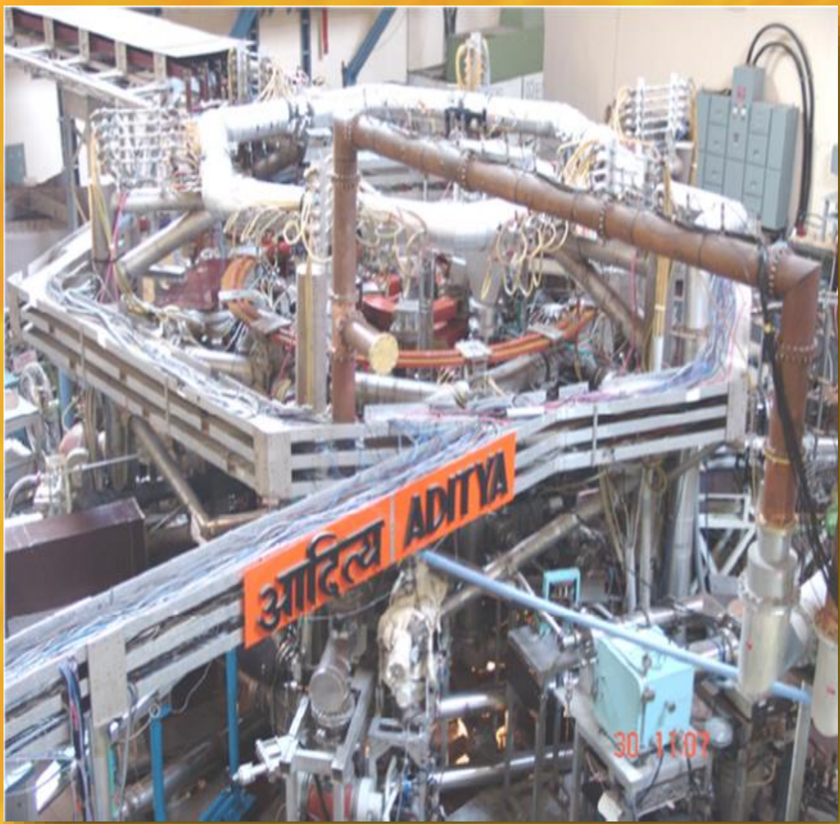
National Symposium for Commemorating 30-years of ADITYA Tokamak

27th and 28th January 2020



Book of Abstracts

Organised by:

Institute for Plasma Research (IPR)
Bhat, Gandhinagar, Gujarat
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Venue
Entrepreneurship Development Institute (EDI) of India, Gandhinagar

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Abstract ID **Abstract Title List**

1. Design, Fabrication and Installation of Support Structure for Reflectometry Diagnostics on Aditya-U Tokamak
Pratibha Gupta, Manoj Kumar Gupta, Bharat Doshi, J.J.U.Buch, S.K.Pathak

2. Charge Exchange Neutral Particle Analyzer (CX-NPA) Diagnostic Measurements in Ohmic as well as Ion cyclotron Resonance Heated Plasma Discharges in Aditya
Snehlata Aggarwal, Santosh Pandya and Kumar Ajay

3. Comparative Study of Perpendicular and Tangential Viewing Soft X-Ray Tomographic Reconstruction for Aditya Tokamak MHD Equilibrium and Stability
Shishir Purohit and Manoj Kumar Gupta

4. Field Simulation of Ohmic Ramp Down in ADITYA – Need For Correction Coils for Improvement of Magnetic Null
Amardas Alli

5. ADITYA Vacuum Monitoring and Control System
Kiran Patel, K.A. Jadeja, Kaushal Patel, H.C. Joshi & Joydeep Ghosh

6. Whistlers: A Probable Mechanism to Mitigate Runaway Electrons in Tokamaks
A. K. Sanyasi, Prabhakar Srivastav, L. M. Awasthi, P. K. Srivastava, R. Sugandhi and D. Sharma

7. Effect of Superthermal Electrons on Ion-Acoustic Wave in Negative Ion Plasmas
J. K. Chawla

8. Effect of Superthermal Electrons and Positrons on Ion-Acoustic Double Layers in Magnetized Plasmas
P. C. Singhadiya and J. K. Chawla

9. Plasma Position Estimation Using Magnetic Diagnostics in ADITYA-U
S. Aich, R. Kumar, T. M. Macwan, D. Kumavat, S. Jha, R. L. Tanna, Sathyanarayana K., J. Ghosh, K. A. Jadeja, K. Patel, Sharvil Patel, Vaibhav Ranjan, Madanlal Kalal, Dinesh Varia, D. Sadharkiya, D. Raju, P. K. Chattopadhyay, C. N. Gupta , Y. C. Saxena and ADITYA-U team

10. Iron Impurity Behaviour Study in the ADITYA Tokamak
A. K. Srivastava, S. Patel, M. B. Chowdhuri, R. Manchanda, J. V. Raval, U. Nagora, P.K. Atrey, R. L. Tanna, J. Ghosh and ADITYA Team

11. 3D Simulation in Aditya Scrape-off Layer Using Steady State Model EMC3- EIRENE
Bibhu Prasad Sahoo, Devendra Sharma and Ratneshwar Jha

12. Tungsten Coating Deposition on Graphite and Process Optimization
P A Rayjada, K. A. Jadeja, N. L. Chauhan, S. B. Bhatt

13. Analytical Model for Erosion Driven Carbon Sources in Tokamaks
P. N. Maya and S. P. Deshpande

14. Infrared Imaging Video Bolometer Diagnostics in Aditya Tokamak
Santosh P. Pandya, Shwetang N. Pandya, Kumudni Tahiliani, S. K. Pathak and Aditya team

15. Investigation of Impurity Seeded ADITYA & ADITYA-U Tokamaks Plasmas
M. B. Chowdhuri, J. Ghosh, R. L. Tanna, K. A. Jadeja, K. M. Patel, R. Manchanda, N. Yadava, S. Patel, G. Shukla, K. Shah, R. Dey, N. Ramaiya, Tanmay Makwan, U. C. Nagora, S. K. Pathak, J. V. Raval, S. Purohit, M. K. Gupta, S. K. Jha, M. V. Gopalakrishna, K. Tahiliani, Rohit kumar, Suman Aich, Suman Dolui, Kaushlender Singh, P. K. Atrey, B. V. Nair, C. N. Gupta, P. K. Chattopadhyay, ADITYA Team and ADITYA-U Team

16. Investigation of Atomic and Molecular Processes in ADITYA and ADITYA-U Tokamak Plasmas
Ritu Dey, J. Ghosh, M. B. Chowdhuri, Ranjana Manchanda, Nandini Yadava, R.L.Tanna, Santanu Banerjee, P. Vasu, Vinay Kumar and Aditya Team

17. Population-Alignment Collisional-Radiative Model for Polarization in Lyman-α Line
Nilam Ramaiya and Motoshi Goto

18. Electronics for Langmuir Probe in Aditya-U for the Measurement in the SOL Layer.
Pramila, Tanmoy, Harshita, Lavkesh, Shwetang, Rachana Rajpal

19. Role of Poloidal Flows on the Particle Confinement in a Current-less Toroidal Device
Umesh Kumar, R. Ganesh, K. Sathyanarayana, Y. C. Saxena and D. Raju

20. Investigation of Neutral Recycling and Ion Temperature of Various Plasma Species in ADITYA and ADITYA-U Tokamak
Nandini Yadava, J. Ghosh, M. B. Chowdhuri, R. Manchanda, Sripathi Punchithaya K, Ismyil, Ritu Dey, Tanmay Macwan, S. Patel, N. Ramaiya, K. A. Jadeja, R. L. Tanna and Aditya-U

21. Study Small Amplitude Ion-Acoustic Solitons in Negative Ion Plasmas with Superthermal Electrons
P. C. Singhadiya

22. Topology Optimization of a Planetary Gearbox for Fusion RH Application
Ratna A. Rajgor, Manoah Stephen M, Krishan Kumar Gotewal, Hitesh .K. Patel

23. Passive Charge eXchange (PCX) Spectroscopy to Measure Plasma Rotation on Aditya-U Tokamak
G Shukla, K Shah, M.B.Chowdhuri, R Manchanda, R.L. Tanna, K.A. Jadeja, K.B.K. Mayya, J Ghosh and Aditya-U team

24. Characterization of Ohmic Breakdown Phase for the ADITYA discharges
Sharvil Patel, Joydeep Ghosh, R. L. Tanna, P.K. Chattopadhyay, Harshita Raj, M. B. Chowdhuri, Ranjana Manchanda, Nilam Ramaiya, Umesh Nagora, P. K. Atrey and Aditya Team

25. An Estimation of the Edge Impurity Transport in the ADITYA Tokamak through Comparison Between the Simulated and Analytical (Model–Based) Edge Impurity Diffusion Coefficients
Amrita Bhattacharya, Joydeep Ghosh, Malay Bikas Chowdhuri, Prabhat Munshi and the ADITYA team

26. ECRH Power Supply System for Aditya Tokamak
Mahesh Kushwah, B K Shukla, KG Parmar, Jatin Patel, Harshita Patel,Dharmesh Purohit and Hardik Mistry

27. Indigenously Developed Data Acquisition System for Plasma Diagnostics in Aditya-U Tokamak
Praveena kumari Shukla, Vismaysinh Raulji, Hitesh Mandaliya, Rachana Rajpal, Pramila, Praveenlal E.V, C.J Hansalia

28. Gas Puff Induced Drift Waves in ADITYA and ADITYA-U Tokamak
Tanmay Macwan, Harshita Raj, Kaushlender Singh, Suman Dolui, J. Ghosh, R. L. Tanna, Rohit Kumar , Suman Aich, Lavkesh Lachhvani, Pramila Gautam, K. A. Jadeja, K. M. Patel, N C Patel, V Panchal, Umesh Nagora, P. K. Atrey, S. K. Jha, D. Raju and ADITYA-U Team

29. Laser Heated Emissive Probes Diagnostic in ADITYA – U TOKAMAK
Abha Kanik, Arun Sarma, Joydeep Ghosh , Tanmay Macwan, R. L. Tanna, Minsha Shah, Ranjana Manchanda, Payal Pandit, Shwetang Pandya, Jayesh Raval, Umesh Nagora and ADITYA-U Team

30. ADITYA Upgrade New Circular Shaped Torus Vacuum Vessel and Pumping System
K. A. Jadeja , S. B. Bhatt, K. M. Patel, J. Ghosh, V. R. Prajapati, Kulav Rathod, K. S. Acharya, Kiran Patel, B. G. Arambhadiya, R. L. Tanna, M. B. Kalal, D.S. Varia, D.H. Sadhrakiya, P. K. Chattopadhyay, Y. C. Saxena, A.Das, D. Bora

31. Control System Development for LIGO outgassing setup
S.Sunil, Faiz Mohammed Masi, Sabbir Ahmed, Gaurav Kumar Singh and Subroto Mukherji

32. Water Cooling System of ADITYA Tokamak
Yagneshkumar Trivedi, M. Vasani, S. K. Sharma, K. Padia and J.M. Gandhi

33. The Refurbishment of Damaged Toroidal Magnetic Field Coils for ADITYA-U
D.H. Sadharakiya, R.L. Tanna, J. Ghosh, B.R. Doshi, P.K. Chattopadhyay, Sharvil Patel, Vaibhav Ranjan, Rohit Kumar, Harshita Raj, K. Sathyanarayana, M.B. Kalal, D.S. Varia, Ram Krushna Panchal, Kulav Rathod, S.B. Bhatt, A.Vardharajulu, Y. C. Saxena, and Shell-N-Tube Team

34. Study of Argon Line Emissions in ADITYA-U Tokamak using Spectroscopic Diagnostic
K. Shah1, M. B. Chowdhuri, G. Shukla1, R. Manchanda, K. A. Jadeja, N. Yadava, N. Ramaiya, K. M. Patel, R. L. Tanna, K. B. K. Mayya, J. Ghosh and ADITYA-U team

35. Real-time Horizontal Plasma Position Control in ADITYA –U
Rohit Kumar, Pramila Gautam, Shivam Gupta, Tanmay Macwan, Praveen Lal E.V., Minsha Shah, Ranjana Manchanda, M. B. Chowdhuri, Nandini Yadav, Kunal Shah, M. N. Makwana, V. Balakrishnan, C. N. Gupta, R. L. Tanna, Suman Aich, Devilal Kumawat, K. Sathyanarayana, S. Jha, D. Raju, Joydeep Ghosh, P. K. Chattopadhyay, Y. C. Saxena and the ADITYA –U Team.

36. Large Amplitude Ion Acoustic Solitons in Warm Negative Ion Plasmas with Maxwellian Electrons.
Kishan Kumar and M. K. Mishra

37. Preliminary Study of Supersonic Molecular Beam Injection in ADITYA-U Tokamak.
Kaushlender Singh, Suman Dolui, Tanmay Macwan, K A Jadeja, K M Patel, Harshita Raj, Suman Aich, Rohit Kumar, B Arambhadiya, Siju George, Y Pravastu, D C Raval, V K Panchal, Jayesh Raval, D. Raju, S.K. Jha, R L Tanna, J Ghosh and ADITYA-U Team

38. Experimental Investigation on Electron Temperature Gradient Driven Instability in the Curvature Magnetic Field of MPD
A. D.Patel, Meenakshee sharma, N.Ramasubramanian, Lavkesh Lachhvani, Y. C. Saxena, R. Ganesh, P. K. Chatopadhaya

39. Spectroscopic Diagnostic for Magnetic Field in Tokamak Plasma
Subir Biswas

40. Study of ADITYA-U Tokamak Plasma using Fast Imaging Camera
Devilal Kumawat, Kumudni Tahiliani, Sameer Kumar, Gopalakrishna M V, Santosh Pandya, S.K Pathak and ADITYA-U Team

41. Design and Development of Different Analog and Digital Electronic Circuits for ADITYA Tokamak
Minsha Shah, Praveenlal E.V., Hitesh Mandaliya, Vismay Raulji, C. J. Hansalia, Rachana Rajpal and Electronics Group, R. L. Tanna, J. Ghosh and Aditya team

42. Homodyne and Heterodyne Microwave Interferometer Systems for ADITYA Tokamak
Umesh Nagora, S.K.Pathak, P.K.Atrey and Aditya Team

43. Non-linear Time Series Analysis of ADITYA-U Plasma discharges
Balamurali Krishna, Mayya Kolake, Sharvil Patel, Tanmay Macwan, M.B. Choudhury, R. Manchanda, J. Ghosh, and Sameer Kumar

44. Investigations of Plasma Disruption Prediction in Tokamak using Machine Learning Tools
Yogesh Meghrajani, Himanshu Mazumdar, Indranil Bandyopadhyay, Satvik patel, Jignasa Patel and Aditya Team

45. X-Ray Diagnostics Systems in ADITYA/ADITYA-U Tokamak
Jayesh Raval, S. Purohit, M. K. Gupta, Y.S. Joisa, Praveena kumari, Minsha Shah,Vishmay Raulji, Imran Mansuri, Manisha Bhandarkar, K. A. Jadeja, K. M. Patel, V.K.Panchal, R.L. Tanna, J. Ghose, R. Rajpal, K.Mahajan, Aditya Team

46. Self-Inductance of Finite Straight Wire using Biot-Savart's Law
Ashvin Mali

47. Self-Inductance of Circular Loop using Biot-Savart's Law
Deenganesh Mali

48. Design of Interlock Systems for Real Time Control of Plasma Events and Experiments in Aditya Tokamak
Praveenlal Edappalli, Minsha Shah, Rachana Rajpal, K.A. Jadeja, Rohit Kumar, Suman Aich, Tanmay Macwan, K.M. Patel, R. L. Tanna, J. Ghosh and Aditya Team

49. Limiter and Divertor of ADITYA-U tokamak
K. M. Patel, K. A. Jadeja, J. Ghosh, R. L. Tanna, S. B. Bhatt, Deepti Sharma, R. Shrinivasan, Y. C. Saxena, Rohit Kumar, Suman Aich, Tanmay Macwan, Kaushlender Singh, Suman Dolui and ADITYA-U Team

50. Forecasting of Disruption in ADITYA-U Tokamak
Suman Dolui, Kaushlender Singh, Tanmay Macwan, Harshita Raj, Suman Aich, Rohit Kumar, K A Jadeja, K M Patel, V K Panchal, D. Raju, Jayesh Raval, S.K. Jha, R L Tanna, J. Ghosh and ADITYA-U Team

51. Recent advances and Upgradation of ICRH System on ADITYA-U
Kishore Mishra, Atul Varia, H. M. Jadhav, Sunil Kumar, and ICRH Team

52. Mechanical Design of a Pressurized Variable Pre-matching Stub for ICRH System on ADITYA-U
Atul Varia, Kishore Mishra, Sunil Kumar, and ICRH Team

53. Spectroscopic Diagnostics for Tokamak - Journey through ADITYA & ADITYA-U
R. Manchanda, ADITYA Team and ADITYA-U Team

Design, Fabrication and Installation of Support Structure for Reflectometry Diagnostics on Aditya-U Tokamak

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Abstract

Reflectometry is a well-known method to measure the electron density (ne) profile of tokamak plasma. The support structure for the Frequency Modulated Continuous Wave (profile) FMCW Reflectometer for electron density profile measurements at Institute for Plasma Research, IPR is described here. The reflectometer system will view port no 7 of Aditya-U. There are four waveguides W90 of copper held in a bracket in two rows and two columns. Waveguides reduce to WR28 and WR42 followed by flexible waveguide before the flange. On the front end there will be an antenna to be mounted on the aluminium flange.

The support structure has aluminium pillars which are integrated with two C channels thus forming a platform. Hylam sheet is mounted on this horizontal platform. The 4 Nos. waveguide brackets holding waveguides in place are mounted on this hylam sheet. The Hylam sheet is kept to isolate the system from the structure. The rubber sheet is kept to absorb vibrations. The design for the support structure for reflectometry for Aditya U was checked for deflection and buckling.

The support structure was fabricated and installed on Aditya-U tokamak along with the reflectometry system mounted on it.

Reference: Design and Analysis of Plasma Diagnostics' support structures and Mechanisms for Aditya-U Tokamak, Plasma 2018

Charge Exchange Neutral Particle Analyzer (CX-NPA) diagnostic measurements in ohmic as well as Ion cyclotron resonance heated plasma discharges in Aditya

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Abstract

Charge Exchange Neutral Particle Analyzer (CX-NPA) diagnostics is a well-established technique to study fusion plasma characteristics [1] and has been effectively used in a number of forms at many magnetically confined plasma machines. CX-NPA measurements in ADITYA-tokamak (R/a=75cm/25cm) are based on ionizing the neutrals in a gas filled stripping cells and subsequent analysis of positive ions [2]. CX-NPA diagnostics system in Aditya, provides the time resolved values of core-ion temperature for ohmic plasma as well as in the presence of Ion cyclotron resonance heating (ICRH). CX-NPA measurements in ohmically heated plasma of Aditya have shown that the core ion temperature lies in the range of 100 to 300 eV, for several observed plasma shots. During ICRH heating also, the core-ion temperature measurements have been performed and reported here in this paper.

Though the idea of a parallel plate electrostatic analyser, which CX-NPA uses, is very simple, there are design challenges to be considered. A few of them are also briefly discussed in this paper.

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Comparative study of perpendicular and tangential viewing Soft X-ray tomographic reconstruction for Aditya tokamak MHD Equilibrium and Stability

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Abstract

The Soft X-ray tomography (SXT) is instrumental in the studies of Magneto-Hydro-Dynamics (MHD) activities within the plasma by providing vital insight into the mode structures, their poloidal rotation, and growth trajectory, Shafranov shift etc. SXT reconstruction are performed by either perpendicular viewing tomography (PVT) or tangentially viewing tomography (TVT), where the plasma viewing is perpendicularly or tangential to **B,** respectively. A comparative study between PVT and TVT is attempted for Aditya Tokamak, with the objective to understand the relative capabilities of the two procedures for the equilibrium plasma and mode structures. The perpendicular viewing tomography (PVT), with three arrays, and the tangentially viewing tomography (TVT), with a single 2D array, has been successful in recovering verity of soft X-ray emission profile and realization the shafranov shift with the increasing β. PVT and TVT are found capable enough to reproduce the m=1/n=1 structure along with the visualization of mode structures poloidal rotation. However, the typical banana shape of the mode structure is only available with the TVT. The reconstruction for higher poloidal mode number was only possible with the addition of two more arrays to PVT system which generated the m=2/n=1 mode reconstruction and the quality of m=1/n=1 improves along with appearances of the typical banana shape. The PVT reconstruction failed for higher mode structures above than m=2/n=1. The TVT reconstruction recovered the higher poloidal modes until m=5/n=2 beyond here the limitation of image pixel resolution came into the picture. The TVT has been found to be a better option for the tomographic reconstruction although being relatively computationally complex than the PVT.

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Field simulation of Ohmic Ramp-down in ADITYA – Need for Correction Coils for improvement of Magnetic Null

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Abstract

The time evolution of magnetic null during the plasma breakdown and current start-up phase in Aditya tokamak is simulated using (please put the name of the code) code. Effect of vertical magnetic field on the dynamics of the null position has been studied in detail and is found to be not conducive for plasma break down and current ramp up in Aditya tokamak. It is shown in the simulation that the magnetic null moves inwards due to vertical field. With vertical field off the null is stable but remains outward. The limitation of generating zero vertical field at the breakdown time as well as limited control in its rise time can be overcome by driving current in additional correction coils with current value derived from the presented simulations.

ADITYA Vacuum Monitoring and Control System

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Abstract

Health monitoring of ADITYA vacuum vessel required pressure monitoring and control at different location of vessel under different condition throughout the duration. That required instrumentation system for pressure measurement on various location of the vacuum vessel, also include precise gas feed system for stabilization of the desired pressure during Pulse GDC, Continuous GDC. The develop instrumentation and control system for monitors vessel pressure at various location, discharge glow detection, cryo pump temperature etc. and automatic gas feed control system during Glow Discharge (GDC). LabVIEW based program developed to control the gas feed system for achievement of desired pressure in vessel. The developed program used for the logging of ADITYA vacuum vessel status recording for 24x7 duration. The manual data logging of the vacuum vessel status and collection of the entire vessel health monitoring is a tedious, time-consuming and inaccurate. To avoid this drawback, the developed vessel health monitoring and control system along with automatic gas feed control system operated for ADITYA-U.

Whistlers: A Probable Mechanism to Mitigate Runaway Electrons in Tokamaks

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Abstract

Whistler waves are commonly identified in ionospheric plasma with free energy attributed to energetic electrons, temperature anisotropy, and loss-cone distribution etc. Recently Spong et al., has shown that whistler activity can have significant role to play in runaways mitigation [1]. They reported that whistler is observed in tokamak plasmas excited by runaway electrons and have correlation to the energy of runaways. Their measurements are mostly based on electron cyclotron synchrotron radiation and hard X-ray diagnostics. The identification of the mode is purely based on frequency spectrum. In tokamaks, these runaway electrons can be mitigated either by enhanced collisional drag force or synchrotron radiation damping or by the excited instabilities. Whistler diagnostics is currently seen as an effective tool to understand the runaway physics and hence opens up possibility of its adaption as runaway mitigation mechanism. In Large Volume Plasma Device (LVPD), oblique whistlers are excited by the reflected particles in presence of energetic electrons [2]. These whistlers are investigated for the frequency – wavenumber spectrum and normalised power spectrum for varying energy of energetic electrons. An inverse correlation between the wave amplitude and the energy of energetic electrons is observed. Also, its parallel wave number starts dominating over perpendicular component with the energy increase of energetic electrons. Despite insignificant energy variation of energetic electrons, appreciable mode damping is observed. These observations are well supported by the theory of whistler growth driven by reflected electrons in a loss-cone in presence of energetic electrons. These concerns to the problem of wave particle interaction and scattering of whistler wave by energetic electrons and vice-versa. Observations made in LVPD can be useful towards developing an understanding on larger prospective of runaway mitigation in tokamaks by (externally injected) whistlers as they can lead to pitch angle scattering of energetic electrons. A comparison on the observation of the whistler mode growth in DIII-D tokamak with LVPD data will be presented.

Keywords: Whistlers, Runaway electrons, Tokamaks

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Effect of Superthermal Electrons on Ion-Acoustic Wave in Negative Ion Plasmas

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Abstract

Ion-acoustic soliton in a plasma consisting of positive and negative ions and superthermal electrons. Using reductive perturbation method, Korteweg de-Vries (KdV) equation is derived for the system. The soliton solution of the KdV equation is discussed in details. Variation of amplitude and width for the solitons are graphically represented to different values of negative ions, spectral index and ionic temperature ratio. The amplitude of the solitons increases with increases in negative ion concentrations and increasing the ionic temperature ratio the amplitude of the solitons decreases. The results obtained in this study may be useful to explain the ion acoustic wave in the astrophysical environments where superthermal electrons, positive ions and negative ions are present.

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Effect of Superthermal Electrons and Positrons on Ion-Acoustic Double Layers in Magnetized Plasmas

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Abstract

Ion–acoustic double layers (DLs) has been studied in plasma with presence of superthermal electrons and positrons. The modified Korteweg-de Vries (m-KdV) is derived using reductive perturbation method (RPM). It is found that for the selected set of parameters, the system supports rarefactive (compressive) double layers depending upon the values of k, δ, σ_e and σ_p. The amplitude of compressive (rarefactive) DLs decreases (increases) with increase in σ_e however increases with increase in δ, σ_p and k keeping other plasma parameters constant. Width of DLs increases with increase in parameters k, δ, σ_e and σ_p. Phase velocity of ion-acoustic waves (s) increases with increase k however decreases with increase in σ_e, and σ_p keeping other plasma parameters constant.

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Plasma Position Estimation Using Magnetic Diagnostics in ADITYA-U

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Abstract

In a magnetically confined toroidal fusion device or tokamak, for the real-time control of the movement of the plasma column, in both horizontal and vertical directions, precise measurement of the location of the plasma column is mandatory [1]. Magnetic pickup coils, which measure the induced voltage due to the change in the flux, linked to them, are widely used to measure the temporal evolution of the position of the plasma column in a tokamak [2]. To measure the position of the plasma column accurately in ADITYA-U tokamak, several types of magnetic probes are introduced. They include Mirnov coils, external pickup coils, Sine-Cosine coils and flux loops [2]. All probes are located near the poloidal periphery of the vacuum vessel, both inside and outside of vessel. In order to have a suitable calibration factor for these probes, which is a necessity to overcome the geometric imperfections and discrepancies introduced during the installations, and the error magnetic fields due to the vacuum vessel, an experiment had been carried out at in-situ environment. A variable current in time had been passed through a rigid copper conductor, placed in different radial and vertical locations within the vacuum vessel. Induced voltages in all magnetic probes were recorded due to the different time profiles of current, driven through the conductor. Analyzing the corresponding data, several interesting results had been obtained, among which the temporal mismatch of the signal from different magnetic probes is one of them. Based on these observations, several numerical codes have been developed that analyse the raw data of these magnetic probes during plasma discharges. After removing all unwanted fluxes, linked to the probes due to induced currents in vessel and other set of magnetic field coils used for plasma operation, the temporal evolution of the horizontal and vertical movement of the plasma column has been estimated in real time. The position of the plasma column measured with these probes coincides quite well with each other as well as with other diagnoses, such as Langmuir edge probes, fast camera images, optical signals etc. Finally, real-time measurements of the horizontal positional movement of the plasma column during plasma discharges have been fed into the plasma position control system for real-time control of the plasma position in ADITYA-U tokamak. This presentation summarizes results from all these magnetic diagnostics along with several interesting consequences.

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Iron Impurity Behaviour Study in the ADITYA Tokamak

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Abstract

Iron (Fe) impurity behaviour and transport in ADITYA tokamak plasma has been investigated by modelling the observed VUV spectral lines at 28.41 nm from Fe^{14+}, and 33.54 nm and 36.08 nm from Fe^{15+}. For this, required photon emissivity coefficient was calculated using the ADAS general collisional-radiative code considering contribution from various atomic processes. In experimental observation, it has been observed that the intensities of the Fe emissions decrease with an increase in plasma electron density. The observed spectral emission and the intensity ratio of Fe^{14+} and Fe^{15+} impurity ions from two discharges having relatively low and high plasma density are modelled using a one-dimensional impurity transport code STRAHL It shows that the observed spectral line emission can be simulated using the same ratio of the convective velocity to the diffusion coefficient v/D radial profile, but with two different iron concentration values. The ratio *v/D* varies from the value of -0.22 m^{-1} at the plasma normalized radius ρ = 0.2 to a maximum value of -0.35 m^{-1} at ρ = 0.6. The obtained diffusion coefficient for Fe impurity ion in the core region is well explained in using neo-classical transport in the Pfirsch–Schluter regime. While the diffusion coefficient in the edge region, which is large by two magnitudes as compared to core region is explained using ion temperature gradient turbulence.

3D simulation in Aditya Scrape-off layer using steady state model EMC3-EIRENE

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Abstract

Numerical simulation of plasma transport along parallel and perpendicular to magnetic field line has finite relevance to many important issues in the Scrape-off layer of tokamaks including medium size devices like Aditya and Aditya-Upgrade. 2D, 3D simulations in support to real experiments has been a good support to understand many important issues in previous machine Aditya and will helpful to Aditya-Upgrade in many aspects [1]. The 3D transport simulations using plasma-fluid and kinetic neutral species are performed for both, the poloidally continuous ring limiter (RL) and a toroidally distributed set of outboard block limiters (BL) similar to Aditya-Upgrade [1, 2]. Initial 3D simulation results using the 3D model EMC3-EIRENE shows the signature of flow shear, regions of long and short connection length and associated diamagnetic drift [1, 2]. The subsequent simulation work shows the perpendicular diffusive transport and recycling flux strong in aditya ring limiter than a symmetric 3 Block limiter configuration similar to Aditya-Upgrade [3]. The recent simulation shows the diffusive particle transport behavior is strong in Aditya ring limiter than block limiter configuration over a range of input edge density at constant input power and diffusion coefficients at both upstream and downstream locations of machine similar to some results in Alcator-c-mod [4].

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Tungsten Coating Deposition on Graphite and Process Optimization

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Abstract

Tungsten and tungsten coating on high melting point materials with high thermal conductivity are extensively explored for their potential as plasma facing components (PFC) in Nuclear Fusion reactors. The impurity minimization in the plasma in such reactors as Tokamak, is of very importance due to energy loss caused by the radiation from the impurities. Hence, ideal PFCs should not be releasing significant impurity material to the hot plasma during its interaction with energetic charged particles from plasma. Relatively very low sputtering yield of tungsten, as compared to the most of other thermally and electrically conductive materials, make tungsten as one of the favourable candidates for this application as PFC. With ultimate intention of studying the behaviour of tungsten film on graphite as plasma facing material, we have carried out magnetron sputter deposition of tungsten on graphite. We optimize the deposition process for typically 1 micron thick, stable films and characterized them using X-ray diffraction and Scanning electron microscopy for its crystal and micro structures. Residual stress, structural defects and resultant physical adhesion are optimized. We will be presenting the details of these results in this paper.

Analytical Model for Erosion Driven Carbon Sources in Tokamaks

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Abstract

Impurities play an important role in the equilibrium, pulse length and plasma performance of a tokamak. The accumulation of impurities in the plasma core can lead to enhanced radiation, fuel dilution as well as can trigger disruptions by edge cooling. On the positive side the controlled presence of impurities helps in creating a radiative mantle and there by helps to spread the heat load evenly on the plasma-facing components. Impurities like carbon are present in the plasma as result of the erosion of plasma-facing components. The uncertainties in the generation of impurities need to be understood so that such models can be used in the numerical simulation of impurity transport as well as in the experiments.

In this article we discuss the erosion driven impurity sources in the case of carbon first-wall.

An erosion model is proposed by incorporating the thermal and energetic hydrogen ions, charge-exchange neutrals, impurity as well as chemical sputtering and chemical erosion.

For a specific case of carbon, we estimate the erosion of carbon tiles by taking into account the tile geometry and the magnetic field configuration. An analysis of the impurity sources will be presented based on a number of factors such as core plasma parameters, transport conditions, heating scenarios etc. We will also discuss the application of the model for ADITYA and other tokamaks.

Infrared Imaging Video Bolometer Diagnostics in Aditya Tokamak

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Abstract

Infrared Imaging Video Bolometer (IRVB) diagnostics is one of the advanced imaging diagnostic for the measurement of total radiated power from the magnetically confined plasmas. It provides a spatiotemporal profile of the total radiated power in a wide energy band [1-3]. The IRVB has been used in several magnetic field confinement devices [4-8]. Several important observations are made using IRVB, namely, investigation of plasma interaction with first wall components, radiation profile measurement during plasma detachment and disruption, radiative collapse study and three-dimensional (3D) radiative structures study in helical devices [9]. The IRVB diagnostic deployed in Aditya [10] views the plasma tangentially. The diagnostic is capable of absorbing radiation from ~5 eV to 8 keV photon energy range using a free-standing ultra-thin platinum foil having a size 6.5 cm x 6.5 cm x 0.00025 cm that views the plasma with a field of view (FoV) of 46° x 46° through aperture size 0.7 cm x 0.7 cm. IRVB images the plasma radiation with an array of 9 x 9 bolometer pixels at a temporal resolution of ~6 ms. The spatial resolution at plasma mid plane is ~7 cm and Noise Equivalent Power Density (NEPD) is estimated to be ~0.250 mW/cm2. The present paper reports analysis of 2-Dimensional radiation brightness images obtained during different phases of a plasma discharge and its comparison with the synthetic diagnostic simulated results, the temporal evolution of total radiated power and its comparison with the measurements obtained through the AXUV bolometer system [11] and total radiated power loss fraction w.r.t. input power during the plasma current flattop. These measurements establish the capability of IRVB method to provide bolometric measurements complementary to existing techniques.

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Investigation of Impurity Seeded ADITYA & ADITYA-U Tokamaks Plasmas

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Abstract

Impurity seeding in the tokamak plasma is investigated for its valuable role in the disruption mitigation and heat load reduction on the plasma facing surfaces. Not only that, it is also studied to achieve the improved confinement in the tokamak plasma, which is known as Radiative improved (RI) mode and considered as an alternative way of operation for achieving fusion grade plasma. This mode is believed to be based on the reduction of growth characteristics of the toroidal ion temperature gradient (ITG) mode due to the increase of Z_{eff} and also because of the suppression of turbulence due to increase of E×B shear rotation in the impurity injected plasma. In the ADITYA and ADITYA-U tokamaks, experiments were carried out to obtain RI mode like plasma using neon and argon gases puffing. In those experiments, the time for the gas puff to start, time gap between gas puffs, number of gas puffs, amount of gas injection by varying pulse width and voltage level in the gas fuelling system were varied. It was found that line average electron density, n_e and central electron temperature were increased after the neon and argon gases puffing. Substantial change in plasma edge properties was observed with the increase of radiation and reduction of hydrogen recycling, which led to better particle confinement. The energy confinement time, τ_e, was increased very much and became almost similar to the value estimated using the H-mode scaling law of ITER93 ELM-free. In this presentation, details on the experiment and obtained results will be discussed.

Investigation of atomic and molecular processes in ADITYA and ADITYA-U tokamak plasmas

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Abstract

We report the analysis of experimental discharges of ADITYA and ADITYA-U tokamak by a thorough understanding of various atomic and molecular process. In these tokamaks, the fuel neutrals enter into the plasma from limiter surface and interact with plasma particles. In addition to that, various impurities such as Carbon, Oxygen can also be present inside the tokamak plasmas. As a result, different atomic and molecular processes occur when these neutrals interact with either plasma charge particles and/or impurities. Due to the atomic and molecular reactions, the radiation comes out from the plasma, which can be measured by spectroscopic means. Hence, the measurement of wavelength spectrum provides a tool to obtain the information about the species present in the plasma.

In this work, the impurity transport behaviour is studied using impurity transport code, STRAHL. The diffusion coefficients are estimated along the minor radius of ADITYA using Oxygen impurity transport, which are found higher compare to that of neo-classical values. The density distributions of different charge state of Carbon are also estimated from STRAHL code to study the behaviour of radial profile of H_α emissivity profile. The behaviour of neutral particle is then studied in details after successfully implementing the limiter geometry in neutral particle transport code DEGAS2. It is found that the neutrals coming from the dissociation process having temperature approximately of 2 eV. Contribution from various atomic and molecular processes is investigated through modelling of experimentally measured radial H_α emissivity profile. It has also been shown that the molecules penetrate quite far inside (~ 4 cm) the limiter radius. It is noticed that the neutrals produced from charge-exchange process penetrate far inside the tokamak plasma. In an another study, the particle source rate is evaluated from the measured emissivity profile of neutral hydrogen for ADITYA and the particle confinement time and diffusion coefficient are estimated, which reveals that the particle transport is anomalous in nature. An attempt is also made to observe the neutral penetration and subsequent changes in the temperature and density profiles due to gas-puffs. For this, the temporal profile of Hα emissivity is simulated with the help of various atomic and molecular processes such as dissociation, ionization, excitations etc. and compared with the experimental measurements.

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Population-Alignment Collisional-Radiative Model for Polarization in Lyman-α Line

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Abstract

In plasma spectroscopy the Collisional-Radiative (CR) model [1] is a versatile tool in dealing with a problem of population. A Maxwellian distribution is implicitly assumed for the electron velocity distribution function (EVDF) in the CR model, and the concept of temperature is used to quantify the activities of electrons. In other words, in the CR model it is assumed that all the magnetic sublevels are uniformly populated. However, in actual plasmas the EVDF may be anisotropic and when atoms/ions are excited by electrons having an anisotropic EVDF it creates non-uniform population distribution over magnetic sublevels. The emission lines originated from such magnetic sublevels are generally polarized. The imbalance of population among magnetic sublevels is expressed by a quantity known as "the alignment" [2]. The Population-Alignment Collisional-Radiative (PACR) model [2] is an extension of the CR model in which the alignment of levels is also treated in addition to the population of levels. The PACR model enables us to interpret the measured polarization degree in terms of the anisotropy in the EVDF.

We have developed the PACR model for Lyman-α line following the methodology proposed by Fujimoto [3]. The present model treats EVDF having different electron temperatures in the directions parallel ($T_{||}$) and perpendicular to the magnetic field ($T_{\perp}$). In this model each energy level is assigned two quantities: population and alignment, and rate equations for the population and alignment are solved under quasi-steady state condition. It has been found that the theoretical polarization degree is negative for $T_{||} < T_{\perp}$ and it is positive for $T_{||} > T_{\perp}$. The main purpose of constructing the PACR model for Lyman-α is to evaluate anisotropy in the EVDF in the edge LHD (Large Helical Device) plasma from the experimentally measured polarization degree in Lyman-α line [4]. The PACR model for Lyman-α will be discussed in details during the symposium.

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Electronics for Langmuir Probe in Aditya-U for the measurement in the SOL layer.

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Abstract

The role of the tokamak edge plasma in influencing the fusion energy yield of tokamaks is now widely recognized and is reflected in the increasing efforts devoted to the experimental and theoretical study of scrape-off layer (SOL) physics. Of particular concern are aspects of the plasma-surface interaction leading to impurity production and the subsequent impurity transport and contamination of the core plasma. The impurity transport depends strongly on the background properties of the SOL plasma, such as the plasma density, potential, electron and ion temperature, ion flows, flow-velocity and their fluctuations and transport coefficients.Understanding of the underlying physics in the edge plasma of Tokamaks requires knowledge of these parameters with a high spatial and temporal resolution. Various types of Langmuir probes, namely limiter flush mounted probes, rack probes and garland probes are installed on Aditya-U to diagnose its edge plasmas.

The signal conditioning electronics is needed for the measurement and validation of the parameters through these probes. The said electronics imparts proper conditioning to the incoming signal. Mainly, there are three different electronics involved viz: sweep-generator and measurement electronics for the ion-temperature measurement; measurement electronics for the ion-saturation current; and measurement electronics for the floating-potential. The measurement electronics is indigenously made and is multi-channel to support the probe-arrays. This poster discusses the design considerations and technical details for different electronics for variety of probes installed on Aditya-U.

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Role of Poloidal Flows on the Particle Confinement in a Current-less Toroidal Device

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Abstract

It is well known that fusion devices such as Tokamaks have complex magnetic geometries and hence performing a detailed study of an individual physics phenomenon is often an involved task. A current-less toroidal device (CTD) provides an alternative scenario to serve as a test-bed to perform studies of basic plasma instabilities which are relevant to Edge/SOL region of fusion devices such as a Tokamak. In a CTD, plasma is confined by the application of toroidal and vertical magnetic field only. Therefore, an effective rotational transform ceases to exist [1].

The device BETA at the Institute for Plasma Research (IPR) is one such CTD with a plasma major radius of 45 cm and minor radius of 15 cm and a maximum toroidal field of 0.1 Tesla. An experimental study of particle confinement with variation in parallel connection length for two different sources has been performed in BETA. Two sources used are hot cathode and Microwave source. In the hot cathode source, a tungsten filament of 2 mm diameter and 20 cm length has been mounted at the minor axis at one particular toroidal location. The discharge is struck between the hot cathode and the vessel wall. It creates a potential well and hence produces a radial electric field. This mean radial electric field provides a poloidal rotation to the plasma. In Microwave source, a 2.4 GHz microwave wave is launched from low field side and resonance occurs at one toroidal plane to produce the discharge and has very weak zeroth order radial electric field.

In the present work, detailed experiments have been performed with the above mentioned sources and with variation in parallel connection length. As reported earlier [1, 2], mean parallel connection length plays a significant role in controlling the nature of equilibrium, fluctuation and flow in a current-less toroidal device. It has been observed that the particle confinement improves significantly with the variation in connection length and it also depends on the presence of poloidal flows [3]. The details of experimental findings from above mentioned experiments will be presented.

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Investigation of Neutral Recycling and Ion Temperature of Various Plasma Species in ADITYA and ADITYA-U Tokamak

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Abstract

The properties of the core plasma of tokamak are related to the overall characteristic of the plasma edge as the particles are transported to the core through plasma edge and vice versa, thus influencing the plasma core. The edge introduces neutrals into the plasma by particle recycling and also impurities through its interaction with Plasma Facing Components (PFCs) via desorption and sputtering. This particle then alters the edge plasma properties by participating into various phenomena inside the plasma, such as instabilities, radiation loss etc. Then to understand the role of the neutrals, recycling, which is basically measure of plasma outflow to particulate matter coming back from the plasma to going out of the plasma, is needed to be studied and it is carried out by measuring influx and out flux using PMT based spectroscopic diagnostic and Langmuir probe kept at plasma edge. Along with that global particle balance has been studied for different discharges of ADITYA and ADITYA-U tokamak. Similarly neutral and impurity ion temperature has been estimated, which is important for understanding the edge plasma through the modelling. However the measuring the temperature from the Doppler broadened of spectral line shape become erroneous without considering Zeeman Effect as the Aditya & Aditya-U tokamak is having magnetic field of 0.75-1.4 Tesla. The Zeeman Effect produces the split of spectral line and this gives extra broadening in measurement. A MATLAB code has been developed to estimate ion temperature after incorporating Zeeman Effect for the neutral hydrogen, carbon and oxygen ions using spectral line at (Hα : 656.28 nm), (C^{1+} : 657.28 nm) and (O^{4+} : 650.02 nm), respectively. This study reveals the poloidal asymmetry of ~ 2 eV in the neutral temperatures in both tokamaks. Similarly, this works also borne out the requisite correction needed in the estimation of impurity ion temperature due to Zeeman Effect as the C^{1+} ion temperature comes to be ~ 70 eV without inclusion of the Zeeman effect but reduces to < 10 eV after the correction. Similarly, ion temperature of O^{4+} ion is also analyzed with the same code.

Study Small Amplitude Ion-Acoustic Solitons in Negative Ion Plasmas with Superthermal Electrons

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Abstract

Small amplitude ion-acoustic soliton in a plasma consisting of positive and negative ions and superthermal electrons. For this purpose, the hydrodynamics equations for positive and negative ions, superthermal electron density distribution and the Possion's equation are used the reductive perturbation method to derive the Korteweg de Vries (KdV) and modified KdV (mKdV equation. The amplitude of the compressive (rarefactive) solitons increases with increases in negative ion concentrations and increasing the ionic temperature ratio the amplitude of the compressive (rarefactive) solitons decreases. The width of the compressive (rarefactive) solitons decreases with increases in negative ion concentrations and decreasing the ionic temperature ratio the amplitude of the compressive (rarefactive) solitons increases.

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Topology Optimization of a Planetary Gearbox for Fusion RH Application

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Abstract

Topology Optimization is a method used to distribute the material properly in the given domain with the predefined constraints without changing the performance of the material. In the presented work, the topology optimization of a planetary gear system is carried out. The payload of the system in which planetary gear train is used is 50 kg. Based on the speed of the actuator the torque on the planetary gear system was obtained and gears were optimized. The static structural analysis of the gear system is also carried out. The main purpose of this optimization is to reduce the mass of gears without affecting the robustness. In this way the load on the actuator could be reduced.

Keywords: *Topology, Optimization, Load*

Passive Charge eXchange (PCX) Spectroscopy to Measure Plasma Rotation on Aditya-U Tokamak

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Abstract

Passive Charge eXchange (PCX) spectroscopy is used in Aditya-U tokamak for the measurement of toroidal plasma rotation and ion temperature [1, 2]. The diagnostic has been recently upgraded with collection optics utilizing a re-entrant view port thus enabling a spatial profile of plasma rotation and ion temperature measurements. The diagnostic comprises of a 1m, *f/8.7* Czerny-Turner arrangement spectrometer along with a fast CCD. The set up allows measurement of spatial profiles by utilizing seven lines of sights to cover complete plasma minor radius towards low field side. PCX line emission of carbon at 529 nm is used for the measurement. Abel like matrix inversion technique is used to obtain spatial profile of plasma rotation and ion temperature. In the presentation results obtained using the upgraded diagnostics are presented.

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Characterization of Ohmic Breakdown Phase for the ADITYA discharges

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Abstract

ADITYA is a mid-size air core tokamak, with major radius 0.75 m and minor radius 0.25 m. Breakdown is achieved via inductive start-up with assistance of filament pre-ionization. The nature of pre-breakdown and its effect on quality of plasma formed later is studied with parameters influencing it, like toroidal electric field E, operating pressure p, toroidal magnetic field B_T, error magnetic field B_{error} and initial electron density n_e. Breakdown failure due to various causes are identified, from which boundary limit and general condition for set of primary operating parameters is derived to achieve successful discharge in ADITYA tokamak. It is learned that breakdown is achieved for operating pressure ranging between $0.5 - 1 \times 10^{-4}$ torr and ratio of toroidal electric field E (V/m) to error magnetic field produced by the Ohmic coil B_{err} (G) above 0.32 V/G-m. Breakdown model is developed to estimate breakdown time by considering electron production and loss mechanism for a given discharge conditions, which agrees well with filament assisted plasma discharges. The condition for successful breakdown $E_\phi\ B_T/B_\perp > 10^3\ \ V/m$ is found to be valid and experimentally confirmed in many tokamaks including ADITYA tokamak.

An Estimation of the Edge Impurity Transport in the ADITYA Tokamak through Comparison between the Simulated and Analytical (Model–Based) Edge Impurity Diffusion Coefficients

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Abstract

Impurities are the non–fuel species in the tokamak plasma which, depending upon the plasma temperature profile, can lead to either radiation losses or plasma dilution disrupting the tokamak operations. Analysis of the nature of impurity transport (neoclassical or anomalous) in the tokamak plasma and their contributing factors, through experiments and modelling, has been an important topic of tokamak research for more than a decade. Experiments for analyzing the nature of edge impurity transport in the ADITYA tokamak have been performed where the chord–integrated brightness of the 650.024 nm ($2p3p^3D_3$–$2p3d^3F_4$) transition of the O^{4+} ions along the inboard/outboard (high/low toroidal magnetic field) regions of the ADITYA plasma have been measured [1]. The 650.024 nm emissivity data from the measured brightness have been obtained using an Abel–like matrix inversion including the uncertainties due to poloidal asymmetry. The radial emissivity profiles of the 650.024 nm O^{4+} transition in both (inboard/outboard) regions of the ADITYA plasma are next determined using the simulated O^{4+} number densities obtained, first using STRAHL and now from the recently developed semi–implicit radial impurity transport equation (SI–RITE) [2], along with the Photon Emissivity Coefficients (PECs) from the ADAS database. The SI–RITE 650.024 nm emissivity profiles are matched here with the ADITYA experimental emissivity data and the STRAHL simulated radial O^{4+} emissivity profiles. Based on the 'best–fit' simulated emissivity profiles with respect to the experimentally obtained data, the corresponding inboard and outboard impurity (oxygen) diffusivity profiles are decided. The (STRAHL and SI–RITE) simulated 'best–fit' diffusivity profiles obtained towards the edge regions (inboard/outboard) of the ADITYA plasma show a dominance of the fluctuation induced transport, conjectured to be contributions from either the Ion Temperature Gradient (ITG) modes or Resistive Ballooning (RB) modes or both at the ADITYA plasma edge. The microinstabilities contributing to the fluctuation induced impurity transport at both (inboard/outboard) edges of the ADITYA plasma are then decided based on the condition for dominance of the ITG modes over the RB modes proposed by Zeiler et al. [3]. The ITG modes are found to be the dominant contributors to the fluctuation induced transport at both (inboard/outboard) edges of ADITYA plasma based on the reported condition. The expression for the ITG driven diffusivity reported by Zeiler et al. [3] however, when used for obtaining the oxygen diffusivities at the ADITYA plasma edge (inboard/outboard), failed to follow the condition reported by Zeiler et al. [3]. A second factor is hence included to the existing expression of the ITG driven diffusion coefficient reported by Zeiler et al. [3] in the present study. The new expression for the ITG driven impurity diffusivity $D_{ITG, FINAL}$ is reported to follow the condition for dominance of the ITG modes and the net impurity diffusivity then matches with the (STRAHL and SI–RITE) simulated impurity diffusivity values estimated at the (inboard/outboard) ADITYA plasma edge.

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ECRH Power Supply System for Aditya Tokamak

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Abstract

Electron Cyclotron Resonance Heating (ECRH) system is used in Aditya and Aditya-U Tokamak for plasma start-up (pre-ionisation), and plasma heating experiments. The 28GHz based ECRH system used for pre-ionisation at low loop voltage at fundamental harmonic at 0.75T operation in Aditya Tokamak. This systems was consisted of Gyrotron source with normal ohmic magnets, transmission line, launcher, power supply system and VME based DAC. This system was capable to deliver 200kW power for 3s duration. Now 42GHz ECRH system is being used for heating and pre-ionisation at second harmonics at 0.75T and fundamental at 1.3T operation in Aditya-U Tokamak. It consists of a Gyrotron source, cryo-magnet, transmission line, launcher, power supply system and PXI based DAC. The Gyrotron delivers 500kW power for 500ms duration at ~50kV beam voltage and 20A beam current.

The 28GHz ECRH power supply system included cathode High Voltage Power Supply (HVPS), Anode Modulator Power Supply (AMPS), Magnets Power Supply (MPS), filament power supply and ion pump power supply. For this system, an unregulated HVPS was used; the main ratings for this power supply are constant 70kV, 10A. An 11kV input voltage variation system was added to get the variable output voltage from 0 to 70kV. That HVPS was a high stored energy power supply so rail gap crowbar protection was used for safe operation of the Gyrotron. The AMPS was a switching Tetrode based power supply and its input voltage was derived from main HVPS. The 4 magnets power supplies were Thyristorized controlled high current power supply (40V, 600A). The filament power supply (30V, 20A) is a voltage controlled regulated power supply floating at -70 kV DC voltage.

The 42GHz ECRH power supply system is an upgraded and new technology based system. The main High Voltage Power Supply is a Pulse Step Modulation (PSM) based power supply (-80 kV, 20A). The key feature of PSM high voltage power supply is fast turn ON/OFF time less than 10 µs, and low stored energy. This type of power supply topology is well suited for safe and reliable operation of microwave tubes like Gyrotron and Klystron etc. The AMPS (30kV, 100mA) is a conventional high voltage low current power supply. The cryomagnet power supply is a commercial off-the-self high current power supply from American Magnetics Inc. The filament power supply (30V, 20A) is a voltage controlled regulated power supply floating at -70 kV DC voltage. A series ignitron crowbar protection is used for safe and reliable operation of Gyrotron, which removes the high voltage within 10µs in an event of fault. This paper presents the highlights the old power supply system and discusses the description of new power supply system for the ECRH used in Aditya and Aditya-U.

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Indigenously Developed Data Acquisition System for Plasma Diagnostics in Aditya-U Tokamak

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Abstract

Aditya Upgrade Tokamak has provided opportunity to implement indigenously developed 64 channel data acquisition system. The system is portable, compact and does not need any external PC for processing. This data acquisition system is more economic and similar in features to PXI based DAS. It is connected to control server through Ethernet. Since the system is stand alone in itself, it is placed in same rack with signal conditioning system of plasma diagnostic. This arrangement reduces the long cables which earlier used to be 20 meters between DAQ unit and signal conditioning system to 1 meter. This setup improved signal to noise ratio and reduced ground loop problems. We have implemented such 5 system among different diagnostic like Soft-Xray, Bolometer, ECE radiometer, EM diagnostic and Spectroscopy. The five units of 64 channel DAQ are functioning successfully since Aditya upgrade phase-I. The inherent features of the board are 16 bit resolution, programmable pre-trigger sample and sampling rate up to 200 kS/s/ch and simultaneous acquisition [1]. The acquisition application is developed in Labview and Linux platform, in SBC. In poster details of system integration, hardware and software architecture and results will be discussed.

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Gas Puff Induced Drift Waves in ADITYA and ADITYA-U Tokamak

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Abstract

The gradients in the radial profiles of density and temperature gives rise to various instabilities which aid in the transport of the particles across the magnetic field lines in tokamaks. These instabilities, signature of whose are embedded in fluctuations observed in various physical quantities like plasma potential, density, etc. are required to be understood properly for mitigating them. Here, we report the excitation of a low frequency mode in the density and floating potential fluctuations in presence of gas puffing in typical discharges of ADITYA-U tokamak, a medium sized air core limiter tokamak. The fluctuations in electron density are measured with microwave interferometry and Langmuir probes. To characterize the observed fluctuations, the Fourier spectrum analysis has been performed on the time-scale data of these fluctuations. The low frequency modes can be characterized as drift waves excited by an external power drive, the ohmic transformer power supply in our case. However, the growth rates calculated from analytical expressions are too less to be observed, as seen in the analysis of density fluctuations also. Incidentally, the presence of gas puff seems to modify the growth rate of the mode. The introduction of neutral particles by the gas puff seems to play a dominant role in the excitation of these modes.

Laser Heated Emissive Probes Diagnostic in ADITYA – U Tokamak

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Abstract

Plasma potential profiles knowledge is of key importance in fusion devices as it governs spatial electric field which in turn determines many turbulent transport phenomena like particle drift, confinement etc. Conventional emissive probes (CEP) are more popular for direct measurement of plasma potential and its fluctuation in plasma devices. However, due to certain limitations of CEP like shorter lifetime of filaments, non-uniform heating and bending under influence of strong magnetic field they ceases to be popular diagnostic tool in fusion devices like tokamaks where frequent vacuum break for burnt filament replacement is not possible. These limitations are eliminated in Laser Heated Emissive Probe [LHEP] that has several advantages over CEP like even electron emission, not influential under $j_{filament}$ x B forces and longer life time.

A Glow Discharge plasma device was designed and assembled to carry out experiments with LHEP using LaB_6, CeB_6 and Graphite as probe tips. Experiments to study temperature response of LHEP probe materials with respect to laser heating in atmospheric and vacuum conditions were carried out that aided in concluding design parameters for LHEP assembly in ADITYA-U. Experiments for recording LHEP V-I characteristic to determine plasma parameters were carried out using different probe materials in Glow Discharge plasma. Based on our knowledge with LHEP experiments, a unique experimental arrangement of the LHEP measurements for ADITYA-U tokamak have been designed and installed. Two circular Graphite probe tips of 6 mm diameter each, separated by ~15 mm and sitting at same radial but different poloidal location were kept in SOL region of ADITYA-U tokamak at 1.2 cm away from limiter. This compact, light-weight, non-magnetic, portable design allows laser to remain focused on probe tip despite of its radial movement while having a possibility of cold and hot probe experiments both separately and simultaneously as an advantage. In this paper we present results of LHEP heating-cooling experiments as well as LHEP experiments in Glow Discharge plasma. Brief account of designing and installation of LHEP for ADITYA-U while giving a detailed summary of initial results of floating potential and its fluctuations recorded with respect to vessel ground using LHEP in ADITYA-U (in cold condition) is also reported here. Data have been further analyzed to find the plasma frequencies to conclude the LHEP experiments in ADITYA-U SOL region.

ADITYA Upgrade New Circular Shaped Torus Vacuum Vessel and Pumping System

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Abstract

The first Indian tokamak, ADITYA had successfully completed 25 years of operation of poloidal limiter plasma configuration in 2014. After the successful achievement of the targeted plasma discharge and performance of many important tokamak experiments, the up-gradation of ADITYA tokamak has been configured for double null divertors and developed accordingly [1]. The up-gradation includes the replacement of rectangular cross section vacuum vessel by circular cross section vacuum vessel without changing the major and minor radius of tokamak to accommodate various divertor coils in space between the new vessel and 20 numbers toroidal field coils [2]. In order to accommodate as many numbers of diagnostics as possible, the ADITYA-U vessel has been designed to have 111 port openings compared to 48 ports in ADITYA vessel. The new vacuum vessel of SS304L was fabricated very precisely in dimension by M/s Godrej & Boyce Pvt. Ltd under the supervision of IPR scientists and as per the IPR design. After successfully completion of the acceptance tests viz., dimension measurement, UHV, leak proof, baking at factory, the final acceptable tests had been carried out successfully at IPR as results of local leak rate < $5x10^{-10}$ mbar.l/s and global leak rate < $5x10^{-8}$ mbar.l/s, UHV test as < 10^{-9} mbar, vessel baking >150° C at 48 Hrs. The new circular cross section vacuum vessel [3] has been installed and tested successfully in ADITYA-U tokamak on machine support structure with corresponding magnetic field coils. After vessel installation, the new diverter coils have been installed on the vessel outer side using outer structure brackets. These brackets have been designed and fabricated on vessel precisely to fasten the divertor coils perfectly according to tokamak machine center.

The ADITYA-U vessel pumping ports have been configured for eight pumping ports at four toroidal locations under consideration of high gas load due to multiple de-mountable joints, high surface area of in-vessel components like Graphite, SS wall, magnetics diagnostics etc. The vacuum pumping system has been designed, developed and installed successfully to achieve various pressure conditions for tokamak requirements like, UHV, vessel wall conditioning and plasma operation. For phase-I and phase-II plasma operation, the four pumping ports of 200CF have been used with two identical turbo molecular pumps and two identical cryo pumps. The four pumping lines have been designed to acquire maximum effective pumping speed at vessel 200 CF ports and prepared using four 250 CF electro pneumatic gate valves and new support structure. The effective pumping speed of N_2 at vessel has been configured after pumping lines installation as ~550 liter/sec and ~650 liter/sec by a turbo molecular pump and a cryo pump respectively. The total gas load of the vessel and in-vessel components generated by global leak rate, total out gassing rate and permeation has been maintained ~10^{-6} mbar.l/s to achieve ultimate vacuum as ~10^{-9} mbar. The total out gassing rate of SS304L vessel, graphite first wall material, in-vessel magnetic diagnostics assemblies etc. has been calculated and studied corresponding to pumping system operation.

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Control System Development for LIGO outgassing setup

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Abstract

A control system is being developed to facilitate controlled heating of SS304L coupons in an outgassing setup. The sample chamber is a 14 liter cylindrical chamber having an internal surface area of 3913 cm^2 pumped through a 9.5 liter cylindrical chamber having an internal surface area of 3166 cm^2. A turbo-molecular pump and an ion pump is used to evacuate the setup to the order of 10^{-10} mbar pressure. Approximately 150 numbers of SS304L coupons of dimension 250mm x 50mm x 3mm can be placed inside the sample chamber and baked to 150°C [1] using baking tapes. A controlled baking procedure is planned, which heat the total setup to the required temperature and maintain it till the desired time. The control system also monitors parameters like vacuum set points, temperature status, real-time pressure values, RGA mapping, gate valve status and give feedback about the total health of the system.

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Water Cooling System of ADITYA Tokamak

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Abstract

ADITYA tokamak (R0 = 75 cm, a = 25 cm) contains three principal sets of magnetic field coils namely the Toroidal magnetic field (TF), Ohmic (TR) and Vertical magnetic (BV) water cooled coils. The 20 nos. of TF coils are cooled by passing demineralized and De-Ionized (D & D) chilled water through the cooling tubes embedded in and soldered to inner surface of each turn. The TR coil system consists of one big Central solenoid [TR1] coil and three pairs of compensating coils TR2, TR3 and TR4 and vertical magnetic field coil system consists of two pairs of BV coils, BV1 & BV2. The TR and BV coils are cooled by passing demineralized and De-Ionized (D & D) water through the central hollow path of the conductor. The water cooling system was designed and installed for ADITYA tokamak magnet system. The chilled water is circulated in the coils at ~ 1 bar pressure through 2 nos. of water headers. Each of the headers is made of SS 304 ERW 150 mm NB. The chilled water enters at the bottom header and after passing through the various coils, returns to the top header. There are 20 parallel paths with valves for the TF coils cooling, 12 numbers of water paths with valves for TR1 coil, 32 for TR2, 04 for TR3, 08 for TR4 and 32 for the BV coils in both supply and return headers. The Braided PVC tube is used for water connections from Header to coils. During ADITYA tokamak operation, higher currents (ranges from lower 3-4 kA to higher 40-50 kA) passing through the multiple magnetic coils. Due to this phenomena, heat is generated in each magnet system during the operation of the electromagnet is to be removed efficiently by continuously passing water through individual magnet coils. Accordingly, cooling water distribution system has been designed to transfer the heat dissipated by each of the magnet coil to the chilled water system. The heat absorbed by cooling water will be dumped into a Plate Type Heat Exchanger (PHE), which in turn will be further dissipated into the chilled water line i.e. chiller system. Water quality of less than 1 µS/cm and temperature less than 24 °C are maintained for operation of ADITYA system. This paper describes the experimental requirement of various cooling water parameters.

The Refurbishment of Damaged Toroidal Magnetic Field coils for ADITYA-U

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Abstract

ADITYA tokamak (R0 = 75 cm, a = 25 cm) is the first Indian tokamak, indigenously built and commissioned at the Institute for Plasma Research, Gandhinagar, Gujarat, India, in September, 1989. After 25 years of successful operation of ADITYA, it has been upgraded from limiter configuration to ADITYA-U tokamak with divertor configuration. There are 20 numbers of toroidal magnetic field coils which produces 1.5 Tesla of magnetic field at plasma centre, when 50 kA of current is passed through them. Each of the TF coils is a picture frame type coil, with a bore of 0.78 m x 0.9 m and outer dimension of 1.03 m x 1.26 m, having weight of 500 kg each. A single TF coil is made up of two C's (Big C and Small C) joined together using 16 bolts. Each C is made up of six C shaped copper plates pressed together with pre-impregnated epoxy glass insulation between the copper plates. For joining the two C's to make one picture frame type TF coil, at the open ends of C, all six plates are tapered to accommodate the respective open ends of small C which has similar tapering. The tapered portion of each C is named as Fingers. Each Finger in both the C's is silver plated having dimension of 160 mm x 160 mm x 6.5 mm thickness and 6.5 mm apart from each other. During the dis-assembly of Toroidal magnetic field (TF) coils, it was realised that 6 numbers of TF coils (Coil No. 2, 3, 8, 9, 17, and 20) are damaged at the fingers joints of two C's sections constituting a TF coil. The copper material have been melted and eroded mainly at the edges of fingers joints of small –C and big C of TF coils especially in the middle fingers. Large depositions of carbon have been found near melted copper. The ADITYA-U team has found in house technique of refurbishing these TF coils. After repairing the damaged TF coils, they are assembled one by one on Test Stand by joining both C's sections and the electrical parameter testing (Resistance and Inductance) of these coils have been carried out. The resistance and inductance measurements of each damaged coil after repairing showed that electrical parameters are within satisfactory limits and are in good condition to be reused again. This remarkable task has saved lot of cost and time for ADITYA-U re-assembly. The details of damaged TF coils refurbishment and its electrical parameters measurements will be discussed in this paper.

Study of Argon Line Emissions in ADITYA-U Tokamak using Spectroscopic Diagnostic

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Abstract

Spectroscopic diagnostics are widely used on high temperature plasma devices to study the impurity behavior inside the plasma as the impurities influence plasma properties through fuel dilution and radiation loss. Spectroscopic diagnostic also provides measurements of plasma rotation velocity and impurity ion temperature from the Doppler shift and Doppler broadening of spectral line emissions, respectively [1]. Impurities starting from low to high Z may be intrinsic, coming out from the plasma facing components due to its interaction with plasma particles, or extrinsic, injected into the plasma in trace amount either for diagnostic purposes or to modify the edge plasma properties. In Aditya-U tokamak [2], a crystal spectrometer has been recently designed and developed to observe Helium like argon spectral line emission at 3.9494 A from the plasma to obtain the core ion temperature and rotation velocity. Considering the Aditya-U tokamak electron density, n_e, (1- 4) x10^{19} m^{-3} and electron temperature, T_e, (300 - 700 eV), the argon gas puffing experiments have been carried out to understand the requirement of the amount of argon puff into the plasma for recording the detectable X-ray emission by crystal spectrometer. During these experiments, the survey spectrum of argon lines in visible wavelength range has been recorded using a 0.5 m spectrometer. Moreover, spatial profile of spectral emission from neutral argon has also been monitored using a high resolution 1 m (*f*/8.7) spectrometer [3]. The influx of argon into the tokamak has been obtained from the observed spectral line. Moreover, the radial profile of emissivity of Ar line has been modelled using 1D impurity transport code, STRAHL. Using this analysis, the estimation of the intensity of Helium like argon line emission in X-ray range is done to understand its detection by crystal spectrometer. In this presentation, the observation and analysis of argon spectral lines in the visible range and diagnostic details of x-ray crystal spectrometer in Aditya-U tokamak will be discussed.

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Real-time Horizontal Plasma Position Control in ADITYA -U

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Abstract

The ADITYA-U tokamak (R_0 = 0.75 m, a = 0.25 m) is designed to have shaped plasmas in both single and double null diverter configurations. It is quite well known that sustaining a shaped plasma in tokamak requires very good plasma column position control, both horizontal and vertical. A FPGA based proportional–integral–derivative (PID) based control system has been designed and operated to achieve horizontal plasma positions control in ADITYA-U tokamak. In this control system, the transfer function model of control power supply and different position diagnostics has been incorporated such that whole system fulfils the stability criteria of the whole control system. Furthermore, the different interlock provisions have been incorporated to take appropriate actions to limit the power supply parameters beyond the designed values. The complete system has been rigorously tested with sample signals before implementing to the ADITYA-U plasma discharges. The control system is integrated to the composite plasma control system of ADITYA –U. Different P, I & D parameters were applied to control the real time horizontal plasma position in ADITYA-U tokamak. The complete design, installation, operation, training of the system along with all the relevant testing will be presented in the paper.

Large Amplitude Ion Acoustic Solitons in Warm Negative Ion Plasmas with Maxwellians Electrons.

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Abstract

In this paper we have presented ion acoustic Solitons (IASs) in multicomponent plasma containing hot negative and positive ions with Maxwellians electrons distribution. We have studied the characteristics and occurrence of IASs in negative ions plasmas employing Sagdeev pseudo potential technique. We have investigated the dependence of different parameters on the characteristic of solitons. It is investigated that the positive (compressive) and negative (rarefactive) potential solitons are simultaneously exist in system. The effect of ions temperature ratio (σ_1 and σ_2) and negative ion concentration (α) on the soliton are discussed in detail. The present model is applied to study the large amplitude IASs in the plasmas containing (H^+, H^-), (Ar^+, F^-), (CS^+, Cl^-) and (Xe^+, F^-). This investigation may be helpful to understand the solitons in laboratory and space plasma, where negative ions are present with thermal electrons.

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Preliminary Study of Supersonic Molecular Beam Injection in ADITYA-U Tokamak

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Abstract

The first indigenously built ADITYA tokamak has been upgraded to ADITYA-U tokamak (R_0 = 75 cm, a =25 cm), having feasibility of having shaped plasma operation with divertor coils. In ADITYA-U tokamak plasma breakdown is generated using hydrogen gas as a fuel and toroidal electric field produced with central solenoid Ohmic transformer coils. Toroidal magnetic field along with some other magnetic field is used to confine the plasma. Along with other issues related to operation of tokamak, fuelling is an important issue. We can fuel the tokamak by mainly three technologies such as Gas Puff, SMBI and Pallet Injection (PI). In ADITYA-U tokamak we are currently having Gas-Puff and supersonic molecular beam injection (SMBI) for fuelling Hydrogen gas in the vacuum vessel. In this paper the details of SMBI system installed in ADITYA-U tokamak along with the preliminary analysis of the data with SMBI discharges will be presented.

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Experimental Investigation on Electron Temperature Gradient Driven Instability in the Curvature Magnetic Field of MPD

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Abstract

This paper presents a detailed study on the controlled experimental observation of temperature gradient driven instabilities in an inhomogeneous six pole cusp magnetic field generated by an in-house developed multi-pole line cusp magnetic field device [Patel et al., Rev. Sci. Instrum. 89, 043510 (2018)]. The device is composed of six axially symmetric cusps and non-cusp (in between two consecutive magnets) regions. The observed instability has been investigated in one of these non-cusp regions by controlling the radial plasma temperature gradient with changing pole magnetic field which is a unique feature of this device. It has been observed that the frequency of the instability changes explicitly with the temperature gradient. Moreover, the scale lengths of plasma parameters, frequency spectrum, cross-correlation function, and fluctuation level of plasma densities have been measured in order to identify the instability [1, 2]. The cross field drift velocity due to fluctuation in plasma parameters has been measured from the wave number-frequency S (kz, ω) spectrum and verified with the theoretical values obtained from temperature scale length formula [2].

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Spectroscopic Diagnostic for Magnetic Field in Tokamak Plasma

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Abstract

Externally applied toroidal magnetic field (B_T) is not sufficient enough for plasma confinement in Tokamak [1]. In order to achieve the equilibrium between the plasma and magnetic pressures, a poloidal magnetic field (B_θ) in addition to the applied B_T is needed. The field B_θ is produced by the plasma current itself flowing in the toroidal direction. Experimental measurement of B_θ is necessary to understand the current distribution and plasma confinement in Tokamak. However, usage of magnetic- or B-dot probe [2] is restricted in the fusion device like tokamak except in the edge region due to the high heat flux of plasma. Spectroscopic method is widely used diagnostics in such case [3, 4].

General spectroscopic method for magnetic field measurement in plasmas is based on Zeeman splitting of spectral line [4]. However, in high temperature Tokamak plasma, Doppler broadening is higher than the Zeeman splitting due to magnetic field and it completely smears out the splitting in the line-shape. In such situation, polarization properties of the σ-component of the emission line (component with obeying selection rule $\Delta m = \pm 1$, where Δm is the difference in magnetic quantum number of the upper and lower levels of the transition) can be utilized for B_θ measurement in Tokamak [5].

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Study of ADITYA-U Tokamak Plasma using Fast Imaging Camera

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Abstract

ADITYA is a first indigenously built medium sized air core tokamak that has been upgraded in ADITYA-U, with a major radius of 0.75 m and a minor radius of 0.25 m. It is routinely operated and produced plasmas are studied using various diagnostics. Fast imaging diagnostic is one of them that enable the viewing of the plasma. The diagnostic system consists of a Phantom v7.1 fast camera coupled with an imaging fiber bundle that views the tangential poloidal cross section of the plasma column. High resolution images (256x256) at frame rate 10000 to 26000 FPS are captured in this diagnostics. The evolution of the plasma and its movement is studied. Plasma interaction with plasma facing components (PFCs) is observed and studied through the enhancement in local visible radiation.

This study involves discussion on plasma wall interaction, estimation of the temporal and spatial profile of the plasma column and location of the resonance layer of ECRH plasma by using post processed images. Dust particle study in terms of their speed and trajectories and pallet injection in ADITYA-U Tokamak are discussed in this work.

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Design and Development of Different Analog and Digital Electronic Circuits for ADITYA Tokamak

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Abstract

Every discharge in ADITYA tokamak is initiated with starting of the toroidal field power supply (TFPS) which also generates trigger pulses at ~ 2.4 second, ~ 40mS and ~ 4mS before loop voltage.
These three trigger pulses operates/controls/initiates various sub-systems of ADITYA tokamak like gas puffing system, data acquisition system, radio frequency pre-ionization and heating system, plasma position control etc. Hence an appropriate trigger pulse from the three fixed time-stamped trigger pulses originating from the power supplies need to be chosen and manipulated according to the experimental requirements. To facilitate the smooth communication among systems lying in tokamak hall and the power supplies kept at ~30-40m from Tokamak hall, an optical fiber (replacing old electrical cables which are prone to noise pick up & false trigger) based trigger transceiver system with programmable delay setting has been designed, developed and installed in ADITYA. An FPGA based (with integrated microblaze soft processor) trigger and timing control system with remotely programmable delay features through LabVIEW based Graphical User Interface (GUI) has been developed which has replaced old CAMAC system.

To measure various plasma parameters, various diagnostics are used. The spectroscopy diagnostic is one of the important diagnostics to know about impurity content in plasma and other plasma parameter like electron and ion temperature, electron density, Z-effective. A chassis based signal conditioning electronics has been developed for spectroscopy diagnostic where signal emanating from Photo Multiplier Tube (PMT) detector is detected. The high voltage control electronics has been designed and developed for biasing PMT detector. A microcontroller based 8 channel standalone controller for 8 Channel compact PMT array for Z-effective measurement has also been developed. The design is equipped with independent bias control of 8 channels of PMT array with potentiometers and displaying control voltage on LCD panel using single 8 channel ADC.

A 3U size 4 channel VFC/FVC based analog/digital fiber optic transceiver modules with high voltage isolation for real time feed-back communication of signals like radial plasma position, plasma current, and loop voltage between ADITYA & APPS has been designed and developed.

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Homodyne and Heterodyne Microwave Interferometer Systems for ADITYA Tokamak.

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Abstract

A microwave Interferometer is a widely used and established system for line integrated plasma density measurement of fusion plasma. It is based on a principle of phase shift, a microwave beam will experience a phase shift while traversing through the plasma with respect to the reference arm. A change in phase is related to the change in refractive index and change is refractive index is related to the change in plasma density. Hence, by knowing the phase shift of the microwave beam coming out of the plasma will provide the information of the plasma density. There are two types of Techniques (1) Homodyne (2) Heterodyne. In a homodyne, only one source of frequency has been used as a probing frequency and for the down conversion of a signal coming out of the plasma. In the case of Heterodyne, two frequency sources of slightly different frequencies have been used. As a result, even in the absence of the plasma the I.F (Intermediate frequency) frequency is not zero while in the Homodyne it is zero. In the case of a Homodyne ambiguity in the phase change direction is remain while in Heterodyne this ambiguity has been resolved. Both the systems have been established for the line integrated density measurement at Aditya Tokamak. A Homodyne System of 100 GHz is a multi-chord system, 7-channels for Aditya and 6-channels for Aditya-U. Due to the multi chord, a radial profile for plasma density can be obtained. A Heterodyne system of 140 GHz is a single channel phase-locked loop system. A heterodyne system gives a real time plasma density. This real time signal can be used for density control and feedback.

Non-linear Time Series Analysis of ADITYA-U Plasma discharges

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Abstract

Tokamak Plasma dynamics are well known to be turbulent and chaotic in their behaviour even in the flat top regime. To study this chaotic behaviour of the Plasma in the upgraded ADITYA-U, we have applied several non-linear time series analysis tools to understand their behaviour. To this end embedding dimension of the data for impurity spectral line emission, mirnov oscillations and floating potential is calculated and analyzed for their non-linear behaviour and compared with each other for obtaining insights into the dynamics.

Investigations of Plasma Disruption Prediction in Tokamak using Machine Learning Tools

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Abstract

Tokamak device is designed to confine the plasma which is the fourth fundamental state of matter. Tokamaks are designed as doughnut-shaped with the aim to retain the heat of plasma and to avoid interaction of plasma with containment vessel materials. However, plasma instability, also known as disruptions can interfere with the process. Disruption causes damage to the confinement of tokamak. To date, disruptions are inevitable in tokamak devices. Prediction of disruption post analyzing different parameters is performed in past years. As disruption prediction is a multi-dimensional problem, it is a complex task and hence, comprehensive model for disruption prediction is not available. Lack of disruption prediction comprehensive model has led the development of data-driven prediction algorithms like machine learning algorithms. Objective of our research is to develop a numerical tool for time series prediction of plasma disruptions for tokamak diagnostic input-output data using state of art machine learning algorithms as proof of concept which will be compatible with the real time hardware based solution. Moreover, to enhance the intuitive understanding and association of input data of disruptions, a 2D visualization tool will be developed. The proposed system will use color coding of normalized parameters, and decision surfaces projecting probable disruption space for offline and real time dataset. The system will be validated for time series prediction of plasma disruption. The presentation will include the demonstration of machine learning tools developed for disruption prediction at Dharmsinh Desai University.

X-Ray Diagnostics Systems in ADITYA/ADITYA-U Tokomak.

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Abstract

X-Ray diagnostics (XRD) systems routinely used in tokomak to study various parameters of plasma discharge. XRD systems have been designed for two energy interval A) soft X-Ray Range (0.1keV to 30keV), B) Hard X-Ray (100KeV to 10MeV), which cover intensity as well as energy measurements in Aditya/ADITYA_U. Intensity measurement of Soft X-Ray provides valuable information about chord average electron T_e measurement, plasma position, magnetic island, mode structure, MHD activity and instability can be studied. Whereas pulse height diagnostics provides valuable information of energy distribution of in soft X-Ray range. In High energy regime, interaction of runaway electron with first wall and limiter component produces hard x rays. It is necessary to Study of temporal evolution of Hard X –Ray photon energy during plasma discharge which is helpful for corrective measures for to stop wall plasma interaction and in improving the quality of plasma discharge. Pulse height measurement of HXR Radiation gives information of maximum energy of runaway electron while studying maximum energy of Hard X-Ray. These reports gives brief introduction of individual diagnostics system and the results obtained from the X-Ray diagnostics system used so far in ADITYA and ADITYA-U.

Self-Inductance of Finite Straight Wire Using Biot-Savart's Law

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Abstract

For most applications the inductance of a straight conductor is ignored as the inductance is very low and for most applications it is too small to have any significant effect on the circuit. However as frequencies rise into the microwave region, even the inductance of short lengths of wire can have a significant effect. A wire can be found in any circuit, and thus, it is important to know how much inductance it is contributing.

The inductance of a straight conductor of finite length is calculated starting from the first principle. It is not easy to find an expression for the inductance of a straight piece of conductor of a finite length. We calculate the internal and external inductance separately and then sum up both to get total inductance. The internal inductance is calculated using the energy stored in the conductor in form of magnetic field. External part of inductance is calculated using usual technique but it leads to little difficult mathematical integration.

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[2]. Introduction to Electrodynamics by David J Griffiths

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[4]. Chapter 9- Sources of Magnetic Fields(MIT Physics Lecture Note)

Self-Inductance of Circular Loop using Biot-Savart's Law

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Abstract

In conventional tokamaks all the required magnetic fields are produced by electromagnets. They are simply coils of different sizes and shapes made up of copper and large electric currents are being passed through them to produce the desired magnetic field. The coils have inductance and which needs to accurately known for designing the power supplies which drives current through them. Therefore inductance values of these coils are required to be calculated in order to design the complete system. Although formulas for inductance calculations are available, but their derivation is not easily available and also the derivations are not very straightforward.

This thesis presents the derivation of inductance of a circular loop coil from the first principle, i.e., from the Biot Savart's Law. Thus, instead of using Magnetic vector Potential, the first principle is used which makes the work unique. It is easy to calculate the magnetic flux penetrating through enclosed surface using Biot-savart's law but the mathematics part which lead to triple integration and elliptical integration is bit difficult to solve using usual integration technique, thus from different sources and books, list of some useful formula to solve some advance integration are listed here and used as well.

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[5]. Chapter 9- Sources of Magnetic Fields(MIT Physics Lecture Note)

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Design of Interlock Systems for Real Time Control of Plasma Events and Experiments in Aditya Tokamak

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Abstract

Aditya-Upgrade tokamak [1] has been operational for the last couple of years. In the upgraded machine many experiments have been carried out successfully. This paper discusses the design and development of cost effective and simple circuits based on 8051 microcontroller for different interlock systems to achieve smooth operation of the machine and thereby by to have long and consistent plasma discharges. The primary focus in the initial phase had been to design and develop an interlock circuit for stopping the pre-determined and repetitive (say one pulse in every …seconds) hydrogen gas puffing pulses upon encountering any kind of disruption in the discharge pulses during the break-down or current ramp-up phases. Continued puffing of the hydrogen gas post-disruption of the discharges causes loading of the vessel wall with hydrogen gas, which subsequently hamper the next plasma discharge, thus resulting in a series of disruptive discharges.

To achieve the requirement of stopping unwanted gas puffing to the chamber, a microcontroller based circuit is designed and developed to control puffing of gas-feed pulses by taking feedback from the discharge itself. After the onset of loop voltage the plasma current reaches to its flat-top value of ~ 100 kA in about 30 – 40 ms in typical discharge of Aditya Upgrade. The circuit developed constantly monitors the value of plasma current (Hardware integrator output of Rogowski coil) at 2 ms time interval after about 30 ms of application of loop voltage. The circuit is programmed in such way that it allows the gas feed valve to insert gas according to the pre-set configuration as long as the plasma current value remained above a threshold value (~ 60 kA for typical discharges of Aditya Upgrade). The steering of the gas-feed voltage to the gas-feed valve is controlled through an analog switch. If the plasma current gets disrupted at any point of time from 30 ms, the circuit shuts off the analog switch and stops the gas feed into the vessel. The circuit is tested successfully for the required stoppage of gas puffing upon encountering any discharge disruption. Further, this deployment of timely stoppage of gas feeding has resulted in drastically reducing the gas loading on the vessel wall, thereby helping significantly in achieving repetitive long plasma discharges in Aditya Upgrade tokamak. Also, various modified versions of the same circuit (modifications hardware as well as software) with an additional trigger and Gate control outputs have been developed for neon gas feeding, ECRH, pellet injection and electrode biasing experiments where similar interlock feature was required. The paper also proposes an FPGA/DSP based advanced design for high speed multichannel interlock matrix system to accommodate all future requirements.

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[1] Ghosh J. et al 2016 Upgradation of ADITYA tokamak with limiter configuration to ADITYA upgrade tokamak with divertor configuration 2016 IAEA Fusion Energy Conf. (Kyoto), 17-22 October, 2016; (FIP/P4–46) Nucl. Fusion

Limiter and Divertor of ADITYA-U Tokamak

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Abstract

The ADITYA tokamak (R0 = 75 cm, a = 25 cm) having a limiter configuration has been upgraded to a state-of-art ADITYA-U tokamak [1] with divertor configuration to support the future Indian Fusion program. Limiter and Divertor are the most important subsystems of any tokamak. They are used to form the plasma boundary inside the tokamak and restrict the high-temperature plasma from hitting the vacuum vessel wall and provide protection to in-vessel components. A much better configuration in terms of energy and particle exhaust can be achieved in the divertor configuration, where the outermost magnetic field flux lines are opened up to make them strike on a chosen divertor target.

The positions of the limiter and divertor plate locations inside ADITYA-U vessel have been determined based on numerical simulation of the plasma equilibrium profile. The new machine accommodates three different configurations of limiter and divertor assemblies. ADITYA-U is having a toroidally continuous inner limiter with a poloidal extent of 1/8 of poloidal periphery of vessel. There are two outer limiter assemblies installed at two different toroidal locations with poloidal extent of 1/4 of poloidal periphery of vessel. The divertor plates are toroidally continuous structures located at upper and bottom halves of the vessel. In addition, one pair of the safety limiter which is a poloidal ring of graphite tiles placed inside vessel (at toroidal) symmetrical locations. Initially graphite will be used as plasma facing material (PFM) in all the limiter and divertor plates. Shaped graphite tiles have been fixed on specially designed support structures made out of SS-304L inside the torus shaped vacuum vessel. The dimensions of the limiter and divertor tiles are decided based on their installation inside the vacuum vessel as well as on the total plasma heat loads falling on them. Depending upon the heat loads; the thickness of graphite tiles for limiter and divertor plates is decided.

As the vessel flange dimensions of ADITYA-U are not suitable for any human to go inside the vessel, installation of limiter tiles along with integration of other in-vessel components on the high field wall side was very challenging. After the installation of plasma facing components and diagnostics in ADITYA-U, the Phase-1 plasma operations were initiated in December 2016. Ohmically heated circular plasmas supported by filament pre-ionization with plasma parameters Ip ~100-177 kA, duration ~100-335 ms, with a maximum toroidal field ~1-1.4 T, chord averaged electron density ~2 - 4 × 10^{19} m^{-3} and plasma temperature 300 – 500 eV, have

been obtained. The divertor plates will be installed during next phase of upgradaton. In this paper, ADITYA-U limiter and divertor conceptual design, fabrication and installation along with challenges faced will be presented.

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Forecasting of Disruption in ADITYA-U Tokamak

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Abstract

The first indigenously built ADITYA tokamak has been upgraded to ADITYA-U tokamak (R_0 = 75 cm, a =25 cm), having feasibility of having shaped plasma operation with divertor coils. In ADITYA-U tokamak plasma breakdown is generated using hydrogen gas as a fuel and toroidal electric field produced with central solenoid Ohmic transformer coils. Toroidal magnetic field along with some other magnetic field is used to confine the plasma. Disruptions in tokamak, is a sudden loss of magnetic confinement of plasma. A huge amount of plasma current abruptly terminates in a few milli-seconds. Therefore, a huge amount of induced electric field is generated, which makes the electron highly energetic (Runaway electrons) and eddy current flows through the vessel. Hence, due to disruption there is a chance of severe damage to the system. Avoidance of disruption [1] and real time mitigation is a very important field of work in tokamak. There are many possible causes for disruption. The most important reason is growth of m=2 mode of instability. Growing instability at q=2 rational surface gives the major disruption. It is observed that higher q-value (q=6) at the edge gives the sharp falling of plasma current. Where the fall of plasma current is slower for lower q-value (q=3) at the edge [2]. The amplitude of oscillation of m=2 mode also decreases with higher q-value at the edge and increases with lower q-value at the edge. The work is focused on finding a strong correlation between q_{edge} and disruptions in ADITYA-U tokamak. Details of the study will be presented.

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Recent Advances and Upgradation of ICRH System on ADITYA-U

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Abstract

ICRH system on Aditya-U is in the upgradation phase. Several improvements are considered and are being carried out at present. To increase the injected power to plasma up to 500 kW, two feeding antenna each injecting 250 kW are considered. Due to space constraint in a single radial port, two different toroidal locations are considered. To feed high power at high vswr, the layout and line size are upgraded. A new lightweight Aluminium coaxial transmission line is developed indigenously with the help of domestic industries and 9 inch size Aluminium 6061 T6 alloy based seamless transmission line is developed for the first time in India. The lines are also tested for 86 kV DC test for electrical breakdown strength and pressurized tests are conducted at 4.5 bar dry air.

A new lightweight fast actuating probe system is developed to support ICRH experiments. It actuates 100mm length in vacuum vessel with a speed of 1.5m/s scanning the last 10mm of scape off layer region in less than 7 ms. The probe measures absolute and gradient of electron density, electron temperature and radial electric field in SOL region of Aditya-U tokamak in a single discharge.

A new center-fed ICRF antenna is designed to excite Fast Waves at plasma edge to propagate in to high density plasma and finally absorbed at various resonances that present inside the bulk. For an efficient antenna plasma coupling, a foremost requirement is to maximize antenna loading. This in turn dictates the maximum instantaneous RF voltage appearing on the antenna structure. Part of the loading goes in to the plasma density and thickness of evanescent layer in front of the antenna, whereas the other prominent part is the design of antenna structure. In the present design, the later part is investigated to optimize various structural components of the antenna so as to maximize the loading, minimize the surface current density, structure of electric field etc. In the present design, poloidal symmetry of wave electric field across the poloidal flux is taken into consideration.

A simplified direct injection antenna feeder is developed and tested to replace earlier multi-component VTL with private UHV pumping mechanism. This concept is being tried to minimize the complexity, reducing presence of dielectrics, and reducing local electric field gradient for better power feeding to antenna. In this paper, these recent developments and current status will be discussed in detail.

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Mechanical Design of a Pressurized Variable Pre-matching Stub for ICRH System on ADITYA-U

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Abstract

High power radio frequency waves in the megahertz frequency range have a wide range of application like tokamak fusion reactor, accelerators, aerospace and defense sector. The RF waves can be transmitted via rigid coaxial transmission lines (TL) at high power. In order to radiate the desired wave spectrum from an antenna and reduce the vswr in the unmatched TL section just after the antenna feeder, a Prematching stub is often required. The prematching stub necessarily required to be variable instead of fixed, to adjust the vswr depending on frequency of operation and plasma condition from shot to shot basis. Furthermore, it is also imperative to pressurize it to reduce the chances of high voltage node arcing in the unmatched TL sections and components.

A new pressurized and mechanically variable Stub is designed to cater this requirement. The pressurizing capability is checked in house and a pressurizable multipin feedthrough is also designed with the help of local vendor. Feedthrough are successfully tested up to 4.5 bar for 6 hours without any detectable leaks or reduction in holding pressure. The detailed design aspect shall be discussed in this poster.

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Spectroscopic Diagnostics for Tokamak - Journey through ADITYA & ADITYA-U

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Abstract

Impurity behavior study in the tokamak plasma is very important to achieve fusion grade tokamak plasma. Spectroscopic measurement of the radiation coming out from tokamak is one of major way to study impurity behavior and then ADITYA tokamak and its upgraded version ADITYA-U was complemented with many such diagnostics. In the earlier phase of ADITYA was complemented visible spectrometer to monitor H_α line radiation. Subsequently, ADITYA tokamak was equipped with space resolved EUV (2 - 70 nm) known as GIM, and VUV (110 - 400 nm), known as NIM for the monitoring of emission from highly ionized low Z, like carbon and oxygen, impurities. The space resolved measurement in both systems was based on scanning mirror and then profile was obtained by scanning the mirror into new radial location in shot by shot basis. Visible spectroscopy system was also later upgraded by including the PMT based system to monitor the H_α, OI and CIII and visible continuum emissions. With the progress of time, 0.5m spectrometer based visible survey spectroscopy system, a high resolution visible spectroscopy system having 1 m long spectrometer and CCD camera capable of multi-track measurement and a VUV survey spectroscopy system (10 - 180 nm) were added up to the spectroscopic diagnostics on ADITYA tokamak. Using these additional diagnostics, the study on drift Alfven mode, oxygen and iron impurity transport and neutral particle dynamic was carried out and interesting results were obtained. The spatial profile of He-like carbon and oxygen were obtained by the NIM system and temporal evolution of spectral line from Fe^{14+} ion was regularly monitored using GIM system. The behavior of effective charge of plasma with various plasma parameter and wall conditioning has been also investigated. In the final phase of ADITYA tokamak, PMT array based diagnostics with fast time response to measure the spatial profile of H_α and impurities emissions was developed and enables to obtain the breakdown location for the ADITYA tokamak Ohmic discharges and also to study the details of the contribution from various atomic and molecular processes in the H_α emission by modelling it using DEGAS2 neutral transport code. In ADITYA-U tokamak, along with those diagnostics, new diagnostics for Z_{eff} profile measurement using visible continuum profile, radial profile of toroidal and poloidal rotation using Doppler shift of carbon emissions and spatial profile of neutral temperature by using Doppler broadened H_α line profile under the influence of Zeeman Effect were implemented. Using these diagnostics, toroidal rotation reversal, poloidal asymmetry in neutral temperatures was found out and investigated. Along with that, new diagnostics, like a VUV survey spectrometer (30 - 300 nm) to monitor VUV spectral emission, crystal spectrometer for the core ion temperature and toroidal rotation and NIR spectroscopy system to measure the electron temperature and plasma flow in the limiter and SOL region is under various stages of development. The details of these diagnostics and the results obtained using those will be presented.

ADITYA → ADITYA UPGRADE TOKAMAK