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Diagnostics for plasma control - from ITER to DEMO

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The plasma diagnostic and control (D&C) system for a future tokamak demonstration fusion reactor (DEMO) will have to provide reliable operation near technical and physics limits, while its front-end components will be subject to strong adverse effects within the nuclear and high temperature plasma environment. The ongoing developments for the ITER D&C system represent an important starting point for progressing towards DEMO. Requirements for detailed exploration of physics are however pushing the ITER diagnostic design towards using sophisticated methods and aiming for large spatial coverage and high signal intensities, so that many front-end components have to be mounted in forward positions. In many cases this results in a rapid aging of diagnostic components, so that additional measures like protection shutters, plasma based mirror cleaning or modular approaches for frequent maintenance and exchange are being developed.

Under the even stronger fluences of plasma particles, neutron/gamma and radiation loads on DEMO, durable and reliable signals for plasma control can only be obtained by selecting diagnostic methods with regard to their robustness, and retracting vulnerable front-end components into protected locations. Based on this approach, an initial DEMO D&C concept is presented, which covers all major control issues by signals to be derived from at least two different diagnostic methods (risk mitigation).

Keywords: ITER, DEMO, tokamak, plasma control, plasma diagnostics

1. Introduction

The European (EU) long-term strategy towards fusion energy foresees the development of a demonstration fusion reactor (DEMO) as the single step between the experiment ITER and a commercial fusion power plant. DEMO should deliver significant net electrical power into the grid by the mid of the 21st century, achieve tritium self-sufficiency, and allow for a safe extrapolation towards the economic viability of a commercial fusion power plant [1]. The baseline concept within the current EU DEMO studies is a tokamak with mainly inductively driven long pulse operation. The development strategy follows a conservative approach [2] assuming only moderate physics and technology extrapolations beyond the status of ITER. This approach is chosen in order to facilitate a timely development under the boundary conditions of limited resources and taking into account the status and

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schedule of ITER development. Some parameters of the EU DEMO baseline concept 2018 are listed in table 1.

Table 1: Main parameters of the current DEMO concept

Parameter	Value
Plasma major radius R_0 and minor radius a	9 m 2.9 m
Toroidal magnetic field B_0	5.9 T
Pulse duration t	2 h
Thermal power P_{th}	2000 MW
Net electrical output power P_{el}	500 MW

The output power for the current DEMO concept has been predicted assuming a standard ELMy H mode scenario with confinement quality H ~ 1.0 [3]. However, the DEMO scenario should have no edge localized modes (ELMs) or only very low energy ELMs below the threshold for wall damage. Furthermore, high core plasma radiation power is required to limit the power flowing towards the divertor, while high plasma edge density is needed to facilitate detached plasma operation in the lower single-null divertor. These additional requirements may cause some reduction of output power as compared to the standard H mode. The time-averaged auxiliary heating power applied will be in the order of 50 MW mainly for plasma control purposes and with only minor impact on the pulse duration via current drive. Some more details on the current physics and technology basis and their gaps have been published by Wenninger et al. [4]. It is important to note that not all of the key features of the DEMO plasma scenario and technology are well defined yet, nor have they been simultaneously demonstrated in large experiments under relevant conditions so far. Thus, at the current stage of DEMO studies, the development of the diagnostic and control system has to be pursued in a generic way, taking into account the significant uncertainties concerning the definition of the plasma scenario and machine properties.

The first tokamak producing significant fusion power will be ITER. Given the plasma parameters and the nuclear environment of the ITER machine, which both represent a large step from smaller experimental tokamaks towards DEMO, the ongoing developments for the ITER diagnostic and control system are an important basis for any considerations towards DEMO diagnostic and control. The ITER diagnostic suite under development [5, 6] has to serve the needs for both plasma control and detailed physics investigations in a burning plasma experiment with predominant alpha particle heating and with moderate neutron fluences up to damage levels in the order of one displacement per atom (1 dpa) near the first wall. Engineering challenges for the realization of ITER diagnostics [7-9] and in particular the nuclear aspects [10] have led to the development of important concepts such as the port plug based integration approach, maintenance of diagnostic components via remote handling and the selection of irradiation-hard functional materials for diagnostics components. Physics requirements are however driving the ITER diagnostic design towards using sophisticated methods with large spatial coverage and high resolution, leading to designs with many vulnerable components mounted in forward positions. In order to protect or refurbish these components, concepts like optical in-vessel shutters [11] and in-situ mirror cleaning [12] are being developed.

In addition to the open issues towards the DEMO physics basis and the definition and validation of the plasma scenario, the development of the plasma diagnostic and control (D&C) system for DEMO is facing a number of significant challenges, which go far beyond the situation for ITER [13-16]. The DEMO D&C system has to provide high reliability, since any loss of plasma control may result in loss of confinement or ultimately disruptions, where the latter may cause significant damage of the inner wall or other components of the machine. On the same time, a high accuracy of the DEMO D&C system is needed in order to reliably operate DEMO near its operational limits, where the power output of the reactor is maximized. Additionally, fast reactions by the D&C system are required in particular in case of unforeseen transient events (e.g. component failure, or radiation increase following impurity ingress into the core plasma). On the other hand, space restrictions for the implementation of diagnostic components in the blanket (for the achievement of a sufficient tritium breeding rate) have to be observed. Moreover, the adverse effects acting on the diagnostic front-end components (neutron and gamma radiation, heat loads, erosion and deposition) will be much stronger than on ITER, resulting in limited performance of measurements, while the capabilities of the available actuators for plasma control (poloidal field coils, auxiliary heating and fueling) are limited as well.

In order to improve the controllability of the DEMO plasma in view of these limitations of available diagnostics and actuators, advanced control techniques will be employed, which aim to provide either a fast state description of the plasma based on the measured data, or deliver model-based predictions towards optimized actuator trajectories [17].

As part of the European DEMO conceptual design studies, the development of the D&C system has been launched within the work package “diagnostic and control” (WPDC) [15]. During the first three years of work, an initial understanding of the prime choices of diagnostic methods and actuators applicable to DEMO has been obtained. In order to prepare the physics models for future advanced control schemes, and to provide some quantitative verification towards the controllability of the DEMO plasma, control simulations are being developed for a number of control issues. In the current status the D&C concept only addresses the stationary burn phase of the discharge. The ramp-up and ramp-down phases, the heating up towards the burn phase, as well as control of instabilities and emergency actions such as disruption mitigation or fast shut-down, will be investigated in more detail in a later stage of the project.

This paper is structured as follows: In chapter 2 we discuss the challenges and main approaches towards integration of diagnostics for plasma control in DEMO. Chapter 3 is the main chapter in which the planned suite of diagnostics for DEMO plasma control is presented, and the mapping between the control issues and the diagnostics is summarized in a table. A short summary and conclusions are given in chapter 4.

2. Challenges and main approaches for the integration of diagnostics for DEMO plasma control

The integration of diagnostics and control systems on a fusion reactor is a challenging task [13-16]. First, the requirements to achieve a tritium breeding rate (TBR) > 1 within the blanket, together with the requirements for first wall and divertor integrity as related to high heat loads and neutron shielding, effectively limit the space available for integration of any other components. Specifically, only a limited number of port plugs and other openings are available for diagnostic integration, which have to be shared with components for the heating and current drive system, remote handling, gas fueling, pumping and other systems. Second, any maintenance of in-vessel components within the nuclear environment of DEMO has to be performed by remote handling and is therefore technically challenging, expensive and time-consuming. Thus, a high overall availability for DEMO can only be achieved by designing all in-vessel components for a high degree of reliability and durability, such that the need for any interventions for scheduled and non-scheduled maintenance is minimized. In the course of these studies we are aiming for a typical lifetime of diagnostic and control components well above the mean time between planned exchanges of the breeding blanket. The target is two full power years (fpy) for the starter blanket which is designed for a neutron fluence of 20 displacements per atom (dpa), and 5 fpy for the second blanket (50 dpa). In this way, together with some redundancy of installed measurements, the need for unscheduled maintenance arising from failure of diagnostic and control components can be minimized.

Nuclear irradiation (neutrons, gammas) of any forward-mounted components on DEMO is resulting in strong volumetric heat loads, transmutation and displacement damage (dpa). For typical diagnostic mounting locations behind the DEMO blanket, the expected lifetime neutron fluence is in the order of 5×10^{21} /cm^2 (outboard midplane) and 2×10^{22} /cm^2 (inboard midplane), which is a factor of 25…100 larger than for typical ITER diagnostic locations [10]. While irradiation testing of diagnostic components for such high fluence is pending, such harsh conditions certainly imply restrictions in the choice of materials, and require specific design choices such as retracted mounting in protected locations, and applying active cooling. Moreover, any in-vessel front-end diagnostic components having an open sightline ("duct") towards the plasma (e.g. diagnostic mirrors) are subject to bombardment by energetic neutral particles originating from charge exchange collisions in the plasma. Fast impinging neutral particles can cause mirror erosion resulting in surface roughness [18], while metal particles released from the first wall or diagnostic duct may be deposited onto the mirror surface and degrade the reflectivity via rough deposited layers. According to the Rayleigh criterion, a typical mirror roughness in the order of $r \sim \lambda / 8 \sin \delta$ is tolerable before the reduction of specular reflectivity for light of wavelength λ becomes noticeable. Here, the angle δ denotes the grazing incidence angle relative to the surface tangent. Both erosion and deposition effects on first mirrors have recently been modelled for DEMO conditions [19]. It was found that with a hydrogen gas density of $n = 3 \times 10^{19}$ /m^3 assumed constant along a duct of length L = 1…2 m and of radius ρ = 1 … 3 cm, both erosion and deposition effects on the first mirror can be reduced down to an affected layer thickness of a few nm per full power year of operation. With this low level of mirror deterioration, the mirror reflectivity could be maintained at a sufficiently good level until the time of the next blanket exchange. The low gas density in the ducts leads to some outgassing towards the main chamber. For a duct with radius ρ = 1 … 3 cm, the outgassing rate is significantly smaller than the plasma fueling rate, such that the installation of a number of small ducts of this type can be afforded. In conclusion, long ducts with gas target and L/ρ ratios of about 40 for infrared, 50 for visible and 80 for vacuum-ultraviolet (VUV) wavelength ranges, will provide a sufficient protection of first mirrors against erosion and impurity deposition.

The boundary conditions discussed above lead to a selection of diagnostic methods for DEMO plasma control according to their robustness, with their front-end components to be mounted in remote (protected) locations. Specifically, it is foreseen to have no diagnostic components in front of blanket and divertor, while within the blanket region only metallic components such as microwave antennae and waveguides, and viewing ducts for any sightlines, shall be implemented. Behind the blanket, metallic optical mirrors and beam paths, magnetic sensors (metallic and ceramic components) and their cabling are to be integrated. Within the port plug regions, components for the further signal routing (mirror labyrinths, cabling) are foreseen, leading to the penetrations (cables, tubes) and/or windows at the port closure plates. Finally, in the divertor design the implementation of a sheath voltage or thermo-current measurement is planned. As compared to ITER, this diagnostic integration concept leads to severe limitations in the applicability and performance (e.g. spatial coverage) of diagnostic methods for DEMO. On the other hand, the complex task of reliably controlling the DEMO plasma near operational limits (e.g. density limit, radiation limit, wall load limit) can only be fulfilled as long as quite accurate and timely information about the actual plasma state in the various regions (core, edge, x-point and divertor) is available. The required spatial coverage of measurements on DEMO will be accomplished by installing a sufficient number of individual sightlines and channels.

For normal plasma operation, a reduced set of measurements may be sufficient if control oriented plasma models are well describing all possible evolutions of the plasma. However, unforeseen events such as the failure of major components (e.g. coolant ingress, or a quench in superconducting magnets), or the increase of plasma radiation following impurity ingress into the core plasma are difficult to capture via predictive models. Therefore, at the current stage of the DEMO D&C project, we have to assume that the information on the plasma state is based on a combination of detailed measurements together with the application of advanced control oriented models.

Since quantitative data on the reliability (mean time between failures) for DEMO diagnostics are not available yet, we currently assume that for all foreseen methods the number of installed channels is typically twice the minimum number of required measurements. Depending on future results on the actually achievable reliability and on the risk for entering into severe damage or even into safety relevant problems, this tentative redundancy factor of two may have to be either increased or decreased.

3. Main diagnostic methods foreseen for DEMO plasma control

3.1 Magnetic diagnostics

The primary diagnostics choice for the equilibrium control on tokamaks are traditionally the in-vessel magnetic coil based diagnostics, and also the control concept for ITER is following this approach, see e.g. [20] and references therein. Coil based measurements provide a signal proportional to the time derivative of the magnetic field, and hence the signals need to be integrated over time. However, during the long stationary burn phase, the raw signals to be integrated are essentially zero, and for typical changes of the plasma equilibrium the resulting magnetic signals are quite small. This makes the magnetic diagnostics sensitive to any spurious voltages that may arise e.g. from irradiation effects [10] acting on conductors and insulators.

The radiation induced electro-magnetic force (RIEMF) effect is due to charged particles generated from gamma and neutron reactions, driving an irradiation-induced current across the cable insulator. At the onset of irradiation, RIEMF is assumed to be the dominating disturbing effect on coil based magnetic signals [10]. Another important effect is the temperature-induced electromotive force (TIEMF) generating a parasitic thermocouple voltage via the thermal gradients inside the sensor arising from nuclear heating [10]. Furthermore, over the period of operation, nuclear reactions lead to transmutation, changing the material composition in the conductors. Thus, in presence of a temperature gradient along the cable, an additional thermo-electric voltage can be generated, which is designated as radiation-induced thermoelectric sensitivity (RITES) [10]. The effects from TIEMF and RITES can be reduced by means of design, specifically via minimizing gradients of irradiation dose and thermal loads over the sensor geometry.

For the application of magnetic sensors at ITER, early studies addressed coil designs based on the mineral insulated cable (MIC) technology [21]. For in-vessel MIC coils it was found that RITES could be the dominant source of spurious voltages for ITER magnetic sensors [21] and it was concluded that additional R&D is needed to reduce the level of disturbances below an acceptable level. Progressing with ITER diagnostic development, the low-temperature co-fired ceramics (LTCC) technology was proposed as an alternative to MIC, and a robust coil design has been worked out [22]. Initial irradiation testing of candidate LTCC coil variants under ITER relevant neutron fluence has recently been conducted on fission reactors [23, 24], but quantitative results on the possible degradation of electrical properties after irradiation are not yet available. In absence of an experimental basis, any quantitative extrapolation towards the possible coil degradation for the case of DEMO loads is not possible at this time. However, considering the large neutron fluence on DEMO, and the long discharge duration over which all the spurious signal contributions will be integrated, there is a high risk that the in-vessel magnetic measurements may degrade over time or even get lost.

As one backup, ex-vessel magnetic coil based sensors are foreseen. These are well shielded by the thick vacuum vessel and hence the irradiation effects are about three orders of magnitude lower than for the in-vessel sensors located behind the blanket [25]. The eddy current shielding by the vacuum vessel will slow down the signals delivered from ex-vessel magnetic sensors, such that the measured signals are too slow for the control of fast vertical displacement events [26]. However, the ex-vessel magnetic signals may be used to correct drifting signals arising from irradiation effects acting on the in-vessel sensors.

As a second backup option, metallic Hall sensors are considered, where no integration of signals is needed and thus any spurious voltages are not accumulated over time. This type of sensors promises some robustness against neutron and gamma irradiation, as compared to semiconductor-based Hall sensors. Metallic Hall sensors provide raw signals proportional to the magnetic field component orthogonal to the sensor supply current, however with only small signal amplitudes which are temperature dependent. For ITER, Bismuth based Hall sensors are under development [27], which provide relatively high signal levels as compared to other metals. However, due to the low melting point of only 271 °C, pure Bismuth appears not suited for in-vessel application on DEMO, where temperatures above 300 °C are expected behind the blanket. Therefore, investigations with Bi based alloys and other metals for Hall sensors are under way [28, 29]. For ex-vessel application on ITER, Bi based Hall sensors have already shown to be resilient against the expected ITER lifetime irradiation fluence [27].

Gold based Hall sensors are under development for application under higher irradiation levels and high ambient temperatures [30]. This type of sensors has been tested under neutron irradiation up to levels of $10^{20}/cm^2$, comparable to the ITER lifetime fluence in the blanket region, without any degradation of signals [30]. For the future, irradiation testing will be needed for a factor 10…100 higher fluence in order to clarify the applicability of these sensors on DEMO behind the blanket at the outboard and inboard side, respectively. Under these high values of fluence, transmutation of Au to Hg may affect the sensitivity of Hall sensors. A general problem for the use of Au based Hall sensors is given by the very low signal levels in the range of only 0.1 mV/Tesla, such that an extremely careful design for cabling and electronics is required.

While further R&D is needed to clarify the range of application for both Bi based and Au based Hall sensors, it is currently assumed that a similar number of Hall sensors and coil based magnetic sensors will be installed both in-vessel and ex-vessel. This comprises up to 240 in-vessel Hall sensors to be integrated into the machine with a similar technical approach like for the in-vessel magnetic coil based sensors. Since each Hall sensor needs to be connected to 6 wires (for signal, supply and measurement of sensor temperature), all in-vessel magnetic sensors will together add in the order of 2000 wires to be integrated and guided to the (vertical) port feedthroughs. The design approach for this cabling will closely follow the ITER developments [10].

An initial concept for the in vessel magnetic coil sensors needed for DEMO plasma control has been developed based on control simulations [26]. As shown in table 2, a number of 30 poloidal positions for one tangential and one normal in-vessel pick-up coils each are foreseen in 4 different toroidal locations behind the blanket, in order to provide high accuracy, noise reduction and redundancy for the control of plasma current, vertical position and plasma shape.

Table 2: Overview on the planned in-vessel magnetic coil sensors and their measurement role.

	Measurement role	Number
Inner vessel tangential and normal pick-up coils	Plasma current and plasma centroid position. Vertical speed. Shape and Equilibrium.	2 x 4 x 30
Inner vessel flux loops	Shape and Equilibrium. Loop voltage. Eddy currents.	8
Diamagnetic loops	Plasma magnetic energy.	4

The suite of in-vessel pick-up sensors is amended by a few flux loops and diamagnetic loops, from which the loop voltage and diamagnetic energy can be derived. In total, 252 coils have to be connected via 504 individual wires (2000 more for Hall sensors), which will mostly be routed along the backside of blanket segments towards the vertical ports, in order to facility a complete exchange of this set of in-vessel diagnostic components together with an exchange of a blanket segment.

3.2 Microwave diagnostics

Microwave (MW) reflectometry will be used for the measurement of the plasma density in the gradient region as well as for the position of the plasma boundary (gap control), while electron cyclotron emission (ECE) measurements will provide the electron temperature profile. Additionally, both measurements have important capabilities for the detection of fast MHD modes and instabilities in the plasma. The front-end components for both MW reflectometry and ECE measurements consist of horn antennae and waveguides, made from EUROFER (ferritic steel) with tungsten coating (for protection, and providing good electrical conductivity). The irradiation conditions, thermal loads and material erosion levels will be similar to the blanket first wall (antennae only slightly retracted against the first wall level), such that the durability of these antennae is expected to be comparable to the blanket first wall.

Microwave (MW) reflectometry measurements are foreseen for 16 different locations surrounding the poloidal plane. These will mainly serve for position and shape control (gap control), as well as the determination of the plasma density profile (control of pedestal top density against the density limit). This MW reflectometry system will be duplicated in a second sector in order to provide redundancy. MW reflectometry will also contribute to MHD detection at least in the outer radial region of the plasma. Near the mid-plane of the plasma, the “single pair” approach for emitting and receiving antennae will provide good spatial resolution. However, near the upper and lower side the curvature of the plasma (incidence angle variations) will cause significant problems for operation and accuracy of reflectometry measurements. Here, each measurement location will require between 4 and 6 antennae to ensure that the reflected beam is captured by at least one of these antennae, even under conditions of larger plasma-wall distance. Assuming on average 5 antennae per poloidal location and adding a factor two for redundancy, we arrive at a total of up to 160 antennae and waveguides for MW reflectometry, to be installed in at least two different poloidal sectors.

The primary integration approach is via the “dummy poloidal section” concept [31], i.e. a full banana-shaped housing with toroidal dimension of about 20…30 cm, carrying the antennae and waveguides, and routing the waveguides towards the vertical port. This dummy poloidal section might be either inserted in between two breeding blanket (BB) sections or laterally integrated into a BB sector. Whenever the blanket will be exchanged, the waveguides would be disconnected near the vertical port and the entire BB sector together with the dummy poloidal sector would be replaced using a similar procedure as for the BB sector exchange. Then, a new BB sector would be inserted and finally the waveguides be connected again to the feedthroughs at the vertical port plate. The final design of the interfaces of the antennas with the first wall will be a compromise between signal to noise ratio, multiple reflections, refraction and heat loads. The development of detailed engineering solutions will be subject to future work.

ECE measurements will be used for the measurement of the electron temperature profile and for MHD control. A sufficient spatial resolution can only be obtained when measuring from the outboard mid-plane side of the plasma [32]. Accordingly, antennae for ECE measurements will be integrated into equatorial ports where two slim “drawers” in different ports are foreseen to host the ECE antennae and routing the waveguides to the backside, with feedthroughs near the port plates. Additional ECE channels are needed in the vicinity of the launchers for ECRH, to serve for local mode detection and control [32]. For this purpose, it is proposed to integrate two ECE channels near each of the ECRH launchers.

3.3 Infrared polarimetry/interferometry

For the measurement of the central plasma density, infrared (IR) laser interferometry and/or polarimetry is foreseen, where the general scheme of the diagnostic layout and beam arrangement will either follow the ITER TIP (Toroidal Interferometer/Polarimeter) [33] or the ITER PoPola (Poloidal Interferometer/Polarimeter) [34] approach.

The front-end components consist of metallic mirrors in retroreflector geometry with high reflectivity in the IR range. For long-term protection against erosion and deposition, these mirrors should to be mounted behind a duct of length L ~ 1…2 m, depending on the required duct diameter as related to the laser beam diameter and optical alignment issues. At the location of the port plates or more outside (e.g. when using a vacuum extension), an IR window (e.g. diamond) will be needed for each laser beam.

In case of choosing the TIP concept, for each laser beam the first mirror and the end mirror (retroreflector) will be located in different ports (eq. and/or vertical port), with oblique sightlines through the plasma and in the port themselves. Restricting to the use of equatorial ports will effectively limit this approach to providing three different sightlines only. A few more different sightlines may become available if additional sightlines with retroreflectors are integrated into vertical ports. These oblique sightlines in the equatorial and vertical ports will occupy relatively large space (blocking the insertion of radial "drawers" into a major part of the port plug). A factor two of redundancy in the number of sightlines should be foreseen.

An installation of the laser beams exactly in the poloidal plane (PoPola approach) would avoid any polarimetric signal contributions proportional to the strong toroidal field. Thus the system could be optimized for high sensitivity for the poloidal magnetic field. A polarimeter beam near the centre of the plasma current would change the sign of the polarization whenever the current centroid moves across the beam, so that this single beam could already provide a useful signal for a basic vertical position control. A more sophisticated scheme would employ additional beams above and below the current centroid. As a caveat, the PoPola installation requires the retroreflectors to be installed at the high field side, where the maximum possible duct length in front of the vacuum vessel would only be in the range of 60 cm, which could be too short for an effective long-term mirror protection in the case of a duct radius of several cm.

The total number of required interferometer/polarimeter beams depends on the control tasks to be covered. Three beams in TIP arrangement would be sufficient for the determination of the central plasma (electron) density, while the edge plasma density is derived from MW reflectometry. A single central interferometer beam would already contain useful information about the spectrum of core plasma instabilities, however without any information on their localization. A set of PoPola beams would be able to localize MHD modes as long as these are rotating or at least crossing the beams. The potential of the PoPola beams for vertical position control in the plasma startup phase, where the plasma diameter is still small, could be essential if the in-vessel magnetic sensors would fail. In this case, a viable solution for the placement of inboard retroreflectors has to be found, e.g. via extending the ducts and retracting the inboard optical components more deeply into the vacuum vessel wall region.

3.4 Spectroscopic and radiation measurements

DEMO will have to be operated at a high core radiation fraction $P_{rad,core}/P_{heat} \sim 0.6 \ldots 0.8$ to reduce the power flow across the separatrix towards the divertor, $P_{div} \sim P_{sep} = P_{heat} - P_{rad,core}$. Technically, the control of core radiation shall be performed by injecting Xenon or Krypton as a radiating impurity into the plasma. At the same time, stable H mode operation requires that the power flow across the separatrix is larger than the H mode power threshold, $P_{sep} > P_{LH}$, which means that the core radiation power should not be too high. A precise measurement of both heating power and core radiation power is required in order to fulfill these two conflicting requirements. While the heating power can be deduced from neutron fluxes (see below) and from the measurement of auxiliary heating power, the core radiation power has to follow from radiation and spectroscopic measurements.

The primary diagnostic foreseen to provide a control signal proportional to the core plasma radiation power $P_{rad,core}$ is the core plasma bolometry. As compared to the ITER bolometry design [35], the detectors for DEMO bolometry will have to be mounted in more retracted locations behind long ducts. The current concept foresees about 10 distributed isolated sightlines from both an equatorial port and a vertical port each within a poloidal plane, in order to obtain a coarse radial profile. Since a long term stable absolute calibration of the detectors appears not viable, a relative calibration of the signal will be gained by analyzing the quantity $P_{sep} = P_{heat} - P_{rad,core}$ at the occasion of H-L transitions and back-transitions, both of which should occur at least once in every successfully controlled plasma pulse. A suitable signal for the fast feedback control of the seeded impurity can be obtained from the intensity of characteristic spectral lines measured with VUV spectroscopy in the plasma edge, using a design similar to the ITER VUV spectroscopy [36]. VUV overview spectroscopy in core and edge plasma also allows to detect any impurity ingress into the plasma via the subsequent occurrence of lines from higher ionization stages. High resolution X-Ray spectroscopy is foreseen to obtain signals proportional to the core plasma density of impurities with high atomic number (Xe, Kr and W), however with a slow response as defined by the radial transport time of the impurities from the edge to the core plasma. For instance, the Ne-like lines of W at 1.3-1.5 Å [37], H-like Xenon lines at 0.39-0.4 Å [38] or He-like Krypton lines at 0.9 Å [39] could be used to monitor the impurity concentrations and provide some of the plasma parameters. Again, an existing design principle developed for ITER [40, 41] can be adapted for DEMO purposes, where a single central sightline is assumed to be sufficient for the DEMO control tasks. Comparing the X-Ray signals with the corresponding edge VUV line intensities will allow to deduce information on the possible accumulation of high Z impurities in the core plasma. This information can be used as input to conduct a real-time modelling of the plasma radiation for the purpose of cross-check with the bolometric signals.

A crucial task for the radiation and spectroscopic measurements is the control of the detachment of the divertor plasma [42-44], which is the baseline approach in the EU DEMO concept to keep the heat flux densities at the divertor target below acceptable levels. The detached plasma state is achieved by enhancing the radiation cooling of the edge and divertor plasma, e.g. by adding gaseous impurities into the plasma [42]. However, the divertor region is an area where the plasma-wall interaction is concentrated and high fluxes of impurity particles (erosion and deposition) are expected. Therefore, long-term durability of any diagnostic components for divertor detachment control on DEMO can only be achieved by locating them as much as possible in protected positions outside the divertor region. Based on this consideration, measurements of divertor detachment from existing fusion experiments [43] have been assessed with regard to their applicability for DEMO control.

First, the onset of detachment near the outboard divertor target leads to the occurrence of a zone of high density low temperature plasma in front of the strike point. Such plasma conditions have been shown to be detectable via the strong intensity enhancement and Stark broadening of higher Balmer lines from hydrogen isotopes, measured by high resolution visible (VIS) spectroscopy [45, 46]. The realization of such measurements on DEMO is technically possible by integrating the first mirror into an equatorial port and looking down into the divertor region under oblique angle with a set of sightlines that are almost parallel to the target when projected into the poloidal plane [47]. For a more detailed analysis of the spatial distributions of plasma density and temperature in the divertor region and the spectroscopic signatures arising from that, SOLPS modelling for detachment on DEMO is under way [48]. Also the impact of light reflections in the divertor region has to be assessed in the feasibility study of this spectroscopic approach for DEMO. As a backup option, divertor VUV spectroscopy (less vulnerable to wall reflections) could be considered, which however would imply larger efforts for optical and mechanical design.

A second approach for detachment control on DEMO could be based on the thermographic observation of the temperature distribution along the target plates [43, 49]. These sightlines can also be realized with optical elements installed in an equatorial port [47]. However, under conditions of detachment the IR intensity emitted from the target plate is strongly reduced while the broadband background radiation is increased. The feasibility of this measurement under detached conditions therefore has to be assessed in more detail.

Measurements of plasma radiation power can also contribute useful signals for MHD control and plasma position control [50, 51]. In order to enhance the redundancy of measurements for these control tasks, the installation of sightlines from both equatorial ports and vertical ports is under consideration. Specifically, this could comprise a set of 2 x 10 horizontal sightlines (equatorial port) and an additional set of 2 x 10 sightlines from the vertical port (wider coverage of radial range, and coverage of Shafranov shift effects), where the factor 2 provides some redundancy. Depending on a further assessment of detector properties, this measurement might be combined with the “bolometry” described above.

A final element within the tasks for spectroscopic and radiation measurements is related to the monitoring of some of the protection limiters which are being designed to protect the first wall (blanket) from overheating [52]. Several observations similar to the ITER wide angle viewing system [53] are under consideration, however with quite limited views due to the retracted mounting position of first mirrors.

A list of all required sightlines and channels for spectroscopic and radiation measurements for DEMO plasma control is presented in table 3. A factor 2 for redundancy has already been included. This reduced list applies for the case that the approach presented above for a spectroscopic detection of detachment will be feasible on DEMO. More details on the suite of spectroscopic and radiation measurements are presented in a separate paper [47].

Table 3: List of channels for spectroscopic and radiation measurements (without limiter observations).

Diagnostic method and target	**Number of channels**	**Integration approach**
Radiation power (core)	2 x 2 x 10	20 in Eq. Port 20 in Vert. Port
X Ray spectroscopy (core)	2 x 3	6 in E.P.
VUV spectroscopy (core)	2 x 4	8 in E.P.
VUV spectroscopy (edge)	2 x 3 x 4	16 in E.P. 8 in V.P.
VIS spectroscopy (outboard divertor and x-point)	2 x 2 x 2	8 in E.P.
Thermography (divertor)	2 x 2	4 in E.P.
X-ray intensity	2 x 2 x 10	20 in E.P. 20 in V.P.
Total		82 in E.P. 48 in V.P.

3.5 Divertor Thermocurrent Measurement

In addition to the spectroscopic and radiation diagnostics, the measurement of the divertor thermo-current at several divertor target plates is a promising approach for power exhaust control. Under conditions of low plasma temperature in front of the divertor target, the sheath voltage should go down to zero. Connecting the divertor target to a shunt resistor, the divertor thermo-current should vanish when going from attached to detached plasma conditions. Plasma detachment control based on this principle has been successfully demonstrated on the ASDEX upgrade tokamak [44].

On DEMO, the integration of this measurement would require the use of ceramic insulators between the divertor target and the divertor cassette or the vacuum vessel [54]. For the measurement of the thermo-current, two options are under consideration: either the divertor targets would be connected via shunt resistors to ground, or the coolant tubes itself may serve as shunt resistors. In the latter case, the feasibility will depend on the amount of currents flowing in the water cooling pipes during disruptions [54], otherwise also ceramic pipe insulation may be required. The durability of insulators to maintain a required minimum electrical resistance under the neutron load conditions, strong material erosion and deposition and high temperatures in the DEMO divertor region has to be verified. Assuming that a technical solution will be found for the implementation on DEMO, the installation of the thermo-current measurement is foreseen for every divertor target plate, such that this will be the only measurement which potentially could provide a complete coverage of the divertor and thus allow to detect any spatial inhomogeneity in the power load distribution.

3.6 Neutron/gamma diagnostics

For the measurement and control of the fusion power, a neutron camera for flux measurement similar to the ITER system under development [55] shall be implemented on DEMO. The system comprises a set of 2 x 10 horizontal sightlines (equatorial port) and an additional set of 2 x 10 sightlines from the vertical port (wider coverage of radial range, and coverage of Shafranov shift effects), where the factor 2 provides some redundancy. The performance assessment (neutron statistics) indicates that the fusion power can be derived with a relative error in the order of 1% at a time resolution of 10 ms. The neutron emissivity profile can be reconstructed up to a normalized poloidal flux coordinate of $r/a \sim 0.9$ with the same time resolution and a relative error less than 1 %. From the measured neutron flux signals the fusion power can be derived, where an in-situ calibration can be accomplished via the calorimetric measurement of thermal power deposited in blanket and divertor, together with an accurate measurement of the auxiliary heating power deposited in the plasma. During the burn phase, the neutron flux measurements may also contribute to the plasma position control with an accuracy in the order of 1 cm (horizontal position) and 3 cm (vertical position), respectively. Furthermore, the D/T ratio and ion temperature can be deduced from neutron spectroscopy with a time resolution of 1 second. An additional spectroscopic measurement of high energetic 17 MeV DT gamma rays from the DEMO plasma is under consideration [56], sharing the same sightlines with the neutron camera similar to the conceptual design for ITER [57, 58], from which an independent value for the ion temperature and fuel DT burning ratio (i.e., Tritium retention) could be obtained, although at low count rates.

The front-end of each channel consists of a long duct with < 7 cm inner diameter. At the far end of each collimator a detector (or series of detectors) will be mounted outside the bio-shield at a distance > 15 m from the front collimator. EUROFER is being considered as the main material of the collimator tube, surrounded by boron carbide B4C. The materials composition of the collimator towards the detector can include material for moderation of scattered neutrons, doped with thermal neutron absorbers and gamma-ray attenuator material.

The sightlines can be integrated in a poloidal plane, such that the space occupation in the ports is minimized. Specifically, the sightlines from the equatorial port can be integrated into slim vertical drawers.

At the location of the detectors (several meters away from the first wall), the irradiation levels are low enough, so that no adverse effects on the detectors are expected.

3.7 Assignment between diagnostics and control issues

The main control tasks for DEMO and the related diagnostic tools currently considered are summarized in table 4. For risk mitigation, most of the control issues are being addressed by at least two independent diagnostic methods. On the other hand, it is evident from the list that many of the measurements are related to multiple control tasks simultaneously. So far, this proposed suite of diagnostics is mainly defined based on the requirements for the flat-top phase of the burning plasma.

Table 4: List of DEMO control issues and the main diagnostic approaches to address them

Control topic	Control quantity	Diagnostics
Equilibrium control	Plasma current	Magnetic diagnostics
	Plasma position and shape, incl. vertical stability	Magnetic diagnostics MW reflectometry, ECE Neutron/gamma/x-ray diagnostics IR polarimetry/interferometry
Kinetic control	Plasma (edge) density	MW reflectometry IR polarimetry/interferometry Plasma radiation
	Plasma radiation, impurity mixture, Z_{eff}	Spectroscopy+radiation meas. Flux loop (loop voltage)
	Fusion power	Neutron diagnostics FW/blanket and div. power (for calibration only)
	Divertor detachment and heat flux control	Spectroscopy+radiation meas. IR thermography Divertor thermo-currents
MHD and event control	(MHD) plasma instabilities	MW reflectometry, ECE IR polarimetry/interferometry Magnetic diagnostics X-ray diagnostics
	Plasma pressure	Magnetic diagnostics Density and temperature meas.
	Unforeseen events (impurity ingress, component failure)	all

4. Summary and Conclusions

Within this paper, the evolution of boundary conditions and requirements for diagnostics from ITER to DEMO has been discussed, and an initial concept for the diagnostic system for DEMO plasma control has been presented. As compared to ITER, the implementation of diagnostics on DEMO is even more limited by adverse effects that degrade the front-end components, in particular by ionising radiation, material erosion and deposition. In order to achieve a high reliability and durability of plasma control, the main diagnostic methods and components for DEMO plasma control have been selected according to their robustness, and front-end components are planned to be mounted in protected (retracted) locations to reduce the loads to acceptable levels. The low space available for diagnostics, remote maintenance and integration issues further reduce the design freedom for the layout of the control system and its components. In the course of the development, a number of critical issues and risks have already been identified. First, the feasibility of in-vessel magnetic measurements in view of high expected neutron fluence can only be clarified by further irradiation studies at DEMO relevant levels. Fast equilibrium control is however not possible with ex-vessel magnetic sensors only. Second, the feasibility and reliability of the proposed approach for power exhaust control (divertor detachment) has to be elaborated further. Third, the need to retract components towards protected locations in the machine is reducing the spatial coverage of diagnostics, and can only be compensated by integrating a large number of individual channels and sightlines, which represents an enormous design effort and will occupy significant space in the machine. Finally, the current analysis has mainly addressed the requirements for the flat top phase, while the detailed treatment of transients and instabilities may reveal additional issues that have not yet been considered.

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References

[1] F. Romanelli, "Fusion Electricity - A roadmap to the realisation of fusion energy," 2013. [Online]. Available: https://www.euro-fusion.org/wpcms/wp-content/uploads/2013/01/JG12.356-web.pdf.

[2] G. Federici et al., Fus. Engin.Des. 109–111 (2016) 1464

[3] E. J. Doyle et al., Nucl. Fus. 47 (2007) S18-S127

[4] R. Wenninger et al., Nucl. Fus. 57 (2017) 016011

[5] A. Costley et al., Rev. Sci. Instrum. 70 (1999) 391-396

[6] A. J. H. Donne et al., Nucl. Fus. 47 (2007) S337–S384

[7] C. I. Walker et al., Rev. Sci. Instrum. 75 (2004) 4243-4246

[8] A. Costley et al., Fus. Engin. Des., 74 (2005) 109-119

[9] M. Walsh et al., IEEE/NPSS 24th Symp. Fus. Engin. Chicago (2011) 1-8

[10] G. Vayakis et al., Fus. Sci. Technol., 53 (2008) 699-750

[11] D. A. Castaño Bardawil et al., Fus. Engin. Des. 96–97 (2015) 746-750

[12] L. Moser et al., Nucl. Fusion 55 (2015) 063020

[13] A.J.H. Donne et al., Nucl. Fusion 52 (2012) 074015

[14] T. N. Todd, AIP Conference Proceedings 1612 (2014) 9

[15] W. Biel et al., Fus. Engin. Des. 96–97 (2015) 8-15

[16] F. Orsitto et al., Nucl. Fusion 56 (2016) 026009

[17] F. Felici et al., Plasma Phys. Contr. Fus. 54 (2012) 025002

[18] A. Litnovsky et al., J. Nucl. Mat. 363–365 (2007) 1395

[19] M. Z. Tokar, Nucl. Fus. 58 (2018) 096007

[20] L. Zabeo et al., Fus. Engin. Des. 89 (2014) 553-557

[21] G. Vayakis et al., Rev. Sci. Instrum. 75 (2004) 4324

[22] S. Peruzzo et al., Fus. Engin.Des. 88 (2013) 1302– 1305

[23] S. Peruzzo et al., IEEE Trans. Plasma Sci. 44 (2016) 1704

[24] G. Vayakis, A. Pironti, private communication (2018)

[25] U. Fischer et al., Fus. Engin. Des. 109-111 (2016) 1458

[26] M. Ariola et al., submitted to Fus. Engin. Des. (2018)

[27] I. Duran et al., Fus. Engin. Des. 123 (2017) 690–694

[28] I. Duran et al., submitted to Fus. Engin. Des. (2018)

[29] S. Entler et al., submitted to Fus. Engin. Des. (2018)

[30] I. Bolshakova et al., Nucl. Fus. 57 (2017) 116042

[31] A. Malaquias et al., IEEE Tr. Plasma Sci. 46 (2018) 451

[32] N. Rispoli et al., Fus. Engin. Des. 123 (2017) 628-631

[33] M. A. Van Zeeland et al., Rev. Sci. Instrum. 84 (2013) 043501

[34] A. J. H. Donné et al., Rev. Sci. Instrum. 75 (2004) 4694

[35] H. Meister et al., Fus. Engin. Des. 120 (2017) 21–26

[36] C. R. Seon et al., Fus. Engin. Des. 109–111 (2016) 656

[37] C. Biedermann et al., Phys. Scr. T134 (2009) 014026

[38] D. B. Thorn et al., Phys. Rev. Lett. 103 (2009) 163001

[39] K. Widmann et al., Rev. Sci. Instrum. 66 (1995) 761

[40] P. Beiersdorfer et al., J. Phys. B: At. Mol. Opt. Phys. 43 (2010) 144008

[41] S. Varshney et al., J. Phys.: Conf. Series 823 (2017) 012055

[42] R. Pitts et al., Phys. Scr. T138 (2009) 014001

[43] J. Boedo et al., Plasma Phys. Contr. Fus. 60 (2018) 044008

[44] A. Kallenbach et al., Plasma Phys. Contr. Fus. 52 (2010) 055002

[45] B. L. Welch et al., Phys. Plasmas 2 (1995) 4246

[46] A. G. Meigs et al., J. Nucl. Mat. 438 (2013) S607

[47] W. Gonzalez et al., submitted to Fus. Engin. Des. (2018)

[48] F. Subba et al., private communication (2018)

[49] A. Kallenbach et al., Nucl. Fus. 55 (2015) 053026

[50] L. F. Delgado-Aparicio et al., Rev. Sci. Instrum. 87 (2016) 11E204

[51] D. Mazon et al., Fus. Engin. Des. 96–97 (2015) 856–860

[52] F. Maviglia et al., submitted to Fus. Engin. Des. (2018)

[53] S. Salasca et al., Fus. Engin. Des. 96–97 (2015) 932–937

[54] S. El-Shawish et al., submitted to Fus. Engin. Des. (2018)

[55] D. Marocco et al., J. Instrum. 7 (2012) C03033

[56] L. Giacomelli et al., Fus. Engin. Des. (2018) in press

[57] M. Nocente et al., Nuclear Fusion 57 (2017) 076016

[58] M. Rebai et al., Rev. Sci. Instrum. (2018) in press