Upgradation of Aditya Tokamak with Limiter Configuration to Aditya Upgrade Tokamak with Divertor Configuration

J. Ghosh[1], R.L. Tanna[1], S.B. Bhatt[1], P.K. Chattopadhyay[1], K. Sathyanarayana[1], Chhaya Chavda[1], C.N. Gupta[1], K.A. Jadeja[1], K.M. Patel[1], V.K. Panchal[1], Vijay Patel[1], Kulav Rathod[1], Sharvil Patel[1], S. Jaiswal[1], P. Chauhan[1], Vaibhav Ranjan[1], Rohit Kumar[1], Harshita Raj[1], Krishna Kumari[1], D.H. Sadharakiya[1], M.B. Kalal[1], D.S. Varia[1], Ramkrushna Panchal[1], K.S. Acharya[1], Nilesh Patel[1], K.N. Chaudhary[1], M.N. Makwana[1], K.S. Shah[1], D. Raju[1], R. Srinivasan[1], Deepti Sharma[1], S. Dutta[1], B.R. Doshi[1], M. Gupta[1], U. Baruah[1], A.Vardharajulu[1], Amita Das[1], Y.C. Saxena[1], D. Bora[1], R. Pal[2], S. Saha[3], A.V. Apte[4], D.R. Patel[4] and Shell-N-Tube Team[5].

[1]Institute for Plasma Research, Bhat, Gandhinagar 382428, Gujarat, India.
[2]Saha Institute of Nuclear Physics, Bidhannagar, Kolkata, India.
[3]Variable Energy Cyclotron Center, Bidhanagar, Kolkata, India.
[4]Space Application Center, Ahmedabad, India.
[5]Shell-N-Tube Pvt. Ltd., Pune, India.

E-mail contact of main author: jghosh@ipr.res.in

Abstract. It is a well-known fact that small / medium-sized tokamaks have enormously contributed in design and development of large size tokamaks such as ITER for building fusion reactors in terms of physics, engineering and diagnostics. Small / medium-sized tokamaks are very convenient to develop and test new ideas, technologies and materials, which because of the risky nature cannot be done in large machines without preliminary studies, such as disruption studies, runaway studies, etc. The worldwide effort on magnetic fusion is devoted to the present generations of large tokamaks like DIII-D, TCV, EAST, SST-1 etc., which are operational emphasizing on divertor and tungsten wall ITER-like operation. However, there are very few (2 –3) small / medium-sized tokamaks operational around the world with divertor facility and technical capabilities to provide able support for operation and trouble shooting of these big tokamaks. Therefore, it has been planned to upgrade the existing ADITYA tokamak (R0 = 75 cm, a = 25 cm) successfully operated over 2 decades with more than 30,000 discharges into a state-of-art machine with divertor operation and very good plasma control to support the future Indian Fusion program. The scientific objectives of Aditya tokamak Upgrade include Low loop voltage plasma start-up with strong Pre-ionization having a good plasma control system. The upgrade is designed keeping in mind the experiments, disruption mitigation studies relevant to future fusion devices, runway mitigation studies, demonstration of Radio-frequency heating and current drive etc. This upgraded Aditya tokamak will be used for basic studies on plasma confinement and scaling to larger devices, development and testing of new diagnostics etc. This machine will be easily accessible and will be very useful for generation of technical and scientific expertise for future fusion devices. The installation of Aditya Upgrade tokamak is completed and plasma discharges will be initiated soon. In this paper, especial features of the upgrade including various aspects of designing and fabrication of new components for Aditya Upgrade tokamak will be presented.

1. Introduction

Confinement of high temperature plasma in a tokamak is a very promising approach towards the realization of fusion reactors for electrical power plants. During the last decade, one of the key drivers for the design and upgrade of the current generation of tokamaks has been the incorporation of advanced divertors. With the ITER [1] in horizon much of the worldwide effort on magnetic fusion is devoted to the present generations of large tokamaks like JET [2], DIII-D [3], JT-60 [4], ALCATOR C-MOD [5], TCV [6], EAST [7], K-STAR [8], SST-I [9], which are operational at present or going to be operational in near future emphasizing on divertor and tungsten wall ITER-like operation. Operational restrictions of large size tokamaks always demand small / medium-sized tokamaks to provide knowledge about different phenomena, which are comparatively less complicated and hence can be easily operated. It is a well-known fact that small / medium-sized tokamaks have enormously contributed in design and development of large size tokamaks for building fusion reactors both in terms of physics and engineering. Small / medium-sized tokamaks are very convenient to develop and test new ideas, technologies and materials, which because of the risky nature cannot be done in large machines without preliminary studies, such as disruption studies, runaway studies, etc. However, there are very few small / medium-sized tokamaks operational around the world with divertor facility and technical capabilities to provide able support for operation and trouble shooting of these large size tokamaks.

Keeping the above mentioned advantages of small tokamaks in mind, the existing ADITYA tokamak [10] with limiter configuration is being upgraded into a tokamak with open divertor configuration to support the future Indian Fusion program. The ADITYA-U will be capable of producing shaped plasmas with the following major technical and scientific objectives: (a) Mid-size tokamak with divertor operation and higher duty cycle, (b) to carry out experiments which are not desirable in large tokamaks such as disruption and runaway mitigation, (c) the upgraded tokamak will be a test bed for new diagnostics mainly divertor diagnostics, (d) replaceable divertor and limiter tile material for plasma surface interaction experiments (e) for generating skilled manpower for future larger tokamaks. The ADITYA-U is designed to produce a circular plasma with plasma current ~ 150 – 250 kA, plasma duration of ~ 250 – 300 ms with electron density and temperature in the range of $3 – 5 \times 10^{19}$ m^{-3} and 500 eV – 1000 eV respectively. Further, it is designed to obtain shaped plasmas with plasma current ~ 100 – 150 kA, elongation (k) ~ 1.1 - 1.2 and triangularity ~ 0.45.

The paper is arranged as follows. Feasibility studies of upgradation are presented in section II. Section III describes the new subsystems. Brief description of disassembly and re-assembly is presented in section IV. Summary is presented in section V.

2. Feasibility Studies of Upgradation

The existing ADITYA tokamak (R0 = 75 cm, a = 25 cm) with a circular poloidal limiter, had been successfully operated over 2 decades producing more than 30,000 discharges. The design of ADITYA tokamak was to produce plasma in a toroidal vacuum chamber of rectangular cross-section with 20 toroidal magnetic (TF) coils capable of producing ~ 1.5 T at machine center and with 5 sets of Ohmic (TR) coils. A two pairs of poloidal field (PF) coils (BV1 & BV2) provided the equilibrium of circular shaped plasma, while two pairs of Fast Feedback (FFB) coils provided the radial control for sustaining it for few hundreds of milliseconds.

To keep the upgradation cost and time minimum, it is planned to use existing sets of toroidal magnetic field (TF) coils, poloidal magnetic field (PF) coils and the Ohmic coils. The main modifications to this existing tokamak are aimed at creating space for placing the extra poloidal field coils to obtain single and double null configuration in the upgraded tokamak. It was envisaged that if the existing vacuum vessel of ADITYA with rectangular cross-section is replaced by a vacuum vessel with circular cross-section, then new PF coils can be accommodated in the space generated near the corners of rectangular cross-sectioned vessel. The schematic drawings of vessel cross-section of existing ADITYA and new ADITYA-U inside a TF coil are shown in Figure 1. As shown in Fig. 1(a), by replacing the old vessel with new vessel of circular cross-section a clear space of dimension 90 x 70 mm can be made available at four corners to accommodate the divertor or the extra PF coils.

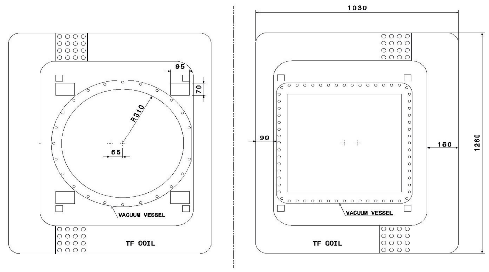

FIG. 1. Space gained in the corners by replacing the (b) existing vessel of rectangular cross-section with (a) new vessel of circular cross-section.

Utilizing this space for placement of new PF coils, the feasibility studies of obtaining single and double null (SN and DN) configuration has been carried out by plasma equilibrium reconstruction with equilibrium codes IPREQ and CEA code. These simulations suggested the possibility of obtaining ~ 150 kA plasma in SN and DN configuration in ADITYA-U by introducing two sets of new PF coils along with the above-mentioned existing sets of magnetic field coils. One set of two new PF coil can be placed in the inboard (high-field) side with an Ampere-turn of ~ 150 kA-turn for obtaining a stable double null configuration as shown in Fig. 2(b). In this case the plasma radius is reduced to ~ 15 cm. To increase the plasma radius in this configuration, 8 – 10 kA-turn is required on the outboard (low field side) at the available space as shown in Figure 2 (a).

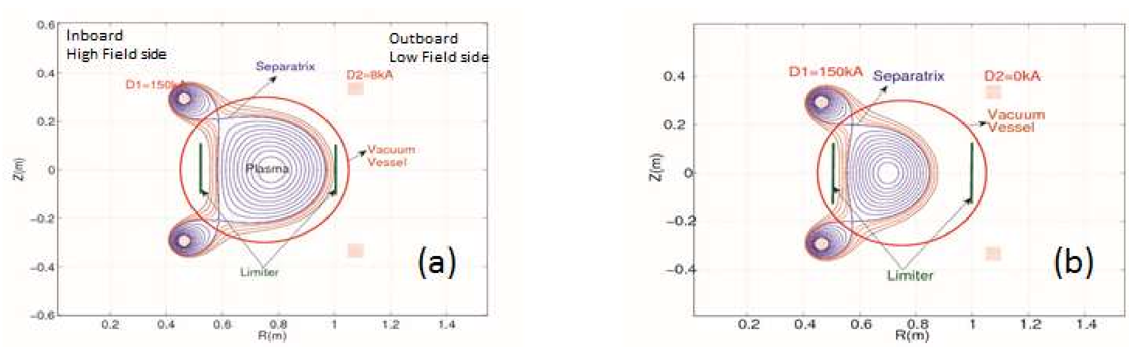

FIG. 2. Equilibrium flux surfaces for double null configuration in Aditya Upgrade (a) case I: 150 kA current in inboard divertor coils and 8 kA of current in outboard divertor coils (b) case II: 150 kA current in inboard divertor coils and 0 kA of current in outboard divertor coils.

Hence, based on the equilibrium calculations, 3 pairs of new PF coils have been installed in Aditya-U, The third pair of PF coil is included to change the location of the null point inside the plasma. In order to accommodate the new set of poloidal magnetic field coils (diverter coils) for plasma shapping, the existing vacuum vessel of rectangular cross-section has been replaced with a new vacuum vessel having circular cross-section. As it is decided to use the same toroidal magnetic field coils and associated structure of the existing machine, the major radius ~ 0.75 m and maximum toroidal magnetic field ~ 1.5 T remains unchanged in the upgraded machine.

3. System description of new subsystems installed in ADITYA UPGRADE

As discussed earlier, the existing vacuum vessel with rectangular cross-section has been replaced with new vacuum vessel having circular cross-section for accommodating the new divertor (PF) coils.

3.1. New vacuum vessel for ADITYA Upgrade

The toroidal vacuum vessel of circular cross section having inner diameter of 610 mm and outer diameter of 630 mm is designed and fabricated at M/S Godrej Industries, Mumbai. The toroidal vessel is made up of stainless steel (SS304L) in two halves. The main features of this vessel include 111 numbers of port openings of different sizes and shapes. These port openings are designed according to the diagnostics requirements. The vessel is bakeable up to 150° C. During designing, standard ASME ultra-high vacuum system design protocol are followed. The design model is developed in CATIA-V5 and analyzed in ANSYS for various mechanical and electromagnetic stress conditions. Tolerance of factor of 2 in wall thickness of torus is opted on the minimum of 5 mm as advocated by the ASME Code. The designed model has been analyzed for stress and deformation analysis in ANSYS workbench under ultra-high vacuum at 150° C with vessel is kept bonded to four fixed gravity supports as shown in Fig 3 (a). The maximum deformation of ~ 3 mm is observed at the upper-outer periphery of the vessel and the maximum stress ~ 128 MPa (SS304L Yield strength: 170 MPa) is developed at vessel-support joint. Further, the vessel model has been subjected to electromagnetic force ~ 0.3 MPa at 22° C ambient temperature under ultra-high vacuum with vessel kept rigidly bonded to four fixed gravity supports. In this case the maximum deformation of ~ 0.45 mm in vessel is observed at upper-outer periphery and the maximum stress of ~ 115 MPa is developed at vessel-support.

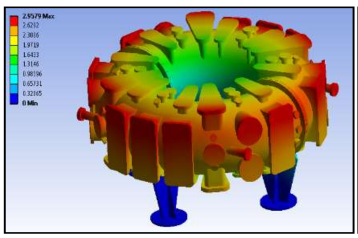

FIG. 3(a) Maximum deformation of the vessel at upper outer periphery.

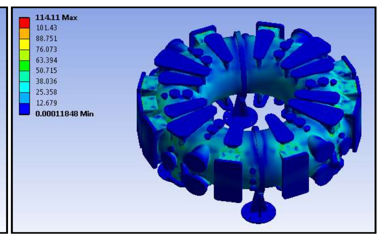

FIG. 3(b) Maximum stress of the vessel at vessel support joint.

The detailed design of the vacuum vessel has been subsequently provided to M/s Godrej for manufacturing. M/s Godrej prepared the fabrication plan and fabrication drawings. After being validated by the IPR personnel, the raw material inspection tests, weld procedure scheme (WPS), submission of PQR (Procedure Qualification Record) and Master Quality Assurance Plan (QAP) has been carried out. The First fabrication Mockup is performed to prepare a quadrant of the torus cell using proper die pertaining to the required dimension and tolerances provided by IPR. Each semi-torus has been fabricated in four stages (i) semi torus formation (ii) radial ports fabrication (iii) Top and Bottom ports fabrication (iv) final fabrication of wire seal ports and others surface treatments. After each fabrication stages of welding, Helium leak tests have been performed to test the leak rates of the fabricated volume.

Final tests have been carried out for complete torus assembly connected by double Viton O-rings using interspaced pumping. Helium leak test of all local joints accepted at Helium background ~ 9.5 x 10^{-10} mbar l/s. This has been followed up with ultra-high vacuum test of vessel and achieved a base pressure of 3 x 10^{-9} mbar using a turbo molecular Pump (1900 l/s). The vessel has been baked @150° C for 24 hours under high vacuum. Further, helium leak test for all local weld joints are carried out after bringing the vessel to room temperature. After carrying out all the required tests successfully, the torus vessel has been installed and positioned at its desired position within +/- 1 mm tolerance. The electrical isolation has been set and tested between two semi torus and torus to gravity supports.

After installation, the Helium leak tests of all local joints of the vessel has been performed and approved under Helium background of 2 x 10^{-9} mbarl/s. After machine assembly, the installation of four pumping lines have been completed. The trial run of vessel pumping has been tested by a single turbo pump achieving a base vacuum of ~10^{-7} mbar in absence of baking.

3.2. Plasma Facing components for ADITYA Upgrade

In the first generation of tokamaks, the plasma edge used to be limited where the outermost boundary is restricted by a material called limiter, with which the plasma remains in contact. However, in such configurations, a considerable amount of impurity is released into the plasma due to erosion of the material limiter caused by bombardment of the hot plasma particles on it, which leads to large impurity radiation losses. 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. Divertor leads to plasma-surface interactions occurring far away from confined hot plasma and particle flow prohibits impurities from entering the confined plasma. Divertors are also very efficient in removing the helium ash from the plasma [11].

ADITYA-U will be operated with PFC's made up of Graphite material during initial phase of operation with hydrogen plasma. It accommodates three different configuration of limiter and divertor assemblies. One is the safety limiter which is a poloidal ring of graphite tiles located at very close to vessel inner surface. The main limiter and divertor plate locations have been decided based on numerical simulation of the plasma equilibrium profile as shown in Figure 4 and 5. ADITYA-U will have 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 location 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. During the installation of PFC's in ADITYA-U, it is ensured that no metallic support structure of PFC's should exposed to plasma except PFC materials. Support structure assembly is made out of stainless steel 304L. Before installation of PFC's into the vessel, Graphite tiles are baked at high temperature of about 1000° C in high vacuum (10^{-5} torr) for 24 hour.

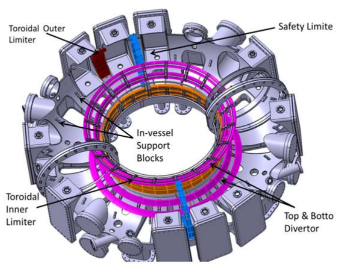

Fig. 4. Assembly of PFC inside ADITYA-U Vessel

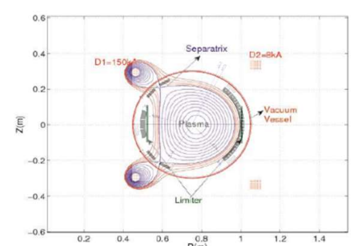

Fig. 5. PFC location based on numerical simulation.

In ADITYA-U, PFMs are designed to withstand ~ 1 MW heat load during pulsed plasma operation. Depending upon the above mentioned heat loads, the thickness of graphite tiles for limiter and divertor plates is estimated. The safety limiter consist of two semi-circular stainless steel rings joined at mid plane of vessel using ceramic insulators. Due to its location it receives low particle or heat load unlike poloidal limiter. 28 numbers of machined carbon tiles are mounted on the ring which face the plasma boundary forming radius of 28 cm. This safety limiter structure is mechanically fixed to the vessel at the four corner of the vessel. The main limiter of ADITYA-U consists of toroidal limiter at high field side with small poloidal extent, while low field side it cover approx. quarter poloidal plane of vessel cross section with small toroidal extent viz., inner poloidal limiter at (50 cm from centre of the machine) and outer poloidal limiter. Inner poloidal limiter is made up of 20 sub-assemblies consisting 60 graphite blocks of ~7 cm in raw. Each sub-assembly is consisting of 6 tiles and is fixed on support blocks, which is welded to vacuum vessel. Graphite cap is designed to avoid direct exposure of any metallic component. Outer poloidal limiter is accommodated between two radial ports which is consisting 9 tiles of 5 cm in poloidal plan in low field side locally with 8 cm of toroidal extent. ADITYA-U will be equipped with divertor during phase-2 machine operation. Divertor support is conceptually identical in numbers and construction with limiter support structure but varies with size and shape. To maintain position and alignment of in-vessel components ~100 support blocks welded to vacuum vessel. More than one third of these support blocks will be utilized to support limiter and divertor support structure.

4. Diverter coils design and installation

The ADITYA-U is equipped with 3 sets of Diverter coils, namely, one set of main diverter coil, one set of auxiliary diverter coil and one set of outer diverter coil. The main and auxiliary sets of diverter coils are placed in the inner (high-field) region of the tokamak. Each set consists of two coils placed symmetrically in the top and bottom locations equidistant from the horizontal mid-plane of the machine. The main diverter coils are made up of copper continuously transposed conductor with short-circuit current carrying capacity of 25 kA per second per turn. Each coil of the main diverter coil set contains 6 layers of conductor and capable of conducting 150 kA-turns for plasma shaping. The complete set of this main diverter coil (top & bottom) has a series resistance and inductance of 2 mΩ and 76 µH respectively.

5. Dis-assembly of existing ADITYA machine and Re-assembly of ADITYA-U

5.1. Dis-assembly of existing ADITYA machine

In order to accommodate new diverter coils and new circular shaped vacuum vessel, ADITYA has been dismantled up to base level. The dis-assembly task involved removal of various sub-systems of existing ADITYA viz., Diagnostic systems and pumping lines, top cooling water header and cooling connections, TF, TR and BV magnetic field coils with their bus-bar connections and support structures, vacuum vessel and its supports etc.

5.2. ADITYA-U Metrology

For performing the metrology, system having features for example, easy and mobile operation, 3D pointing accuracy ~ ±0.5 mm, robust geometric data analysis software support, online error evaluation, commercially approved is required. Such requirement is fulfilled by Advanced Optical Metrology System. Machine origin is derived using precise magnetic field measurement and best cylindrical fit. Magnetic center and geometric center derived agree well with each other within accuracy of ± 1 mm. For carrying out reliable measurements, a set of reference network is established for dedicated application.
Primary reference network: 4 target plates each at $0°, 90°, 180°$ and $270°$ on wall with known (x, y, z) / (r, θ, z) coordinates known with respect to machine origin, with accuracy $\pm$ 0.5 mm. This network is mainly used for deriving machine origin. References are accessible from any position in tokamak hall.
Secondary reference network: Toroidal angular degree reference network consist of 80 anodized plates embedded on hall surface, each $9°$ apart relatively. Angular degree accuracy $0.01°$ at the radius of 5000 mm and 6000 mm from origin. This reference network is used for alignment of diagnostics system.

Z-elevation reference network: Z mid-plane reference network consist of permanent punch mark on six rigid permanent structural columns of machine. Machine is symmetric about this mid-plane. Relative discrepancy between elevation references is ± 0.5 mm. The achieved assembly tolerances are tabulated in Table 1.

TABLE 1: ACHIEVED ASSEMBLY TOLERANCES

Key Components	Z Elevation	Concentric about origin	Angle
Magnetic field coils (TF, TR, BV, Divertor, Feedback coil)	± 1.5 mm	± 1 mm	Toroidal field coil $\pm \frac{1}{2}°$
Vacuum Vessel	± 0.5 mm	± 0.7 mm	$\pm 5'$
Structural Components (Central Structure, I-beams)	± 1 mm	± 1.5 mm	$\pm 30'$

5.3. The Assembly of ADITYA-U and Integrated power testing of coil systems

The assembly of ADITYA-U is successfully completed. The Re-assembly activities involved, installation of new circular shaped vacuum vessel, new buckling cylinder, refurbished TF coils (20 nos.), TR coils (11 nos.), BV coils (4 nos.), FFB coils (4 nos.), Diverter coils inner (02 nos.), Divertor coils outer (02 nos.) and auxiliary Diverter coils (02 nos.). The coils are accurately positioned within an accuracy of +/- 2 mm using ECDS. The busbar connections of all the coils (new and old) are assembled and clamped with proper supports. The TF, TR and BV coils cooling connection installation work is completed. All 20 nos. of TF coils, TR and BV coils cooling connections are tested with D.M. water at 1.5 kg /cm^2 inlet pressure.

The TF, TR and BV coils are successfully charged during integrated power testing. The TF coil assembly has been tested ~ 1.5 Tesla, The Ohmic coil assembly has been tested ~ 12.5 kA producing Loop Voltage ~ 20 V. The Vertical coil assembly has been tested ~ 3 kA. The TF coils displacement, fault current monitoring and magnetic field measurements is carried out during current charging. The movement of outer vertical leg of TF coils was recorded below 0.2 mm at full TF current. There was no fault current observed during the test. All the CTC conductor based coils (main Diverter coils, Aux. Diverter coils, outer diverter coils, FFB coils and single turn correction coils) are successfully heated up to 120° C for insulation curing.

6. Summary

The existing ADITYA operated over 2 decades with limiter configuration is being successfully upgraded into a tokamak with open divertor configuration to support the future Indian Fusion program. The new toroidal vacuum vessel has been designed with two semi-tori having electrical isolation at two junctions by keeping the major radius (0.75 m) and minor radius (0.25 m) similar as previous machine. The vacuum vessel is fabricated at M/s Godrej, India and tested successfully of having a local leak rate < 5 x 10^{-10} mbar.l/s and global leak rate < 5 x 10^{-8} mbar.l/s, UHV is tested up to < 10^{-9} mbar with vessel temperature > 150° C. After external testing, the vessel is installed successfully in ADITYA-U within ± 1 mm tolerances.

The existing ADITYA dis-assembly task had several critical challenges. Foremost, disassembling components like TF coils and its support structures, TR coils, buckling cylinder and vessel with great precaution, without damaging any component of the machine. Position, alignment and electrical parameters of every component was documented at every stage of disassembly in order to have a benchmark for re-assembly. Secondly, the few TF coils were found damaged due to long operation and repairing them posed a great challenge for them to be reused. The ADITYA- U team has successfully repaired those TF coils in house. After repairing the damaged TF coils, they are assembled one by one on Test Stand and the electrical parameters measurements of these coils after repairing are within satisfactory limits and are in good condition to be reused again. These coils were successfully charged close to design parameters after re-assembly. This remarkable task has saved lot of cost and time for ADITYA-U re-assembly. Last but not the least in-situ winding of new diverter coils was a big technical challenge, which was overcome by designing a diverter assembly line on site. Apart from overcoming all these technical challenges, we have put a lot of effort in meteorological studies of each component. All the major machine references like machine center were properly mapped. Each and every component was strictly placed back with a precision tolerances of ± 1 mm. A good deal of research was called out in order to minimize the error field with the help of various simulation tools as well. Finally, due to a rigorous team effort and great coordination, we were able to re-assemble entire machine including power testing was successfully completed in a year's time.

References

[1] LOARTE, ALBERTO, et al., "Effects of divertor geometry on tokamak plasmas", Plasma Phys. Control. Fusion 43 (2001) R183–R224.

[2] JET TEAM et al., "Fusion Energy Production from A Deuterium-Tritium Plasma In The Jet Tokamak", Nuclear Fusion, Vo1.32, No2 (1992).

[3] LUXON, J.L., et al., "A design retrospective of the DIII-D tokamak", Nucl. Fusion 42 (2002) 614–633.

[4] ISHIMOTOA, H., et al., "Advanced tokamak research on JT-60, Nucl. Fusion 45 (2005) 986–1023.

[5] GREENWALD, M. et al., "H mode confinement in Alcator C-Mod" Nuclear Fusion, Volume 37, Number 6.

[6] CODA, S. et al, "Electron cyclotron current drive and supra thermal electron dynamics in the TCV tokamak", Nucl. Fusion 43 (2003) 1361–1370.

[7] SONGTAO, W.U., et al., "An overview of the EAST project", Fusion Engineering and Design 82 (2007) 463–471.

[8] LEE, G.S., et al., "Design and construction of the KSTAR tokamak", Nuclear Fusion, Vol. 41, No. 10, (2001), 1515-1523.

[9] SAXENA, Y.C., et al., "Present status of the SST-1 project", Nuclear Fusion, Vol. 40, No. 6, (2000), 1069-1082.

[10] BHATT, S.B., et al., "ADITYA: The First Indian Tokamak", Indian J. Pure Appl. Phys. 27, 710 (1989).

[11] REITER, D et al., "Helium removal from tokamaks", J. of Plasma Physics and Controlled Fusion, Volume 33, No. 13, pp 1579-1699, 1991.