Magnetic Fusion Energy Program of India

Abhijit Sen

Institute for Plasma Research,

Bhat, Gandhinagar 382428

5th INTERNATIONAL CONFERENCE ON THE FRONTIERS OF PLASMA PHYSICS AND TECHNOLOGY
April 21, 2011, Singapore

Outline

- Early history – the underpinnings in Basic Studies
- Tokamak program – Aditya and SST-1
  - Some highlights
  - present status
- Technology program
- ***ITER-India***
- Future roadmap - DEMO

- Beginnings:

  Plasma Physics Program nucleated at the Physical Research Laboratory, Ahmedabad in 1982 by Dept. of Sc. & Tech. (DST)

- Became an Independent and autonomous grant-in-aid Institute of DST in 1986 and moved to the present campus

- Transferred to Dept. of Atomic Energy in 1996 under an MOU between the two departments

Institute for Plasma Research

Institute for Plasma Research

- 1986: Set up by the Department of Science & Tech
  - *R & D in Plasma Science and associated Technologies with special emphasis on thermonuclear fusion*
  - ADITYA tokamak
- 1996: Became a DAE funded institute
  - SST-1 tokamak
- 2006: Domestic Agency for ITER
- 2007: Approvals obtained for
  - Test Blanket Module Activity
  - Prototype Activity
  - Starting the National Fusion Program

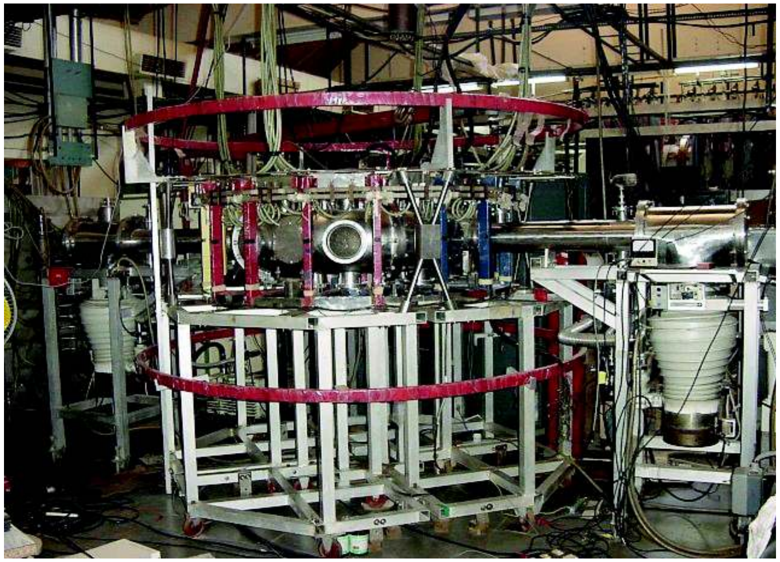
Programs at IPR

- Fundamental Plasma sciences
- Industrial applications
- Fusion Technology Development
- ITER Collaboration

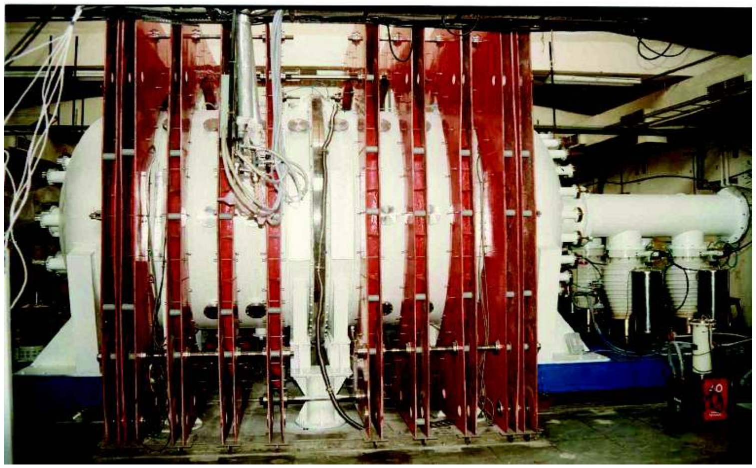
Fundamental Plasma Sciences

Theoretical and experimental studies

- Theory & Simulation : laser-plasma interaction, nonlinear dynamics, tokamak physics
- Experiments : BETA, Non-neutral plasma, LVPD
- Developing human resources for the Indian fusion program



BETA



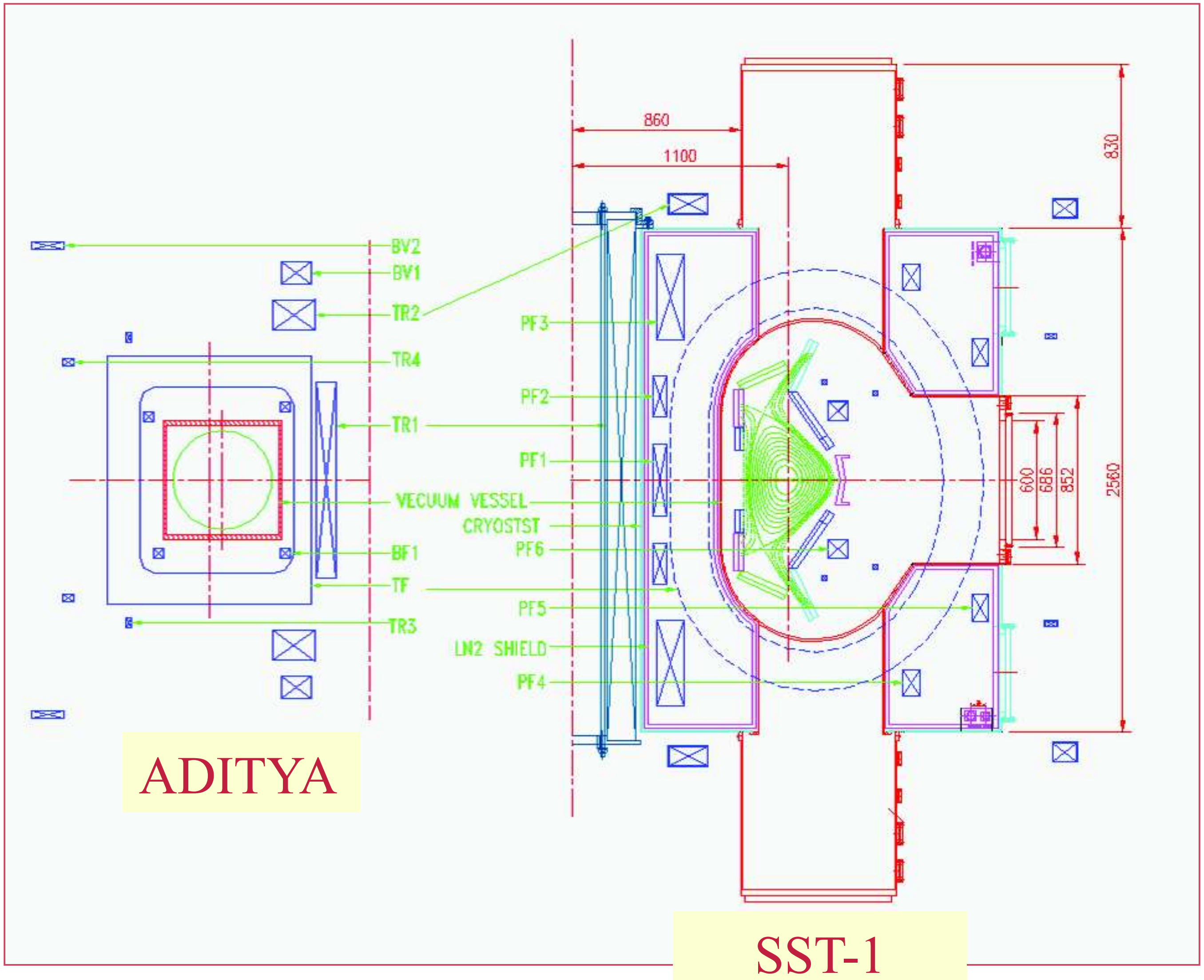
LVPD

Fusion Research

Focused on DEMO and beyond

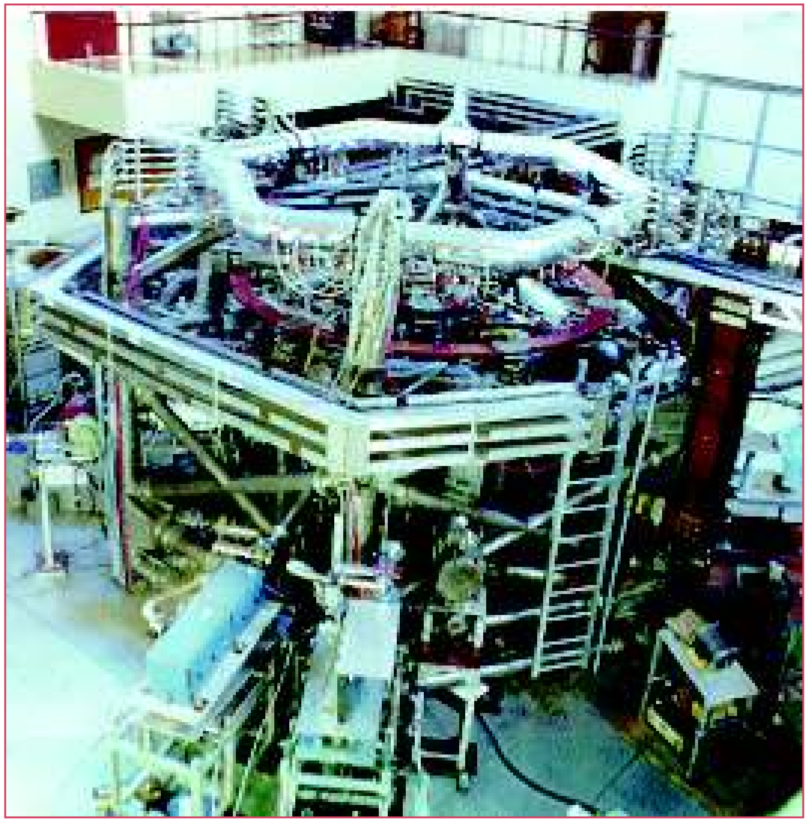
- Tokamak devices for resolving issues expected in future reactors
- Various plasma diagnostics and plasma control for operation
- Enabling technologies needed for the steady state operation of tokamak
- fusion blanket development
- Human resource development

Existing tokamak devices ADITYA and SST-1



ADITYA TOKAMAK

- First Tokamak designed by IPR and fabricated in India.
- Commissioned in September 1989



Achievements

- Physics :
  - Plasma edge-sol experimental study led to the discovery of plasma phenomena “intermittency” and later to observe, blob transport, etc.
- Realized technologies:
  - Large volume UHV toroidal vessel,
  - large copper magnets,
  - various plasma diagnostics, position and current control systems
- Technologies for next device:
  - RF heating and current drive systems developed and tested in this machine

ADITYA EXPTL. RESULTS

- In tokamaks edge and core plasma are integrated and phenomena in edge influence core transport profoundly . ADITYA dedicated to edge turbulence studies
- Discovered intermittency in tokamak edge turbulence (Phys Rev Letters 69 , 1375 1992)

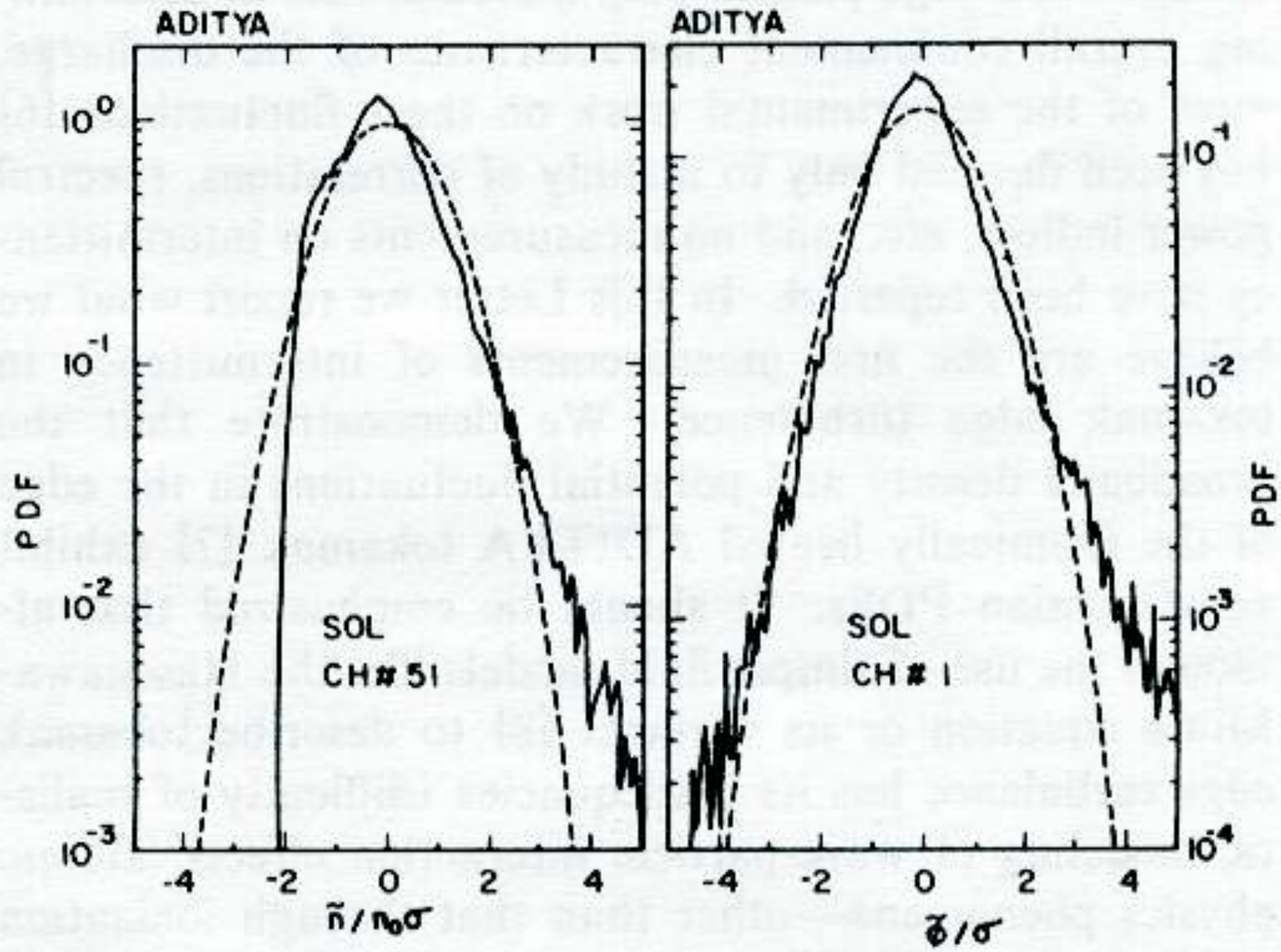
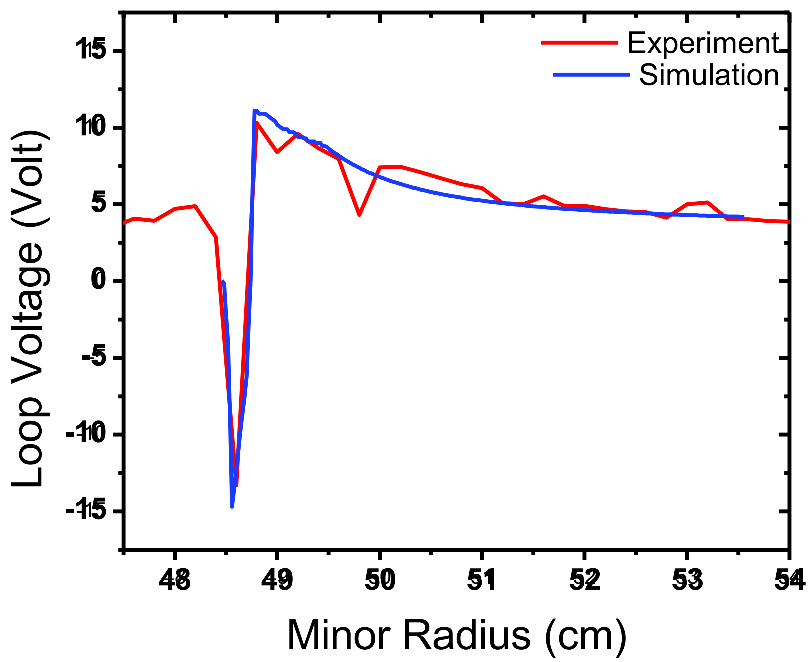


FIG. 3. The PDFs of $\tilde{n}$ and $\tilde{\phi}$ in the SOL plasma as a function of fluctuation amplitude normalized to the respective standard deviation (σ). The dashed curves represent Gaussians with the same σ.

Recent experiments in ADITYA

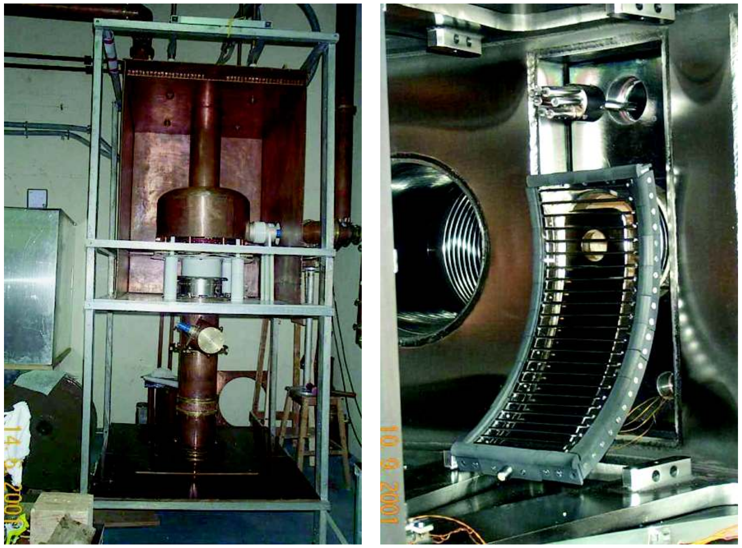
- Fluctuation suppression using gas puff [PPCF 51 (2009) 095010]
- Discharges with –ve spikes followed by +ve spikes [Phys. Plasmas (2010)]



ADITYA TECHNOLOGIES

- Large Volume UHV Toroidal Vessel
- Large Copper Magnets
- 50 Kiloamp shaped power supplies
- Novel Diagnostics
- Data Acquisition and Control Systems
- IPR and industries like BHEL, L & T , HBB, NGEF, Laxmi Vijay ..... Worked together

ADITYA RF Systems



A 20-40 MHz 200 kW ICRH system for heating

A 28 GHz 200kW Gyrotron based ECRH system

ADITYA AS A WORKHORSE

- ICRF Heating Experiment   200 Kwatts
- LHCD Experiment
- ECRH Preionization Experiment 80 KW
- Diagnostic Developments :
  MW Interferometer , Laser Thomson Scattering, Soft X-Ray imaging, ECE Radiometer , Probes, Visible and UV Spectroscopy , Charge Exchange , ...

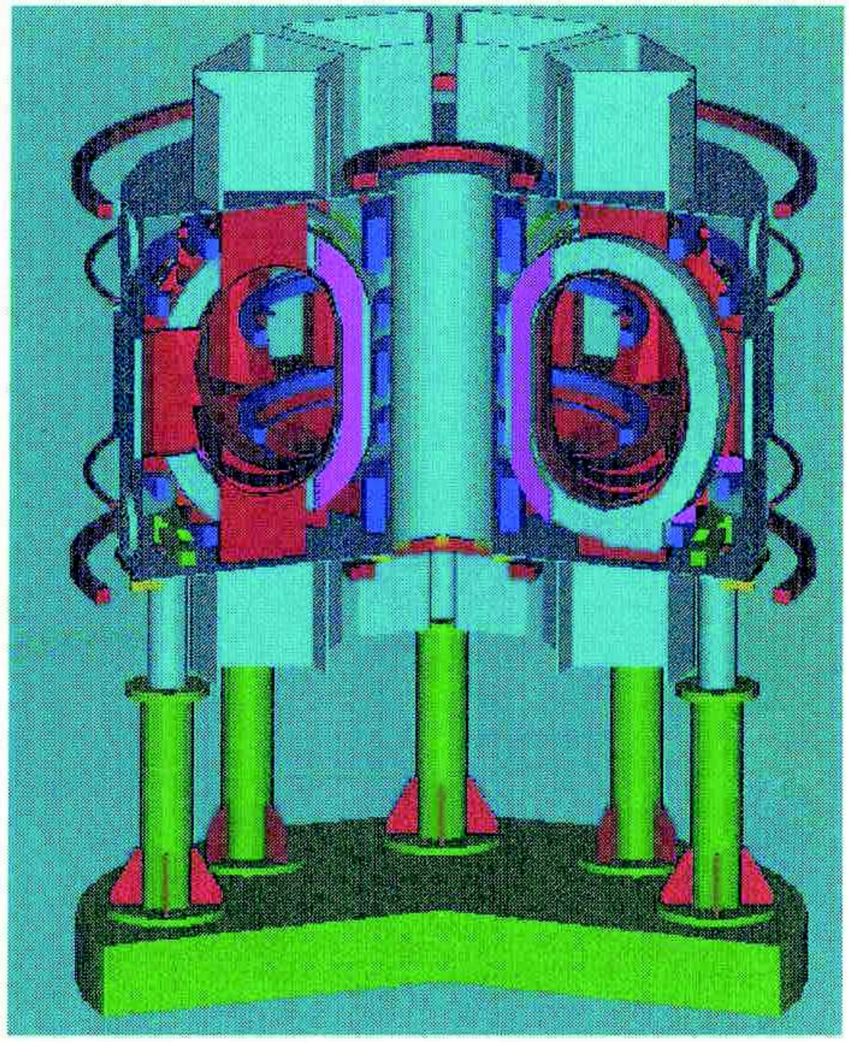
SST 1

- State-of-the-art Superconducting Steady State Tokamak with shaped plasma & divertors
- Designed for a 1000 second pulse
- In the same class as EAST, KSTAR , JT-60 SU

SST-1 TOKAMAK

PARAMETERS

- Major Radius : 1.1 m
- Minor Radius : 0.2 M
- Elongation : 1.7-2.0
- Triangularity : 0.4-0.7
- Toroidal Field at R_0 : 3T
- Plasma Current : 220 kA.
- Plasma Specie : Hydrogen
- Pulse Length : 1000S
- Configuration : Double Null, Poloidal Divertor
- Current Drive & Heating:
  - Lower Hybrid : 1.0 MW
  - Neutral Beam : 0.8 MW
  - ICRH : 1.0 MW
  - Total Input Power : 1.0 MW
- Fuelling : Gas Puffing



Expected parameters

- $n \sim 2 \times 10^{19}$ m^{-3} ($n_{Greenwald} \sim 1.8 \times 10^{20}$ @ 220kA)
- $T_e \sim 1.5$-3 keV
- $T_i \sim 1$ keV
- $\tau_E \sim 10$-20 ms
- $\tau_p \sim 30$-60 ms
- $\tau_{cd} \sim 1$-10 sec (current diffusion time)
- $\tau_{wall} \sim 100$-1000 sec

SST TECHNOLOGIES

- Superconducting Magnets
- High Power CW RF Technologies

  ICRF ~ 45 and 90 MHz 1.5 MWatts

  LHCD 3.7 GHz 1 MWatt

  ECRH 82 GHz 0.2 Mwatt
- Neutral Beams ~ 50 KeV 2.5 Mwatt
- High heat flux removal ~ 1MWatt/m sq

Technology achievements

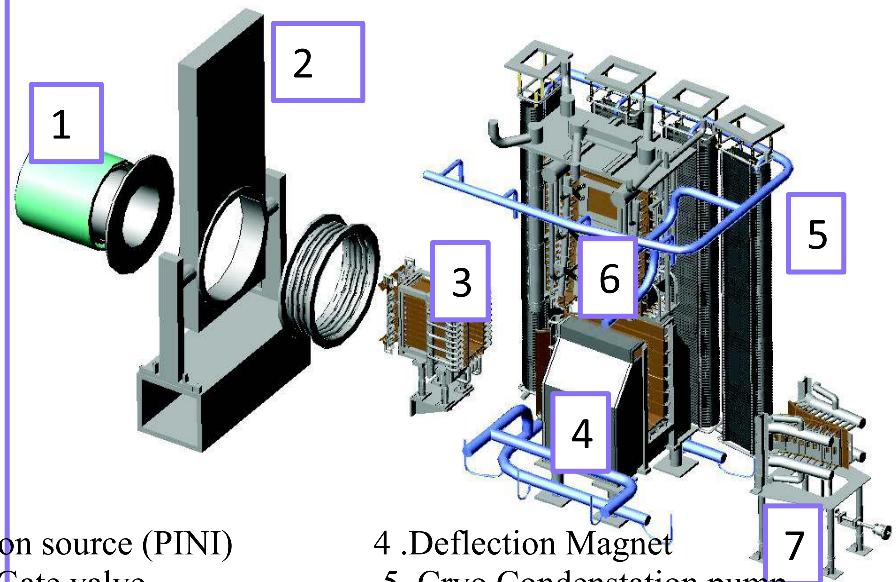
Developed :

- Large superconducting magnets, cryogenic system, thermal shields
- Auxiliary systems like NBI, ICRH, ECRH, LHCD etc.
- Diagnostics, Data acquisition and control systems for steady state operation of tokamak
- Plasma facing components like limiter (short duration), passive stabilizer and Divertor for steady state operation

To be achieved for future reactors: Operational experience, plasma control, coupling of heating and current drive system, particle and heat handling for steady state operation

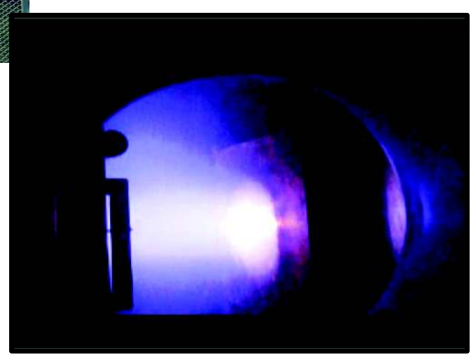
SST-1 NBI system

Beam Voltage (kV)	Extracted current (A)	Extracted Ion Beam Power (MW)	NB Power (MW)
30 (H)	35	1	0.5
55 (H)	90	5	1.7
80 (H)	60	4.8	1.5

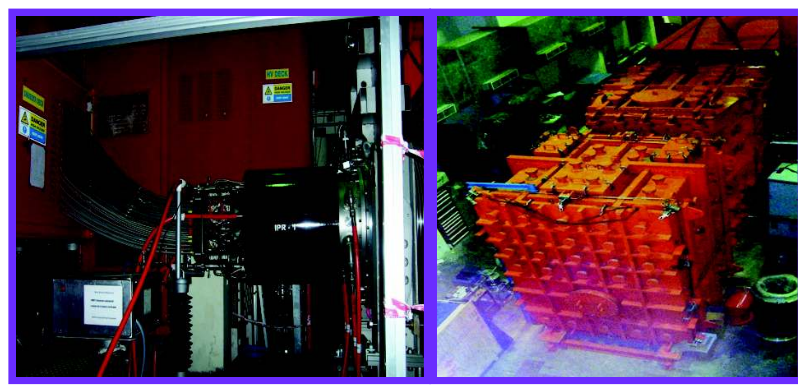


1. Ion source (PINI)
2 . Gate valve
3 .Neutralizer IInd stage
4 .Deflection Magnet
5 .Cryo Condenstation pump
6 .Ion Dump 7. Beam dump

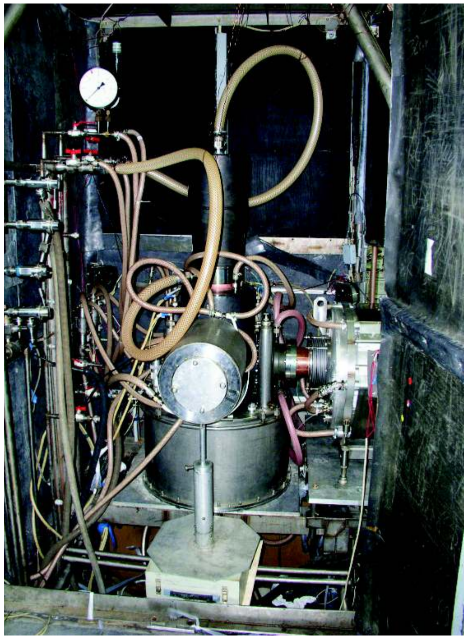
High voltage power supplies



100kW beam intercepted by a beam dump

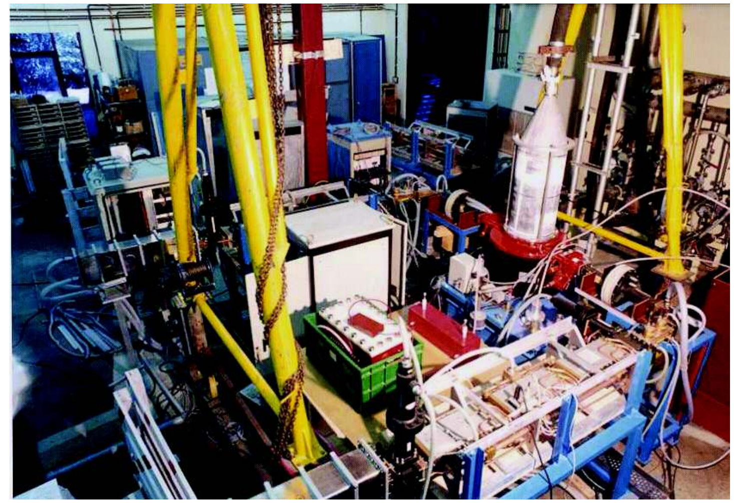


SST-1 RF system

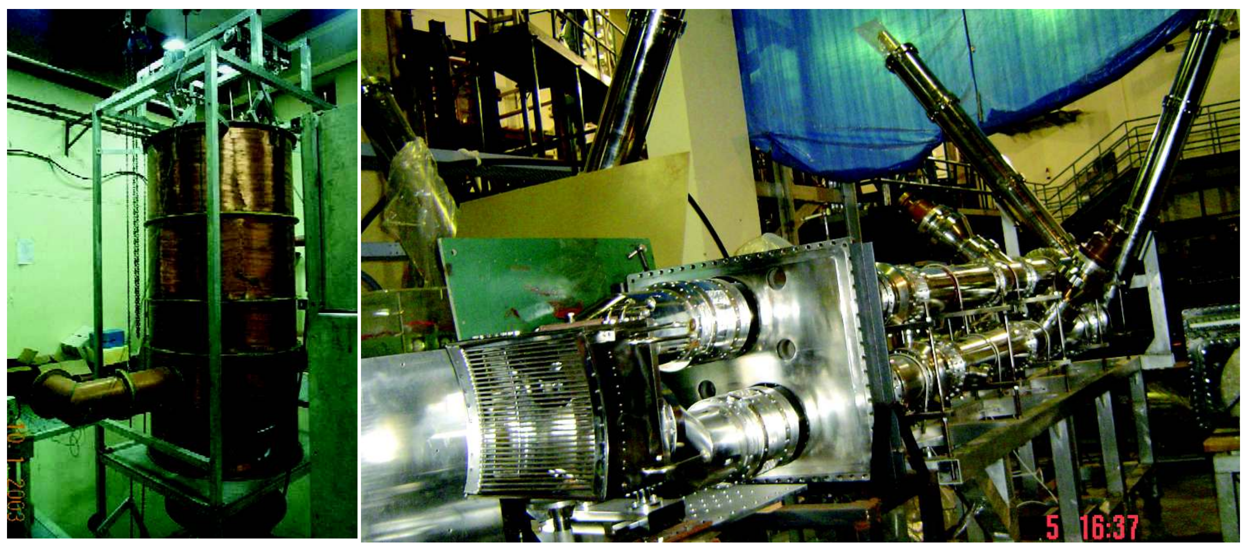


ECRH system 82.6 GHz

ECRH LFS Launcher



LHCD system 3.7 GHz



ICRH system 20-40,91.2 MHz

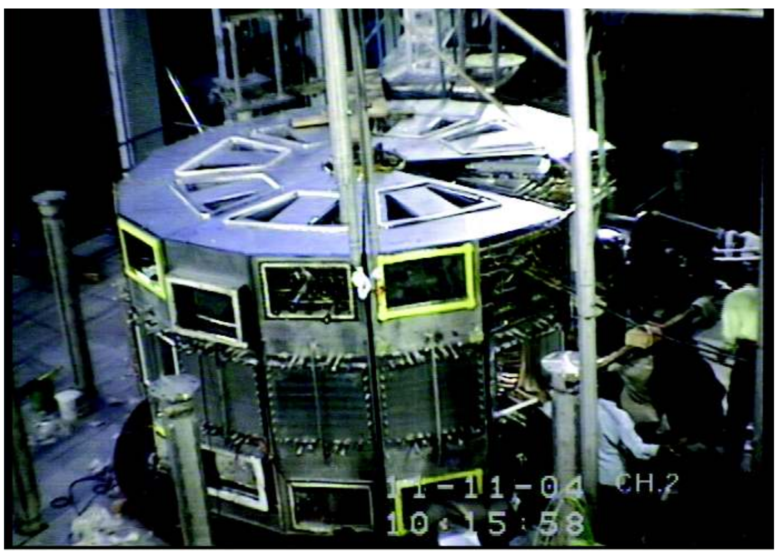
Summary of research program of SST-1

- SST-1 tokamak will explore a unique regime of long pulse operation with reactor-like first-wall geometry, time-integrated heat and particle fluence and similar q-profiles
- How to retain good confinement modes in steady-state will be a major focus
- The key issues in AT :maintaining a stable equilibrium by feedback stabilization of locked modes, RWMs, NTMs and control of ELMs for enhancing performance in steady state – will be researched upon for generating a database for future reactors

SST-1

To Study Physics of Plasma Processes under steady-state conditions

- Particle Control
- Heat removal
- Current maintenance



Progress:

- First assembled in 2004
- Integral testing continued up to 2007
- Leaks were identified and analyzed
- New techniques developed to resolve this and refurbishment was initiated
- Joints with sub nano-ohms developed, tested & implemented
- 5 K SHe cooled bubble type shields developed
- TF coils tested successfully
- Expected first plasma – **Dec. 2012**

TF Magnets are through !

Progress in SST-1 TF Magnet System

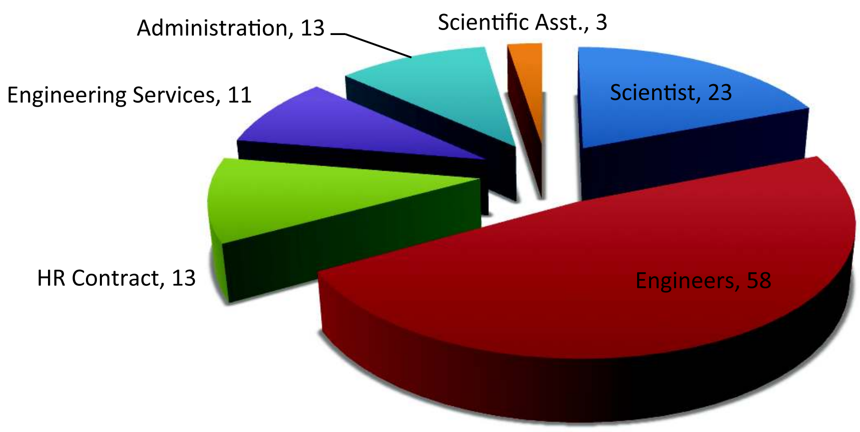
- SST-1 TF # 16 was fully & successfully tested on Jan 24, 2011 in supercritical helium flow conditions.
- The inter-pancake DC joint resistances were measured to be in the range of 02-06 nano-ohms.
- With these tests been done, SST-1 Mission announced (on Jan 24, 11) **the successful completion of testing all the SST-1 superconducting Toroidal Magnets.**

SST-1 Newsletter, February 2011

ITER-India

General

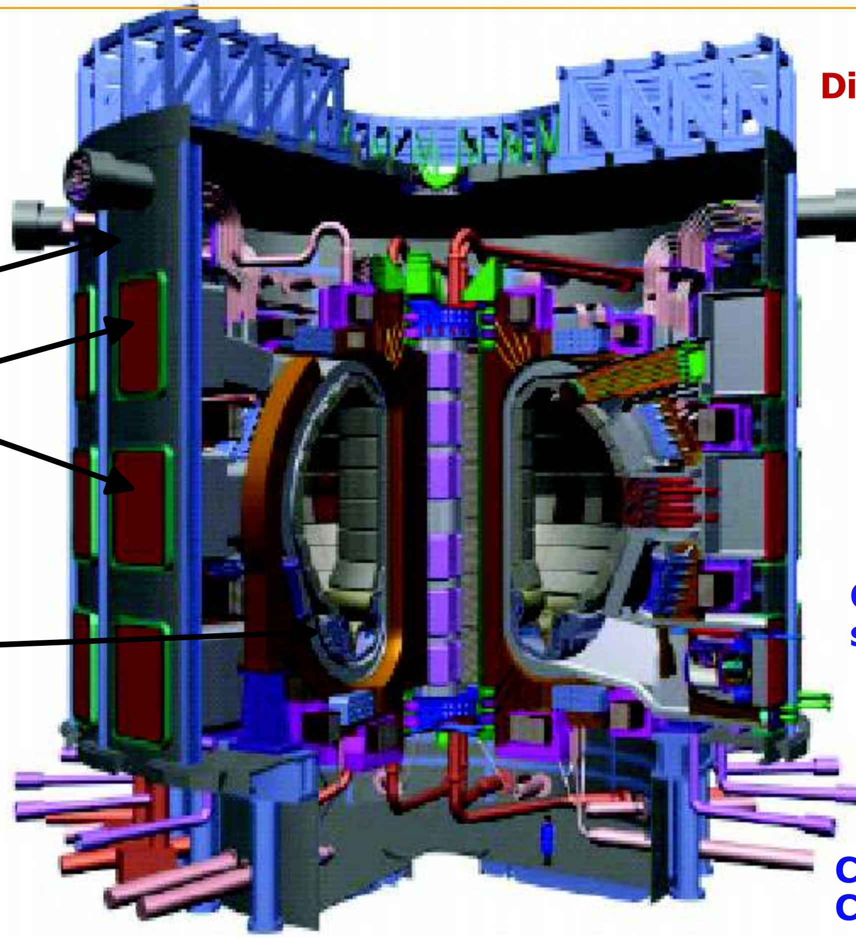
- India formally joined the ITER Project in the year 2005.
- ITER-India is an empowered group created in Institute for Plasma Research (IPR) to deliver the Indian commitments to the ITER project.
- IPR is committed to the long term fusion reactor programme.



ITER-India Personnel Distribution

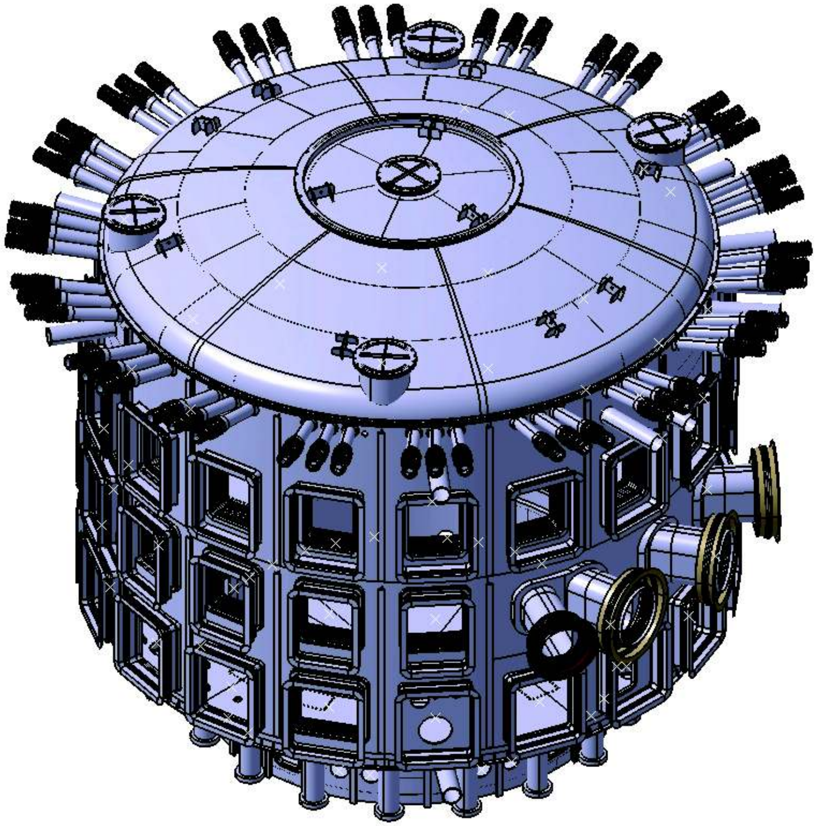
Procurement Packages

FS: Functional Specification; BTP: Built to Print



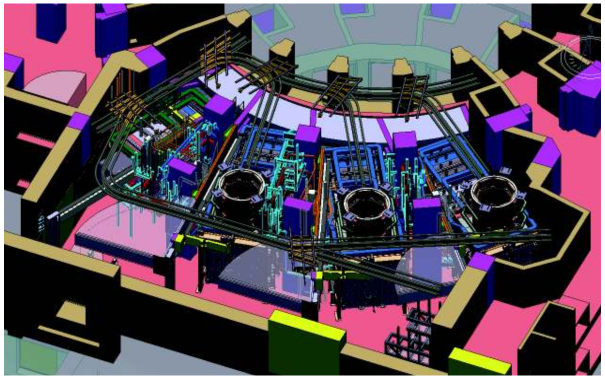
CRYOSTAT & VVPSS

- Reinforced, single walled vacuum vessel with dome-shaped top and flat-bottom head
- Overall Dia. ~ height ~29 m, about 400 penetrations, fully welded 50 mm thick SS304
- Provides a vacuum environment to avoid excessive thermal loads to the cryogenically cooled components
- Transfers all loads to the floor through pedestal columns
- Provides a secondary confinement barrier
- Passive removal of decay heat by gas conduction and convection

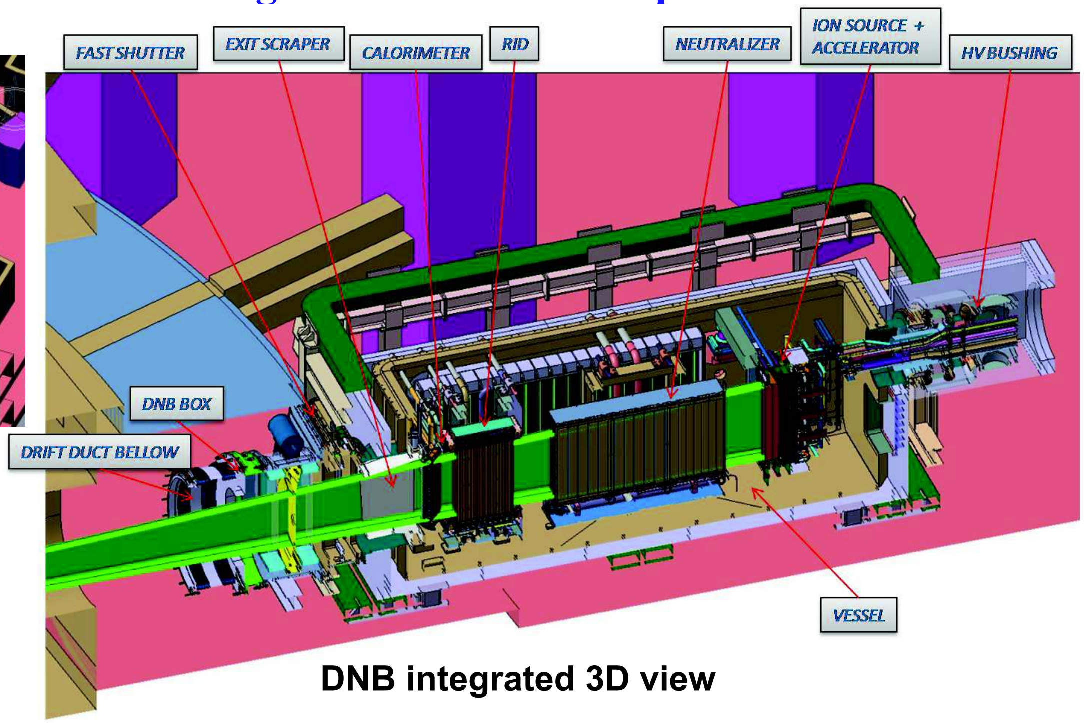


DIAGNOSTIC NEUTRAL BEAM INJECTOR

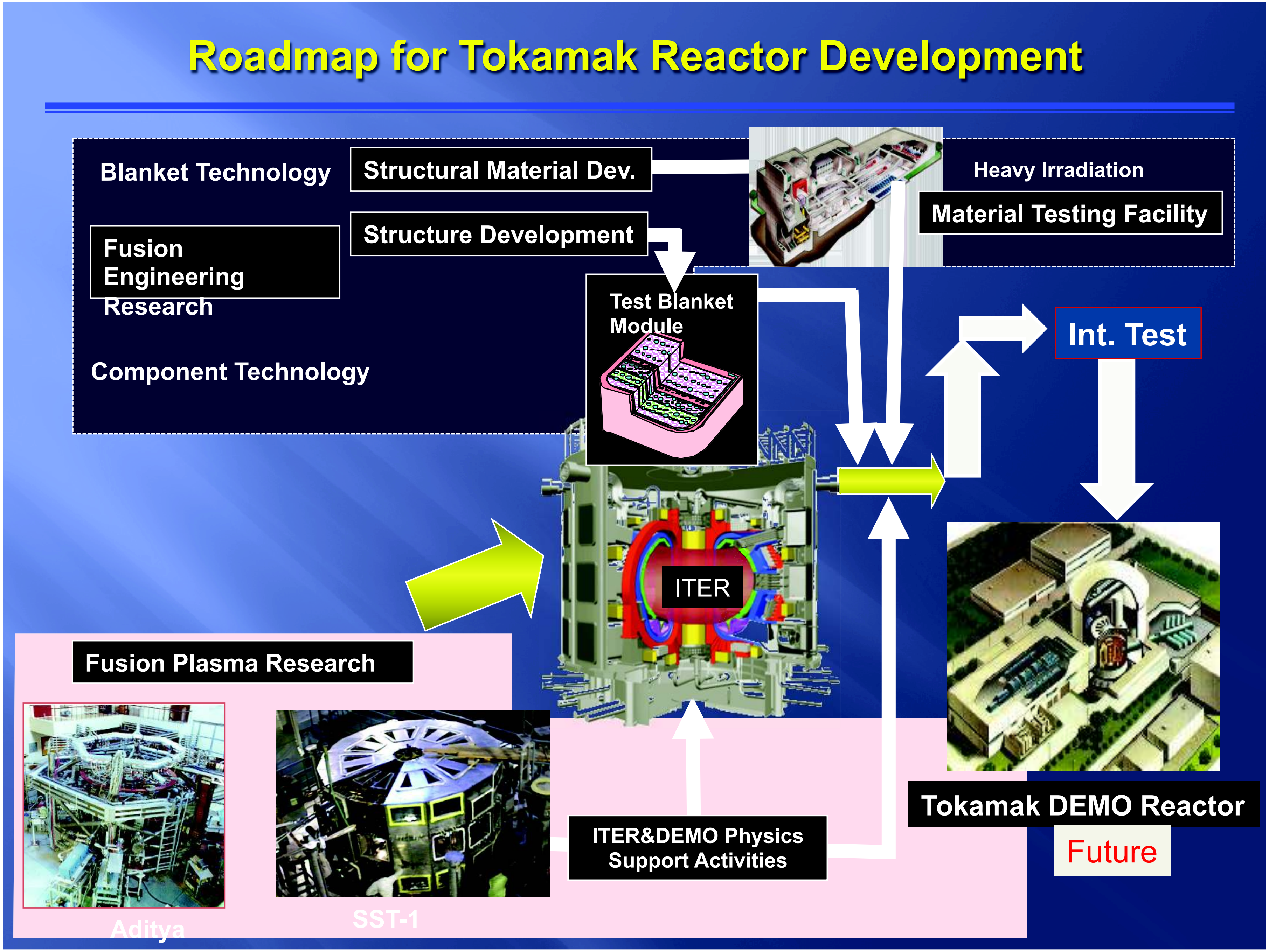
- Only diagnostics for Helium ash measurement. Supports CXRS, BES Diagnostics
- Package Type BTP with some FS & DD components
- Negative H beam, ~60 Amp accelerated & ~20 Amp neutral current injected to ITER at 100 keV. Development of Beam Source & transport involves R&D
- Configuration similar to HNB enabling same maintenance protocol



NB Cell



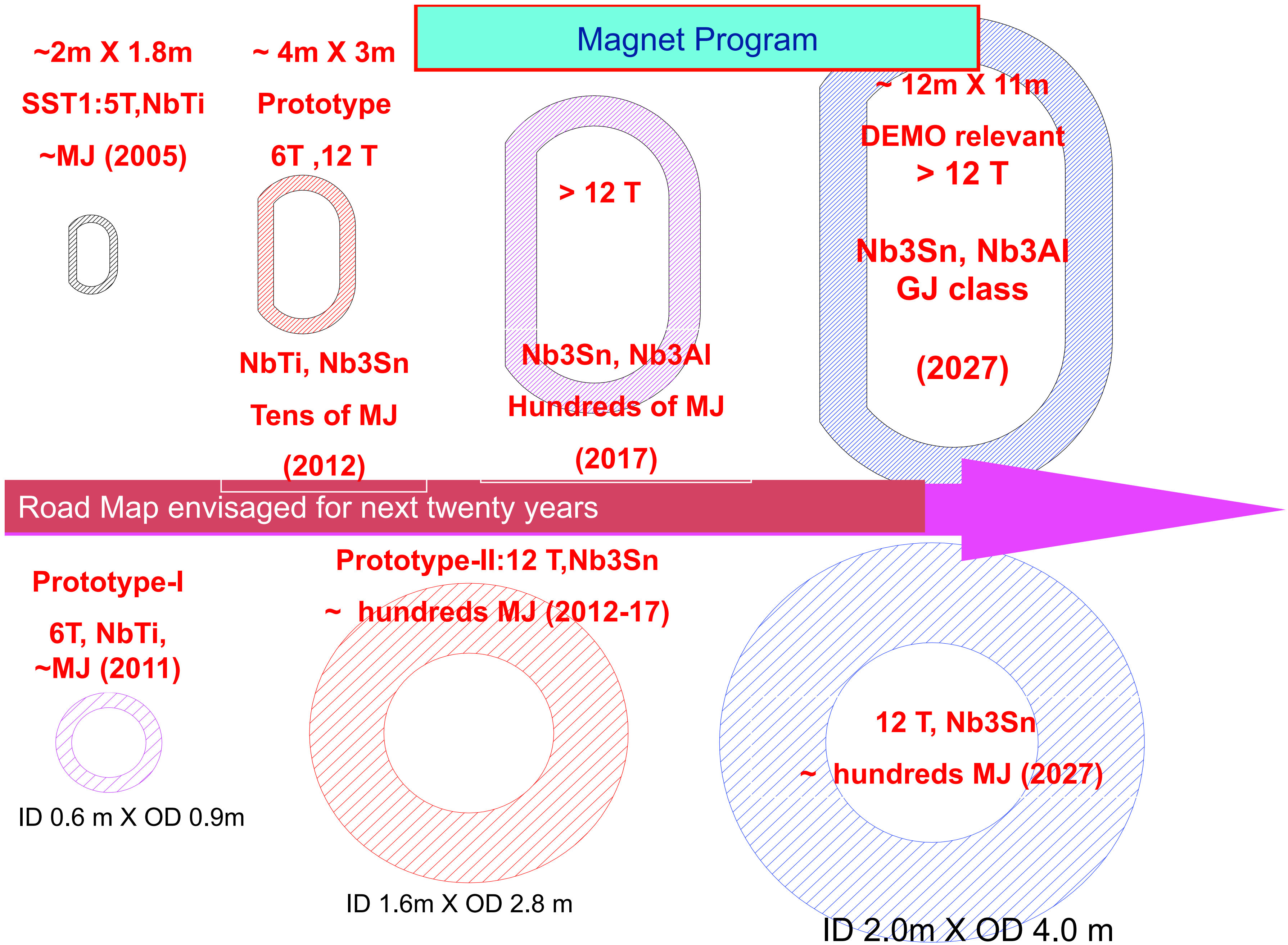
DNB integrated 3D view



Roadmap for Tokamak Reactor Development
Blanket Technology
Structural Material Dev.
Heavy Irradiation
Material Testing Facility
Fusion Engineering Research
Structure Development
Test Blanket Module
Int. Test
Component Technology
ITER
Fusion Plasma Research
Tokamak DEMO Reactor
Future
ITER&DEMO Physics Support Activities
Aditya
SST-1

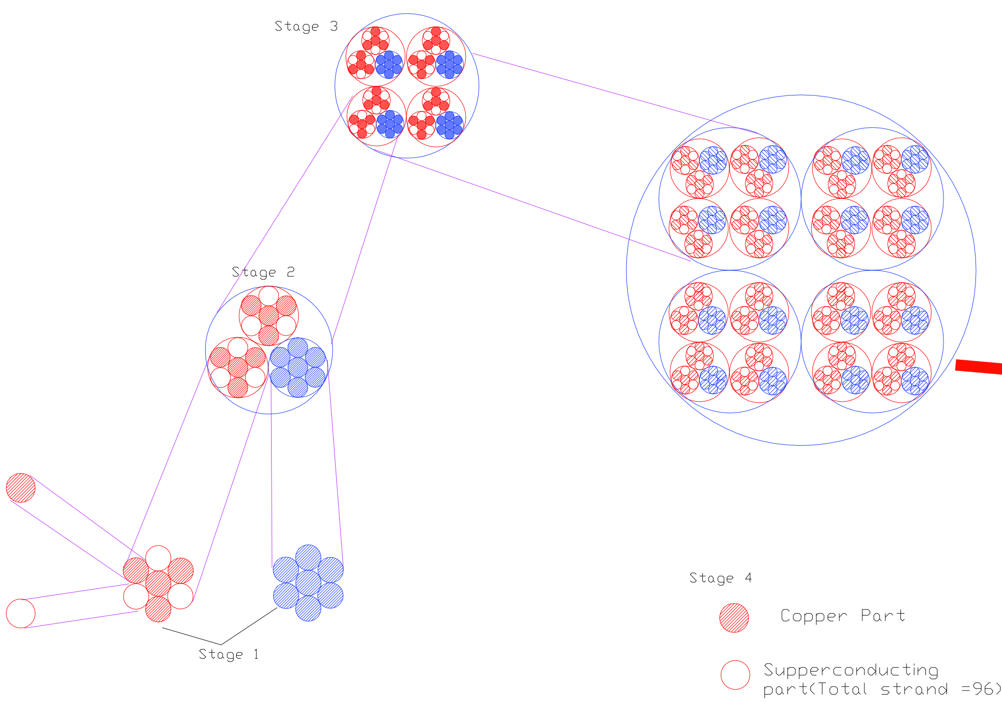
Fusion Technology development Program

- Design of DEMO reactor
- Integrated physics modeling
- Magnets development program
- Developing Divertor and its related technologies
- VV, Cryopump and fueling technologies
- Materials development program & Test facilities
- NNBI system
- RF system
- Blanket system

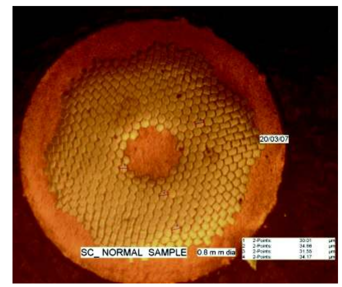

Magnet Program
~2m X 1.8m
SST1:5T,NbTi
~MJ (2005)
~ 4m X 3m
Prototype
6T ,12 T
NbTi, Nb3Sn
Tens of MJ
(2012)
> 12 T
Nb3Sn, Nb3Al
Hundreds of MJ
(2017)
~ 12m X 11m
DEMO relevant
> 12 T
Nb3Sn, Nb3Al
GJ class
(2027)
Road Map envisaged for next twenty years
Prototype-I
6T, NbTi,
~MJ (2011)
ID 0.6 m X OD 0.9m
Prototype-II:12 T,Nb3Sn
~ hundreds MJ (2012-17)
ID 1.6m X OD 2.8 m
12 T, Nb3Sn
~ hundreds MJ (2027)
ID 2.0m X OD 4.0 m

Prototype Magnet

IPR & AFD(BARC) have launched a joint initiative towards realizing fusion grade superconductors (NbTi) & Nb3Sn & Cable-in-conduit-conductors (CICC) since Sep 2006. Several critical technologies & mile stones have been achieved since then (validated experimentally)



Fusion grade CICC designed (NbTi)



Strand X-section (NbTi, 0.8 mm dia)

Strand (NbTi)

Cabled strand NbTi

CICC (NbTi) 30 kA, hybrid

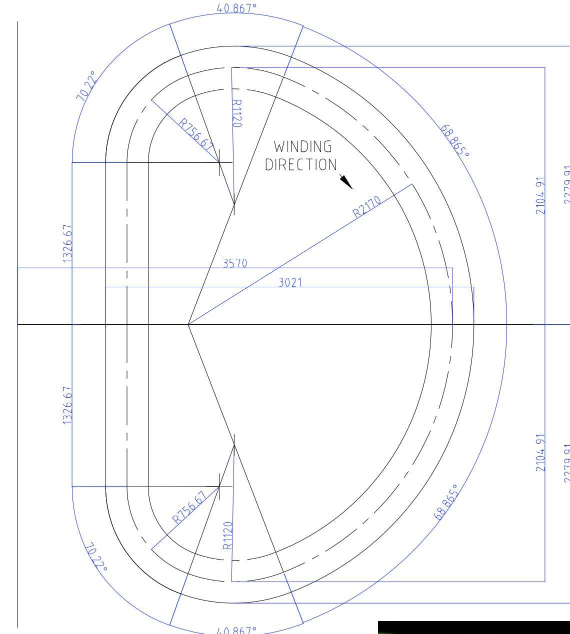
Fusion grade Nb3Sn strand and CICC realization have also been initiated through `Internal Tin' route for possible PROTOTYPE magnets

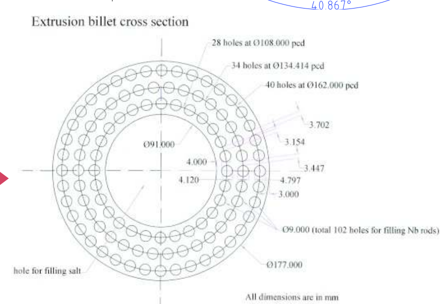
Prototype magnet has been designed

Nb3Sn cables has been designed

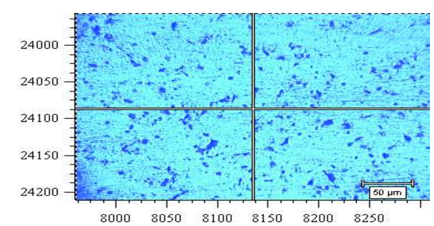
Nb3Sn strands Trails have begun

Hi Ho Nb rods, OFE Cu are being developed





Billet design & billet



Micrograph of pure Nb sample developed

OFE Cu billets

Divertor program

- Divertor should take heat load of 15-20 MW/m^2 for DEMO
- Materials to be developed are W & W-alloys, CFC, Be
- Remote handling tools to be developed to inspect, handle and repair the divertor cassette
- Developing fabrication technology and test facilities to qualify the divertor components
- Develop advanced materials for DEMO and develop innovative concepts to handle high heat flux

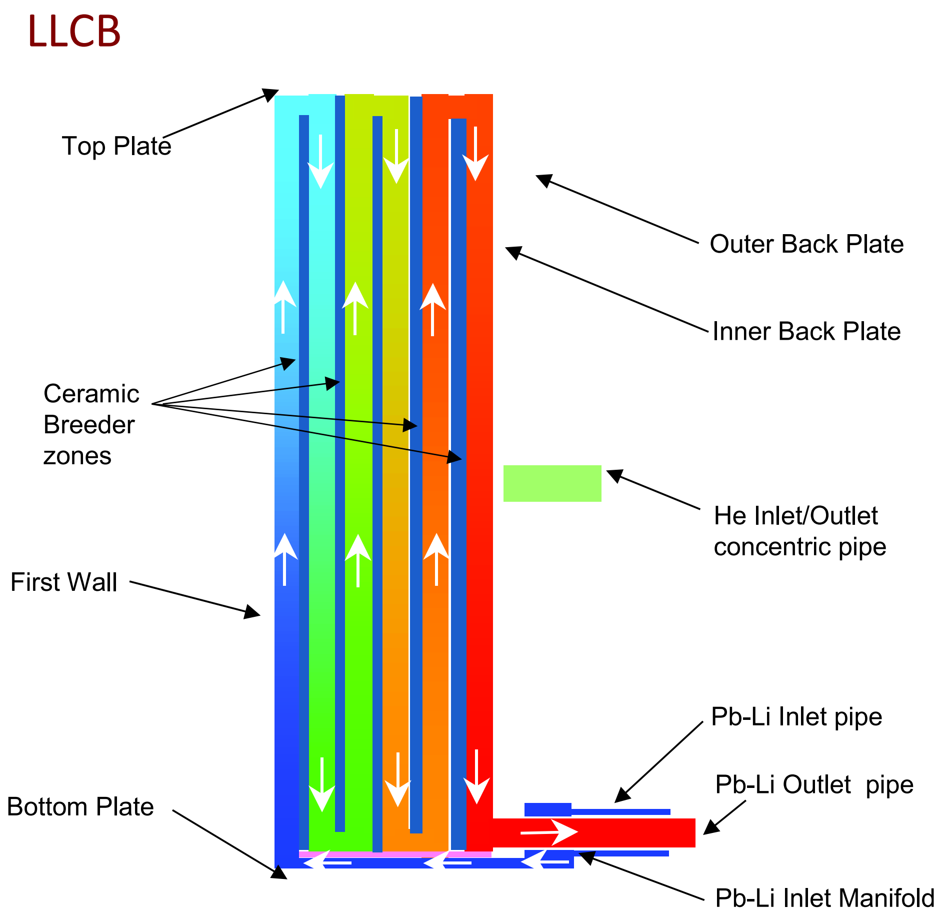
TBM program

- Indian TBM team is developing both Liquid and Solid Breeder blanket concepts:

  - **Liquid Breeder type:** Lead-Lithium cooled Ceramic Breeder **(LLCB)** **(To be tested in one half of ITER port no-2)**

  - **Solid Breeder type**: Helium Cooled Solid Breeder **(HCSB)** **(Sub-Module type testing in collaboration with other ITER partners)**

LLCB TBM Parameters



1.66 m (h) x 0.484 m (w) x 0.54 m (t)

Structural material	IN-RAFMS
Breeder	Pb-17Li, Li_2TiO_3
Neutron reflector / shield	SS 316 LN IG
MHD insulation	Al_2O_3 or Other choice
Primary coolant	Helium and Pb-Li
He inlet/outlet	350 / 480 oC
Helium pressure	8 MPa
He pressure drop in module	0.3 MPa
Pb-Li inlet/outlet	350/480 oC
Li-6 enrichment	90 % in Pb-Li

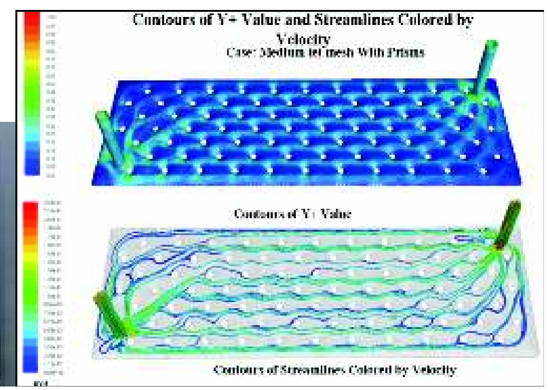
Helium as Purge gas for Tritium

Other systems

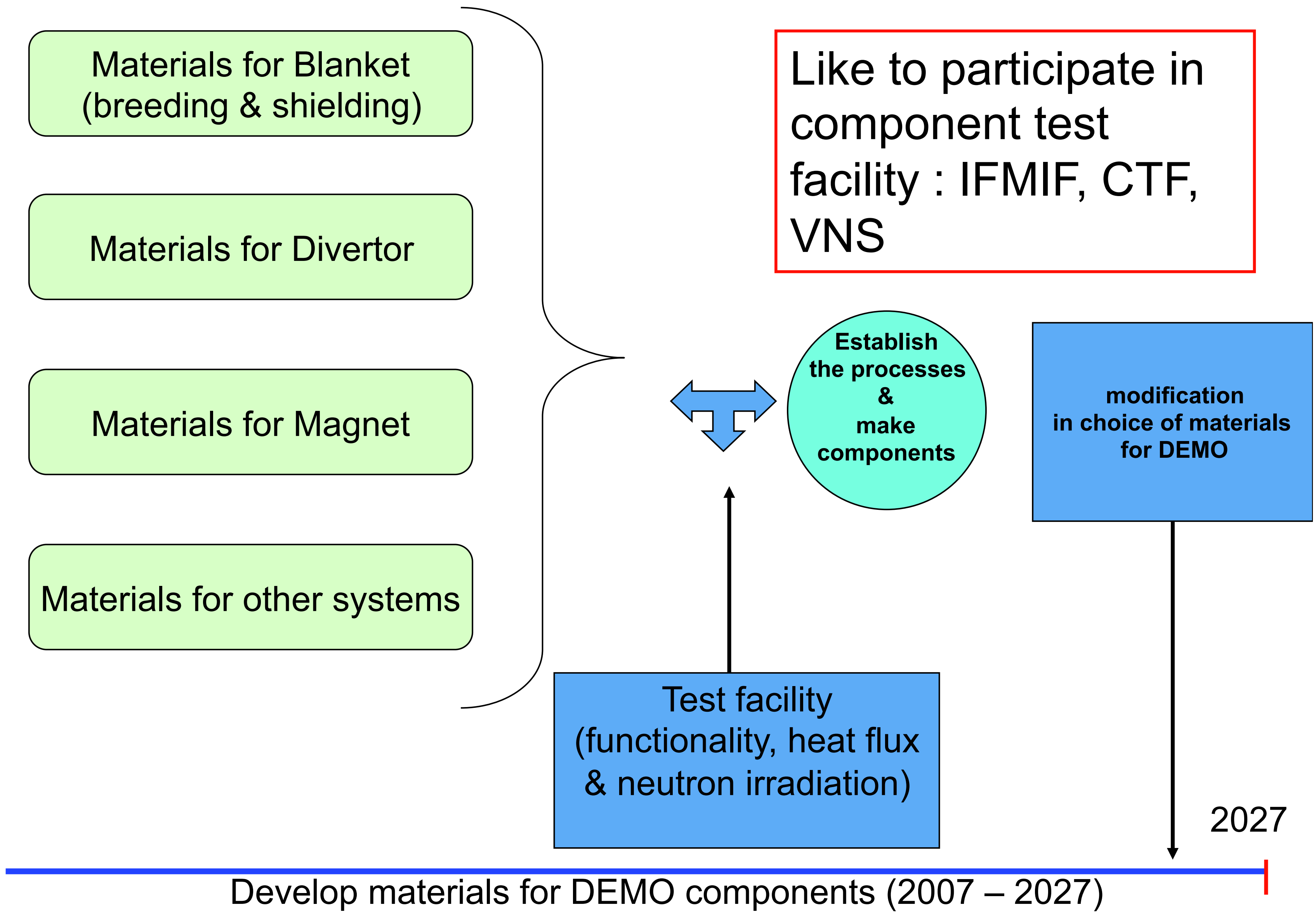
- Fueling system
  - Pellet injector system to fuel the high density reactor grade plasmas
  - Inject pellets at a faster rate to have continuous fueling
- Vacuum vessel & Cryo pump
  - Double walled Vacuum Vessel
  - Pumping requirement of DEMO will be of 3 x 10^5 liters/s (Need of cryopump with 10^5 liters/s)

Pellet injector under commissioning

Single and double embossed Cryo-panel for cryo-pump developed and tested

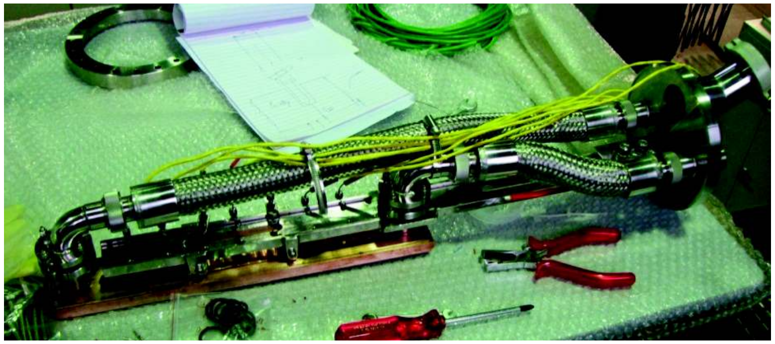


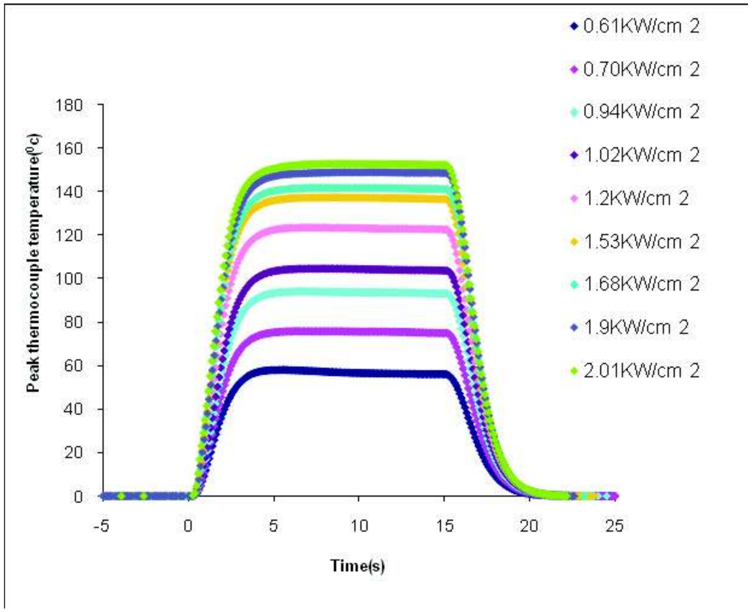
Materials program



NBI program

- Developed NBI system for SST-1 (1.7 MW)
  - Heat transfer systems (10 MW/ m^2)
  - Fast switching RHVPS (8 MW, 80 kV)
  - Cryocondensation pump (10^5 liters/s)
  - Technologies related to ion source
- Providing DNBI for ITER (~4 MW)
- Developing NNBI for DEMO (70 MW)
  - Power supplies of 1MV with required current





Neutral Beams : Negative ion beams

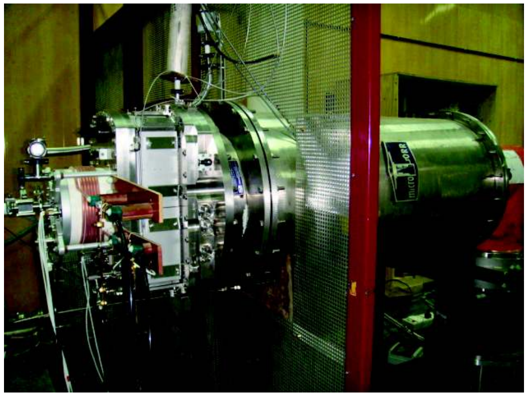
- **Experimental program of production of RF based Negative ion**

Objectives:

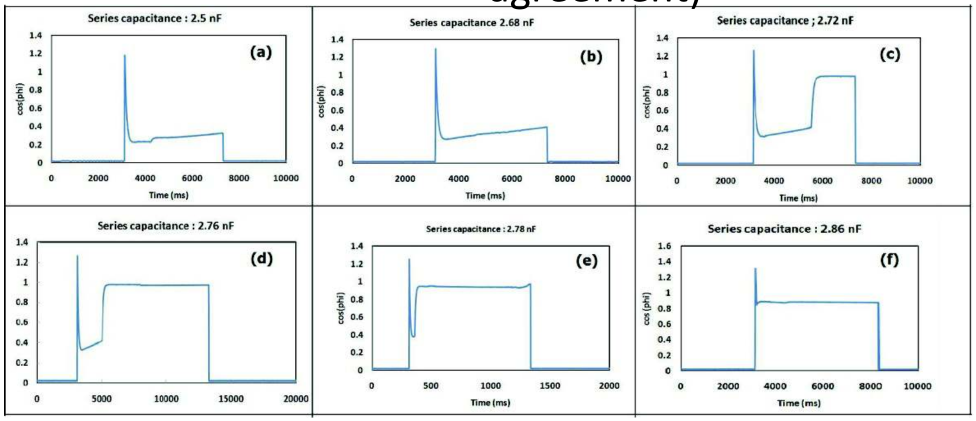
- **To Learn coupling of RF power to produce plasma in the source; Characterization of plasma**
- **Study various filter field configurations for optimal solution**
- **Beam extraction , acceleration & characterization**

View of RF matching circuit

RF based Source



Integrated source in operation (source under IPP agreement)



- **Present Status:**
  - –Lab established
  - –CODAC developed inhouse
  - –Plasma produced by coupling power from 100 kW, 1 MHz RF generator

RF system

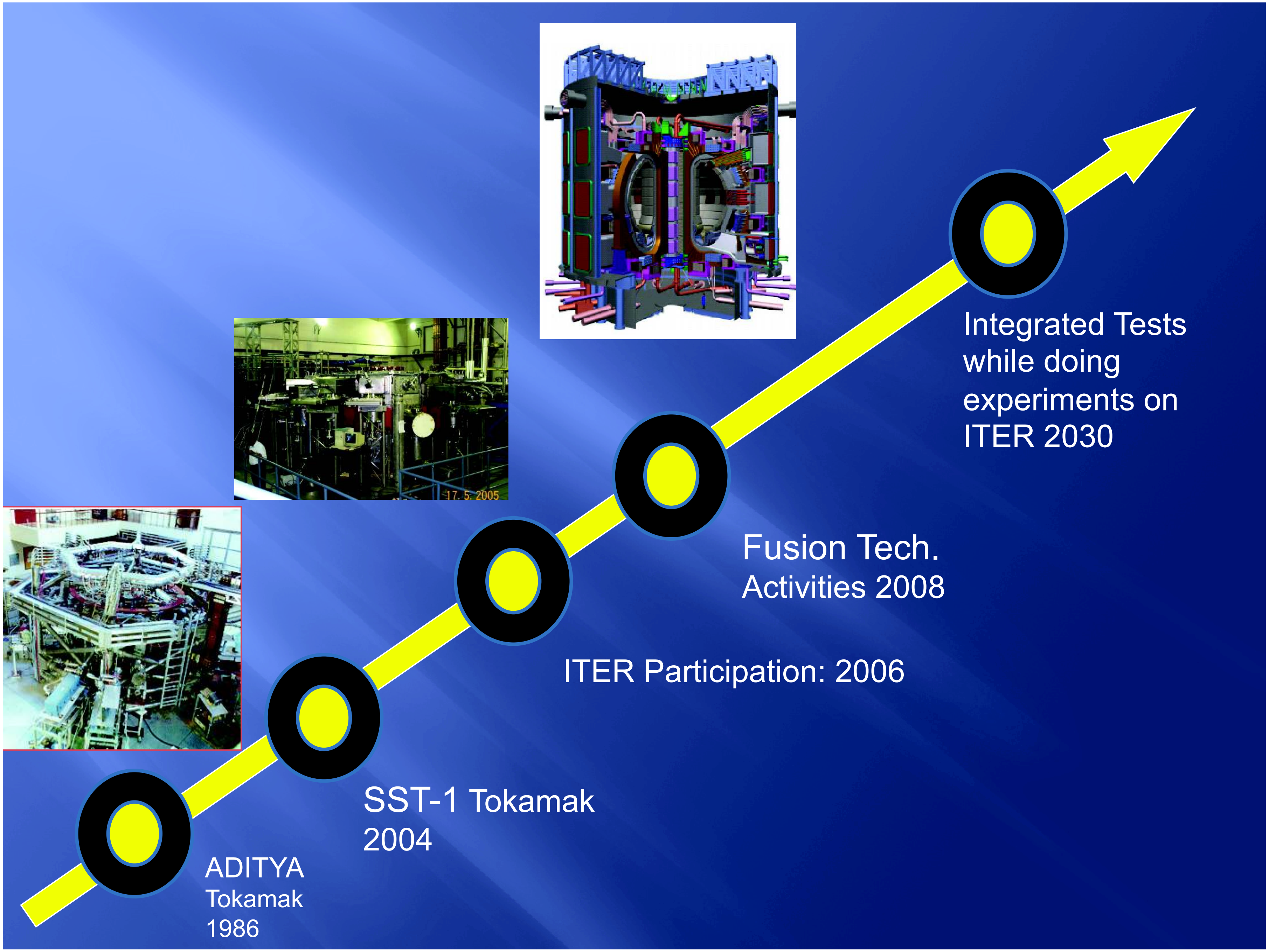
Activities by RF group

ITER ICRH
20 MW

	SST-1	DEMO
ICRH	20-40 & 91.2 MHz, 1.5/3 MW	35 MW
ECRH	82.6/140 GHz, 0.2/0.5 MW	20 MW
LHCD	3.7 GHz, 1/2 MW	20 MW

Human Resource Development Program

- Initiated a program to bring various labs, universities and industries to participate in the R&D program of fusion reactor
- Providing engineering services to many ITER tasks and available for our own program
- Such activities will nucleate various working groups required for the fusion reactor
- Future human resources for fusion will be developed through this program


17. 5. 2005
Integrated Tests while doing experiments on ITER 2030
Fusion Tech. Activities 2008
ITER Participation: 2006
SST-1 Tokamak 2004
ADITYA Tokamak 1986