ADVANCES IN EUROPEAN IN-KIND CONTRIBUTIONS TO PLASMA DIAGNOSTICS AND PORT INTEGRATION FOR ITER

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Abstract

EURATOM contributions to the ITER Project include plasma and first wall diagnostics providing around a quarter of the primary measurements needed to avoid damage, to control the plasma and to explore its physics. These contributions include eight diagnostic systems, as well as six port plugs (housing a variety of diagnostics and other ITER systems) and extensive in-vessel electrical systems, providing connections to in-vessel Diagnostics. The design phase for these systems is close to completion and many are already in manufacture. This paper summarises the designs developed, and the challenges addressed.

1. INTRODUCTION

Fusion for Energy (F4E) is the European Agency delivering in-kind contributions to the ITER Project on behalf of EURATOM, which includes a wide range of diagnostics and ancillary systems. In most case, F4E has had responsibility for the design development. These designs must be compatible with high levels of radiation, limiting material options, and both thermal and electro-magnetic loads, all far greater than in the current generation of tokamaks. In-vessel components are subject to strict ultra-high vacuum requirements and many components form part of the ITER tritium confinement barriers, requiring extensive qualification subject to regulatory control. Lastly, despite the large size of the ITER tokamak, the space available for Diagnostic integration is remarkably limited and many systems are essentially inaccessible for the ITER lifetime, resulting in demanding reliability and availability requirements. In addition to these 'generic' challenges, the system designs faced many specific challenges to meet the ITER performance requirements. The following section describes how a few of these challenges were met during the design.

2. SYSTEM DESCRIPTION AND CHALLENGES

2.1. Collective Thomson Scattering system

The Collective Thomson Scattering (CTS) system will measure the density, and velocity distribution function of fast ions (p, D T and He3) in the range 0.1 to 1 MeV and confined alpha particles in the range 0.3 to 3.5 MeV; these having much higher relative densities than in present fusion machines, creating a substantial effect on the overall dynamics of the ITER fusion plasma [1]. ITER requires that these measurements are provided with 100 ms time resolution and 50 cm spatial resolution at the plasma core.

The CTS will introduce a 1MW, 60 GHz mm-wave 'probe' beam into the plasma, where it will be Thomson scattered from collective electron motion driven by the fast ions. The scattered radiation will be collected along seven lines of sight, intercepting the probe beam at different radial locations in the plasma. An eighth line of sight does not intercept the beam, and will be used for subtraction of background, mm-wave radiation from the plasma. The spectrum of the scattered radiation yields information on the 1D projection of the fast particle velocity distribution, from which advanced tomographic inversion techniques will be used to derive the required measurements [2].

The 1MW, 60GHz source will be provided by a gyrotron and transmission line, entering the ITER vacuum through a CVD diamond window at the rear of ITER Equatorial Port Plug #12. The port plug will house waveguides, mitre-bends (to avoid neutron streaming) and quasi-optical mirrors, that focus and launch the beam into the plasma through an aperture in the Diagnostic First Wall facing the plasma [3]. Scattered radiation from the plasma will pass through a second aperture onto a quasi-optical mirror that directs it to an array of focusing quasi-optical mirrors, defining the 8 lines of sight, and through waveguides, mitre-bends and quasi-optical mirrors to fused silica windows at the rear of the port plug. From there, the radiation passes through a transmission line to detectors.

The CTS port plug sub-system is currently in manufacturing. The design faced many, specific challenges. For example, most components in the port plug must be actively cooled because of high operational loads that combine those from the plasma with losses from the 1MW beam. Accessible surfaces inside the port plug must be covered with microwave absorbing coating to reduce coupling of stray mm-wave radiation from the plasma (e.g. from the

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EC heating system) to the fused silica windows, which have limited power handling capability. One particularly notable challenge related to the location of the fundamental electron cyclotron resonance at 60 GHz, which for the 5.3 T ITER baseline scenario lies inside the port plug [4], where the neutral gas density is poorly controlled and plasma breakdown leading to damaging arcs cannot be ruled out.

Avoidance of arc formation at the resonance is achieved by introducing a split, biased waveguide (SBWG) into the beam launcher line across the region of concern. The SBWG is a longitudinally-split waveguide, with the halves electrically isolated and biased with 1 – 2 kVDC. Particle-in-cell (PIC) calculations show that the bias causes free electrons to diffuse to the wall before they can generate an ionization-avalanche [4], for a broad range of likely environmental conditions (i.e. gas pressure, species and temperature). In addition, two dedicated systems are incorporated in the beam launcher line to monitor the beam parameters that will provide a signal to an interlock system for shutting down the gyrotron in the case of abnormal values.

2.2. Tokamak Services system

Tokamak Services provide the electrical infrastructure serving diagnostics mounted on the vacuum vessel and divertor cassettes. The scope is extensive, with four main sub-systems: in-vessel cabling (IVC), in-vessel supports (IVS), in-divertor electrical services (IDES) and in-vessel electrical feedthroughs (IV-EFT) [21].

IVC comprises around 1,200 mineral insulated (MI) cables, each approximately 10m in length, with a variety of cores (single, twisted twin, twisted quad, thermocouples, high current, coaxial, triaxial and RF power). Whilst a commercial product, significant adaptation was needed for ITER: use of low activation stainless steel for the outer jacket, development and qualification of a vacuum termination compatible with the radiation and thermal environment and a robust, homogenous 5µm thick copper coating (to minimise absorption of stray mm-waves, e.g. from the Electron Cyclotron heating system). IVS comprises thousands of cable clips, cable clamps and junction boxes (allowing permanent inter-connection between MI cables). Manufacture of IVC and IVS is almost complete, as these were one of the first diagnostics components required for ITER assembly.

IDES comprises more UHV-terminated MI cables, mechanical supports and electrical junction boxes, as well as electrical connectors. As the cassettes are designed for periodic replacement by remote handling (RH), the connectors must be also RH compatible. Space for the connectors in the divertor region is extremely limited, which represented a significant integration challenge. For example, a connector at the inboard side of the divertor had to be designed within a volume of 0.4 x 0.25 x 0.25 m^3 to fit up to 90 MI cables and 110 pairs of pins and sockets. Commercial components will be used for the pins and sockets but with material adaptations to avoid adhesion (e.g. cold welding) in the ITER environment, which required extensive qualification via prototyping. The IDES is in the final stage of design, prior to the start of manufacturing.

Finally, IV-EFT comprises around 80 electrical feedthrough assemblies, enabling up to 5,000 signals to cross the vacuum vessel boundary. The IV-EFT is currently in manufacturing. Since the IV-EFT represent a barrier to tritium escape from the vessel, ITER requires inclusion of a ‘double vacuum barrier’ so that the interspace (i.e. the volume between the barriers) can be connected to the ITER secondary service vacuum system and monitored for possible leaks. Glass-to-metal seals (GTMS) were selected during the design phase as the most suitable technology for the IV-EFT vacuum barriers, which comprise a structural body made of low activation stainless steel (the bulkhead), molybdenum pins (the electrical feedthrough) and a pair of alkali-barium glass seals, defining the interspace. The IV-EFT includes multiple GTMS assemblies integrated into the bulkhead, welded to a vacuum extension that supports an MI cable tail assembly connected to the pins.

The GTMS seal materials have significantly different thermal coefficient expansions and development of a precise process was required to optimally soften the glass and allow proper flow over the surface, with carefully controlled duration and temperature steps in the furnace from room temperature up to 950 C, to ensure a tight seal reaching the ITER leak rate requirement of around 10^{-10} $Pa.m^3/s$. Qualification of the GTMS process identified several critical factors for the IV-EFT seals: the pins must be in a vertical orientation, placed on a horizontal surface to guarantee specimen stability, and with a limited number of units per batch to avoid disturbing furnace conditions. The pin geometry evolved during the design from a straight rod to a structure incorporating undulations in the interspace region to relieve thermal stresses during sealing.

2.3. Diagnostic Pressure Gauges

The Diagnostic Pressure Gauges (DPGs) will provide data related to the pressure of neutral particles in ITER during plasma operations, at seven poloidal locations in the divertor cassettes (DC), in the ducts of the lower ports (LP) and in the equatorial ports (EP). ITER requires that the DPGs in the EP are calibrated for a pressure range from 10^{-4} to 1 Pa, and those in the DC and LP from 10^{-2} to 20 Pa, with an accuracy of 20 % and time resolution of 50 ms.

In the design phase, several technologies were studied for their suitability to the environment and measurement requirements. Ultimately Bayard-Alpert ionization gauges were selected, with adaptations to allow for operation in a magnetic field up to 8 T, as experienced at the inner divertor on ITER. In these gauges, neutral particles are ionised by a source of accelerated electrons. The ions are collected by a biased electrode and the ion current, for a given current of electrons, is well correlated to the neutral density. The adaptation for high magnetic field was originally developed for the ASDEX tokamak [5] and uses planar electrodes to accelerate the electrons and collect the ions. Whilst well suited to the ITER environment, this approach measures the neutral density rather than absolute pressure and interpretation of the measurement is dependent on knowledge of both the neutral species and temperature, as well as calibration for each species.

The electron source, electron acceleration grid and ion collector form the DPG 'head'. The head is mostly enclosed in a metal box to protect it from direct exposure to charged particle and photon fluxes from the ITER plasma, which would impact the measurement, and to form a local environment where the neutrals can thermalise to the box temperature, measured by a thermocouple. This is a standard component, oriented to the magnetic field at the different DPG locations.

The DPG are now in the final stages of design, prior to manufacturing. Development of a stable, long-lasting electron source represented the most notable design challenge. Thermionic emission was selected as the most suitable technology, given constraints on the space available and the high magnetic field. Simple filaments, however, required high currents that generate significant Lorentz forces. Prototyping demonstrated that these forces would result in creep that excessively limited the lifetime. Instead, a 1mm thick zirconium carbide crystal is used [6]. The crystal is stacked between pyrolytic graphite plates and a current up to 15A is passed through the stack via 0.1 mm thick iridium foil electrodes. Ohmic heating of the graphite raises the crystal to around 2,000 K. The stack is held under compression by a sprung tungsten-rhenium frame, from which it is thermally and electrically isolated using, respectively, Zirconium Dioxide and Aluminium Dioxide plates.

Prototyping has demonstrated that the electron emitter stack will have an operational lifetime of 860 hours, or around 7,000 ITER pulses, relative to the envisaged 30,000 pulses over the ITER lifetime. In addition, a stack at operational temperature would not be expected to survive any loss of coolant events that may occur during early ITER operations. In order to meet the ITER reliability and availability requirements, therefore, a high degree of redundancy of DPG is required; 52 DPG are distributed across the seven locations, with only 1 DPG operational at any time in any location.

2.4. Radial Neutron Camera

The Radial Neutron Camera (RNC) will measure the line-integrated flux of un-collided 14 MeV (DT) and 2.5 MeV (DD) neutrons across a fan array of poloidal chords viewed from the outer mid-plane. The measurement, together with that from a companion system viewing vertically, will be used to derive the neutron emissivity and alpha particle source profile through tomographic inversion. ITER requires that these measurements are provided with 10 ms time resolution and 20 cm spatial resolution.

The poloidal chords will be defined by collimating structures made of neutron absorbent materials. To achieve the required spatial coverage and resolution, the RNC design includes an array of 16 collimators viewing the core of the plasma from the ex-vessel (i.e. air side) of ITER Equatorial Port Plug #1, and a second array of 6 collimators viewing the plasma edge from inside the plug ('in-vessel') [7]. Neutrons arriving to these collimators will pass through apertures in the Diagnostic First Wall facing the plasma and, for the ex-vessel, transit lines of sight through the interior port structures and pass to the air side via a thinned region in the stainless steel at the rear of the port plug. The in-vessel collimators will be formed from stainless steel whereas a combination of stainless steel and concrete will be used ex-vessel. Neutron flux detectors will be located at the rear of the collimators. Despite local neutron shielding (e.g. Boron Carbide for the in-vessel detectors), these detectors will be exposed unavoidably to large neutron fluences over the ITER lifetime that can cause significant degradation.

The RNC ex-vessel systems are now in the final stages of design, prior to manufacturing, whilst manufacturing has already begun for the in-vessel systems. Challenges encountered during the design include the need for mechanisms to re-align the massive (17 t) ex-vessel collimator array to the port plug between operational and maintenance periods (i.e. due to differential thermal expansion), and for short cable runs to neutron detector signal conditioning electronics therefore needing bulky neutron and magnetic shielding. Developing a design for the neutron detectors with sufficient sensitivity but also adequate lifetime represented, however, the most notable challenge.

For both the in-vessel and ex-vessel systems, a combination of neutron detectors will be deployed on each collimator. For the in-vessel, a single crystal CVD diamond (sCD) matrix made of 32 diamond pixels of 50 micron thick will allow discrimination between DD and DT neutrons during early ITER operations (DT-1), but is expected to fail soon after full power DT operations (DT-2) begin [8]. For this phase and beyond a high-purity

U238 fission chamber (FC) is included in the design. The FC cannot discriminate but is a robust technology and prototyping has demonstrated that it is compliant with reliability and availability requirements over the ITER lifetime. The FC and sCD will be arranged colinearly and housed in a single unit. Since the FC is pressurised to 4 bars, all the detector modules are housed in a sealed 'cassette' connected to the ITER secondary service vacuum system that monitors for possible leaks.

For the ex-vessel, three detector types will be used: a plastic scintillator, a He4 scintillator operating at 80 bars and a 4-pixel sCD, again arranged colinearly and housed in a single unit together with photomultipliers for the scintillators. This combination offers a variety of sensitivities chosen to ensure sufficient redundancy and to cover the wide range of DD and DT neutron emissivity in the plasma core region. The plastic scintillator will degrade in DT-2 but is compensated by the He4 scintillator. Prototyping of the He4 scintillator, which is a novel technology for fusion, demonstrated good intrinsic radiation resistance with good linearity and gamma-ray rejection properties [9].

2.5. Bolometer diagnostic

The bolometer diagnostic will provide measurements of line-integrated, total plasma radiation along chords distributed poloidally and toroidally, from which parameters such as the total radiation emission profile will be derived by tomographic inversion. ITER requires that this is provided with 10 ms time resolution and 5 cm spatial resolution in the divertor region.

To achieve the spatial resolution the diagnostic will use 467 chords, these being defined by collimators or pin-holes forming part of bolometer cameras that also house the bolometer sensors and electrical connections: 22, 5-channel cameras mounted on the vacuum vessel, viewing the plasma through gaps between the blanket modules, 40 cameras mounted on five divertor cassettes, viewing through gaps between divertor tiles and dome, and in six, multi-channel (10 – 60) cameras in three ITER Port Plugs, viewing through apertures in the Diagnostic First Wall facing the plasma. The sensors are sensitive from IR to x-ray and measure up to 95% of the total power radiated by the plasma (depending on plasma conditions, such as impurity content and temperature).

The camera subsystems and electronics are in the final stages of design prior to manufacture. Designing cameras compatible with the high loads from plasma radiation, nuclear heating and disruptions (i.e. EM forces) has been a particular challenge, see e.g. [10]. The design and prototyping of electronics that can cope with signal cables of over 200 m in length are ongoing. The data analysis also poses challenges due to many effects that impact the measurements to a much higher degree than on present-day fusion experiments (e.g. 100's °C variation in sensor temperature and local neutral pressures up to 20 Pa). A simplified method of predicting performance of the derivation of total radiated power has been developed [11].

The sensors integrated in cameras represented a notable challenge due to high levels of nuclear radiation, accelerations during plasma disruptions, pressure transients, risk of steam exposure (during possible loss of coolant events) and operating temperatures up to 400°C. They must also cope with very high, pulsed heat loads when the disruption mitigation system is activated. These sensors are delicate, micromachined components but must operate for 20 years in these conditions, whilst maintaining sufficient dynamic range.

An extensive prototyping and testing programme [12][13] resulted in the qualification of a sensor design using a balanced, resistive bridge on the surface of a 3 μm thick silicon-nitride substrate. This sensor is similar to those in use in present-day fusion experiments, but with a 20 μm thick gold absorber on the substrate to capture adequately x-rays from high-temperature ITER plasmas. Tests under neutron/gamma irradiation in a research fission reactor, which took several years to design and execute, reproduced lattice damage in the sensor substrate broadly equivalent to the 0.05–0.25 dpa level expected in the various bolometer camera locations over the required lifetime. The sensors did not show any damage in post-irradiation examination after exposure to 0.09 dpa ±7%, but on-line resistance measurements and calibrations showed that their functioning is impaired after 0.023 dpa ±7% has been reached. Despite extensive measurements (including of insulation resistance) and thermal cycles before, during and after irradiation, the fission reactor environment is very challenging and the results are not conclusive in how far the effect is caused by flux, fluence or temperature. Nevertheless, the sensor functionality was restored after the reactor cycle. It is possible that the magnitude of effects observed resulted from the irradiation flux in the accelerated fission-reactor tests, which is far higher than expected in ITER. It may be expected that the frequent thermal cycling during normal ITER operations will anneal (i.e. repair) some of the lattice damage. There are thus good prospects that the sensors will function for a considerable time in ITER, with risk increasing beyond 0.023 dpa.

2.6. Core Plasma Thomson Scattering system

The Core Plasma Thomson Scattering System (CPTS) will measure the radial profile of electron temperature (T_e) and density (n_e) in the core plasma by spectral analysis of light elastically scattered by free electrons in the plasma from a powerful, near-IR laser. Doppler broadening of the narrow laser line being related T_e whilst intensity of the scatter light is related to n_e. ITER requires that this is provided with 10 ms time resolution and 20 cm spatial resolution in the core of the plasma, for $0.5 < T_e < 25$ keV and $3x10^{19}$ m^{-3} $< n_e < 3x10^{20}$ m^{-3}.

The CPTS design [14] envisages use of commercial-grade laser sources: a main laser (Nd:YAG : 1064 nm, 500 W, 100 Hz x 4 ns) will be optically combined with a supplementary laser (Nd:YAG : 1320 nm, 20 W, 10 Hz x 4 ns), for calibration and measurement of T_e above 25 keV, and several alignment lasers. The combined source will be generated in the ITER Diagnostic building and relayed (mirrors, flight tubes) over 50 m, entering the ITER vacuum through a fused silica window at the rear of ITER Equatorial Port Plug #10, passing into the plasma through an aperture in the Diagnostic First Wall and finally arriving at a beam dump embedded in a blanket at the inner wall. A set of focussing and planar mirrors mounted in the port plug will view the intersection of the laser with the plasma, collecting scattered light from across the outboard radius and guiding it through a series of periscopes (to avoid neutron streaming) through a window in the rear of the plug. Collection optics (mirrors and lenses) will focus the light onto a fibre bundle, that delivers it to polychromators in the Diagnostic building for spectral analysis.

The design of the CPTS is nearing completion, and has faced many, specific challenges. For example, the need to minimise power density on the laser windows, which are tritium confinement barriers, required beam expanders and avoidance of irregularities (i.e. peaks) in the laser beam profile due to e.g. diffraction effects in the optical relay. Since almost all the beam crosses the plasma without scattering, the beam dump will receive high power loads and needed careful choice of materials (Molybdenum was selected, for its combination of thermal conductivity and high damage threshold/melting point) and geometry (i.e. maximisation of the absorption surface, to reduce the power density, by inclusion of triangular 'blades'). One of the most challenging aspects of the CPTS design is design of the ex-vessel collection optics and ancillary systems.

The collection optics design [15] addresses a broad range of design constraints. Maintaining high resolution over a wide spectral range is achieved by use of compound achromatic lenses, involving a variety of refractive materials. These materials must be resistant to neutron-induced UV darkening, which limits the achievable achromaticity. Active alignment of the system must account for any misalignments that may arise from the relative displacement between the in-vessel and ex-vessel zones of the collection optics due to thermal and mechanical loads that arise during operation. This is accomplished by actuating two ex-vessel mirrors (in a dog-leg formation) at the rear of the port plug, and the fibre bundle support structure, during operation. The actuation is governed by a laser-based monitoring system that measures the relative position between the port plug and the ex-vessel structure.

2.7. Diagnostic Ports Integration

For the six EU ports the design of the diagnostic ports encompasses both in-vessel and ex-vessel structures as well as design integration of the Diagnostic systems housed in the ports. The in-vessel structures, situated within the port plugs, include the Diagnostic Shielding Modules, which are regarded as the most critical elements due to the challenging environment and the complexity of the Diagnostic sub-assemblies that they host. Additionally, these components feature cooling water, gas, and electrical services, including feedthroughs installed at the primary confinement barrier, which provide first confinement barrier for tritium and other hazardous substances, classified as a safety function. The ex-vessel components consist of the Interspace, Bioshield Plug, and Port Cell structures, which also accommodate many Diagnostic sub-systems [22].

The EU Diagnostic Ports (EU Ports) play an essential role in the ITER Tokamak, providing a platform for housing advanced diagnostics systems that enable assessment and control of plasma and overall reactor performance. Situated at the L1 and L2 horizontal levels, the EU ports are engineered to withstand harsh environments characterized by high radiation, vacuum, electromagnetic forces, and seismic activity. Their design reflects a careful balance of robust structure, safety assurance, and the flexibility required for integrating a wide variety of diagnostic technologies.

F4E's mandate within the EU Diagnostic Ports framework encompasses the full engineering lifecycle: from supplying specialized steel grades and manufacturing port structures to fabricating water, gas, and electrical feedthroughs. Key deliverables include, design and integration of in-vessel and ex-vessel structures, production and assembly of Diagnostic Shielding Modules (DSMs), port plugs, interspace, and port cell assemblies. Development and supply of 24 diagnostic payload subsystems, designed and manufactured by up to five different Domestic Agencies (DA), includes the implementation of safety-critical features such as tritium barriers and radiation shielding, qualification and factory acceptance testing through a dedicated Port Plug Test Facility, ongoing support for installation, operation, maintenance, and disassembly.

The project is moving into procurement, which will result in a contract covering materials, tooling, engineering, testing facilities, and on-site installation.

The complexity of EU Diagnostic Ports is underscored by several engineering and operational challenges, including the requirement for material and manufacturing precision with strict tolerances below 50 microns for large structures, integration of advanced diagnostic payloads systems within limited spatial constraints, implementation of radiation shielding using boron carbide, Novashield, Polybore, and concrete-filled steel assemblies, application of feedthrough technology including glass-to-metal seals, ultra-high vacuum mineral insulated cables, and specialized sealing methods like Helicoflex, as well as the maintenance of cleanroom conditions during assembly with integrated functional testing protocols to ensure system reliability.

Some design challenges include achieving efficient cooling in the DSM while maximizing neutron shielding. This has been addressed by incorporating 3D cooling channels machined into the front part of the SS 316 L(N) IG DSM's where greater heat dissipation shielding is required, steel and water ratios are 75 and 25 approximately. A conventional pipe design is used in the back section of the DSMs. The design also involves developing suitable laser welds to seal the machined cooling channels, with extensive testing conducted to optimize both geometry and welding technology.

Shielding design improvements have been achieved by adding B4C pellet boxes to DSM structures (weighing 3–7 t, measuring 1–2 m) to increase neutron absorption and reduce weight while allowing flexible configurations. Characterizing pellet composition and shape by specifying relevant parameters and their ranges to optimize heat dissipation has also been an important activity. Test have been performed to characterize B4C pellet geometry proposals, along with detailed analysis.

2.8. Magnetics Diagnostic

The Magnetics Diagnostic includes (but is not limited to) a wide variety of coils mounted on the interior (IVC) and exterior (OVC) surfaces of vacuum vessel, on divertor cassettes (DVC) and inside the TF coils (CER). These coils, around 900 in total, measure fluctuations in the local magnetic field, which are electronically integrated in some cases to provide an absolute measurement of the field. These coils, and the associated electronics and software, contribute to measurement of the plasma current, halo currents, MHD and reconstruction of the plasma equilibrium. Delivery to ITER is almost complete.

The Magnetics Diagnostic electronics architecture includes more than 1600 state-of-the-art electronic integrators, achieving a drift well below the ITER requirement of 500 µVs over 3,600 s [16]. The electronics are installed in 17 dedicated I&C cubicles, forming a distributed design where tens of aggregator boards equipped with FPGA system-on-modules (SoM) and embedded processors enable real-time processing of all channels at 2 MHz sampling rates. A modular network architecture with <100 µs latency streams the data to GPU-equipped servers, supporting the computation of critical plasma parameters, including plasma current, velocity, and shape [17], essential for the real-time operation of ITER.

The CER are wound Rogowski coils housed within the toroidal field (TF) cases and surrounding the vacuum vessel. They measure net current through their loop, which is dominated by the plasma current but also includes currents flowing in the vacuum vessel. The IVC, OVC and DVC are mostly pick-up coils (either wound or printed), strategically placed on the vacuum vessel surfaces and divertor regions and oriented to measure fluctuations in the radial, poloidal and toroidal fields.

Each pick-up coil is integrated into a platform tailored for its location, providing the necessary support, electromagnetic shielding (to prevent damage from stray mm-wave radiation), thermal conduction, electrical connectivity, and remote handling capabilities. Local conditions and the specific measurement objective for the coil constrained design choices. For example, high-frequency coils (for MHD measurement) use glipcop Al-15 wound on shapal Hi-M formers to achieve the frequency response. Diamagnetic coils, which measure the variation of toroidal flux induced by the plasma, are wound from mineral insulated cables brazed to a CuCrZr bobbin. In some cases, the available space was so limited that producing a coil with adequate sensitivity (i.e. number of turns x coil area) required them to be printed on multi-layer low-temperature cofired ceramics (LTCC).

Numerous challenges were encountered during manufacture of coils: handling fragile materials such as alumina mandrels and AlN ceramics, high-precision machining of thin-walled Inconel covers, and specialized welding techniques for sensor integration were critical hurdles. Calibration processes, especially for OVC, had to account for differential thermal expansion and delicate coil movements, necessitating redesigns and careful interpretation of measurements.

The LTCC design posed notable challenges, including the need for qualification prototyping of the chosen technology in a fission reactor, to simulate the ITER neutron fluence. Manufacturing required precise control of shrinkage between ceramic and metal layers, robust joint designs to prevent delamination, and optimization of thermal behaviour for operational reliability. These challenges were solved by implementing precise process controls during the cofiring stage, optimizing joint designs to prevent delamination, and rigorously monitoring thermal behaviour to ensure sensor reliability. The control of thickness and joint robustness was achieved through iterative prototyping and close collaboration with suppliers, resulting in high-quality sensors that met the demanding operational requirements of the ITER environment.

2.9. Wide-Angle Viewing System

The equatorial visible and infra-red (IR) Wide-Angle Viewing System (WAVS) provides images of the plasma and of the plasma facing components (PFC). The images in the IR range are used to determine PFCs surface temperatures and to avoid potential damages (machine protection function) and contribution to ITER physics studies. The system uses 15 lines of sight in 4 equatorial ports, targeting a 80% coverage of the in-vessel PFCs. Components for Equatorial Port Plug #12 are in manufacture, whilst the design of the remaining components is nearing completion.

The Port Plug houses the First Mirror Units (FMUs), each containing two actively water-cooled mirrors (M1 and M2) with integrated pneumatic shutters and Radio Frequency (RF) cleaning systems to mitigate co-deposition and to keep acceptable optical performances under ITER conditions. The active cooling is key to maintain temperature below the minimum temperature of PFCs. Both mirrors are Rh coated, the first of which has an off-axis ellipsoidal shape, the second one being flat. Those shapes are a consequence of the optical requirements (such as field of view, resolution etc.) The light is then transferred to a so-called Hot (as it is not cooled) Dog Leg (HDL) composed of two flat mirrors, allowing to transfer the light to the Vacuum Window while minimizing the neutron streaming [18]. The FMUs and HDLs are embedded within a Diagnostic Shielding Module (DSM), which provides structural support and neutron shielding.

Beyond the Port Plug, the ex-vessel optical chain extends through the Interspace, Bioshield, and Port Cell, relaying the image over nearly 10 meters to the diagnostic cameras. The Interspace contains the Optical Hinge (OH) and Optical Relay Unit (ORU), both mounted on a common mechanical supporting structure. The OH compensates for the differential displacements between the vacuum vessel and the building. This OH compensation function relies on a piezo-actuated mirror system, validated for radiation hardness and remote operation. The ORU uses off-axis aspherical mirrors to produce a collimated beam, which is then relayed by refractive afocal modules (IAM) and further folded by Cold Dog Leg (CDL) mirrors in the Bioshield. In the Port Cell, the beam passes through additional afocal modules (PCAM) and enters a shielded cabinet (SC) that houses dichroic beam splitters, filter wheels, and four cameras per line of sight (two visible, two IR) [19]. The SC provides neutron, gamma, and magnetic shielding, and includes an active water-cooling circuit for the IR cameras, ensuring operational stability and protection of sensitive electronics.

The development of WAVS faced several challenges, like defining a robust opto-mechanical design to ensure appropriate optical performance under demanding nuclear, radiative and electro-magnetic loads; integrating active cooling, shutter protection, RF cleaning, and remote handling features into compact FMUs assemblies; extensive R&D and prototyping activities in order to select suitable materials for withstanding harsh ITER environmental conditions and ensuring a reliable motion compensation in the OH. This was solved by selecting a custom piezo actuator (PI N-216) capable of delivering suitable force and movement accuracy. That solution was validated under gamma irradiation and tested with a 110-meter distance between piezo actuator and the remote controller.

2.10. Charge Exchange Recombination Spectrometer

The core plasma Charge Exchange Recombination Spectrometer (CXRS) is an optical, port-based system providing spatially resolved measurements of the fuel ion temperature, plasma rotation, and impurity concentrations, by analysing light emitted during charge exchange between plasma ions and energetic neutrals, injected by a dedicated neutral beam system. The system includes collection opto-mechanical components (e.g. mirrors and lenses) and calibration and alignment systems, a shutter and a mirror cleaning system, an optical fibre bundle, high resolution spectrometers and instrumentation and control. The design of the CXRS is at final design phase level.

The system starts with a front-end optical assembly in ITER Upper Port #3, where a water-cooled first mirror (M1) collects light emitted, specifically, from plasma-neutral beam interactions. M1 assembly is protected by a pneumatically actuated shutter, which also holds a retroreflector for calibration purpose; and hosts an RF plasma cleaning system. The light collected by M1 is transmitted through a labyrinth of four mirrors (M2–M5), with M1 being flat and M2 to M5 being toroidal to optimize image quality and minimize aberrations. The optical path continues through a vacuum window assembly, which uses dual fused silica disks with monitored interspace for leak detection.

Outside the vacuum vessel, the optical transport system is composed of four main subsystems: the Long Focal Telescope (LFT), the Cold Dog Leg (CDL), the Optical Bench Assembly (OBA) and the Optical Fibre Bundle (OFB). The LFT and CDL use precision-aligned, motorized mirrors to compensate for thermal expansion and ensure alignment during operation. The OBA, mounted on a hexapod, injects the collected light into a fibre bundle with minimal loss. This fibre bundle delivers the signal to the Tritium Building, where it is distributed to a calibration bench and to high-resolution spectrometers. The calibration bench enables in-situ radiometric and wavelength calibration, ensuring measurement accuracy. The spectrometers, equipped with custom optics and high-sensitivity cameras, analyse the spectral lines to extract ion temperature, rotation velocity, and impurity densities with high spatial and temporal resolution [20].

The development of CXRS faced several challenges. First, the optical design required extremely tight tolerances for mirror alignment and positioning. Minor deviations could degrade spatial resolution and measurement accuracy. Second, the system had to be robust against thermal and mechanical deformations caused by neutron and gamma heating, requiring motorized adjustment mechanisms and extensive thermo-mechanical modelling. Third, the integration of the radio-frequency cleaning system requires respecting the current and voltage limits of the electrical feedthrough and cabling, while ensuring adequate power delivery to the mirror surface. Additionally, the need for remote handling compatibility imposed constraints on component interfaces and maintenance procedures, particularly for the replacement and fine alignment of mirrors within the Diagnostic Shield Module (DSM).

One of the most critical challenges is the need for remote handling compatibility within the DSM, as manual adjustment is not possible in the activated environment. This imposes constraints on component interfaces and maintenance procedures, particularly for the replacement and fine alignment of mirrors within the Diagnostic Shield Module (DSM) To solve this issue a custom remote handling tool has been designed, to be used in the Hot Cell to position the new mirror against three lateral contact points within the DSM, using a cam dowel mechanism operated by a remote bolt runner. This approach guarantees repeatable, unambiguous positioning, preserving the optical axis and system performance without the need for post-installation manual adjustment.

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