Mexican Design of a Tokamak Experimental Facility

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Abstract. Mexico presents its proposal Tokamak Experimental Facility design under the necessary effort to develop Science and Technology into the thermonuclear magnetic confinement fusion area. This Research and Development Project (R+D) was approved by the Mexican Education Ministry (SEP, spanish acronyms) in 2007 for its actual development stage at Facultad de Ingeniería Mecánica y Eléctrica (FIME) - Universidad Autónoma de Nuevo León (UANL). We have made this effort in order to unify and consolidate under a tokamak experimental configuration the Mexican Energy Fusion Program and generate an attractive scientific technological proposal to the mexican research centres with main objective to participate in ITER (International Thermonuclear Experimental Reactor) development. This present design aims to generate, innovate, understand and develop scientific and technological fusion knowledge, also to form researchers in fusion confinement area. We consider at this time that nuclear fusion represents an attractive, powerful and clean energy source. This R+D Project involves multidisciplinary physics and engineering areas that coexists into a nuclear fusion reactor, science and technology works together to establish a natural symbiosis between theory and experiments. Our tokamak facility design, at this time, is being developed and simulated under the use of COMSOL Multiphysics and 3D CAD software, all programs running under Gentoo GNU/Linux installed in our SGI Altix XE250 Platform. With this resources we can develop entirely systems involved in our magnetic confinement fusion research line, focused in the stronger application of engineering, technology and science concepts, developing systems and devices into this energy source generation. Taking advantage of this computational infrastructure we have proposed the possibility to participate throught this tokamak facility, studying and developing research over plasmas stability, confinement regimes and resonant magnetic field perturbations.

1. Introduction

The earth energetic future platform over this XXI century, represents one of the most important challenges that can defy all mankind energy concepts, the renewable efforts are taking great interest to develop an attractive clean energy source, within there is more parameters that we must consider over this energy development, not only to minimize the global pollution, but also is important it sustainability. In the modern cities where millions of people live, the need to maintain thousands of consumption energy systems is inevitable, the effort that could handle this necesities is called nuclear fusion energy. The latest mexican efforts on the tokamak configuration area were made in the later 80's of the XX century, almost 27 years have passed since the born of Novillo and in this 2010, a new effort grows up in a Mexican northern city: Monterrey. The non nuclear mexican tokamak design proposal has the main objective of establish an intense dialog to develop in the Mexican Research Universities and Institutes, one national fusion criteria, applying a unique long term conclusion: Mexico must participate in ITER [1].

2. Characteristics design

This R+D Fusion Tokamak Project involves multidisciplinary physics and engineering areas applied into the magnetic confinement field with a scientific proposal scopes over: plasma stability (understand turbulent transport effects and the mechanisms proposed to explain the edge transport barriers [2]) and the effort into the plasma dynamics introducing deviations in the usual axisymmetric confinement field with resonant magnetic field perturbations (RMP) [3, 4, 5]. We consider fundamental that Mexico's contribution issue must be attractive into confinement plasma stability research for the international fusion community, this mexican facility has a fundamental objective: science and technology must work together to obtain a natural symbiosis between theory and experiments. Our engineering and physical efforts estimate projections exploding several campaigns using hydrogen plasmas, advanced research could be done through the maximum operational conditions in medium experimental tokamak devices.

TABLE I. SMALL TOKAMAKS AROUND THE WORLD

Country	Designation	R(m)	A(m)	Ip(kA)	B_T(T)	T(ms)	Year
Russia	Globus-M	0.36	0.24	305/500	0.5	300/300	1999
Russia	TUMAN-3M	0.55	0.24	180	1	150	1985
UK	Compass-D	0.557	0.232	350	2.1	1 s	1992
Japan	TS4	0.55	0.45	300	1.4	30	2000

Table I shows a general comparison with others confinement facilities, our "T" tokamak is similar but not equal to UK Compass-D tokamak, transferred actually in Prague, but developed in United Kingdom (UK) [6].

3. Confinement Systems

At this first design stage we consider to use copper and ferromagnetic materials for all confinement systems coils criteria. This facility has been designed to provide in the future a material migration to superconducting ones, thus we could obtain a long-term functionality on this device. The present section concerns the physics and engineering designs mainly on the mechanical structure and the confined criteria arguments; the plasma toroidal section relationship is given by the following equation [7, 8]: $A = R/a$. The safety factor (q), is given by $q = (a \cdot B_T)/(R \cdot B_P)$, where B_T and B_P are the toroidal and poloidal fields, respectively. In a magnetic configuration tokamak, the toroidal field requires to be larger than the poloidal field ($B_T \gg B_P$). An important factor in the confinement regime, is the determination of the maximum plasma current (I_p), we use $I_p < (2\pi \cdot a^2 \cdot B_T)/(\mu_0 \cdot q \cdot R)$ [7]. B_P and B_T values are 0.3 T and 1.3 T respectively, with a found value of 35.808 kPa we have established a β value of 0.05326. For ionic temperature determination (T_i) we establish our calculus in the Artsimovich studies [9, 10], where $T_i = (1.29 \pm 0.11)(I \cdot B \cdot R^2 \langle n \rangle)^{1/3}/A_i^{1/2}$ eV, the ion temperature (T_i) is 280.53 eV. For the electron temperature, we have considered that ion energy balance can be analyzed independently of the electron energy balance, thus applying $f(T_e) = T_e - (3.7 \cdot T_e^{3/2})/(11 \cdot T_i^{1/2}) - T_i = 0$ and solving numerically, with a range of values of (T_e) from 300 to 3000 eV using (T_i) value, we found a (T_e) of 516 eV. The experimental campaign will cover times from 300, 500 and 700 ms, however this experimental facility desires a maximum of 1 s discharge time, the project has designed a D-shaped toroidal coil to develop certain experimental similarities toward ITER. The number of coils inside the facility: 1 Central Solenoid (CS), 12 toroidal coils (B_T), 6 poloidal coils (B_P), 2 small coils for the divertor, 10-12 correction coils [11].

TABLE II. MAIN CHARACTERISTICS PROPOSED FOR THIS MEXICAN "T" TOKAMAK EXPERIMENTAL MAGNETIC CONFINEMENT FACILITY.

General features of "T" Tokamak	
Chamber material	Stainless Steel 304LN
Wall thickness chamber	$7-10$ mm
Chamber Height	1 m
Chamber sections	6, with 3 upper windows for the injection of solidified pellets located at 120° each of one, with interchangeable carbon tiles
Pellets Insertion speed	$500-800$ m/s
Divertor	Designed with replaceable carbon tiles
Support structure	Internal/primary & external/secondary, stainless steel 304 LN
Major radius, R	41.0 cm
Minor radius, a	18.5 cm
Aspect ratio, A	2.2162
safety factor, q	1.9552
Magnetic toroidal field	1.3 T
Maximum plasma current	277 kA
Heating system	Ohmic heating (OH) in the first stage, injection microwaves and/or neutral particles at second one.
Duration of plasma	Studies from 300 ms to 1 s
Electronic density	2 to 3×10^{13} cm^{-3}
Electron temperature	approx. 516 eV
Ion temperature	approx. 280 eV

At FIG. 1, we can see a geometric distribution of the toroidal coils, the back cover coils has 1.1 m high and tentatively the total generated magnetic field has to be close to 1.6 T (the back covers coils are designed to make an easy disassembled scope due our future migration to superconducting coils materials). To calculate the main conductor toroidal coil area, we realized analysis in Fortran applying short-circuit times (300, 500, 700 ms to 1s) and an electrical current range (10-30 kA), we have identified a certain number of conductor areas depending on $(I^2/A^2)t = 0.0297\cdot\log([T_2+234]/[T_1+234])$, that could generate a maximum toroidal magnetic field B_T. The equation established for the toroidal magnetic field (B_T) through a D shaped almost semicircular coil, at the center of the vacuum chamber is as follows:

$$\mathbf{B} = \frac{\mu_0 I}{4\pi}\Big[\frac{\sin\,\phi_1 - \sin\,\phi_2}{r_l} + \sum_i^3 \frac{\tilde{a}_i}{r_i^{\frac{3}{2}}}\int_{c.s.}\frac{\tilde{a}_i - r_i\,\cos(\theta_i-\phi)}{[r_i - 2\tilde{a}_i\,\cos(\theta_i-\phi)]^{\frac{3}{2}}}d\phi\Big]\hat{\mathbf{z}}$$

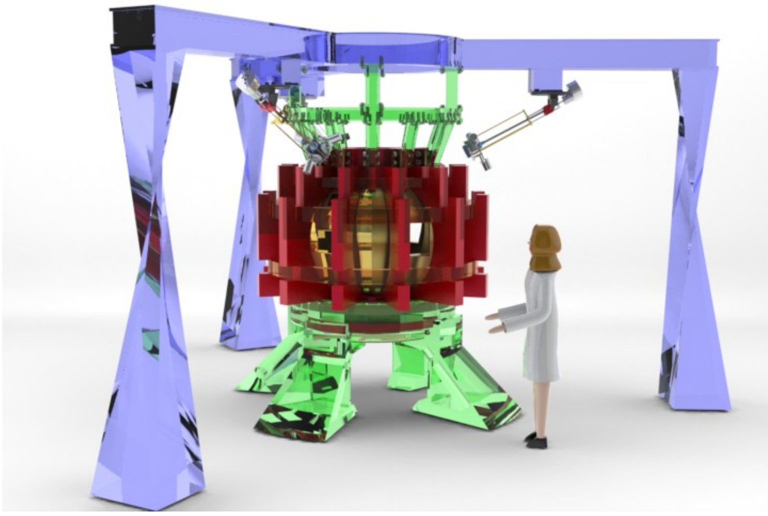

FIG. 1. Schematic 3D "T" Tokamak view

4. Vacuum chamber

In the present design (FIG. 2) we propose as a structural vacuum chamber material, stainless steel 304 LN (1), exploring the use of interchangeable materials inherents in an experimental major facility, using a remote handling system. We can make research on the complex phenomenon of plasma-wall interaction through theoretical irradiation material models [12]. Our design has three upper windows used for solidified hydrogen pellets injection (6) (external systems), localized at 120° each one, improving the global dynamic confinement scheme. The stainless steel 304 LN, is easily cleaned and has a low electrical conductivity, in the equatorial zone we expected install the diagnostic probes, and allow the entry of the maintenance systems.

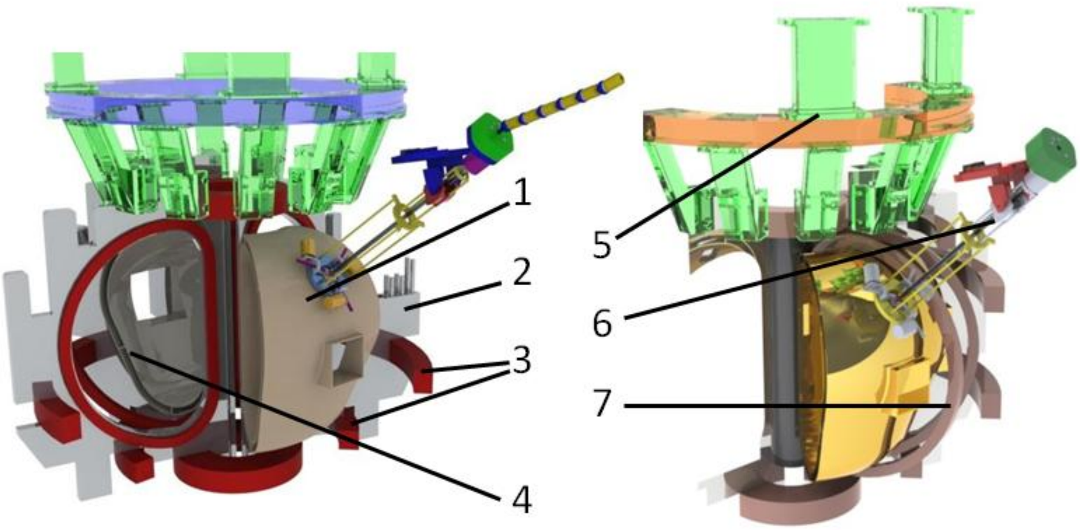

FIG. 2. Schematic 3D transversal section "T" Tokamak view

Has six toroidal sectors and between the walls outer and inner (4) stainless steel walls (5 cm) use a circulating flow coolant. The design is such that the crossing external magnetic fields, especially the vertical help us to distancing the plasma, the thickness of the stainless steel material rise from 7.0 mm up to 10.0 mm maximum. This facility has two defined structural support, internal/primary has an upper connection (5) with the second one. Back cover coils (2), poloidal coils (3), toroidal coils (7) are shown in FIG. 2.

5. Generation Vacuum Systems and Hydrogen Supplies

The vacuum pressure could be generate by diffusion pumps, mainly reason is its economic accessibility and manufacture, these pumps are going to be located at 120° symmetrically to each other and will be placed in the low part of our tokamak at 60° from the equatorial windows we use 3 diffusion pumps that would have an individual capacity of 1.33×10^{-4} Pa. The pressure measurement system would be controlled through thermocouple gauges calibrated to 1 μm. The pumping capacity has an estimate of about 0.04 to 0.045 m^3/s, the internal volume of our chamber round the amount of 0.838364598 m^3. There will be cleaning discharges to eliminate impurities is estimated 25 shots per day, with optimal intervals between puffs of 15 minutes, in this operation time we can retrieve the status of our tokamak and verify its operational capability, the daily volume of hydrogen consumption is approximately 40 m^3; in the week: 200 m^3, per month: 800 m^3 for our tokamak. Respect with the load system, we employ piezoelectric valves to allow hydrogen supplies. The hydrogen generation system has a load volume: 5 m^3/h and an outlet pressure of 0/20 MPa, is a HGH85000 model, with this we would ensure not only a minor investment but our own hydrogen generation system.

6. Feeding, supplement of power

We studied the state of the Electrical Grid of the Central University Campus (UC) that is supported by the Federal Electricity Commission (CFE, Spanish acronyms). The analysis uses a medium voltage (13.8 kV nominal supply), using the Energy Quality Analyzer AEMC Model 3945-b Power Pad as measuring equipment, give us satisfactory results of the Internal Electric Grid behavior [13] over: Unbalanced of Phases (Voltage and Currents), Total and Individually Harmonica Distortion (Voltage and Currents), Captured of transients, Register of re-closed and cut of electric supplies, Measure and register of the quality of power system (kW, V·A, V·A·R), Factors of crest for voltage and current, Voltage and current of peak, Measure RMS true of currents and voltage. Actually UC has its electrical support with two substation power: MUN-T2 and UNI-T1 both with 30 MV·A, presents five internal circuits those would be re-designed. This study produced two proposals over the management electrical grid, **First external electrical support**: is considered to have several support electrical external links and equipment, with the main objective of not affecting the electrical charge in one only circuit, sectioning or feedback the energy of the UC circuits, supplying energy from several external circuits with a separately maintenance on each circuit, using a remote supervisory control. **Second external electrical support**: this consider a new Substation Power (30 MV·A), on the High Tension side (115 to 13,8 kV) in which we would redesign the exit of the all internal circuits of UC, to enable the Federal Electric Commission to handle a positive impact in Monterrey City area.

7. Supercomputing system, data processing and control

This Research and Development Project has exclusively a SGI Altix XE 250 platform (POLAR), with 2 $\times$ 2.66 GHz Xeon Quad Core processors, 64 GB DDR2 667 MHz of RAM, 3.7 TiB (5 HD 75 GiB), Gentoo OS. Theoretical 3D CAD models and numerical applications has been carried out like powerful physic-mathematics solver tools. Research fields could be developed through supercomputers validating physics models like research of fusion materials [12]. COMSOL software is widely used applied for conducting studies on each "T" tokamak system design we also used the SolidWorks software like powerful 3D CAD tool. For the control system we have considered the Logix Platform from Allen Bradley, is a new generation PAC (programable automation controller): 32 simultaneous tasks, 100 programs (by/for) task and 100 rutines (by/for) program, gateway of Device Net standard comunication, Control Net, Ethernet, DH+ wich includes direct conectivity with wireless systems directly from the chasis.

8. Heating, coolant and divertor Systems

This facility consider auxiliary heating systems developing ECRH and NBI. The research will be made applying these two heating systems like those used in Compass-D facility [14]. Our Fusion Research Group desires create absolute interest to develop with others Mexican research institutes options that will let us a better study of the plasma and the confinement conditions. The main advantage using a divertor driving the plasma discharges in our "T" Tokamak is considered because we desires extrapolate our basic design to ITER, where divertor has a great importance to the plasma research [ITERdivertor]. The facing materials in consideration over our primarly divertor design are tungsten and carbon capable of withstanding high temperatures. Cooling systems, each back-cover-coils and mainly the vacuum chamber, are designed to use a coolant fluid, this facility desires extrapolate in a short time to investment expertise, knowledge in technological and scientific scopes. Is considered the use of threaty water with coolant air forced flow to the system.

9. Pellet injector, external systems, and diagnostic

"T" Tokamak will be equipped with a pneumatic pellet injector with a selectable output angle (SICIAV from spanish acronym). The main functional feature of this design is its angular and articulated displacement over the torus volume plasma. This pellet injector has being designed taking advantage of proven results, as the screw or piston extruders plus a brand new electronic angle controller from the guide line of high pressure, insulated and articulated pipe extruder, to its extremely shooting line. With this design we can control not only the pellet direction, but also its insertion speed into the vacuum chamber with different programmable shots on the evolution discharge plasma. Additionally, the angular position controller operatively links a net through a new protocol called Fusion Serial Bus, able to connect with our tokamak peripherals (speed 8 Mbps). The Injected pellets have 3-4 mm diameter with a range speed insertion from 500-800 m/s. This pellet injector has been presented at Symposium of Fusion Technology 2010 [15]. FIG. 2 shows a brief description of all components.

Telecontrol System Maintenance, in larger facilities as ITER, maintenance has great importance, this due the neutron irradiation, therefore the need of systems that can perform the process of handling components remotely. ITER includes the use of an automated system consisting of a remote manipulator arm. In the same way our facility considers the use of remote-Maintenance System, which will allow us to projecting both the remote system as the maintenance program to larger installations. This system consists of a robotic arm which will be remotely operated, composed of two appendages, which will allow us to maneuver the components, each one will have video cameras which will give us a real time image of the working area. [16, 17], As system actuators is considered the use of servomotors thanks of the own internal control advantage , that control subsystem has inputs and outputs wich are easily manipulated. [18] Using a PAC (Programmable Automation Controller) [19] which are multifunctional modular controllers based on open protocols, which will provide us with the ease of the process without limitations of networks and protocols.

The diagnostic systems are important on magnetic confinement devices because they can help us to obtain vital information from several variables, establishing passive and active diagnostic, over the last area, we design an interferometry technique to measure the electronic density, is a multichannel (ten channels) taking account of previously researchs works where: its cutt-off density is higher, the refractive effect caused by density gradient is smaller, using a modulation Doppler shifted frequency technique with a device of the rotating gratting difraction. This interferometer has a great simplicity system optics, with these characteristics more interpretated measures can be obtained on the electron density profile, this technique was used in [20], the interferometer array is a Mach-Zehnder arrangement, laser HCN of 337 μm [21, 22].

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