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for the Ignitor Experiment

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Highlights of the Physics and Technology for the Ignitor Experiment*

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Abstract

The highlights of the most recent developments in the physics and technology of the Ignitor program are presented. These include the investigation of quasi-stationary nearly-ignited regimes and of ignition conditions achieved accessing the H-regime (most recent scalings) with "divertor" configurations. The ICRH system, the plasma control system, the construction of the high-speed pellet injector, the remote handling system, the fabrication procedures of the toroidal magnet plates, the detailed design of the plasma chamber, of the extended first wall and of the incorporated diagnostic system with the relevant R&D are among the engineering highlights.

1. Introduction

The main purpose of the Ignitor experiment is that of establishing the "reactor physics" in regimes close to ignition, as required for realistic and economical reactors, where the "thermonuclear instability" can set in with all its associated non-linear effects [1]. The driving parameter of the machine design ($R_0 \cong 1.32$ m, $a \times b \cong 0.47 \times 0.83$ m^2, triangularity $\delta \cong 0.4$) is the poloidal field pressure [$\bar{B}_p^2/(2\mu_0)$] that can contain, under macroscopically stable conditions, the peak plasma pressure $p_0 \cong 3 - 3.5$ MPa needed for ignition with central densities close to 10^{21} m^{-3}. The maximum design magnetic field on axis, excluding the paramagnetic contribution, is 13 T, and the plasma current $I_p \lesssim$ 11 MA, with a magnetic safety factor $q_\psi \cong 3.5$. The validity of the scientific objectives of the Ignitor Program and of key design solutions adopted for it has been recently reaffirmed [2], including the fact that reactor relevant physical regimes with $Q > 50$ have to be produced and investigated, and that the most appropriate solution at this time is the adoption of normal-conducting magnets. Furthermore, experiments that do not include a conventional divertor chamber can sustain, for equal overall sizes and magnetic field values, higher currents and therefore achieve better confinement parameters [2]. Ignitor is designed to operate with an "extended first wall" configuration, and with double X-points near the first wall and lower currents($I_p \cong 9$ MA). In fact, H-mode regimes become accessible according to recent scalings on the power threshold for similar configurations. Since the process of attaining ignition has been investigated extensively, a particular effort has been devoted to identify the conditions aided by the auxiliary heating system where the thermonuclear instability is barely prevented over the entire length of the current pulse but the factor Q is high. While tritium is the necessary step forward of any advanced fusion facility, Ignitor can provide novel and important results that justify its construction even when limited to operate with (non burning) H, D, and He plasmas in the early phase of its experimental life.

A significant amount of R&D activities is underway. For example, a pellet injector capable of injecting pellets of variable sizes with velocities in the range 1-4 km/sec, is nearly completed through a joint effort by ENEA-Frascati and the O.R.N.L. in the U.S. The injection of pellets is needed to provide proper control of the plasma density values, and of its radial profile. Fabrication techniques of magnetic diagnostics with inorganic insulator coatings capable of sustaining the expected neutron and gamma radiation background are being established; two prototype pick-up coils are ready for testing on present experiments. The construction of "second generation" toroidal magnet plates has been completed and has led to optimize their design and relevant fabrication procedures.

On the engineering side, the upgraded designs of the Plasma Chamber (PC) with variable thickness and of the First Wall systems have been completed together with those of the relevant remote handling and electrical diagnostic systems. The welding procedures for the final assembly of the plasma chamber and for the toroidal magnet plates are being developed. Updated scenarios for vertical plasma disruption events (VDE) have been considered [3] using an appropriate dynamic elasto-plastic structural analysis to verify the reliability of the design. The structural analysis of the machine Load Assembly has been improved including the friction coefficients at the interfaces between the significant components. The Poloidal Field Coil system, as presently designed, has been shown, using the CREATE_L response code [4], to be suitable to ensure the vertical stability of the plasma column and of its shape.

2. Steady State Operation, H-mode Regime, and Reduced Parameters Regimes

In addition to the reference "extended first wall" configuration for which Ignitor is designed in order to reach ignition conditions, we have investigated: i) nearly stationary, slightly sub-ignited regimes where the thermonuclear instability is not allowed to develop and the relevant plasma parameters are maintained by a modest amount of ICRH injected heating; ii) access to ignition with double X-point configurations, $B_T \cong 13$ T, $I_p \cong 9$ MA and entering the H-mode regime on the basis of recently established scalings; iii) reduced parameters, long pulse regimes with $B_T \cong 9$ T that are considered for "preparatory" experiments and are directed at surpassing the (significant) ideal ignition condition at high plasma densities where the α-particle heating compensates the bremsstrahlung emission.

A report of the positive results obtained from analyzing nearly ignited regimes (point i) is given in ref. [5]. Referring to point ii), given that new (more favorable) scaling laws for the access to the H-mode regime have been identified [6], the operation of Ignitor in this regime, with a double X-point configuration has been analyzed. Adopting $B_T \cong 13$ T and $I_p \cong 9$ MA (this "divertor" configuration does not allow higher currents) ignition can be achieved as the relevant power threshold is overcome by the combined contributions of Ohmic, α-particle and ICRH heating. The operating space corresponding to $Q \cong 10$ is found to be relatively broad, even considering the pessimistic case of rather flat density profiles. Moreover, considering in particular the JET experimental data [7] indicating that relatively peaked density profiles (e.g. $n_0/\langle n \rangle \cong 1.5$) can be obtained in the H-regime, values of Q considerably larger than 10 can be attained. Note that the adoption of recent scalings [8] with a weak dependence on β does not change the overall operating space significantly in the case of Ignitor.

Referring to point iii), two scenarios with magnetic fields up to 9 T and appropriate levels of injected ICRF heating were investigated: “extended first wall” (no X-point) configurations with plasma currents up to 7 MA and double X-point configurations with plasma currents up to 6 MA. In both cases, the compatibility of the poloidal flux consumption with the flux available from the Poloidal Field system was verified and the constraints on long pulse flat-tops related to the thermal conditions of the toroidal magnet system, were taken into account. By decreasing the peak density below the optimal value for ignition, and maintaining the plasma temperature by modest amounts of ICRH heating (below 5 MW) during the rise of the plasma current, steady state conditions are attained. Peak temperatures around the ideal ignition temperature (6 keV $\lesssim T_{e0} \cong T_{i0} \lesssim$ 8 keV), at which the plasma density can be increased without encountering a bremsstrahlung barrier, and a significant amount of α-particle (Q ~1) are produced (Figure 1). To this end many simulations were carried out with the JETTO code, which is capable of taking into account self-consistently the free-boundary plasma equilibrium evolution. Sawtooth oscillations are included, their onset being related to a critical peaking factor of the plasma pressure ($p_{kc}=p_0/<p>=2.7$).

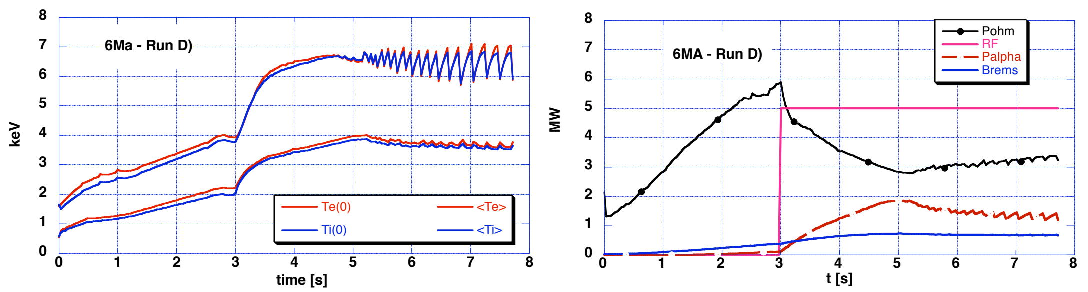

Figure 1. Peak and volume average temperature (left), heating and radiated power (right) for a 6 MA, 9 T, double X-point discharge with no edge transport barrier (JETTO).

3. Pellet Injector

The four barrel, two-stage pneumatic pellet injector for the Ignitor experiment is under construction in collaboration between ENEA - Frascati and O.R.N.L.. The goal is to reach pellet velocities of about 4 km/s (capable of penetrating the central region of the plasma column when injected from the low field side), in order to control the density profile, to provide a means of fuelling, and to interfere with the onset of the thermonuclear instability. Two important innovations at the basis of the Ignitor Pellet Injector (IPI) design are the fast shaping valves, allowing high pellet speeds, and the separation of the gas removal system into four identical and independent systems to avoid the need of large expansion volumes.

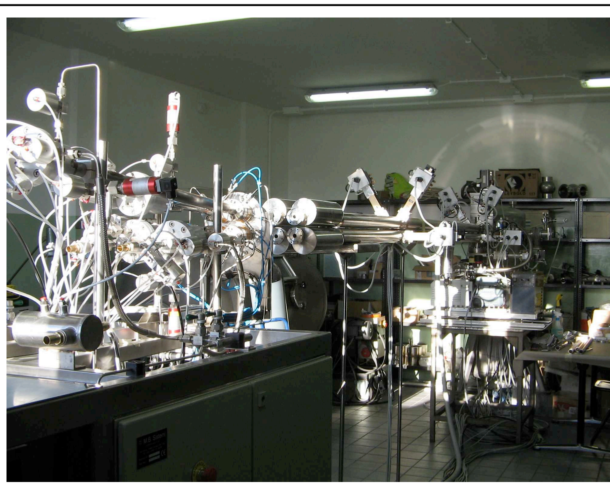

Figure 2. The Ignitor Pellet Injector propelling sub-system built by ENEA at Criotec Impianti.

The IPI consists of four independent injection lines, each including a two-stage pneumatic gun (TSG), a pulse shaping relief valve (PSV), a pipe-gun barrel, a propellant gas removal line and related diagnostics, sharing a single cryostat, a common pellet mass probe, and an accelerometer target. Two independent sub-systems have been built by ENEA and ORNL separately. The ENEA sub-system [9] includes the pneumatic propelling system and related diagnostics as well as its own control and data acquisition system. The equipment is ready for shipping to ORNL for final joint experiments.

Figure 3. The pellet injector cavity built at O.R.N.L.

The ORNL sub-system consists of the cryostat and pellet diagnostics (Figure 3). The assembly of most of its components was recently completed, and initial testing with D_2 pellets were carried out [10] at speeds of about 1 km/s. More than 30 pellets have been successfully formed and launched using the 3 mm bore barrel. Soon the ENEA and the ORNL systems will be integrated, and testing at high pellet speeds (up to 4 km/s or higher) will be carried out with a wide range of operating parameters explored. If results from testing at high speeds indicate that lower pellet temperatures than those provided by the present pulse tube cryo-refrigerator would be useful, the injector is equipped with components to accommodate supplemental cooling from a liquid helium dewar. This would allow pellet operation at temperatures approaching 5 K.

An analysis of D_2 pellet penetration for Ignitor was carried out, using the NGS ablation model [11], for the pellet sizes and speeds that the IPI is capable of producing and plasma temperatures and densities described by parameterized profiles, with central values ranging from 1 to 13 keV for the temperature, and 0.5 to 12.5 × $10^{20}m^{-3}$ for the density. Thanks to the compact size of the machine, it was found that pellets of 4 mm at 4 km/s could reach the central part of the plasma column for the typical plasma parameters at or near ignition conditions. At the lower parameters, smaller and slower pellets can be used. In order to assess the possibility of testing the new injector on existing experiments, the same model was used to simulate the penetration on JET, for a range of relevant parameters, assuming an injection from the low field side mid-plane and no mass drift effect. A more detailed analysis was carried out for a particular case of a plasma with an internal barrier [12], for different pellet sizes and speeds. Injection is from the low field side mid-plane and no mass drift effect is included in the calculation. The results show that a 5 mm pellet at 4 km/s can reach the region inside the ITB, where a higher peak density could produce considerably higher pressures [10]. Such an experiment would be useful also because future burning plasma experiments (e.g., Ignitor and ITER) will operate at a relatively low value of the dimensionless parameter ν^* and, for both devices, relatively peaked density profiles are expected to be beneficial from several perspectives; in particular these profiles can represent a stability edge against the so-called ITG modes that enhance the ion thermal energy transport.

4. ICRH System and Physics

The Ion Cyclotron Heating system is an integral part of the Ignitor experiment as it provides the flexibility to reach ignition regimes following different paths in parameter space and, in particular, by shortening the time needed for this. Another important use of the ICRH is to maintain the plasma in a slightly sub-ignited state, avoiding the excitation of the thermonuclear instability, under quasi-stationary conditions, for the entire duration of the plasma current flat-top. The ICRH system is structured with a modular configuration and launches the power into the plasma through RF strap-antennas based on 4 straps, grouped in two poloidal pairs, per port. The system is designed to operate in the frequency band 80 - 120 MHz. Each module consists of 4 high power generators whose power is split over two ports (8 straps) in order to keep the maximum electric field (especially in the vacuum region of the straps and transmission line) below 5kV/cm. A 30 Ω vacuum transmission line, including the feedthrough, transfers the power to each strap. The RF configuration of the modules allows a full phase controls (toroidal and poloidal) of the straps though a Phase Lock Loop (PLL) control. Two modules, distributed over 4 ports, can couple about 6 MW to the plasma under reliable conditions, in order to attain ignition with a limited RF pulse during the plasma heating phase. Two additional equatorial ports are designed to accept another module if an increase of the heating power is called for. When applying a pulse of ICRH in the process of reaching ignition the optimal frequency is about 115 MHz.

For the preparatory experiments at reduced machine parameters, the appropriate application of ICRH is studied in order to identify the power deposition profiles to be used in the transport analysis. In particular, a parametric study of the power deposition profiles as function of the minority concentration, minority species, frequency band for both configurations, has been carried out by using a full wave code in plane and toroidal geometry. An optimal frequency band is found in the range 85-95 MHz with a delivered power of 8 MW (extended wall configuration) and 5 MW (X-point). The results show that the power is essentially absorbed by the minority and redistributed by collisions to the main ion species of the plasma column.

5. Plasma Chamber, First Wall, Welding Procedures, and Thermal Wall Loads

The 12 D-shaped toroidal sectors of the Ignitor plasma chamber are made of Inconel 625 (forged or rolled) welded together by automatic remote equipments. A fabrication procedure of the plasma chamber [13] has been developed to cover all the manufacturing phases: the raw material specifications (including metallurgical processes), the machining operations required to achieve the final dimensional accuracy, the acceptance procedures and the vacuum tests. The design of the vertical supports of the plasma chamber has been upgraded to comply with the latest estimates of the electromagnetic loads due to eddy and halo currents plus the net horizontal force resulting from VDEs. In addition, the design of the radial support has been improved to allow better handling and maintenance. In particular, servicing of the radial supports can take place hands on with direct access from outside the cryostat without affecting the cryostat vacuum. The plasma chamber is maintained at room temperature by a dedicated duct system carrying helium gas. The system makes it possible to bake the plasma chamber at 200 ^{o}C and to cool it down prior to operation (about 18 hours), and provides the cooling after a current pulse (about 5 hours for a plasma discharge at full parameters).

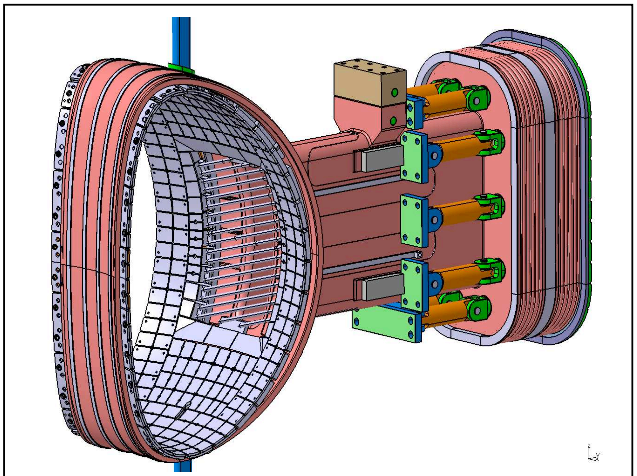

Figure 4. One sector of the plasma chamber including the ICRH Faraday shield and the first wall.

The plasma chamber is assembled into the complete torus by sequential welding of individual sectors already mounted onto the respective magnet sections. The sectors are first joined together by laser butt-welding that ensures vacuum tightness. Once the torus is completed, the welding groove is then filled by TIG-NG (Narrow Gap) to strengthen the joint. The welding process has been tested on a simple plate model, to validate the numerical simulations, carried out with the ABAQUS code, that allow the assessment of the residual stresses and deformations due to shrinkage [13].
The entire inner surface of the plasma chamber is protected by TZM (Molybdenum) tiles supported by inconel plates (tile carriers) attached to the PC. The electromagnetic loads on these elements during a reference VDE disruption have been evaluated with a detailed 3D finite element model. The thermo-structural non-linear analysis (using the ANSYS code) of the most stressed tile carrier under cycled loads shows that temperature increases, stresses and deformations are within the allowable values. The design of the first wall has been finalized taking into account the positioning of the electromagnetic diagnostics and the requirements for the Faraday shields of the ICRH antennas (Figure 4).

Considering the "extended first wall" configuration, the design of the First-Wall Limiter (FWL) is optimized to spread the plasma heat flux onto a relatively large area, thus reducing the peak values. An accurate estimate of the heat load distribution onto the FWL cannot be carried out by numerical codes with magnetic fitted coordinates like SOLPS [14] for cases such as that of Ignitor [15] in view of its geometry. This can be a relevant issue also for configuration with conventional divertors during the limiter start-up phase. Thus, a new code, ASPOEL [16], to model the edge of the plasma column is under development.

The present version of ASPOEL solves a set of single-fluid equations, neglecting neutrals and impurities. The power distribution onto the FWL is computed with the assumption that a given fraction of the input power is radiated. Instead of a classical Finite Elements technique, a triangular mesh, on which the Control Volume Finite Element (CVFE) technique [17] is applied to enforce local energy and particle conservation, has been adopted. The power conducted/convected onto the first wall can be separated into parallel and radial contributions. The former peaks where the LCFS-to-FWL distance is small, but vanishes where they become tangent, while the latter becomes maximum. Radial contributions to the heat flux onto the wall are confirmed by experiments. In fact, these had been taken into account empirically in previous estimates [18]. The peak values self-consistently computed with the new code (1.6 MW/m^2) are comparable with the older ones, although the maxima occur at the inner midplane rather than near the top and bottom of the plasma chamber.

6. Structural Analysis, Toroidal Magnet Plates and Remote Handling System

A Finite Element ANSYS model was used to analyze the non-linear mechanical behavior of the entire (machine) structure. The relevant calculation lead to find stresses that remain within the allowable limits at the relevant operating temperatures. The values of the interlaminar shear stresses values on the insulators of the toroidal field coils have been validated by the results of experimental tests performed by the Ansaldo Group. The non-linear analysis takes into account both the in-plane and the out-of-plane loads. Under normal operating conditions, the assumed friction coefficient on the wedging surfaces is adequate to assure the structural stability of the machine load assembly. Furthermore, once unloaded, the structure comes back without any permanent deformation. The safety factors of the average shear stresses against the insulation shear rupture strength at the beginning of the life of the machine is always greater than 3, while at the end of life this is reduced to about 2 because of

expected degradation of the mechanical properties due to the estimated neutron dose. Keys of proper dimensions between the 30^o extensions of the C-clamps modules have been adopted to ensure structural stability.
The Bitter type Toroidal Field Coils (TFC) adopted for Ignitor consist of plates that are cooled down to 30 K by Helium gas. Copper OFHC has been selected for these plates, allowing for an Electron Beam (EB) welding solution of the cooling channels. The Kabel Metal Group has defined the welding parameters and qualified the welding procedures. The qualification covers both the welding of the cooling channels and the inlet/outlet tube made on two full size samples. A metallographic examination and vacuum and pressure tests have been preformed to validate the basic suitability of the EB welding process.

The detailed design of the in-vessel Remote Handling System, based on the "two port concept" with two operating booms, has been completed. A 3D mock-up of the plasma chamber (PC) has made it possible to simulate the boom (Figure 5). This validates the ability of the boom, equipped with the attached end-effectors, to reach any in-vessel zone by 180^o on each side without interferences. Thus, the operating procedures applicable to several interventions have been established. Furthermore, a failure analysis of the boom components has been carried out in order to identify a recovery procedure. The design of the ex-vessel cabin with the function of holding the boom apparatus and managing the removal and installation of in-vessel components has been completed. The boom is made up by a sliding straight arm and articulated links. A structural analysis of both components under a maximum payload of 25 kg has estimated an acceptable deflection of about 7 mm.

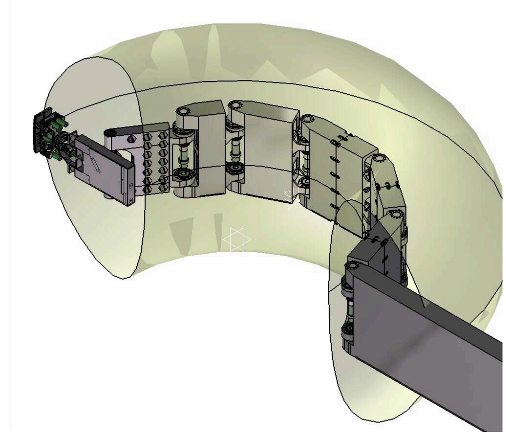

Figure 5. Fully extended boom inside the plasma chamber.

7. Plasma Control, Diagnostics, Site

The control of the plasma position and shape is a crucial issue in IGNITOR as in every compact, high field, elongated toroidal experiment. The capability of the Poloidal Field Coil (PFC) system, as presently designed, to provide an effective vertical stabilization of the plasma has been investigated [19 using the CREATE_L response model [4]. This linearized MHD model assumes axisymmetric deformable plasmas described by few global parameters.
An optimization of the vertical position control strategy has been carried out and the most effective coil combination has been selected to stabilize the plasma while fulfilling engineering constraints on the coils and minimizing the required power and voltage. The two pairs of coils selected for the vertical control will carry up-down anti-symmetric currents, superimposed to the equilibrium currents and provided by a dedicated power supply. The growth rate of the vertical instability and the power required by the active stabilization system have been estimated with this model, indicating that it is possible to design a control system able to guarantee a stability region that includes the most interesting operation conditions. A realistic description of the power supply system has permitted to carry out the optimization of the PID (Proportional-Integrative-Derivative) controller, both with a voltage and a current loop control scheme. In addition, an assessment of the requirements for the plasma cross section shape control has been carried out considering independent perturbations of the plasma global parameters as disturbances and showing that the undesired shape modification rejection is pos-

sible with the present PFC and power supply systems. The PF coils have been ranked on the basis of their capability to restore the shape modifications due to different plasma disturbances and the coil combination that minimizes recovery time and voltage required, has been selected.

The full set of electromagnetic diagnostics needed to measure plasma current, loop voltage, horizontal and vertical position, plasma shape, plasma beta, toroidal and poloidal modes has been incorporated in the detailed design of the plasma chamber and of the first wall systems. Given the neutron fluxes to which the electrical diagnostic systems of Ignitor will be exposed, new bonding techniques for inorganic insulators are being developed. A suitable MgO coating of the thin wires to be used as pick-up coils has been adopted. Other types of insulator are being molded to provide the support structures for larger in-vessel coils. At the same time, the possible failure of the relevant electromagnetic diagnostics has been taken into account, evaluating the robustness of the plasma position reconstruction strategy and investigating the possibility to use additional means to monitor the position of the plasma column, under demanding conditions, for example through the plasma X-ray emission or fast thermometric diagnostics.

An in depth analysis of the connection at Caorso, a former nuclear power site with suitable facilities, of the Ignitor power supply to the European electrical grid was carried out by the appropriate government authority (GRTN). It shows that the power requirements (max active and reactive power, max active power negative/positive steps) are consistent with the characteristics of the combined two terminals of the 400 kV grid at the site.

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