beam-beam interaction taking into account space charge effect among the beam.

6.4 Negative Ion Source for ITER DNB

A negative-ion-based NBI system is inevitable for a large-scaled experimental fusion machine such as ITER, because at energies > 100 keV/nucleon, the neutralization efficiency for positive ions decreases drastically while holding at around 60% for negative ions. To optimize the negative ion formation and extraction, plasmas with different characteristics are required in the different regions of the same source chamber and for that reason a special kind of magnetic field topology is present in the source and in the extractor system. A strong transverse magnetic filter field virtually divides the chamber into two parts. One part contains high energy (~20eV) electrons (Driver region) and the other part, which is closer to the extraction grid, contains low energy (~ 1eV) electrons (Extraction region).

The negative ions are mainly produced in the Extraction region. During extraction of the negative ions, electrons are also co-extracted due to similar charges. To filter out those co-extracted electrons, the extractor system also contains strong magnets (~ 1kG). There are two routes for Negative ion formation. (1) Volume production, where high energy electrons collide with background gas molecules to form vibrationally excited molecules and then these excited molecules are converted into negative ions through dissociative attachment process with low energy electrons. (2) Surface production, where nascent atoms after dissociation and the nascent ions after ionization are converted into negative ions while bombarding and getting reflected from a surface having low work function (Cs coated). Surface production dominates in Cs injected sources. The destruction mechanisms of negative ions primarily depend on the presence of hot electrons (electron detachment reaction) and the positive ions (mutual neutralization reaction).

NIS 1: Study on plasma dynamics through a transverse magnetic filter field in a negative ion-source.

The diffusion of the low energy electrons through the transverse magnetic field is either due to the elastic collision with the neutrals or the e-e coulomb collision or some turbulence created by **E**×**B** force or combined. The magnetic filter helps to reduce the electron temperature near the grid to facilitate the negative ion formation and reduce their destruction. But the transverse magnetic field reduces also the electron density near the extraction region. The electron density reduction near extraction is not desirable as far as negative ion production is concerned. Surface process relies on the flux of the H atoms and positive ions, which suggests that more plasma density near the plasma grid is better for negative ion production. The presence of transverse magnetic field reduces the plasma flow towards the grid and effectively reduces the flux of the H atoms and positive ions. Due to these merits and demerits of the transverse magnetic filter, understanding of the physics behind the magnetic filter and the optimization of its topology are very important.

NIS 2: Study on plasma sheath near a conducting wall having cusp magnetic field due to the plasma confining magnets in an Ion-source.

The presence of strong magnetic field, electric field and the presence of different ions having huge mass difference like Hydrogen ions, Cs+ ions and electrons the sheath physics on the plasma grid is complex. The sheath also plays a crucial role in the extraction of negative ions.

NIS 3: Study on Negative ion surface production mechanism on a conducting wall having low work function and having cusp magnetic field due to the permanent magnets in a Negative ion-source.

The study correlates between the negative ion extraction and the Cs contains in the plasma, the work-function of the plasma grid and different plasma parameters. The study can determine the fraction of surface produced negative ions, which are able to be extracted and optimize the Cs evaporation rate into the source. Excess Cs is can sit inside the cooled grid plates and starts breakdown in lower voltages. This problem reduces the possibility to obtain a higher energetic negative ion beam.

NIS 4: Effect of plasma grid shaping on Negative ion extraction through simulation and experiment.

Near the extraction grid surface produced negative ions are dominant which has a directional velocity towards the plasma and not towards the hole due to the sheath potential (plasma is +ve with respect to the wall or the grid). But by shaping the plasma grid holes it is possible to have a preferential starting direction towards the grid hole and get better extraction capability.

NIS 5: Optimization of extraction magnet field and its effect on Negative ion beam through simulation and experiment.

Strong magnets (~1kG) are present in the extraction grid to filter out the co-extracted electrons from the negative ion beam. Due the presence of magnetic field, the negative ions also can deviates from its path and increase the divergence of the super beam. Surface produced negative ions on the plasma grid have a directional velocity towards the plasma and not towards the hole due to the sheath potential. The extraction magnetic field has a potential to bend the negative ion trajectory and helps to bring the ions near the grid-hole center. Therefore optimizing the strength of the magnets inside the grid system is necessary.

NIS 6: Laser-Photo detachment diagnostic for Negative ions: The effect of laser shock and ablation on the laser-photo detachment signal

In a plasma where negative ion is only ~ 10% of the electron density, photo-detachment technique is used to find out the negative ion density and temperature. In this diagnostic method, a high power laser pulse is launched into the plasma having negative ions and due to the photo-detachment; there is an increase of electron density without an immediate increase of positive ion density in the laser beam column in the plasma. The outcome of the reaction consists of two particles - an electron and a hydrogen atom. By measuring the increase of electron density or H atom density, the negative ion density can be found. The increase of electron can be measured by observing the variation of electron saturation current (Δi_e) of a positively biased Langmuir probe which is place co-axial to the laser beam. The negative ion velocity information is present in the time evolution of the laser photo detachment signal.

High power strong laser pulse is associated with strong electromagnetic field which has a potential to disturb the plasma by creating shock and ablation and therefore must have some correction effects on the Langmuir probe based photo-detachment signal. This is so far not quantified.

NIS 7: Low work-function (~ 1.7eV) (solid, conducting and high temperature ~ 300 ^{o}C resistant) material for plasma grid manufacturing for Negative ion source.

Cesium is a very important substance as far as negative ion surface production is concerned because it can reduce the surface work function, which is a favorable condition for surface production. But Cs dynamics is uncontrollable in a negative ion source as it sits wherever it finds cold surfaces. Therefore in spite of closing the Cs oven, it is not possible to operate the source “Cs free” once Cs is injected. Cs molecules come out from those places, where it was deposited earlier and takes part in the plasma discharge. Sometimes, because of its presence inside the extraction grid structure, it is not possible to raise the beam acceleration voltage further due to early breakdowns. This problem can be solved if we can make a low work-function solid material for plasma grid manufacturing.

6.5 Ion Source Engineering

The negative ions (H^-/D^-) are extracted and accelerated through number of grids each having 1280 apertures each diameter of 14 mm. The reference design of the extractor system for H&CD NB injector (1 MeV beam energy) is multi-aperture multi-grid (MAMuG) type with active cooling. One such grid is called plasma grid facing plasma side in ion source is shown in figure.1. Where as DNB injector is a simplified structure with 3 grids system due to its low voltage of 200 kV.

ISE 1: Precision CNC machining:

Critical issues are: drilling of shaped aperture with position tolerance of 4 micron, maintaining flatness of about 150-200 micron and milling of cooling grooves in OFHC copper plate.

ISE 2: Copper Electro-forming

This is one of the important area by which embedded cooling channels are made by deposition of 4-6 mm thick OFHC copper on grids base plate which is also OFHC copper. The required joint strength should such that it should sustain more than15 bar water pressure. Thermal, electrical conductivity & hardness values should be close to parent base material

ISE 3: Friction Welding of dissimilar metal joint of SS304L stub rod with OFHC copper plate.

The friction welding is required for joining of 15 mm rod at various locations on large size OFHC copper plate as shown in figure.1. This rod later on shall be drilled for making pipe, which will supply water to the grid system. The required penetration depth of the rod would be 2-3 mm, during friction welding perpendicularity and specified center-to-center distance of the friction welded rod have to be maintained.

7. Power Engineering Technologies

Development of indigenous technologies for the following: High power vacuum tubes: Triode/Tetrode, High Power Transmission lines,

PET 1: Power supplies:

Power supplies 100 kV/60 A and other auxiliary power supplies for DNB require a study to be conducted on the dynamics due to modulation requirements for operation of DNB. This is expected to form a significant R&D issues for the following: Fast transients (multi megawatt), Limits on energy dissipation (< 5 J) during break down, special protection

8. Advanced Data Acquisition and Control Engineering

In future, four major components like intelligent sensors (smart sensors), signal-processing power, real time communication and distributed intelligence (Artificial Intelligence, GRID computing and Neural Networks) will be important in the Tokamak control as well as monitoring. SOC (System On Chip) will provide the base for the above technologies. Next generation of FPGA technology will support large amount of processing power, flexible real time communication and support with interface to smart sensors, all with the reduced size and operating power. Thus, enough resources will be available due to SOC technology. Hence next step is to provide the information management as well as resources management freed from complexity of the hardware and software.

ADC 1: On-line Fast Feedback Control

The challenges in the discharge control is of the fast control, and the use of technology in noisy environments, use of SOC technology, use of real time fiber optic communication, use of distributed control methods using real time communication, resources management and information sharing among distributed units. Hence major challenges come in planning and management rather than technology. SOC with real time communication are already available that may fulfill current requirements of the Tokamak control and monitoring and in future there will be large amount of resources will be available in the single chip of SOC.

All the future reactor scale tokamaks are envisaged to have Integrated Plasma Control (IPC), which means controlling the plasma with the understood models of plasma behavior. IPC essentially refers to both a scheme and a design approach: it will have Actuators, Sensors, Plant physics and Controllers. Here Controllers are multi-variable, control algorithms with knowledge of coupled physics subsystem states from many diagnostics. Actuator commands which take into account predicted cross system effects.

IPC design is model-based control design, where models have been validated by experiments. It contains multi-variable design techniques, which include performance optimization and methodology for systematically confirming control performance using realistic and accurate simulations.

The requirements for the Real Time Control are (1) Loop time on the order of 0.1 to 1 milliseconds (2) 1 to 10 GFLOP processing power 3) Tera bytes of data generation and management

ADC 2: Development of Control System Hardware & Software:

FPGA / SOC based cluster, Real Time Fiber Optic based Multichannel Communication networks (using wavelength division multiplexing), Intelligent sensors which includes sensor preconditioning circuit, digitizers, pre-processing the raw data, communication interface etc., Standard methods for the distributed intelligent like (AI, Neural networks, GRID, Computing etc. that will be ported to any FPGA/SOC cluster, Remote operation methods using robotics, Synchronization between cluster units, Development of generic Interface methods, Real Time data management methods and storage, Interface and resource management using XML/DXML, HTML/DHTML software technology with data centric applications and service oriented architecture.

ADC 3: Wireless Sensor Networks for Future Fusion Grade Reactor

The crucial component for wireless sensor network is wireless nodes (called Motes) and the physical size as well as the cost of manufacturing of these motes. While deploying, these sensors are dispersed in the field very densely and communicate to one master transponder. Because of its distributed nature it is very convenient to collect data from very wide area. So in the proposed fusion reactor these wireless motes can be deployed to monitor remotely any sort of radiation leaks. In addition to this, health monitoring of the reactor is also possible using wireless sensor network. For developing wireless sensor networks we need to develop our own wireless motes with sensors (electronics) according to the requirement of fusion reactor and the communication protocol for transferring and processing the data from mote to mote, mote to master and from master to host PC.

ADC 4: Control Systems using Advanced Technology for Future Fusion Grade Reactor

At present for SST-1 we have implemented control systems using ready-made electronic modules as well as indigenously developed electronic modules. For the former, we are dependent upon the foreign companies. In addition to this there may be a problem of compatibility between different modules of various vendors and also the support of vendors in case of obsolete modules. To avoid this dependency, it is a good idea to develop our own control cards using SOC and FPGA, which provides at most freedom of logic- reconfigurability. Use of FPGA and SOCs shrinks the board size and makes it very compact. The biggest advantage of homemade system is we have strong support and upgradeability. Modules for development are DSP card with ADC, DAC, Analogue and digital I/O modules, event synchronization, timer, trigger, memory modules. All these modules need to be developed using a single technology of SOC and FPGA with universal compatibility for all the platforms like VME, CPCI, PXI..

ADC 5: Remote Participation In The Machine Operation For Fusion Grade Reactors

The World Wide Web Technology and embedded system make it possible to participate in Tokamak operations from any location. The machine status, diagnostic data, subsystem status can be monitored and machine operated remotely. The scientists also need to communicate through Audio/Video conferencing during the machine operation. They would like to visualize, the online activities at tokamak hall, subsystems and central control rooms remotely. We must develop these systems while ensuring that these work with diverse hardware technologies, networking protocols, and heterogeneous operating systems with ease and scalability. Remote monitoring and operation of fusion reactors over the Internet requires strict security measures. Selection of the appropriate software, hardware technology communication protocol and development tool is a challenging job.

ADC 6: Development of Artificial Intelligence Systems for Fusion Reactor

Fusion devices remain inaccessible to humans during experiments due to radiation, high magnetic fields and other hazards. Any small change in physical setup required during operation brings down the system, thereby incurring huge loss. Materials made brittle by neutron irradiation will have to be replaced. This will entail shutting down the facility for several months so that the interior can be completely rebuilt. This can be achieved in shorter periods with the help of in-vessel remote-controlled robots and thus minimizing the downtime. Efficient and reliable robotic techniques are crucial to the successful operation and for improved reliability and maintainability of a burning plasma experiment.

Several features of maintenance necessitate use of robots in the reactor environment. First is the low frequency of the operation, which calls for a general-purpose system capable of doing an array of maintenance tasks. Second, maintenance and repair require high levels of dexterity. Third, the complexity of these tasks may be unpredictable because of the uncertain consequences of a failure.

The precision with which certain components in burning plasma experiments need to be manipulated is beyond the realm of the state of the art. Robots play a major role in the integration of scanning techniques, remote focusing and position sensing of the laser optical components; compatible in burning plasma environment particularly, high radiation, ultra-high vacuum, high temperature and high magnetic field.

ADC 7: Development of Real Time Network

Fusion Reactors have to be controlled and monitored in real time for successful operation. However, as the sources of information spread out and the amount of information generated increases, the burden on the communication infrastructure grows, posing many technical challenges such as those related with achieving real-time operation, dependability, scalability and flexibility over different media.

Real Time Network focuses on the current technological challenges of developing communication infrastructures that are real-time, reliable, pervasive and inter operable. These infrastructures are the bottom layer that supports a myriad of services upon which communication network is based. The more abundant, accurate and deterministic source of information results in better management, planning and control. The challenge remains in establishing the right combination between real-time operation, dependability, flexibility and the cost.

ADC 8: Real-Time Database Manager/Server for Time Critical Applications

Compared with traditional databases, database systems for time critical applications have the distinct feature that they must satisfy timing constraints associated with transactions. Transactions in real-time database systems should be scheduled considering both data consistency and timing constraints. Since a database system must operate in the context of available operating system services, an environment for database systems development must provide facilities to support operating system functions and integrate them with database systems for experimentation.

Conventional database systems are typically not used in real-time applications due to the inadequacies of poor performance and lack of predictability. Current database systems do not schedule their transactions to meet response requirements and they commonly lock data tables to assure only the consistency of the database. Locks and time-driven scheduling are basically incompatible, resulting in response requirement failures when low priority transactions block higher priority transactions.

New techniques are required to manage the consistency of real-time databases, and they should be compatible with time-driven scheduling and meet both the required system response predictability and temporal consistency.

Nuclear Fusion Control Algorithms demands high time criticality and needs information retrieval in real-time. To fulfil the requirement one has to design and develop real-time database server or manager.

ADC 9: Real-time Data Grid

Data Grids provide access to distributed data stored in heterogeneous resources in a scalable network. They provide a uniform access mechanism for discovery and browsing of datasets using common well-defined processes for data organization, authentication, authorization and server-side processing in distributed administrative domain.

Real-time Data Grids add additional requirements for timely access, fault-tolerance and fail-over and bring challenges in fusing real-time data from different networks/sensors from diverse domains. A real-time data grid with a well-designed architecture will seamlessly fuse and access archived (static) data, real-time data and other dynamic databases systems.

Major components requires for real-time data grid: are a distributed set of participating real-time, or near real-time data sources, as well as archives, file systems and databases, a distributed network servers, transparent access, authorization, resource and data discovery and data management. Data-client interface for which the client/user need not know exactly where, how, or by whom the requested data is being served and a set of client APIs, libraries, web services and GUIs that enable application support and user access.

ACD 10: Tomography for Magnetic Field Measurements

Magnetic field profile must be accurately known at every stage of the tokamak operation. The dynamics of the magnetic field distribution must be known that will demand real time communication for large volume of data and huge amount of processing power. Magnetic Induction Tomography (similar to the technique already used in the medical field called the MRI) can calculate Magnetic Field distribution. In future, the technology will be available like SOC that will provide support for the calculation of real time magnetic profile within the machine. By knowing the magnetic filed profile within the tokamak machine, the operation and control of the tokamak can be easily managed.

9. Nuclear Technologies

Till now, there are very few Tokamak experiments with neutron environment. It is not still clear how the various components of different systems will behave in the high flux neutron (~14.5 MeV) environment. So it is needed to start to study the effect of neutron dosages on various components.

Since there were no previous circumstances to design test and use fusion product diagnostics, there is a tremendous need to do that now. Some of the envisaged Fusion products diagnostics are:

NCT1: Alpha knock-on neutron tail - Nuclear emulsion track detector

NCT 2: Escaping alpha-particle - On-line monitor of gamma-ray emission from nuclear activation of foils mounted near first wall.

NCT 3: Fusion reaction rate - Time resolved Neutron Act. System based on flowing water.

NCT 4: Alpha-particle detectors - Ceramic Scintillators, Micro-Faraday Cup arrays , Imaging Bolometers

NCT 5: Gamma-ray diagnostics, gamma-ray detectors - ***Scintillation detectors, Semiconductor detectors***

NCT 6: RF Antenna and LHCD Couplers in neutron environment.

The functioning of various microwave and RF Antennas, LHCD Couplers have to be tested and documented in neutron environment. It is to be noted that Neutron Damage studies need to be done with high energy, high dose-rate, and some specific components with high material temperature (because of their proximity to plasma)

NCT 7: Testing of x-ray detectors in Neutron environment

It is to be seen whether the materials of the soft X-ray detectors such as Si, GaAs (Gallilium Arsenide) may undergo transmutation with nuclear process. So it is necessary to test the soft x-ray detectors such as Silicon Surface Barrier diodes (SBDs), and PIN diodes in the neutron environment and to see their response before and after the exposure. It is also important to use different x-ray filter materials such as Be, Al, Mo and one can observe whether their actual material remains same or changes with nuclear transmutation

9.1 Blanket Technologies

Breeding Blanket is one of the major technological breakthroughs required to pass from ITER to the next step called DEMO Fusion Power Reactor (FPR) which need to breed the tritium required for D-T reaction and to convert nuclear energy into heat extracted by a coolant under pressure and temperature conditions appropriate to drive an acceptable thermodynamic cycle. Tritium Breeding Blanket is a critical component exposed to severe working condition, which is required for DEMO power reactor. ITER is a unique opportunity to test the Blanket mock-ups, which is known as Test Blanket Modules (TBMs). Two types of blankets are considered for testing; Solid and Liquid type breeder blankets.

The solid type module will be Helium Cooled Pebble Bed (HCPB) ceramic blanket. In this concept, the ceramic breeder and the beryllium multiplier in form of flat pebble beds, which are separated and cooled by cooling/stiffening plates. The helium coolant flows at high pressure (80 bar) and high temperature (350 – 500 C) in the first wall and through small channels in the breeding zone, while the beds are purged by a low-pressure helium flow. This independent purge flow removes the tritium produced in the beds, carries it to a tritium extraction system and keeps low the tritium partial pressure at the interface with the coolant channels reducing the permeation flow to the main coolant system.

The liquid type blanket has two options as breeder material (Pb-Li or Pure

Lithium) with helium gas to cool the first wall and main structures of the TBM. The Low Activation Ferritic/Martensitic (LAFM) steel is considered as the structural material. The FCIs (Flow Channel Inserts) e.g. SiC_f/SiC, are designed as the thermal and electrical insulators inside the Pb-Li flow channels to reduce the magneto-hydrodynamic (MHD) pressure drop of LiPb flow and the maximum temperature in structural walls. Coating may be considered as the tritium permeation barrier if necessary. For the lithium system Vanadium alloys (V-4Cr-4Ti) are being considered as the structural material and special coatings (Erbium oxide/ Yttrium oxide/AlN) for reducing MHD pressure drop will be needed.

FBT 1. Development of Structural Materials (Low Activation Ferritic-Martensitic Steels)

The selection and development of Ferritic Martensitic 9-12%Cr steels with low activation (LAFM-steels) for nuclear fusion is based on the excellent experience with commercial 9- 12%Cr steels in conventional power plants and on a promising irradiation performance under high neutron dose exposure in Fast Breeder Reactors. The possibility to achieve reduced neutron-induced long term activation by an appropriate chemical modification of major alloying elements and a consequent limitation of harmful impurities, which worsen the long term radioactivity properties as well as the potential for recycling were further reasons for the selection of this material group.

FBT 2. Fabrication Technologies

The FW and cooling stiffening plates with coolant channels are too complicated to fabricate using the ordinary manufacturing or welding techniques. So the Hot Isostatic Pressing (HIP) diffusion bonding is generally considered as the primary fabrication technique for these components due to its unique advantages. However, further investigation on the effects on the structural material of the HIP process and how to obtain the permitted tolerances of the components is necessary for hipping these components. In addition, great efforts are required to investigate the joining of the TBM components e.g. the joining of FW and cooling plates. In the process of the components joining, the difficulties are how to make qualified joints without hurting the nearby channels, and how to control and decrease the residual deformation caused by the welding. Currently, researches on EBW (Electron Beam Welding), LBW (Laser Beam welding) and TIG (Tungsten Insert Gas welding) of RAFM steels are greatly needed to explore suitable techniques for the joining of the blanket components.

FBT 3. Development of Lithium Ceramic pebbles

In the development of tritium breeding blankets for fusion reactors, lithium-containing ceramics such as Li2O, LiAlO2, Li2ZrO3, Li2TiO3 and Li4SiO4 were quickly recognized as promising tritium breeding materials. Particularly, Lithium Titanate (Li2TiO3) and Lithium Silicate (Li4SiO4) have attracted an attention of many researchers because of easy tritium recovery at low temperature, high chemical stability. The wet process and sol-gel method are the most advantageous for fabricating small Li_2TiO_3 pebbles from the reprocessing lithium-bearing solution. The required particle size of the Li_2TiO_3 powder is about 10 µm on an average.

FBT 4. Liquid Metal Technologies

Liquid metals such as Pure Lithium or Pb-Li eutectic are the promising candidates as the liquid breeders and coolants for heat extraction. The Development of coolant quality standards, analysis of impurities condition, their sources and accumulation rate in the circuit; analysis of corrosion and mass transfer; development of methods and equipment for keeping impurity content in the coolant within acceptable limits and analysis of different operating procedures and evaluation of related impurities input;

Coolants produced by the industry do not always meet technological requirements. Therefore additional procedures are performed prior to filling circuit with the coolant in order to bring the coolant to required condition. Another cause of changes in the coolant composition is corrosion, resulting in deterioration of mechanical properties of structural materials. Corrosion products are transported along the circuit, and depositions are formed that can affect hydrodynamics and heat transfer, and hence reliability of facility. All circumstances mentioned above lead to the necessity of permanent control over the impurities content and corrosion processes in the circuit.

FBT 5. Hydrogen Isotopes Recovery Process

Hydrogen/Tritium recovery process including the helium purge gas conditions for tritium extraction from the blanket can be investigated and demonstrated by the in-ITER module tests. On the other hand, tritium release from solid breeder materials is strongly related to operational temperatures of the breeder materials. The breeder materials should be maintained within an appropriate temperature range to enhance the tritium release.

FBT 6. Coolant Purification System

A purification system should be provided for the HCS to purify the helium coolant stream keeping the partial pressure of HT lower than the design level. The CPS will extract other impurities (solid, liquid and gaseous) from the main coolant system. The fraction of the coolant gas leaving the cooling system downstream of the coolant blower is sent through a water separator to remove condensed water that may be present as a consequence of water leakage in the heat exchanger. Then, an oxidizer unit should be employed to convert all molecular hydrogen isotopes into water (Q_2 ® Q_2O, Q = H, D, T). This water is frozen out in a cold trap operated at -100°C while the remaining impurities are adsorbed on a molecular sieve bed operated at LN_2 temperatures. The pure helium is warmed up again and returned into the main coolant loop upstream of the blower.

FBT 7. Structural integrity Studies

As for the structural integrity, synergetic effects of thermal, mechanical and electromagnetic loads can be investigated with the TBM. The same coolant and purge gas pressure loads as of the DEMO reactor will be applied to the TBM. Supplementary analyses can compensate the dimensional difference between the TBM and the DEMO blanket. Electromagnetic loads are dependent on plasma parameters including the method of stability control, which are not well defined for the DEMO reactor at present. In any case, prediction methods for the induced electromagnetic loads and the mechanical response of the blanket structure are expected to be established through the in-ITER module tests and will be applied to the DEMO blanket. Thermal stresses are caused by temperature gradient across the material. Heat loads such as surface heat flux and nuclear heating cause the temperature gradient. Since the heat loads in ITER are lower than those in the DEMO reactor, an "act-alike" design of the test module can be considered to simulate thermal stresses of the DEMO blanket. For the "act alike" design, modification of material thickness and cooling conditions can be taken into account.

FBT 8. Neutron Irradiation effect studies

Irradiation effects on the materials properties and the blanket performance are a most critical issue for the testing. Some changes in thermo-physical properties of non-metals, e.g., thermal conductivity, will occur below neutron fluence of 0.1 MWa/m2. Several important effects become activated in the fluence range of 0.1-1 MWa/m2. These effects include solid breeder sintering/radiation cracking and possible onset of breeder/multiplier swelling. Therefore, the blanket behavior relating these effects and their interactions can be investigated by the in-ITER module tests. However, irradiation effects, especially on structural material characteristics including DBTT change, helium embrittlement and swelling, will appear in the fluence range of 1-3 MWa/m2 or higher. Therefore, these irradiation effects on structural materials need to be evaluated separately in out-of-ITER R&D programme.

FBT 9. Blanket Modeling

Neutronic Analysis using MCNP code and FENDEL 2.1 data

(a) Tritium Breeding Ratio

(b) Nuclear Heat deposition estimation

(c) Activation analysis

(d) Decay heat analysis

RELAP/ATHENA codes have been used to simulate the whole He loops (including a relatively detailed simulation of the TBM flow schema) for transient evaluation.

Engineering Design and Analysis requires ANSYS, CATIA, Solid Modeling for Structural mechanics, Thermal hydraulics, Thermo-mechanical and Electromagnetic Analysis. 3D Fluid dynamic calculations (with STAR-CD) have been used for analysing detail of the flow in the Manifolds and in the FW channels, resulting in a evaluation of velocity distribution, pressure drops and heat transfer coefficients. 2D and 3D (up to ¼ of the TBM) ANSYS models with different detail have been developed for calculating temperature distribution and stresses

FBT 10. Helium Cooling System (for high grade heat extraction)

Possibly electricity generation or at least the extraction of high-grade heat can be demonstrated by using high temperature coolant. To reach thermal steady state, or at least quasi-steady state, of blanket materials and outlet coolant temperature, plasma burn times longer than 400-500 sec for blanket front part and 3000-5000 sec for blanket back part are required. Heat removal performance and thermal-hydraulic characteristics can be also investigated and demonstrated with some supplementary analyses for overall dimensional difference between the TBM and the DEMO blanket.

FBT 11. Heat Extraction from Pb-Li

A heat exchanger has been designed mainly for the demonstration of Pb-Li blanket. It allows rejection of the full TBM thermal power from the LiPb stream to a secondary helium cooling system that leads to the tokamak cooling water system (TCWS) building. The heat exchanger is located close to the test module and will handle the liquid LiPb at the temperature up to 700 C and it has the following unique features: reducing length of the liquid LiPb pipes running from the TBMs to the heat exchanger in the TCWS and then minimizing the amount of liquid LiPb and the corresponding loss of tritium to the surroundings

FBT 12. Remote Handling Technologies

The development of remote handling systems is key requirements for maintenance, repair and modification of fusion power reactor components. The removal and replacement of components of blankets are critical. And their in situ maintenance tasks requires cutting welded joints, unfastening bolted flanges and couplings, and disconnecting water and gas pipes, as well as lifting and manoeuvring of extremely heavy components weighing several tons.

10. Plasma Diagnostics

Apart from diagnosing fusion products, the expertise for all the basic to advanced level plasma diagnostics needs to be nurtures for the national programme. For this the following interesting problems has been proposed for immediate action

DIG 1: Laser-Produced Plasma Interactions

The basic physics of high-beta, spatially localized plasma expanding into ambient plasma and the consequent radiation of wave energy is of interest in many areas of space and plasma physics. In solar physics, coronal mass ejections play a key role in the dynamics of space weather and the origin of turbulence in the solar wind. In laboratory fusion experiments, tokamak pellet injection severely impacts the background plasma, radiates waves, and sometimes causes disruptions. Also in the ionosphere, barium release experiments created another example of high-density plasma interacting with dilute background plasma.

The laser-produced plasma is generated by the irradiation of solid targets with a 1.6 J, 9 ns, Nd-YAG laser focused to a spot (size, ~1 mm). The laser is fired with 1 / 30 Hz, repetition rate at the plasma. To present a fresh surface to the laser, target is rotated and /or translated along its axis. The expansion of laser-produced plasma into vacuum / plasma is diagnosed using an angular array of Faraday cups, pulsed magnetic-field coils, fast-gated channel-plate framing cameras, spectrometers. This program requires first, development, testing and calibration of above diagnostics. After the development, the study of laser-produced plasma interaction will be taken up

DIG 2: Measurement of plasma flows in tokamak edge plasma

Increased poloidal plasma flows can suppress turbulence in the edge plasma and hence increase particle and energy confinement. Recent studies have shown that plasma flows can drive turbulence also. It is therefore necessary to get a better understanding of energy transfer between large-scale plasma flows and small-scale turbulence. New measurements should be conducted on SINP, Aditya and SST-1 tokamaks for this purpose. These measurements will require deploying Mach number probes, RPA etc for determining plasma flows, Langmuir probes and optical diodes for fluctuation measurements and new schemes for altering flows (e.g., biased probes, molecular beam injection, RF heating of the edge plasma).

DIG 3: Turbulence imaging by gas-puff and high speed camera in SST-1

The edge plasma is very hot in diverter configuration (e.g., in SST-1) and hence static probes cannot withstand the heat load. The edge turbulence in SST-1 therefore must be studied using probes on fast retracting drives and other remote techniques. We expect that plasma imaging will become an indispensable tool for future tokamaks like ITER and DEMO. Turbulence imaging using local gas-puff to enhance emission has been used in some recent experiments. We are in the process of developing such a technique for imaging turbulence in the edge plasma on SST-1. Small scale blobs of plasma pressure can be measured by gas-puff imaging using fast framing camera and Mach probes on fast drive can measure flows. These measurements can be combined to study relationship between large scale flows and small scale turbulence.

DIG 4 : Development of radiation hardened bolometer

The measurement of radiated power in a broad VUV and X-ray spectral range is necessary for safe operation of thermonuclear reactors like ITER and DEMO. Since these reactors will have high neutron fluence, the bolometer that will measure radiation power should be resistant to high doses of neutrons. At present, development of a radiation hard bolometer head has become a priority for ITER. It requires developing special absorber/ resistor layer, substrate material and electronic circuits for zero-maintenance operation.

DIG 5: Plasma position and shape imaging

Plasma position and shape are measured by magnetic induction coils in present day tokamaks. These coils are going to suffer from saturation in long pulse discharges of steady state tokamaks (like SST-1 and ITER). It is therefore necessary to develop complementary techniques. One such technique is the imaging of plasma position and shape. For this purpose, the X-ray continuum from plasma centre (or full plasma cross-section) will be converted into visible light by using a micro-channel plate (MCP) and recorded using a CCD or Digital Camera.

DIG 6: Spectroscopic study of gas injection physics into tokamak plasma

Spectroscopic measurements of spectral line intensities in small devices with controlled gas flow and verifying the corona and the CR models with measurements with probes.

DIG 7: Abel Inversion calculations

A deconvolution technique (Abel inversion) is required to obtain emissivity using the line-of-sight intensity measurement. There are several versions of inversion methods available for the deconvolution of intensity data. The aim of each of these methods is to reduce the amplification of the random errors inherent in the data to obtain actual emissivity. The accuracy of the inversion method can be measured by its ability to reconstruct the radial profile of emissivity with minimum error using chord integrated simulated data as input. At the same time, the robustness and reliability of the inversion under a wide range of plasma conditions are desirable. The error propagation is also to be evaluated in order to infer a meaningful radial profile. Therefore an error propagation study is highly desirable. Also, the experimental verifications are needed to confirm the simulation results.

DIG 8: Diagnostics developments for Disruption identification and using them for mitigation

Disruption not only destroys plasma, but also the immediate plasma facing components, which might need to be replaced for further operation. So it is vital to develop diagnostics to forecast disruptions and help to mitigate the uncontrollable disruptions.

DIG 9: Development of He-beam diagnostics for SST-1 & Aditya tokamak

The edge plasma region in tokamaks exhibits rapidly varying parameters, in both the temporal and spatial domains. In particular, the phenomena accompanying H-modes, ITB, gas puffing, includes steepening of gradients in electron density n_e and temperature T_e. The regions of interest extend over a few cm's and changes occur within time scale of ~1 ms. In past, line intensity ratios from a neutral He jet are used to measure n_e and T_e profile at the edge plasma These measurement make use of emission rates based on a collisional - radiative model of neutral helium.

Calculations for line intensity ratios are to be carried out using the atomic data and analysis structure, ADAS, which provides all the atomic coefficients needed. The line intensity ratio 706.1/ 728.3 nm for T_e and 667.8/728.3 nm for n_e determination are to be computed using ADAS for SST-1 and Aditya injection geometry. Experimentally measured helium line ratios, may be compared with calculated ratios, with known n_e and T_e dependence, allowing values of electron temperature and density to be derived.

He-beam diagnostics system consists of two major components, 1. The helium introduction system (Pulsed supersonic He atom beam injector) 2. The observation system (a. UHV Compatible Imaging optics and linear array optical fibers, b. Multi-channel spectrograph with EMCCD detector). Following task will be undertaken: calculation of line intensity ratio, integration, calibration, testing of both experimental system, and operation of He-beam diagnostics system to study required physics issues.

DIG 10: Thomson Scattering Diagnostics for SST-1

Thomson scattering has long been a standard diagnostic for measuring the electron temperature and density in tokamak plasma. SST-1 Thomson scattering systems uses multiple 30 Hz Nd:YAG lasers (6 in number) to measure the electron temperature and density profile periodically throughout, the 1000 sec plasma discharge of SST-1. The system is designed for high spatial resolution (1.0 cm) and wide dynamic range (T_e: 20ev –2 keV, $n_e > 1 \times 10^{12}$ cm^{-3}). The present design uses multiple lasers. These laser beams are combined into a common beam path. The laser beams are transported and aligned over 30m of distance by 5 beam folding mirrors. During beam transport, beam parameters are to be optimized very carefully. A beam feedback system corrects the beam position at different locations. The scattered signal is collected by three different optical imaging lens systems, with magnification of 0.2 to provide spatial resolution of 1.0 cm. The scattered photons are dispersed by two different sets of five-channel interference filter polychromator to cover the dynamic range of divertor, SOL, and main plasma. The scattered light is detected by cooled Si-APD. Laser control and data acquisition is performed by PXI based system throughout the plasma discharge.

Following tasks are to be undertaken: development, testing and calibration of, feedback system, laser beam clustering system and transport, controls, integration with SST-1 tokamak, calibration and its operation to study required physics issues.

DIG 11: Development of Fast CDD / CMOS camera and CCD like technology for sensor electronics

Plasma position during plasma formation, disruption and flattop has to be monitored for good discharge with sufficient time resolution (~ 2-4 ms). Commercial CCD cameras at > 200 Hz are available with fix memory on camera board. It makes it difficult to acquire images for full duration of plasma shot (1000 sec). These images can also provide information about flying object during plasma shot. The proposed project is to develop 500 frame/sec CCD/ CMOS camera. Development involves time sequencing electronics & data acquisition system for variable frame rate acquisition. The developed camera circuit is versatile in terms of different frame rate, remote operation, optical coupling & DAQ free from bus, with LabView® as application software.

With time, number of data acquisition channels required for operation & monitor of tokamak, plasma diagnostics, and sub-systems are increasing enormously. A new technique is needed to minimize DAQ channel requirement and reduce data analysis part. CCD like technology for sensor electronics and DAQ seem to be a solution to this problem. A proto-type sensor electronics and DAQ is under development for Thomson scattering diagnostics system. We plan to develop first as general and then compatible to large no of channels.

DIG 12 : Simulation of Thomson scattering diagnostic signal conditioning and data acquisition

We propose to develop a hardware based simulation program for signal conditioning and data acquisition system of a diagnostics system before it is developed in the laboratory. It will help us in evaluating the developed system in terms of it performance. Depending on the performance of the developed system in comparison to simulated one, system will be accepted or will be send for further improvement.

SST-1 Thomson scattering has ~250 signal channel and ~ 100 control signals. These numbers of channels are ~ 25 times larger than Aditya Thomson scattering system. It will be very difficult to do the performance test before taking plasma shots daily. Also, it will be very difficult to find the fault in such large no of channels in short time. It is proposed to develop a hardware based simulation program along with test setup to do the fault finding and daily performance test.

A diagnostic system consists of signal conditioning, data acquisition and control, acquisition and analysis software, storage of data etc. Thomson Scattering system consists of APD detector signal conditioning electronics, data acquisition, control, acquisition software and analysis software. Each part of the system is to be tested for its performance and reliability first. We propose to develop simulation of each subsystem of signal conditioning, data acquisition and control system of TSS using software and hardware. The simulated results will be compared with the actual system's performance. The performance test can be performed for individual subsystems and for integrated system also. At the later stage, it will be gernalized for other diagnostic system.

DIG 13: To develop 500 frame/sec CCD/ CMOS camera.

Development will involve time sequencing electronics & data acquisition system for variable frame rate acquisition. The developed camera circuit is versatile in terms of different frame rate, remote operation, optical coupling & DAQ free from bus, with LabView as application software.

With time, number of DAQ channels required for operation & monitor of tokamak, plasma diagnostics, and sub-systems are increasing enormously. A new technique is needed to minimize DAQ requirement and reduce data analysis part. CCD like technology for sensor electronics and DAQ seem to be a solution to this problem. A proto-type sensor electronics and DAQ is under development for Thomson scattering diagnostics system. We plan to develop first as general and then compatible to large no of channels.

DIG 14:Ellipsometry Technique for Plasma Diagnostics

Ellipsometry technique can be used to determine the magnetic field distribution and hence the safety factors q profile as well as the particle density. Upon entry into the plasma medium, an electromagnetic wave is split into two *characteristic waves* of fixed orthogonal polarizations ('Cotton-Mouton effect' – perpendicular to field and 'Faraday effect' – parallel to field) that traverse the layer at different phase velocities. In the former effect, the characteristic modes are linearly polarized and in the later case they are circularly polarized. At an arbitrary angle to the magnetic field, the two effects combine in a rather complex way and characteristic modes are elliptically polarized. After passage through plasma, they recombine into a single wave. The evolution of the state of polarization of this outgoing wave gives the information of the plasma properties. It, therefore, is necessary to evaluate the plasma effect on evolution of the polarization state of an electromagnetic wave that can be used as a very useful plasma diagnostics.

This project will involve in numerically solving the evolution equation describing the change of polarization. The evolution equation is obtained using Stokes formalism - where input and Stokes vectors represent output polarization states and a transition matrix called Mueller matrix represents the plasma. Since plasma is inhomogeneous, the Mueller matrices evolve as the radiation propagates through plasma. The propagation in plasma involves ray tracing inside the plasma, which in turn, is used to derive the Mueller matrices in each plasma slab. Further, these matrices can be used to calculate the evolution of the polarization state. The comparison of the results will be done with the special cases for which exact analytical solution are possible to validate the numerical calculations.

DIG 15: Plasma position measurement using microwave swept reflectometers

This diagnostic is to be designed to act as a stand-by gap measurement, in order to correct or supplement the magnetics for plasma position control, during very long (>1000 s) pulse operation, where the position deduced from the magnetic diagnostics could be subject to substantial error due to drifts.

Reflectometry relies on the total reflection of electromagnetic waves by a plasma when the local refractive index μ equals zero. For the ordinary mode waves (**E** || **B**), this occurs when the wave frequency $f = f_p$. A beam of microwave is launched from an antenna outside the plasma, parallel to the density gradient. The reflected waves are received by the same or an adjacent antenna and are combined with a reference beam, which has a fixed optical path. Changes in the phase of the reflected beam with respect to the fixed reference result in an interference signal which is detected at output. If the input frequency f is swept over a range Δf, this produces changes in phase even when plasma conditions are in steady state and gives information on the absolute position of the reflecting layer. By using four or more swept reflectometers at different poloidal positions in Tokamak like SINP/ADITYA, the plasma position can be measured.

For any clarifications regarding the contents of this document, please contact ;

Professor P.I. John,
Meghnad Saha Chair in Plasma Science and Technology,
Institute For Plasma Research,
Bhat, Near Indira Bridge,
Gandhinagar 382 428, Gujarat

Tel : 079-23969001 to 015 , Ext. 255 Mobile : 98253 72431
Fax : 079 - 23969016
E-mail : john@ipr.res.in

Annexure – I : Template for submission of research proposal under the National Fusion Programme

1. Title
2. Key words
3. Category (theory, experiment, modeling)
4. Principal Investigator (PI)
5. Collaborating personnel
6. Participating Institution
7. Recommendation from Head of Institute
8. List of Referees with affiliation
9. Objectives
10. Benefits to IPR/National programmes
11. Deliverables
12. Project Personnel to be recruited
13. Work Description
14. Major Work Elements and Identification of Responsibility
15. Critical Tasks and Back-up plan
16. Schedule of Work with Milestones
17. Facilities available
18. Salaries
19. Indigenous Equipment with list of suppliers and estimated price in INR
20. Imported equipment with list of suppliers and estimated price in INR
21. Fabricated items with list of fabricators and price in INR
22. Consumables list
23. Software list
24. Procurement restrictions if any
25. Budget (salaries, equipment, consumables, software, travel etc.)
26. Budget year wise break-up
27. CV of the PI with publication list
28. CV of collaborating personnel with publication list