The ETE spherical tokamak project

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Abstract

This paper describes the general characteristics of spherical tokamaks, or spherical tori, with a brief overview of work in this area already performed or in progress at several institutions worldwide. The paper presents also the historical development of the ETE (Experimento Tokamak Esférico) project, its research program, technical characteristics and operating conditions as of October, 2002 at the Associated Plasma Laboratory (LAP) of the National Space Research Institute (INPE) in Brazil.

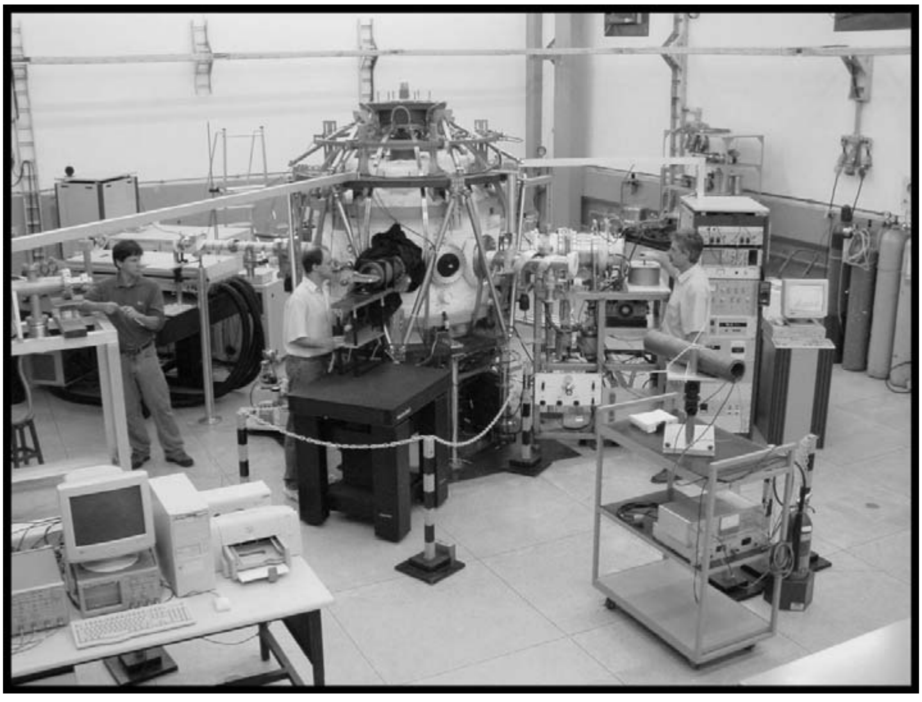

Figure 1: Overall view of the ETE spherical tokamak experiment.

1. INTRODUCTION

The ETE spherical tokamak (Experimento Tokamak Esférico) started its operational phase in late 2000 at the Associated Plasma Laboratory of the National Space Research Institute in Brazil (ref. Figure 1). It is devoted to the study of the operating regimes, confinement properties and current drive schemes in a low aspect ratio tokamak plasma configuration. This hot plasma magnetic confinement configuration [1] has been a subject of experimental study for a little over than one decade, offering the prospect of compact, low-cost fusion power plants. Besides exploring the properties of low aspect ratio tokamaks, the ETE experiment will allow diagnostics development and training in tokamak operation, with a research program open to collaboration with other institutions. The initial operation of ETE is limited to the study of inductively heated plasmas. In the near future, with the implementation of auxiliary heating methods, it will be possible to explore improved confinement regimes and means of current drive for steady-state operation. The large number of physics problems and technical issues that remain to be explored and solved in the spherical torus configuration allow to perform, even in a modest device, a research program that can contribute effectively to the international fusion research effort.

2. CHARACTERISTICS OF SPHERICAL TOKAMAKS

The principal characteristic of spherical tokamaks, when compared to conventional tokamaks, is the small value of the aspect ratio, that is, the ratio $A = R_0/a$ of the major to the minor radius of the plasma torus. Figure 2 compares a low aspect ratio torus ($A = 1.5$) with a large aspect ratio torus ($A = 3.4$). For values of $A < 2$ this simple geometric characteristic leads to several distinct plasma equilibrium properties.

The compact geometry has a small external inductance, with naturally elongated plasmas and expanded flux tubes at the up and down extremities, as illustrated in Figure 3. The magnetic field lines in this configuration run almost toroidally in the inner edge of the plasma and poloidally in the outer edge, as shown in Figure 4, leading to large values of the safety factor. Moreover, the lengths of the field lines are short in the bad curvature region (outer edge) and long in the good curvature region (inner edge), when compared with a conventional tokamak. Finally, the large magnetic field variations along the field lines result in a large fraction of trapped particles.

The characteristics of spherical tokamaks result in a stable plasma configuration with high plasma pressure in a low magnetic field. The heat and particle fluxes are naturally dispersed and there is the prospect of nearly fully self-generated current drive. These features, combined with

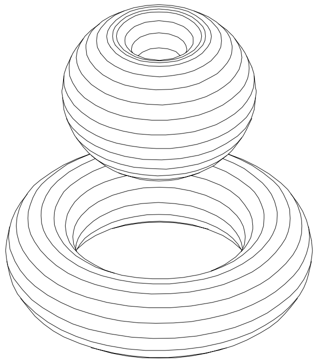

Figure 2: Comparison between low aspect ratio ($A = 1.5$, top) and large aspect ratio ($A = 3.4$, bottom) tori. The large tokamaks in operation have an aspect ratio $A \simeq 3.0$, a value considered large for the present standards.

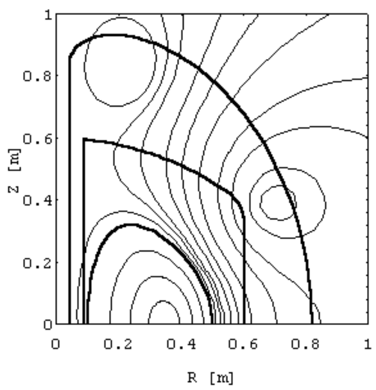

Figure 3: Equilibrium flux contours for a 0.22 MA plasma current and a 0.101 T vertical equilibrium field in ETE. The separatrix lies between 1.1 and 1.2 times the poloidal flux at the plasma edge (heavy line defined by the limiter position). The centerlines of the vacuum vessel and of the toroidal field coil are also shown.

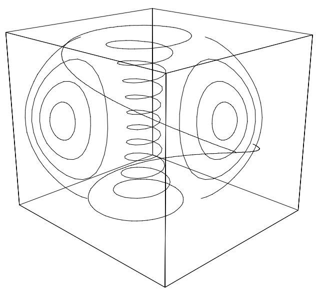

Figure 4: Magnetic field line on the boundary of a low aspect ratio tokamak plasma.

good energy confinement qualities in a small size device, make the spherical tokamak an attractive candidate for neutron sources (offers the possibility of trying different reactor components in a number of cheaper sources) and future fusion power reactors (leads to a cheaper and faster high gain path to reactors).

From the scientific point of view, the spherical torus provides unique conditions to studying the physics relevant to all magnetic confinement configurations, notably tokamaks. The good field curvature that dominates over most part of the particle orbits leads to magnetohydrodynamic stability at high plasma pressure. The longer time spent by trapped particles in the favorable curvature region and the small width of the banana orbits at high β (ratio of the plasma kinetic pressure to the magnetic pressure) may limit the effect of micro-instabilities. The large fraction of trapped particles results in a high bootstrap current contribution, and may lead to a fully bootstrapped configuration at high elongation. The high pressure driven sheared flow is favorable to the formation of transport barriers with reduced turbulence. The natural high elongation and dispersive scrape-off layer allow to address the questions of heat and particle handling at the plasma edge. Finally, the spherical torus at aspect ratio near unity provides an overlap with the spheromak and field-reversed configurations. Figure 5 shows the location of the spherical torus (ST) in the three dimensional (A, q, β) space, relative to the spheromak, field-reversed configuration (FRC), conventional tokamak, and reversed-field pinch (RFP) devices. The safety factor q is related to the rate of change of toroidal flux with poloidal flux.

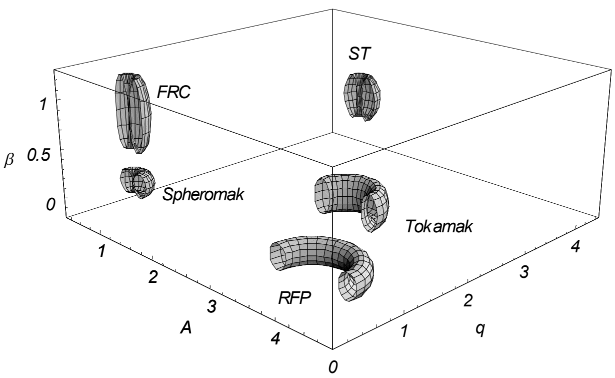

Figure 5: Axisymmetric configurations. Spherical tokamaks present both the strong toroidicity effects of compact tori (notably magnetic shear) and the good stability properties provided by the external toroidal field of conventional tokamaks.

Many of the properties of spherical tokamaks have been confirmed by recent experiments, yet the database is poor and much of the physics remains to be studied in detail. The expected high bootstrap current regimes have to be confirmed and the disruption immunity limits have to be established. Appropriate methods of pressure and current density profile control have to be developed in order to attain the highest β values. Notwithstanding the encouraging results of current drive by coaxial helicity injection, other means of efficient and localized current drive have to be investigated. The mechanisms of heat and particle transport are poorly understood and the physics of the low-high (LH) transition must be clarified, an important issue for the general tokamak effort. In particular, it is necessary to understand the balance between non-diffusive cross-field transport, associated with radially elongated convective cells (streamers), and transport barriers with reduced turbulence, associated with self-generated localized poloidal velocity shear (zonal flows).

3. SPHERICAL TOKAMAK EXPERIMENTS WORLDWIDE

The Heidelberg Spheromak [2] , the Lucas Heights Rotamak-ST [3] and the SPHEX Rodomak [4] were the first experiments that tested the spherical tokamak concept, in the late eighties, by inserting a current-carrying central rod into small existing devices. These experiments were limited to cold plasmas (electron temperatures of a few tenths of eV) but showed promising equilibrium and stability results.

The START (Small Tight Aspect Ratio Tokamak) experiment began operation in 1991 at Culham Laboratory, UK, and was the world's first tokamak to study hot plasmas ($> 100\,\mathrm{eV}$) at aspect ratios $A < 2$ [5] [6] . In this device, the low aspect ratio configurations were obtained by a dynamical method using external induction coils and a vertical field to compress the plasma towards the central column. A small solenoid was wound round the central toroidal field rod in late 1991 and upgraded in late 1992 to rise the plasma current up to 300 kA and assist confinement studies. The good stability characteristics expected for the spherical torus were confirmed by this experiment. The installation of a 500 kW, 30 keV neutral beam injector in 1995 considerably improved the plasma parameters of START to reach record values of volume averaged beta about 40% during the 1996 to 1998 campaigns [7] . The experiment was discontinued in 1998 to be replaced by the present MAST device, but some of its results are still under study [8] .

The main objective of the CDX-U (Current Drive Experiment-Upgrade) device in the Princeton Plasma Physics Laboratory (PPPL), USA, is to explore concepts and physics of non-inductive current drive. Initially, CDX-U was operated with dc helicity injected and neoclassical pressure driven currents up to 10 kA. An ohmic heating solenoid was installed in 1993 to increase the plasma current to the 100 kA level and compare the efficiency between inductive and non-inductive current drive techniques. Effective operation in low aspect ratio $A \cong 1.5$ tokamak mode began in 1994. At this time the

TS-3 (Tokyo University Spherical Torus No. 3) device was conducting tokamak-merging experiments to investigate the relaxation and reconnecting mechanisms for future current drive amplification. In TS-3 the toroidal plasma is typically formed starting from the arc current between the external electrodes of two plasma guns. Some of the first results for CDX-U and TS-3 were jointly reported in the 15th IAEA Fusion Energy Conference [9] . In 1997, a high-harmonic fast wave heating system was installed in CDX-U to test this concept for later application in the NSTX spherical tokamak. Recently, the CDX-U device has been used in experiments involving the use of liquid lithium as a plasma facing component [10] . Meanwhile, the high power heating of the magnetic reconnection has been used in TS-3 and the larger TS-4 device to study the high-β stability of spherical tori [11] [12] .

The first plasma in the Tokyo Spherical Tokamak (TST), which was dedicated to investigate the basic physics of plasma confinement and stability at low aspect ratio $A <$ 1.2, was obtained in 1995. This device had to be modified (TST-M) to circumvent problems in the toroidal field coil current feedthroughs. The present TST-2 configuration, with an aspect-ratio $A = 1.6$, is an upgrade of TST-M that started operating in late 1999 reaching plasma currents in the 100 kA range. Internal reconnection events, excitation and propagation of high-harmonic fast waves, and emission of electron Bernstein waves have been recently studied in TST-2 [13] .

The Helicity Injected Tokamak (HIT-II) experiment at the University of Washington, USA, uses transformer action and coaxial helicity injection to drive toroidal current in the spherical tokamak geometry. From 1998 onward this device extended previous results of HIT [14] by using a double-null diverted flux boundary and producing discharges with toroidal current near 200 kA [15] . The HIT-II experiment provides support for the NSTX start-up and current profile control program [16] .

The Pegasus toroidal experiment at the University of Wisconsin-Madison, USA, is an extremely low aspect ratio device with the goal of minimizing the central column while maintaining good confinement and stability. The technology and infrastructure for the Pegasus tokamak were provided in great part by previous experience with the tabletop, $A = 1.5$, $\sim$20 kA of plasma current Medusa device. Tests of Pegasus started in mid 1998. Following major upgrades in early 2001, Pegasus achieved aspect ratio $A \geq 1.15$ and a ratio of plasma current to toroidal field current $I_P/I_{TF} < 1.2$ in ohmic plasmas. Using up to 1 MW of radio frequency power this experiment aims to demonstrate current drive and heating with high-harmonic fast waves [17] .

The objective of the Proto-Sphera experiment at the Centro Ricerchi Frascati - ENEA, Italy, is to test an ultra low aspect ratio tokamak, with the central conductor replaced by an arc discharge, as a prototype for a larger Sphera device. In this proposal an initial screw-pinch is driven unstable by increasing the arc current and then relaxes to a stable flux core spheromak state. Proto-Sphera aims to extend the results of the similar TS-3 experiment, to effectively test the stability of the screw-pinch/spherical tokamak configuration [18] [19] . A Proto-Pinch experiment was constructed in 1998 and tested through 1999 to investigate the geometry of the extended electrodes needed to maintain the high-current steady state arc.

The objectives of MAST (Mega Ampere Spherical Tokamak) in UKAEA Fusion, Culham Science Center, UK, are to enhance the tokamak database, to test and develop scaling, and to explore the physics of the spherical torus ($A > 1.3$) at plasma currents in the 1$\sim$2 MA level. MAST uses the same inductive start-up method employed previously in START. Two systems of additional heating will be used: neutral beam injection (5 MW, 70 keV) and electron cyclotron heating (1.5 MW, 60 GHz). Results of MAST will be compared with present medium-sized tokamaks (ASDEX-Upgrade, DIII-D). MAST produced its first plasma in early 1999 and 1 MA plasma current in early 2000. The Greenwald limit has been exceeded, and higher (H) mode achieved with modest additional neutral beam injected (NBI) power. In 2001 the NBI was raised to 2 MW and the database of H-mode discharges has been extended, with studies of the LH transition, plasma edge behavior and magnetohydrodynamic activity [21] . By the end of 2001, H-mode was attained in purely ohmic discharges in MAST.

The NSTX (National Spherical Torus Experiment) located at the Princeton Plasma Physics Laboratory is a multi-institutional project in the USA designed to explore advanced spherical torus regimes ($A > 1.3$) in the 1 MA range of plasma current. Very large power auxiliary heating and current drive systems are available: neutral beam injection (5 MW, 80 keV, 5 s), high harmonic fast waves (6 MW, 30 MHz, 5 s) and electron cyclotron heating (400 kW, 28 GHz). NSTX started operating in early 1999 and reached the 1 MA plasma current level by the end of 1999. During the year 2000 the machine operation improved significantly, the current level was raised to 1.4 MA and the best confinement data exceeded the ITER PB Y98 H-mode scaling. In 2001, coaxial helicity injection successfully drove up to 0.4 MA of toroidal plasma current with a plasma current multiplication factor of 14 [20] .

The Globus-M project is dedicated to basic spherical tokamak ($A = 1.5$) physics research up to the 0.5 MA plasma current level, with emphasis in auxiliary heating and current drive using various radio frequency methods: ion cyclotron heating (1-1.5 MW, 10-15 MHz), high harmonic fast waves (0.7 MW, 30-50 MHz), lower hybrid schemes (<0.5 MW, 2.45 GHz) and neutral beam heating ($\sim$1 MW, 30 keV). Operation of Globus-M started in early 1999 at the Ioffe Institute in St. Petersburg, involving the participation of several Russian Federation institutions. The experimental campaigns of 2000 and 2001 concentrated in optimizing the machine performance, reaching a plasma current of up to 0.25 MA in ohmic operation [22] . In 2002 the current was raised to 0.36 MA.

SUNIST (Sino United Spherical Tokamak) is a new facility under construction in Tsinghua University, Beijing, People's Republic of China, which will be operated by an experienced group on turbulence, alternating current operation, and current startup with electron cyclotron wave and electrode assisted heating on small tokamaks. The vacuum vessel of the SUNIST device has toroidal and poloidal electrical breaks to reduce eddy current effects during the formation phase [23] . In a second phase, electron cyclotron resonance start-up will be combined with fast wave current drive and, if successful, should permit non-inductive operation.

4. HISTORICAL DEVELOPMENT OF ETA, PROTO-ETA AND ETE SPHERICAL TOKAMAKS

The Associated Plasma Laboratory (LAP) at INPE was established in 1978. For over two decades LAP carried out various experiments and activities in such diverse areas as: simulation in the laboratory of space plasma phenomena, development of plasma technologies for ion propulsion and materials processing, and fusion plasma research, including spherical torus, high-power microwave sources and plasma diagnostics development.

In the area of controlled thermonuclear fusion LAP was one of the first research centers to develop the spherical torus concept. Starting in 1986, the Laboratory was responsible for the conceptual design of the Experimento Toroidal Avançado - ETA (Advanced Torus Experiment), a tokamak with aspect ratio $A = 1.5$ and plasma current near 1 MA dedicated to explore means of radio frequency current drive. The Ministry of Science and Technology of Brazil approved the ETA experiment in 1987 as the principal experiment of the future National Plasma Laboratory. The proposal had wide international exposure when presented in the Energy Independence Conference on Fusion Energy and Plasma Physics sponsored by the International Union of Pure and Applied Physics and held in August 1987 in Rio de Janeiro [24] [25] .

When it became clear that there would not be sufficient funding available to build ETA, the Associated Plasma Laboratory researchers started to design smaller versions of the experiment, firstly presented in the 1988 IAEA Technical Committee Meeting (TCM) on Research using Small Tokamaks [26] . The original design of the ETA prototype was considerably improved during the first and second US-Brazil Experts' Workshop on Spherical Torus Experiments, held in May 1990 in São José dos Campos, SP, Brazil and September 1990 in Oak Ridge, TN, USA, respectively. These meetings had the participation of experts from several Brazilian and North American laboratories and universities, besides invited specialists from laboratories in Europe, Russia and Japan. The final conceptual design of Proto-ETA was presented in the 1990 IAEA TCM on Research using Small Tokamaks [27] .

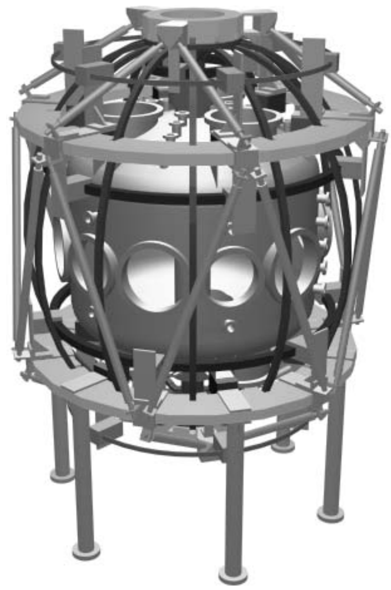

Figure 6: Artistic view of the ETE spherical tokamak.

While waiting for the approval of funds to build Proto-ETA, the Associated Plasma Laboratory collaborated during 1991 in the conceptual and engineering design of the TBR-E (Brazilian Spherical Tokamak) device. The TBR-E was a flexible design developed to explore different inductive and non-inductive tokamak regimes by varying the aspect ratio $1.5 \leqslant A \leqslant 2.2$ and the magnetic field $B_0 \leqslant 0.63\,\mathrm{T}$ [28] . This tokamak should be installed in the campus of the University of São Paulo (USP), as a joint effort between several Brazilian plasma physics groups. However, due to the lack of sufficient funding, the USP group decided in 1992 to acquire the deactivated TCA (Tokamak Chauffage Alfvén) device from the Ecole Polytechnique Fédérale de Lausanne, in order to continue the research in plasma heating using Alfvén waves. The development of theoretical and experimental tools by the USP group in this area has become decisive for the implementation of auxiliary heating methods in the final version of the spherical tokamak in Brazil.

Finally, a minimal design for a spherical tokamak was completed in 1993 [29] leading to the present ETE device, which is illustrated in Figure 6. Parts (mostly used) and equipment for this experiment were gradually purchased using limited funds from the home institution (INPE). The detailed engineering design was then carried out, with manufacturing of some components beginning in mid 1995 [30] [31] . Additional funding by a Brazilian federal government agency (FINEP) was approved for the period 1997-1999, leading to the effective fabrication of hardware. Assembly of ETE began in mid 1998 after completion of the new experimental hall built at INPE. Tests of the machine started in late 1999 and the first tokamak plasma was attained in 28 November 2000 [32] .

The conceptual and basic engineering design of ETE is relatively old, having been completed in 1993, but still presents some innovative technological characteristics that result in a very compact and simple design, with good plasma accessibility. This experiment is expected to make an effective contribution to the understanding of basic spherical tokamak physics, as well as to the development of hot plasma diagnostics.

5. RESEARCH PROGRAM OF ETE

The experimental program of ETE focuses on the study of the basic physics of spherical tokamak plasmas. The program will start with plasmas in the ohmic regime, optimized by good vacuum conditioning and pre-ionization. An operation parameters space study will be carried out, looking for high-density limits and lowest ratio of the central column current to plasma current. The implementation of a fast neutral lithium beam probe and an array of electrostatic probes will allow a detailed study of the plasma edge conditions. The particle recycling conditions at the edge is crucial for a low impurity discharge. Furthermore, the plasma boundary processes have a strong influence on the core plasma behavior. Magnetic activity will be monitored during the entire research program.

The plasma diagnostic system of ETE also includes the basic electromagnetic diagnostics, optical spectroscopy, optical imaging and Thomson scattering, already installed, and a far infrared laser interferometry system in the near future. Soft X-rays imaging in three planes will be attempted in order to estimate the position of the magnetic axis and the elongation at the axis, which should improve the results of the magnetic reconstruction procedure, still to be implemented. Neural networks will be gradually applied for future feedback control of the vertical equilibrium and fast reduction of diagnostics data.

By increasing the available capacitor banks the plasma current can be extended from the presently attainable 0.2 MA range to nearly 0.5 MA with a 50 ms pulse duration (solenoid stress and heating limits). The maximum allowable value of the magnetic field is 0.8 T (mechanical stress limit), depending also on an upgrade of the capacitor banks. Means of auxiliary heating and current drive, which will lead to studies of improved plasma confinement, have been just recently addressed, in collaboration with the University of São Paulo in the area of Alfvén wave heating. Auxiliary heating and current drive is envisaged up to the 1 MW level (2 MW as a machine limit).

Theoretical work has been carried out during the last few years notably in the areas of self-consistent equilibrium calculations including neo-classical effects, plasma start-up models including eddy current effects, and, looking to applications of high-power microwave sources in ETE, propagation and interaction of intense electron beams with radio-frequency fields in resonant cavities and corrugated wave-guides. Modelling of the scrape-off layer and studies of the plasma edge physics will be undertaken in the near future.

In general terms, the objective of the ETE experiment is to explore the new physics of plasmas in the spherical torus geometry and to investigate means to attain a compact, stable, efficient, steady-state configuration that can lead to a viable fusion reactor. In this way, the ETE experiment can contribute effectively to the spherical tokamak database. Besides research in low aspect ratio tokamaks and diagnostics development, another important objective of the ETE program is training in tokamak operation. The ETE tokamak is open to collaboration with other laboratories and universities, both national and international, providing the only spherical tokamak facility available for training in the southern hemisphere.

Table 1: Main parameters of the ETE tokamak.

	First phase	Upgrade
Major radius R_0	0.30 m	-
Minor radius a	0.20 m	-
Elongation κ	1.6	1.8
Triangularity δ	$\sim$0.3	-
Toroidal induction B_0	0.4 T	<0.8 T
Toroidal plasma current I_T	0.22 MA	0.44 MA

6. TECHNICAL CHARACTERISTICS OF ETE

The main parameters of ETE are listed in Table 1. During the initial operation a modestly high plasma current $I_T \cong$ 0.22 MA ($\sim$15 ms) will be produced in a 1.5 aspect ratio configuration with a toroidal field $B_0 \leq 0.4$ T ($\sim$100 ms). These parameters should be attained with fully inductive current drive using the capacitor banks presently available for the ohmic and toroidal field circuits. However, the maximum parameters will be reached only after upgrading the capacitor banks. By increasing the stored energy one expects also to increase the pulse length to values that will permit a better study of the plasma confinement properties. The ultimate parameter values are limited by mechanical stresses in the demountable joints of the toroidal field coil ($B_0 < 0.8$ T) and by stresses and heating in the ohmic solenoid ($\sim$ 0.25 Wb, $\sim$ 180 ms). The design of ETE incorporates some innovative technological features that resulted in a compact and light weighted device with good plasma accessibility. In the following sections some of the technical characteristics of the main components of the tokamak will be described.

6.1 VACUUM SYSTEM

Figure 7 shows a drawing of the vacuum chamber of the ETE tokamak. It is manufactured from Inconel 625 alloy. The relatively high resistivity of Inconel helps to weaken the eddy current effects in the continuous vessel. The thickness of the inner cylindrical wall is 1 mm and the thickness of the torispherical heads and outer cylindrical wall is 6.35 mm. The internal diameter of the vessel is 0.178 m, the external diameter is 1.219 m and the height is 1.200 m. The inner cylindrical wall of the vacuum vessel can be removed by grinding and replaced by a smaller diameter tube with the removal of the ohmic heating solenoid. In this way the aspect ratio of ETE can be reduced from 1.5 to 1.3, if necessary in the future.

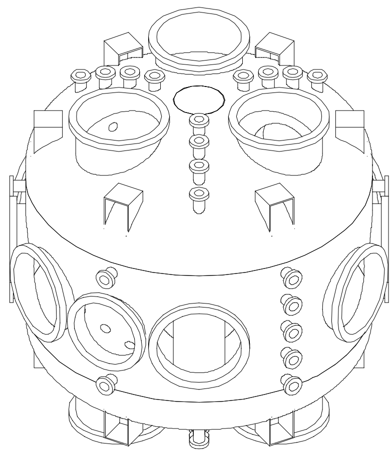

Figure 7: Vacuum chamber of the ETE tokamak.

Standard ConFlat flanges are used so that adequate baking and discharge cleaning up to 200°C can be implemented. Short bellows were roll-formed at the extremities of the inner cylindrical wall in order to reduce the thermal stresses during baking. The flanges are distributed around the vacuum chamber in the following configuration: 6×14CF and 4×250CF lateral ports, 3×14CF upward facing ports, 3×14CF downward facing ports and 42×40CF small flanges. Several supports welded in the inner wall of the vacuum chamber are used to install rail limiters and diagnostics. The vacuum system consists of a 1500 ℓ/s turbodrag pump backed by a 4 m^3/h diaphragm pump, plus a 30 m^3/h rotary pump for maintenance procedures, two piezoelectric valves in parallel for gas injection (hydrogen, helium, argon, nitrogen) trough a stainless steel line with flow and time control, and a residual gas analyzer (RGA). The vacuum chamber is covered by hot tapes for baking (16 kW) and a thermal insulation blanket. A base pressure of 8×10^{-8} mbar is easily achieved with baking only up to 110°C. A butterfly valve placed in front of the turbodrag pump is used to control the gas flow during glow discharge cleaning (maximum 1 kV/10 A in two symmetrical anodes).

Electric fields as low as 2 V/m are sufficient for breakdown at $2{\sim}3\times10^{-4}$ mbar hydrogen pressure with pre-ionization provided by hot filaments (4 A×47 V, 75 V polarization). Pre-ionization by electron cyclotron resonance heating will be tried in the future to improve the start-up performance of ETE. For this application a monotron, which is the simplest microwave tube, is under development by the high-power microwave sources group of LAP. The monotron experiment currently under way consists of an electron beam ($E_0 = 10$ keV, $I_0 = 20$ A) that traverses a standing-wave cavity resonator ($f = 6.7$ GHz in the mode TM_{020}) producing 30 kW of output power in the mode TM_{01} (the electronic efficiency is $\eta_{elet} =$ 0.20 and the circuit efficiency is $\eta_{circ} =$ 0.75). Figure 8 shows a particle-in-cell code simulation of the electron beam bunching inside the cavity. Presently, the prototype electron gun and resonant cavity are being characterized.

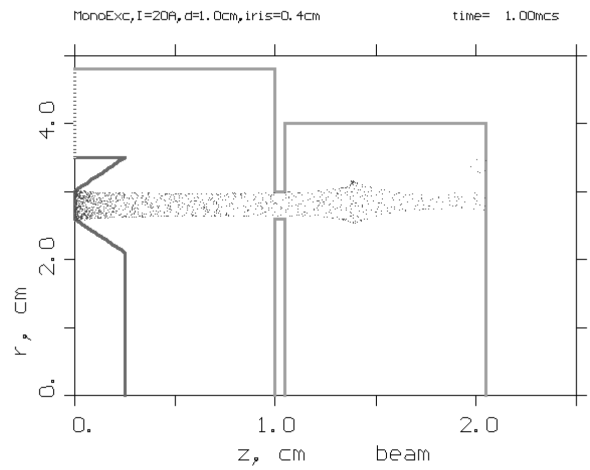

Figure 8: PIC simulation of the monotron showing bunched electron beam.

6.2 TOROIDAL FIELD COILS

The toroidal field (TF) coil uses a D-shaped minimum stress design shown in Figure 9. Each one of the 12 single turns of the TF coil is free to expand in the poloidal plane. The 12 turns are connected in series by a system of current feed rings, illustrated in Figure 10, that compensate the stray magnetic field. The height of the TF coil is 1.68 m and the diameter is 1.64 m, leading to a magnetic field ripple < 0.3 % at the plasma edge.

Electron beam welding was used in all the fixed joints to maintain the half-hard condition of the 7.3 cm^2 cross section copper bars, which constitute each turn of the coil. The external legs of the TF coil were bend and machined to the bending moment free shape, and the internal legs were machined to their final trapezoidal cross-section. The center stack, with a diameter of 12.4 cm, is water cooled between shots. Since the current density is constant over the length of the coil, each turn is water cooled in its entire length in order to reduce the stress end effects due to temperature variations. Copper tubes with an internal diameter of 5.75 mm were pressed into slots machined in the bars of the TF coil and soft-welded or fixed to each bar for water cooling.

The demountable single bolt joints used in the TF coil are illustrated in Figure 11. Prototypes of these joints were submitted to static stress tests to determine the actual stress limit. Figure 12 shows the stress-strain diagram for the external demountable joint. Both the external (superposition) and internal (insertion) joints yielded at a stress level equivalent to $\sim$ 1 T magnetic field, as predicted in the design (a 1 T toroidal magnetic field corresponds to a thin-shell torus constant tension of $\sim$28 kN). This result indicates that the operation of the TF coil should be limited to a maximum toroidal field of 0.8 T, to keep a safety margin greater than 50 %.

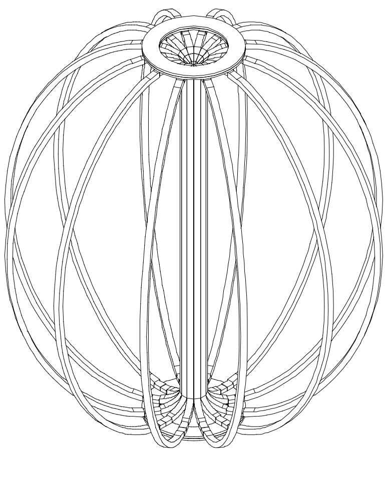

Figure 9: Toroidal field coil of the ETE tokamak.

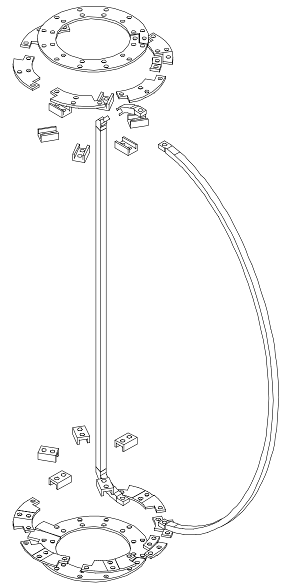

Figure 10: Exploded view of one turn of the TF coil and of the current feed rings.

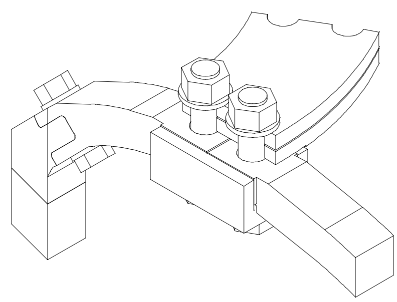

Figure 11: Single bolt demountable joints of the TF coil.

6.3 POLOIDAL FIELD COILS

In the design of the ETE tokamak a minimal set of coils was adopted for the poloidal field (PF) coils system, as illustrated in Figure 13. It comprises: (1) the plasma magnetizing coils system, formed by the ohmic heating solenoid in series with two pairs of compensation coils, (2) a pair of equilibrium field coils, and (3) a pair of elongation coils. Table 2 summarizes the final geometric parameters of the PF coils, measured during the fabrication procedure. The mean radial dimension indicated for the ohmic heating solenoid corresponds to the equivalent radius for an infinitely long solenoid. Some of the coils allow small adjustments in the vertical position, indicated in the column labelled Z_{range}. The width (ΔR, ΔZ) corresponds to the region filled by the conductors.

The position of the PF coils relative to the main components of the tokamak is shown in Figure 14. The location of all the PF coils was carefully studied to allow good diagnostic access. The ohmic heating solenoid

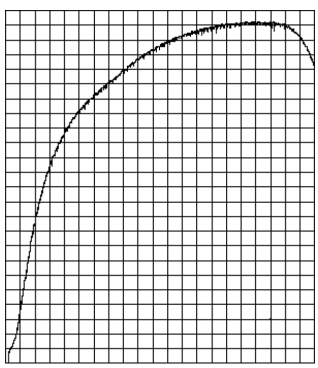

Figure 12: Stress-strain diagram for the external (superposition) demountable joint of the TF coil. Each horizontal division corresponds to a displacement of 1 mm and each vertical division to a force increment of 3.07 kN.

Table 2: Geometrical parameters of the poloidal field coils in ETE.

Coil denomination	$\overline{R}$ [m]	$\overline{Z}$ [m]	Z_{range} [m]	ΔR [m]	ΔZ [m]	$NR \times NZ$
Ohmic Heating Solenoid	0.07246	0	-	0.0192	1.3054	2×130
Internal Compensation Coils	0.1022	± 0.6606	-	0.021	0.100	2×10
External Compensation Coils	0.6512	± 0.8797	$0.87 \sim 0.96$	0.010	0.020	1×2
Equilibrium Coils	0.7002	± 0.3900	$0.32 \sim 0.45$	0.040	0.040	4×4
Elongation Coils	0.2001	± 0.8300	$0.79 \sim 0.87$	0.040	0.040	4×4

(OH) fits the narrow gap between the outer radius of the TF coil and the inner wall of the vacuum vessel. In ETE the OH solenoid is formed by two layers of a square cross-section (8.81 mm×8.81 mm), water cooled hollow conductor (4.93 mm diameter central hole) and was wound around the central column of the TF magnet. The remaining PF coils were wound using a slightly different square cross-section (9 mm×9 mm), water cooled hollow conductor (5 mm diameter central hole). All the PF coils are insulated with Kapton, fiberglass tapes and final epoxy resin impregnation. The magnetizing coils system is capable of providing a 0.248 Wb flux variation with double swing operation within the admissible stress limit for half-hard copper, for a maximum current of about 30 kA.

In the initial phase of operation the solenoid will be driven to about half its maximum flux swing capability. In the extended phase the solenoid operation will be limited essentially by the maximum adiabatic temperature rise during each shot. The two pairs of passive compensation coils have their location and number of windings optimized to produce, together with the OH solenoid, a minimum error field over a large region near the midplane of the plasma (hexapole field cancellation), as shown in Figure 15. These coils must also be water cooled since they are fabricated with basically the same conductor used in the OH solenoid and are connected in series. The remaining two pairs of PF coils supply the time-varying vertical field necessary for plasma position control and for some limited control of the plasma elongation and dispersion of field lines. The central stack ($\ell = 1.87$ m, $d = 168.8$ mm) was the most challenging piece completely manufactured in the workshop of the Associated Plasma Laboratory, as well as all the PF coils.

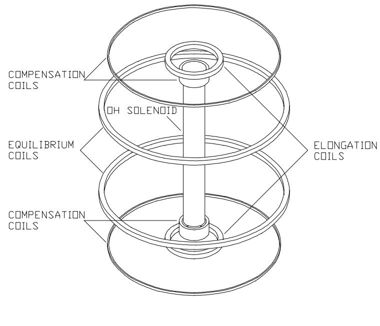

Figure 13: Poloidal coils system of the ETE tokamak.

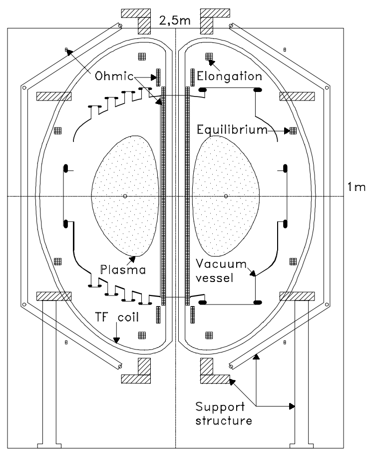

Figure 14: Schematic cross-section of the ETE tokamak.

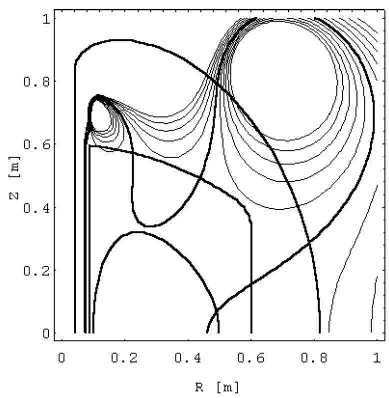

Figure 15: Magnetizing flux contours on the poloidal plane. The heavy contour indicates a poloidal flux of 0.25Wb (2×7.8MA-turns in the OH solenoid for double-swing operation) with a 2% flux increment between contours. The plasma outline, and the vacuum vessel and toroidal field coil centerlines are also displayed.

6.4 MECHANICAL STRUCTURES

Figure 16 shows a drawing of the support structure of the ETE tokamak. It consists basically of two crowns and two rings of insulating material (Tufenol) placed symmetrically with respect to the midplane of the device.

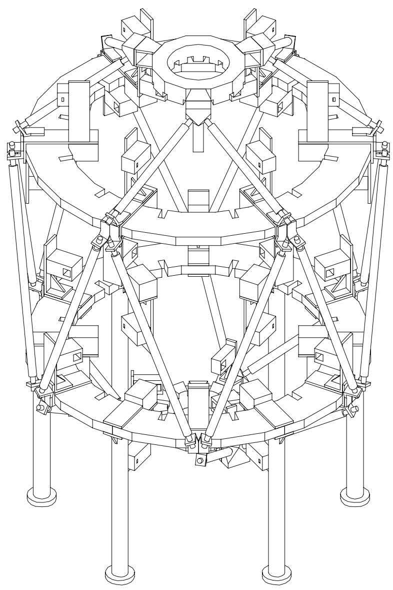

Figure 16: Structure for supporting the vacuum vessel and coils of ETE.

The TF coils are inserted in slots cut in the crowns and rings, which leave the coils free to move in the poloidal plane but resist the out of plane forces. Adjustable stainless steel rods, manufactured from mechanical tubes, connect the crowns and rings, forming a rigid truss structure. The PF coils are attached to the crowns and rings by simple L-shaped supports and clamps which allow some control of the coils position. The vacuum vessel is placed on plastic shock absorber blocks attached to the support legs and to the upper and lower rings.

The entire assembly is modular and allows very good access for diagnostics. Simple engineering calculations show that the structure is capable to resist the normal and fault electromagnetic loads, but a more careful analysis is necessary to assess the operating limits of the TF coil ($B_0 \leq 0.8$ T) and the structural response of the vacuum vessel to the dynamic forces during a large plasma disruption. The support structure is mounted on a strong aluminum sustentation structure (with toroidal break), which is fixed to a concrete block on the ground level of the tokamak laboratory, giving access to the capacitor banks, power supplies and water cooling equipment. All the mechanical structures have been fabricated in the Institute's workshop, and have been assembled and adjusted in the tokamak hall with an overall precision of 2 mm.

6.5 POWER SUPPLIES

The power supplies system for ETE is based entirely on capacitor banks for energy storage. There are four independent banks: toroidal, magnetizing, equilibrium and elongation.

The toroidal capacitor bank has a total energy of 1.13 MJ and is designed to supply a current of $\sim$50 kA during about 100 ms to the toroidal field coil, producing a magnetic field of $\sim$0.4 T with $\pm 1\%$ fluctuations. Low voltage electrolytic capacitors, high current thyristors and diodes are used in a two modules configuration: fast bank (1.25 kV, 309 kJ) and slow bank (360 V, 821 kJ fired sequentially in 8 stages). Figure 17 shows the current waveform in the toroidal field coil, obtained by numerical simulation.

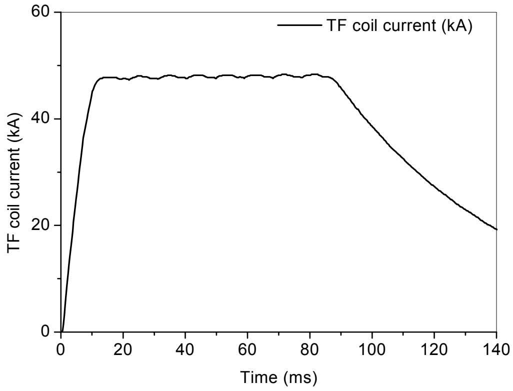

Figure 17: Current waveform in the TF coil, obtained by numerical simulation.

The magnetizing bank or ohmic heating bank has a total energy of 988 kJ and is designed for double swing operation. This bank uses high voltage oil capacitors in three modules: charge (8 kV, 467 kJ), startup (8 kV, 296 kJ) and flattop (7.5 kV, 225 kJ, presently fired in 2 stages). In this configuration the magnetizing bank has the capability of driving and maintaining a plasma current of $\sim$220kA during 15 ms. Figure 18 shows the current waveforms in the plasma and in the double swing OH solenoid, obtained by numerical simulation.

The equilibrium bank has a total energy of 140 kJ and comprises two modules: fast bank (7.5 kV, 90 kJ) and slow bank (360 V, 50 kJ fired in 5 stages), which use an ignitron triggering circuit and thyristor triggering circuits, respectively. The capacitor bank is designed to supply a current of $\sim$6 kA during 25 ms to the equilibrium field coils. The elongation bank is still in the design phase since its implementation is not essential during the initial operation of the tokamak.

All activities involving the reconditioning and testing of capacitor banks, fabrication of support structures, and development of triggering and safety circuits used in ETE were conducted by the electronics workshop of LAP.

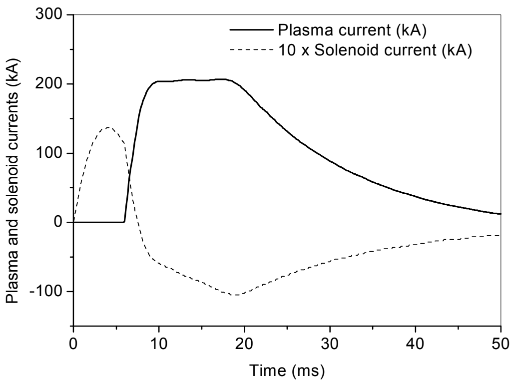

Figure 18: Waveforms of the current in the plasma and 10 times the current in the double swing OH solenoid, obtained by numerical simulation.

Presently, the energy of the capacitor banks is being continuously increased by adding more capacitor modules as well as by rising the voltage rating to its maximum to reach the first operational phase listed in Table 1.

The magnetizing bank is being provisorily operated in low energy single polarity mode while the effects of eddy currents induced on the vacuum vessel are evaluated and compensation for error fields in the start-up phase is introduced. With this low energy configuration the plasma current is limited to the 30 to 60 kA range corresponding to a pulse duration in the 10 to 5 ms range.

6.6 CONTROL AND DATA ACQUISITION SYSTEMS

The ETE tokamak is computer operated with an electropneumatic safety system and complete galvanic insulation. The control is based on CAMAC modules and data acquisition on a VME bus standard. An optical link provides an insulation of 2 kV (15 kV in the near future) between the control computer and the CAMAC modules. The galvanic insulation between the CAMAC crate and the hazardous environment is achieved by optical and pneumatic systems. All the components of the control system, including triggering circuits and pneumatic actuators, are functional and the control software is under continuous development using the C language. The VME modules will be introduced progressively with the installation of the diagnostic systems.

6.7 DIAGNOSTIC SYSTEMS

A complete set of fundamental diagnostics is planned for ETE. Table 3 lists the diagnostics proposed for implementation during the initial (inductive) phase of operation, together with the plasma parameters to be measured.

An initial set of electromagnetic diagnostics is already installed, comprising: three Rogowski coils to measure the currents in the toroidal, equilibrium and ohmic circuits, one Rogowski coil placed inside the vessel to measure the plasma current and a second one outside the vessel to measure the induced eddy current, twelve loop voltage coils in different positions (one placed inside the vessel), four fixed magnetic pickup coils (B_r, B_z) protected by the graphite limiter for magnetohydrodinamic and magnetic field measurements, two movable magnetic pickup coils (B_r, B_z and B_ϕ) at the midplane, and one movable electrostatic probe also at the midplane. A visible light spectrometer (12 Å/mm) for impurity emission detection and an H_α detector with interference filter ($\Delta\lambda$ = 13.52 nm) have been installed. The hard X-ray detector is being used to map the radiation pattern around the machine. One fast CCD camera with speed up to 500 FPS and frame velocity up to 1/10,000 has been used on loan.

Table 3: Plasma diagnostics to be installed in ETE.

Diagnostic
Magnetics
H_α detectors, bolometers
Hard X-ray monitors
Optical spectrometers
Thomson scattering
Fast CCD camera
Electrostatic probes
Low-energy lithium beam probe
Far infrared laser interferometer
Arrays of soft X-ray diodes
Plasma Parameters
Magnetic reconstruction, magnetic activity
Radiated power
Fast electrons, magnetic activity
Electron temperature and impurities
Electron temperature and density profiles
Optical imaging
Plasma edge parameters
Plasma edge parameters
Integrated density
Electron temperature imaging

A Thomson scattering system consisting of a 10 J, 20 ns Q-switched ruby laser, a 5-channels polychromator filter with avalanche photodiodes, and collecting lenses capable of scanning 22 different plasma positions along 50 cm of the laser beam trajectory at the midplane is operational. In the initial tests, plasma temperatures ranging from 20 to 160 eV and densities up to $0.4\times10^{20}\,\mathrm{m}^{-3}$ have been measured in hydrogen discharges with plasma current of 60 kA amplitude and 5 ms duration. The present single channel Thomson scattering system will be gradually upgraded to obtain electron temperature and density profiles during each shot.

A 10 keV fast neutral lithium beam (FNLB) probe with glassy β-eucryptite source (up to 1 mA/cm^2) is presently under development at LAP and will be used for measuring several parameters in the outer region of the plasma. The FNLB, which is currently undergoing final tests, consists of an ion gun (ion source and 3 electrostatic lenses), a sodium based neutralization chamber, a flight line with differential pumping, and a detection system.

Design of a multipass Michelson interferometer system has been recently completed. This system uses a 50 W CO_2 laser ($\lambda_{CO_2} = 10.6\,\mu$m) and a 20 mW HeNe laser ($\lambda_{HeNe} = 0.6328\,\mu$m). The CO_2 laser is injected in the tokamak through a set of flat mirrors and is kept in the plasma by multiple reflections between two spherical mirrors (R =1.5 m). Simulations indicate that it is possible to have 22 passages through the plasma, which correspond to more than two fringes of wavelength shift considering a plasma length of 60 cm and peak density 0.5×10^{20} m^{-3}. The HeNe laser will be used for alignment purposes and to eliminate errors due to mechanical vibration of the system. The arrays of soft X-ray diodes, which are in the design phase, will be used to produce images of the temperature in three planes, providing additional information for the magnetic reconstruction procedure, still to be developed.

6.8 INFRASTRUCTURE

The main building for ETE was finished in mid 1998. The installations comprise: the tokamak hall (150 m^2 containing a 2 ton crane and a 800 kg lift), the power supplies area (225 m^2 for the capacitors banks), the control room (35 m^2), the engineering and physics room (30 m^2), the computer room (18 m^2), the support systems area (50 m^2 reserved for the water cooling system - 10 bar, 6 m^3/h, 6-20° C - and the dry compressed air system - 7 bar, 0.5 m^3/min), the electrical supply room (50m^2 containing transformers with ultra insulation - 450 kVA, plus transformers with standard insulation - 300 kVA). Finally, the ETE workshop, with 250 m^2 in two stories, is used for the fabrication of coils and mechanical parts plus the development of electronic circuits.

ACKNOWLEDGMENTS

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