Fusion Plasma Physics
Annual Report 2020

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1. Introduction

Fusion is the energy source that powers the Sun and other stars in the universe. In the core of the Sun, hydrogen nuclei collide, fuse into helium and release enormous amount of energy. In order to use the same process for a power producing reactor on earth, the hydrogen fuel in the reactor has to be heated up to extreme temperatures of the order of a hundred million degrees. At these temperatures, the hydrogen fuel is in the form of a plasma. The devices needed to achieve the required conditions are extreme in several ways. An example of this is the ongoing construction of the ITER tokamak in France. The superconducting magnets that provide the strong magnetic field needed for the plasma confinement, the vacuum vessel containing the plasma, the cryostat providing the thermal insulation for the superconducting coils, etc, are all components or the tokamak device that represent a big technological challenge due to their size and complexity.

The assembly of the ITER device is now progressing largely according to schedule, and the first plasma is expected in 2025. If everything goes well, ITER will be the first fusion device to produce net energy. The aim is to test the integration of technologies, materials and physics required for a fusion reactor. This is a huge project that is carried out in an international collaboration involving seven partners – EU, USA, Japan, China, Russia, India, and Korea. The name – ITER – stands for “the way” in latin, and it is to be understood as “the way to fusion energy”.

The Division of Fusion Plasma Physics is actively participating in the international fusion research effort as well as educating students in fusion related topics at KTH. During the fall semester of 2020 a new Introduction Course in Fusion Technology was given, designed for physicists and engineers who want to learn about technology and properties of wall materials exposed to the extreme conditions in a fusion device. The topic is highly relevant for ITER and the planned following demonstration reactor DEMO.

Fusion research in Europe is coordinated through the EUROfusion consortium, which has been formed by research institutes and universities involved in fusion throughout Europe. Vetenskapsrådet is the Swedish member of the EUROfusion consortium representing research groups at Chalmers, Uppsala University and KTH.

The activities of the Division in the EUROfusion work programme has continued in 2020 in spite of the difficulties posed by the covid-19 pandemic. The scope of the Division contribution to the programme is focussed on code development, research education, enabling research projects, experimental campaigns at JET, JET plasma components, fusion materials, experimental campaigns at the medium-size tokamak facilities AUG and TCV, and plasma facing components.

Per Brunsell
Head, Fusion Plasma Physics
2. Research projects

2.1. EXTRAP T2R device

The EXTRAP T2R device at KTH is utilized for front-line research on MHD active stabilization as well as for hands-on practical research training for fusion students. The relatively simple geometry of the device allows for straightforward interpretation of experiments and easy access for diagnostics and peripheral equipment installation. The plasma has a toroidal, circular cross-section with major radius 1.24 m and minor radius 0.183 m. The device operates in the reversed-field pinch magnetic field configuration with plasma current of the order 100 kA, a plasma density around $10^{19} \text{ m}^3$, an electron temperature around 300 eV and pulse lengths up to 0.1 s. A number of plasma diagnostic systems are available at EXTRAP T2R, such as a large array of magnetic diagnostics, a two-colour laser interferometer, a single-point Thomson scattering system, various SXR detectors, and a range of visible and ultra-violet light spectrometers, bolometers, Langmuir and collection probes.

Fusion research activities at the EXTRAP T2R are focused on active MHD control, in particular magnetic feedback control of resistive wall modes (RWM). A main aim of the research is to develop methods and physics understanding required for MHD control, while a particular goal is to develop control methods suitable for tokamaks that incorporate MHD control coils, such as ASDEX Upgrade and JT60-SA. The underlying motivation for the activity is the need for active MHD mode control in ITER advanced operation scenarios and in future tokamak reactor designs. The active control system has also been utilized for studies of applied Magnetic Perturbations (MPs) and for development of Error Field (EF) detection.

The EXTRAP T2R device produces a hot and dense plasma that can be utilized also for basic plasma science, research on space plasmas, investigation of technical plasma applications, or plasma based material research. The device has excellent capabilities for material studies, which are not yet fully utilized. There are a number of access ports, through which material samples can be introduced into the plasma edge. Translation systems for insertion and extraction of material samples are available, also enabling in-vacuum transport of samples for post-exposure analysis elsewhere. Several ion beam based analysis methods are available through our collaboration with the Tandem Accelerator facility in Uppsala University.
The main components of the MHD mode control system at EXTRAP T2R is an array of active control coils placed outside the conducting shell, and a corresponding array of sensor coils placed inside the shell.

The main features of system are:

- **128 magnetic flux loop sensors** at 4 poloidal and 32 toroidal positions inside the shell.
- **128 active saddle coils** at 4 poloidal and 32 toroidal positions outside the shell.
- **Pair-connected** at each toroidal position to 64 independent “m=1” coils
- **Integrated digital controller unit** including CPU board, ADCs and DACs.
- **Control algorithms implemented in software.**

During 2020 a major control system upgrade has been initiated involving both the control coil power amplifiers and the integrated digital controller. The upgrade is motivated by the need to sustain error field control at long pulses and to enhance the control system computation capability for GPU implementation of complex control algorithms.

The active coils are driven by a set of 32 audio amplifiers. A new set of power amplifiers manufactured by DAP Audio have been acquired. The new model HP-3000 2-channel PA audio amplifier has output power of 1500 W/channel (3000 W per unit), higher than the current set of audio amplifiers with 400-600 Watt/channel (800-1200 Watt per unit). The higher power is primarily required to dynamic field error correction in long plasma discharges.

The integrated controller has been upgraded to a new system acquired from D-TACQ Solutions. The new system design is based on the D-TACQ ACQ400 Series and includes four 32-channel ACQ423ELF ADC units and two 32-channel AO424ELF DAC units, integrated in one ACQ2106 appliance unit. The appliance unit uses data communication over a fiber optic link to the host computer allowing data transfer on the PCIe bus at a high data rate of 400 Mb/s, demanded for multi-channel real-time control operation.

A new host computer of model SuperMicro SuperServer 5019GP-TT with an Intel Xeon 6-core processor has been acquired. The new host computer includes a HP Nvidia model Tesla P40 Graphic Processing Unit that will be used for demanding control computations. Feedback control algorithms are written in C or C++. Various types of controllers are implemented, such as the conventional Proportional-Integrating-Derivative (PID) controller, the Linear-Quadratic-Gaussian (LQG) control as well as modern model based controller algorithms such as the Model Predictive Controller (MPC).

The new system control is fully integrated allowing direct data transfer over the PCIe bus between the ACQ2106 memory and GPU memory. This direct communication link over the PCIe bus is a key to enable efficient parallelized computations of complex control algorithms utilizing the GPU. An example of algorithms that are computationally demanding are the model based controller algorithms such as the Model Predictive Controller (MPC).
2.2. Development of resistive wall mode control models  

_P. Brunsell, E. Saad (PhD student)_

Recent developments of resistive wall mode (RWM) feedback control utilize modern state-space controller designs that are suited for the multiple-input, multiple-output type of control system that results from the distributed magnetic field sensor and control coils arrays. Model-based control designs have potential to perform better under realistic conditions in which for example the capability for mode discrimination is important. Models applicable for control implementation of the RWM instability are then needed. The models should accurately describe the plasma response to the applied external control fields, including the 3D structure of the conducting wall and a realistic plasma description. Both theory based first-principle (“white-box”) and experimental data-driven (“black-box”) approaches to control modelling, and the combinations of the two (“gray-box”) are currently being developed.

The model-based approach for RWM control design involves model development, algorithm design, system simulation and experimental validation. The availability of the model allows a simulation test to be performed before implementation of the algorithm in the physical system, increasing the confidence level of the algorithm before experimental validation.

The first basic reference plasma model used in this project is the linearized ideal MHD model in cylindrical geometry. For the slowly evolving RWM mode, the marginally stable ideal MHD eigenmode is a sufficient approximation for the RWM. It is obtained by solving a one-dimensional, linearized first-order differential equation, the so called Newcomb equation subject to the thin shell boundary condition at the resistive wall radius:

$$\gamma_{mn} \tau_w = \frac{r}{b_r^{mn}} \frac{d(r b_r^{mn})}{dr} \tau_{w+} - \tau_{w-}.$$

The experimental Fourier decomposed radial magnetic field may be compared with the model time evolution without plasma (vacuum) and with plasma. For the case with only vacuum the time evolution of the radial magnetic field at the radius of the resistive wall may be expressed according as

$$\tau_{mn} \frac{db_r^w}{dt} + b_r^w = b_r^{ext}.$$

This equation dictates the exponential settling with the rate $\sim -\frac{1}{\tau_{mn}}$ and

$$\tau_{mn} = -\tau_{w}^{w'} K^{w'}_m (1 + (mR_0)^2 (n r_w)^{-2})^{-1}.$$

The corresponding relation for the radial magnetic field at the radius of the resistive wall for the plasma case is, introducing the RWM growth rate:

$$\tau_{mn} \frac{db_r^w}{dt} + \gamma_{mn} \tau_{mn} b_r^w = b_r^{ext}.$$

Experimental validation of the model proceeds by applying a step coil current waveform to all actuators with the objective to excite a given toroidal mode number $n$. The direct excitable mode numbers in this device are $n \in \{-16,\ldots,15\}$ and due to sideband it is possible to excite higher order modal numbers. The signal from the sensor array may be Fourier decomposed according to the
expansion $b^w_r(r, \theta, \varphi, t) = \sum_m \sum_n b^{w,m,n}_r(r, t) e^{i(m\theta + n\varphi)}$ for poloidal mode number $m$ and toroidal mode number $n$. Experimental data. The resulting comparison of model calculated and experimentally measured data are shown in Figure 2.2.1 for the wall time constants in the vacuum case and in Figure 2.2.2 for the resistive wall mode growth rates in plasma discharges.

Figure 2.2.1 Comparison of model calculated and experimentally measured resistive wall time constants in EXTRAP T2R for different poloidal $m=1$ and $m=-3$ and toroidal $n$ mode numbers.

Figure 2.2.2. Comparison of model calculated and experimentally measured resistive wall mode growth rates in EXTRAP T2R for different poloidal $m=1$ toroidal $n$ mode numbers.
2.3. Plasma - wall interactions

M. Rubel, P. Petersson, L. Dittrich (Ph.D. student), Sunwoo Moon (Ph.D. student)

In collaboration with
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Plasma-wall interactions (PWI) comprise all processes involved in the exchange of mass and energy between the plasma and the surrounding wall. Two inter-related aspects of fusion reactor operation - economy and safety - are the driving forces for studies of PWI. The major issues to be tackled are: (i) lifetime of plasma-facing materials (PFM) and components (PFC), (ii) accumulation of hydrogen isotopes in PFC, i.e. tritium inventory; (iii) carbon and metal (Be, W) dust formation. PWI is one of the primary areas where integration of the Physics and Technology programmes is being achieved. The work at KTH in the field of PWI and fusion-related material physics has been fully integrated with the international fusion programme: (i) EU Fusion Programme, (ii) International Tokamak Physics Activity (ITPA), (iii) International Atomic Energy Agency (IAEA), (iv) Implementing Agreements of International Energy Agency (IEA). It is demonstrated by the participation in:

- EU Fusion Work Programme:
  - Work package JET2: Analysis of Plasma-Facing Components from JET
  - Work package PFC: Plasma-Facing Components for ITER
  - Work package MAT: Functional Materials for DEMO
- Broader Approach Work Programme:
  - IFMIF Project Committee, Rokkasho, Japan
  - Material studies at Int. Fusion Energy Research Centre (IFERC), Rokkasho, Japan
- ITPA, IAEA and IEA activities.

Experimental work was carried out at home laboratory, JET at Culham Science Centre, Forschungszentrum Jülich, Ruder Boscovic Institute in Zagreb, Warsaw University of Technology and at IFERC. The research programme is concentrated on:

- Material erosion, migration and re-deposition.
- Fuel retention studies and fuel removal techniques.
- Dust generation processes in fusion devices.
- Characterization of plasma-facing materials including testing of high-Z metals.
- Development and testing of diagnostic components: first mirror test at JET for ITER
- Development of diagnostic methods for PWI studies.

2.3.1. Search for mobilized dust in operation with remote handling equipment

The presence of dust in fusion devices is an unavoidable consequence of mechanical in-vessel operations and material erosion-deposition processes. Risks associated with the generation and accumulation of dust particles originating from erosion of plasma-facing components (PFC) in controlled fusion devices have been identified and discussed for the last two decades. The issue was especially severe in carbon-wall tokamaks where thick layers of co-deposits were formed and then peeled-off. This had major consequences for the wall technology: beryllium and tungsten in the JET tokamak with the ITER-Like Wall (JET-ILW). The operation of JET-ILW has clearly proven low amounts of dust in the presence of metal PFC: around 1 gram per campaign (19-23 h plasma operation) in comparison to over 200 g removed by vacuum cleaning of the divertor after the last JET-C operation. From the safety point of view there is also another aspect which is to be thoroughly addressed in the licensing procedures: dust mobilization and
transfer outside the vacuum vessel. This is connected with Be and radioactive matter with tritium and activation products. Therefore, on the direct request from the ITER Organisation, the accumulation of dust on the remotely handled (RH) equipment was studied in JET with the ITER-like wall (JET-ILW) during the shutdown after the third ILW campaign. The exercise carried out for the first time-ever aimed at answering three basic questions: what, how much and where on the arm is deposited and transferred outside the vessel? Specific emphasis has been given to the transfer and sticking of beryllium and tungsten particles, and to the contamination of the RH equipment and in-vessel air by tritium and beryllium.

**Experimental** To facilitate the task, ten adhesive carbon pads (stickers of 1 inch diameter) were placed in different locations along the robotic arm (boom) and on the multifunctional robot (Mascot). A drawing in Figure 2.3.1-1(a) shows schematically the structure of the multi-segment boom. It also informs about the extent in remotely controlled operations ensuring access to each place in the entire tokamak chamber. There is a skirt at the boom enclosure door which rubs on both the upper and lower surfaces of the plastic gaiter covering the metal structure. The equipment is made of aluminum (Al) and boron nitride (BN) is used as a lubricant of joints. Details of the construction and the location of stickers on the gaiter protecting the boom are presented in Figure 2.3.1-1(b).

![Figure 2.3.1-1. (a) Extent of boom in vessel and (b) location of carbon adhesive pads, 1 – 10, and boom joint numbering used to define smearing locations. Joints A1 – A6 are shown.](image)

The stickers were on the equipment for nearly four months with 672 hours of RH operation. This has also allowed for collecting airborne dust. The stickers were analysed by means of microscopy and X-ray spectroscopy. The cellulose filtration papers from the air samplers and gaiter smearing were analyzed for the content of beryllium and tritium. The specific tritium activity (kBq m⁻²) and specific beryllium mass (µg m⁻²) were then calculated using the smeared area, typically 1000 cm².
**Dust accumulated on adhesive stickers** A collection of micrographs in Figures 2.3.1-2 and 2.3.1-3 are surveys for two samples from distinctly different locations: the top surface of the boom (Sample 9) and the Mascot wrist (Sample 4), respectively. In both cases observations were performed at several places on the sticker. There are islands with a fairly high density of particles exceeding 1000 mm$^{-2}$. On all other exposed stickers one also finds species of various size and shape but the areal density is lower than on those shown above. The composition of individual grains is very diverse and one finds a mix of low-Z and medium-Z elements: B, C, N, O, Na, Mg, Al, Si, S, Cl, Ca, Cr, Fe, Ni. Aluminum and carbon particles constitute the majority. It becomes evident that Al originates from the structural material of the RH equipment, as there are no other sources of pure Al in the vessel. The presence of boron nitride is associated with the boom: a component of the lubricating agent. The origin of some species, e.g. Ca or bits of Si, has not been identified; the source remains unknown.

![Figure 2.3.1-2. SEM image and an example of X-ray spectrum for dust particles accumulated on Sample 9 from the top surface of the boom A3-A3B.](image)

![Figure 2.3.1-3. SEM and EDS data for various dust particles accumulated on Sample 4 from the right wrist of the Mascot: (a) and (b) survey images in different area; (c) X-ray spectra for selected single particles; numbers in the spectra correspond to individual grain marked in frame (a).](image)
However, the least frequent are those particles for which the whole reported search has been designed and performed: (a) Be flakes from co-deposited layers or tiny droplets from melt-damaged limiters; (b) W flakes or droplets. Only one flake of a co-deposit rich in Be and C has been identified on Sample 4 (Mascot wrist). That object with splitting strata is shown in Figure 2.3.1-4. One cannot exclude that there were also some other tiny Be-rich species sticking to the pads, but one can state that the RH operation did not lead to massive mobilization of co-deposits. In the case of tungsten only very few tiny (less than 1 µm) species have been found. In general, metal particles (Fe, Ni, Cu, W) identified on various samples are of micrometer size.

Figure 2.3.1-4. A flake of a stratified co-deposit containing mainly beryllium and carbon; a single object found on Sample 4 from the right wrist of the Mascot.

**Boom gaiter contamination by beryllium and tritium** The contamination was assessed using a smearing survey procedure in which two cellulose based filter papers are rubbed across the surface to be tested; one for beryllium assessment and one for tritium measurements. The size of the smeared area is also recorded to enable determination of specific values of contamination per unit area. For the boom gaiter a series of Be and T smears were taken between the joints A-0/A-1, A-1/A-2 etc., along the top and bottom surfaces. The results of the smear analyses are shown in Figures 2.3.1-5(a) and (b) for T and Be, respectively. They indicate that in most cases the values are higher on the upper side than the lower side, although the values on the upper side seem to be higher from A-3B to A-5. In this type of exercise it is not possible to determine the direct cause for that increase but one can consider two highly probable reasons: (a) light contact of RH (touching) with the vessel wall and (b) relative proximity to the Mascot operation area in comparison to other samples from A-0 to A-3B. In general, the contamination by beryllium is considered to be low, < 1.5 µg/m² for 91% of results and < 4 µg/m² for all results,
as shown in Figure 2.3.1-5(b). This is below the 10 μg/m² threshold used whereby additional controls such as personal protective equipment, personal air sampling and specified working procedures are required to work with Be. The detection limit is 0.06 μg.

Figure 2.3.1-5. Results of smear test on the gaiter: (a) specific tritium activity and (b) specific mass of beryllium in twelve examined positions.

Concluding remarks

Until now the examination of dust on the RH equipment has been the only exercise of that kind. The results inform in detail about the character of contamination during a long in-vessel work with many tasks performed. Already with the current set of data one can talk about a truly new insight and valuable contribution into at least two areas of dust research: (i) assessment of hazards related to particles’ mobilization in RH operation and (ii) aid in critical view and formulation of conclusions regarding the origin of matter retrieved from tokamaks. The study confirms earlier experimental evidence that Be-rich co-deposits (and also W coatings on CFC) adhere well to plasma-facing components. It has become evident that in-vessel activities generate particulates. The most important is that the accumulation of the most "fearsome" species, i.e. Be and W, is negligible. Very few small pieces of W from the coatings and only one Be-rich flake from peeled-off co-deposits have been detected.

2.3.2. Fuel inventory and impurity deposition castellated tungsten tiles

Plasma-facing components (PFC) in ITER will have a castellated structure in order to improve thermo-mechanical durability and integrity under high heat flux loads. However, such structure may act as a trap for deposited species thus leading to enhanced fuel inventory in grooves of the castellated tiles and between the tiles. Eroded and transported impurity atoms will be co-deposited with fuel species in the grooves which may be considered as remote areas from the direct plasma flux. The surface area of grooves in ITER will be approximately three times larger than plasma-facing surface (PFS). In addition, remote area cleaning is difficult by currently known method. This study is focused on the impact of tile shaping and misalignment on the
impurity deposition and fuel retention in castellated tungsten tiles exposed in KSTAR (Korea Superconducting Tokamak Advanced Research) with the major radius $R = 1.8$ m and the minor radius $a = 0.5$ m. The inner wall is fully covered by around 3400 graphite tiles.

**Experimental** In order to study the impact of tile shaping and misalignment on fuel retention a number of castellated tungsten tiles were fabricated. Two sets of tungsten tiles were mounted on a stainless-steel base and installed in the divertor, as shown in Figure 2.3.2-1. Based on the ITER divertor mono-block design, the tiles were designed as cuboids with surfaces of 30 mm $\times$ 12 mm and 20 mm high. As shown in Figure 2.3.2-2, the top 4-5 mm region is tungsten and then 2 mm pure copper as intermediate layer based on a 13 mm plate made of copper-chrome-zirconium (CuCrZr) alloy. The tiles were installed with perfect alignment and some with intentional misalignment. The misaligned tiles are taller than the neighboring ones to create shadowed area from the plasma flux. In addition, the height of one side of the chamfered tiles was elevated to 0.5 mm. In this study, four different poloidal gaps were analyzed: one aligned gap and three misaligned gaps considering chamfer effect as misalignment.

![Figure 2.3.2-1. Location of the tungsten tiles from the top view on the divertor and two sets of the exposed castellated tungsten tiles in the cassettes.](image-url)
The tungsten setup was located in the central divertor. As shown in Figure 2.3.2-1, there are four lines of rectangular graphite tiles named CD1 to CD4 in poloidal direction. The setup was installed in CD2 in red dot box, and blue dotted box marks two graphite tiles which were designed and installed in order to eliminate any shadowing to the tungsten tiles. By the X-point radial position control, the outer strike point was located either CD1 or CD2 on central divertor or on the outer divertor. Tungsten tiles were exposed to L- and H-mode plasma for 4364 seconds during whole 2015 KSTAR campaign (total plasma operation times is 12,411s). Central divertor temperature ($T_e$) and density ($n_e$) were calculated by scrape-off layer plasma simulation (SOLPS) based on midplane temperature. $T_e$ is about 50-100 eV near the outer strike point and 1-10 eV in the private flux zone. The average values of plasma parameters near the castellation are: $T_e = 20-60$ eV, $n_e = 8-25 \times 10^{17}$ m$^{-3}$ and $T_i = 1.5 T_e$. The temperature of bulk tungsten is about 100-500 °C while the surface has reached to 1200 °C, according to preliminary tests. And the base temperature of tiles between shots are 50-80 °C without active cooling.

**Surface analysis** was performed using nuclear reaction analysis (μ-NRA) and particle-induced X-ray emission (μ-PIXE). From the energy distribution, the concentration of carbon and deuterium was calculated with the SIMNRA code, while the metal concentrations were calculated by GUYLS using the PIXE results. **Modelling** was done with the 3D-GAPS impurity transport code for selected poloidal gaps. Impurity penetration into gaps along magnetic field lines was assumed, so that plasma-exposed areas were defined by geometrical shadowing. In the following one complete example is presented.

**Results and discussion.** The geometry of the chamfered and misaligned gap is shown in Figure 2.3.2-3(a) while the appearance of surfaces inside the gap is in Figures 2.3.2-3(b) and (c). The gap has a different deposition level and trend than on the shadowed side has several features: (i) erosion at the top edge, (ii) bold black belt and (iii) hue gradation. Analysis results match this trend as shown in Figure 2.3.2-4(d) and (e). The amount of deposition increases exponentially from the entrance to the highest value of carbon $2.0 \times 10^{18}$ cm$^{-2}$ and deuterium $4.9 \times 10^{17}$ cm$^{-2}$ at about 0.2-0.3 mm from the plasma-facing surface and then decreases within 0.5 mm into the groove to $1.0 \times 10^{16}$ cm$^{-2}$ (carbon) and only $5.0 \times 10^{15}$ cm$^{-2}$ (deuterium) into the groove. D/C ratio of shadowed side is about 0.2 near the top and decreases to 0.1 after 1 mm, whereas over 0.3 in all range at open side. Modelling result, Figure 2.3.2-4(f) has the same trend and the decay length of 0.5 mm on the shadowed side. In contrast, the open side has the highest value at the top edge. The gradation of about 1 mm from the groove entrance in Figure 2.3.2-4(c) is confirmed by NRA: C and D decrease exponentially in about 1 mm in Figure 2.3.2-4(e). In summary, the chamfered gap has less deposition than that for the flat and aligned case.
Figure 2.3.2-4. (a) Geometry of the chamfered and misaligned gap for modelling. The left side is shadowed and the right is open to the magnetic field line. Optical microscopy of surfaces in the poloidal gap: (b) shadowed and (c) open side. C and D deposition profiles in the (d) shadowed and (e) open side. Modelling of carbon deposition profiles: (f) shadowed side, (g) open side.

Figure 2.3.2-5 shows the distribution of metallic impurities in the groove of chamfered and misaligned gap. The iron (Fe) and chromium (Cr) deposition profiles follow those for C and D seen in Figure 2.3.2-4(d). The greatest value at the entrance, $2.5 \times 10^{17}$ Fe cm$^{-2}$ and $8.0 \times 10^{16}$ Cr cm$^{-2}$, and then one observes a decrease by two orders of magnitude deeper into the gap.

Figure 2.3.2-5. Deposition profiles of Fe and Cr in the groove of chamfered and misaligned gap.
**Concluding remarks.** The work constitutes a contribution to the assessment of relation between the arrangement of PFC tiles and fuel retention in grooves of castellated structures. The main value is in an integrated approach to achieve the research goal with a dedicated experiment performed in KSTAR: (i) careful design and precise manufacture of the tiles, (ii) definition of experimental condition, i.e. strike point position, (iii) ex-situ analyses of all deposited species in an analytical system and (iv) modelling using a relevant code. As could be expected, carbon is the main deposited species, but the impact of the shaping and alignment/misalignment is huge taking into account the chamfer of only 1° and misalignment of even 0.3 mm. It is remarkable that such small modifications change the deposition by 50 times ($283 \times 10^{17} \text{ cm}^{-2}$ versus $6 \times 10^{17} \text{ cm}^{-2}$) with the greatest value in the flat and aligned tile setup, while a deliberate misalignment and/or chamfer have a positive impact on the decrease of deposition. Carbon fluxes drive co-deposition of deuterium with the deuterium-to-carbon concentration ratio in a range from 0.1 to 0.4.

From the quantitative point of view integrated deuterium contents in differently shaped gaps are to be considered: $3.64 \times 10^{18}$ in the flat aligned and $0.57 \times 10^{18}$ which correspond to 12 µg and 1.9 µg, respectively, on the area of around 1 cm$^2$ (gap length $\times$ deposition width). These small amounts would lead to huge retention if they are extrapolated to a full-scale operation in a carbon machine with millions of castellated tiles. However, with the elimination of carbon in next-step devices the risk of massive carbon-related retention also would be strongly reduced. The most positive aspect is connected with the coherence of experimental and modelling data revealing one very positive trend: sharp decrease of deposition and related retention with the depth in the gap. These results of the carefully prepared experiment consistently match observations that the deposition inside castellated metallic structure is very shallow. The data also indicate once again the need for tile shaping in a reactor class-machine.
2.4. Theoretical fusion plasma physics

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In collaboration with the EUROfusion member CCFE, CEA, CIEMAT, ENEA, EPFL, IPP, IST, LPP-ERM-EMS, VTT, Wigner RCP, with PPPL, F4E and ITER-IO.

The fusion plasma physics theory group is focused on studying wave-particle interactions and integrated modelling relevant for present day fusion experiments and future reactors. The group is particularly active in developing numerical models and codes for studies of Ion Cyclotron Resonance Heating (ICRH), and in validating them against experiments. This work is well integrated into the European fusion program through participation in the integrated modelling activities in EUROfusion/WPCD and the exploitation of the European experimental facilities. Recently the group has become involved in integrated modelling, and in particular with the development of the European Transport Solver (ETS) with the EUROfusion/WPCD.

2.4.1. Integrated modelling within the EUROfusion-WPCD

The group participates in integrated modelling within EUROfusion under the Work Package for Code Development (WPCD), where Thomas Jonsson is the coordinator for the development of the ETS in IMAS, see the figure below, as well as the activities related to Heating and Current Drive (H&CD).

Six deliverables related to the ETS were set and achieved for the end of 2020, including a commission deliverable on tutorials, training, documentation on and release of the core-edge transport simulator in IMAS. The ETS combines state-of-the-art models from many fields, including four types of heating and current drive, turbulent and collisional transport, plasma equilibrium and stability, plasma control systems and electron-run-away processes, as well as data management and visualization. These codes are connected in a framework that evolves the plasma state in time, while ensuring consistency between the evolution of the magnetic field, the thermal and the non-thermal plasma populations. Consequently, the ETS development team combines experts from many fields, as well as a core-team of four people focusing on the integration, the testing, the release-cycle, the development of the transport solver and the visualization. In addition to managing the ETS team, KTH contributed significantly to the integration, the development of the solver for the transport equation and the visualization.

In order to achieve these six deliverables one Code Camp (2 weeks with more than 20 people) and 8 working session (3-5 days with 4-15 people) were arranged. In addition, weekly group meetings were arranged with 15-25 participants.

Within the heating and current drive activities the group was responsible for delivering workflow that could simulate the real-time control of so-called neoclassical tearing modes (NTMs) using electron cyclotron current drive (ECCD). This was a collaboration between four groups (from KTH, EPFL and ENEA) with expertise in NTMs, ECCD, real-time control and the IMAS infrastructure. Here KTH was leading the project, implemented the real-time control system and performed the modelling.
2.4.2. New ICRF code FEMIC

The main codes developed by the group are PION, FIDO, SELFO, SELFO-light, RFOF, FOXTAIL and FEMIC. PION was the first self-consistent code for modelling ICRH and NBI heating using simplified models and is used routinely at JET. For more advanced modelling the Monte Carlo code FIDO was developed to calculate the distributions of resonant ions taking into account effects of finite orbit width, RF-induced spatial transport and interaction between MHD waves and fast ions. By coupling the FIDO and wave code LION the self-consistent ICRH code SELFO was developed. The latest development is the FEMIC code which combines state-of-the-art numerical techniques available in COMSOL in combination with advanced plasma physics implemented in MATLAB.

The new ICRF wave solver FEMIC, has been further developed during 2020. The FEMIC code uses COMSOL Multi-physics for calculating wave field, in combination with a plasma model implemented in MATLAB and connected to COMSOL using Live-Link for MATLAB. The use of COMSOL provides FEMIC with flexible adaptable gridding and an optimized electromagnetic solver. However, the wave equation relevant for ICRF are more complex than conventional electromagnetic waves due to non-local terms in the dielectric response kernel. In order to include the non-local response, a novel technique has been developed based on a spectral representation of the wave and iterations using the Andersson-acceleration technique. As a proof of principle, this technique was used to describe ion-Bernstein modes in a 1D representation of the tokamak. The Bernstein waves in FEMIC were benchmarked against the TOMCAT code.

In addition, non-local effects in 2D geometry have been implemented to resolve the impact of poloidal fields on the Doppler-shift. This work is still in progress, but promising results have already been obtained, showing that the poloidal field can modify the balance between ion and electron heating.
2.4.3. JET Work programme

During 2020, the KTH group has participated in the analysis of JET experiment on plasma transport and in the preparation for the coming JET D-T campaign. The work has been performed through ICRH modelling with the SELFO code and with the European Transport Solver, ETS. For the JET D-T campaigns, the ETS was run with the wave code Cyrano and the two Fokker-Planck codes StixRedist and FoPla to quantify the impact of ICRF on the neutron yields and the synergy between ICRF and NBI. In addition, the accumulation of tungsten in JET hybrid discharges were studied, including the impact from ICRF.
2.5. Computational methods for fusion plasmas, with a focus on time-spectral methods

J. Scheffel
In collaboration with:
PhD student K. Lindvall, KTH
Prof H. Nordman, Chalmers University of Technology, Gothenburg

The present project concerns time-spectral modelling. In the computational approach termed the Generalized Weighted Residual Method (GWRM), traditional finite time differencing is replaced by a spectral representation of the time domain. The computed solutions are truncated, approximate analytical Chebyshev polynomial series valid for all time, spatial and, optionally, physical parameter domains and are immediately tractable for mathematical analysis. Scalings in physical parameters are thus potentially obtainable in a single computation. The GWRM has during 2020 been successfully employed for solving an extensive set of 2D problems, including drift wave turbulence in tokamak geometry. We have also, for the first time in numerical modelling, successfully solved problems in kinetic plasma theory employing time-spectral methods.

The GWRM is expected to be of commercial interest; it has the potential to provide the engine in a new generation of time-efficient solvers for nonlinear large scale simulations relating to subjects like fluid mechanics and meteorology. In a recent article we have shown that chaotic elements of numerical weather forecasting may be modelled more efficiently with the GWRM than by standard time-marching methods. Now facing challenging fusion physics implementations, we develop essential elements like adaptive time and spatial domains in 2D and strong parallelisation of the time domain.

2D Nonlinear hydrodynamics / plasma physics modelling

For benchmarking the GWRM for 2D nonlinear fluid modelling, the compressible Navier-Stokes partial differential equations have been solved. The particular problem addressed was the Kelvin-Helmholtz instability, in which two fluids with different densities and with oppositely directed velocities are oriented next to each other. For adequate resolution, and satisfying the CFL stability criterion, explicit finite difference methods require a grid of approximately 500×500 grid points. The time development of the instability, as solved using the GWRM, can be seen in the adjacent graph.
The spatial domain was subdivided into $16 \times 16 = 256$ subdomains, each using eighth order Chebyshev polynomial solution expansions. The time intervals of the GWRM was about 5 times longer than finite differencing time steps. Two different root solvers were used; the rapidly convergent Quasi Semi-Implicit root solver (developed by us, mainly for 1D problems) and Anderson acceleration. Generally Anderson acceleration has been found to be best suited for 2D simulations because of the very low memory consumption of the method. For more advanced problems, with higher order spatial derivatives, the Quasi Semi-Implicit root solver is preferable.

2D Nonlinear magnetohydrodynamics / plasma physics modelling
A second benchmark test has been to solve the full nonlinear compressible MHD equations for the Orszag-Tang Vortex (OTV) problem. This is also a difficult test problem, since steep gradients quickly evolve as shock waves and shock-shock interactions. It is thus an excellent test of a number of different algorithms that should be included in numerical simulation like divergence cleaning, shock capturing, adaptive mesh control and post-processing filters.

The problem was solved by the GWRM using $40 \times 40 = 1600$ spatial subdomains and $6^{th}$ order Chebyshev polynomials. A corresponding finite difference domain would constitute some $700 \times 700$ grid points. The time interval used by the GWRM was a factor 10 longer than that of finite difference methods. A time series of a solution can be seen in the graphs above.
2D Nonlinear magnetohydrodynamics / Tokamak turbulence

The diffusion of plasma across flux surfaces can be partly explained by classical/neo-classical effects. It is, however, widely known that also anomalous transport, due to turbulence, plays an important role. One of the unstable modes that leads to microturbulence is the Ion Temperature Gradient (ITG) drift mode, also called the $\eta_i$ (ratio of density scale length and temperature scale length) mode. It is excited by universally present temperature and density fluctuations in the plasma.

To analyze the ITG drift wave modes, we employ the two-fluid Weiland model. In the figure (calculated in a box of 100×100 Larmor radii and with Cyclone base case parameters; $\eta_i = 3.14$, $\tau = 1.0$, and $\epsilon_n = 0.909$) we show a 2D GWRM computation of the perturbed electrostatic potential. From the computational results we can verify the fastest growing ITG mode $\gamma_{ITG} = 0.3$ and the existence of so-called “streamers”. These are long radial structures in the linear growth rate phase that lead to enhanced turbulence. It has been shown that zonal flows tend to break these structures and dampen the turbulence to a saturated state.

A massive improvement is given by the potential for parallelization so that multiple CPUs/computers can solve for the individual spatial subdomains in parallel. Recently we have invented a method to drastically reduce the number of computational operations and the memory requirements for computing the Jacobians required for the GWRM root solver. For guaranteed convergence, the GWRM requires simultaneous information from all parts of the computational domain. A primitive algorithm would then simultaneously involve all the Chebyshev coefficient equations of all the subdomains. This would be costly - with $N$ being the total number of Chebyshev modes the memory requirements scale as $N^2$ and computational CPU time scales as $N^3$, being prohibitive for advanced problems. With present optimization of the GWRM we have however obtained the very favourable scaling $N^{1.4}$ for CPU time.
2.6. Confinement physics

L. Frassinetti, E. Stefániková, H. Nyström in collaboration with JET, AUG and TCV teams.

2.6.1. Pedestal properties and confinement in JET

Operations in JET resumed in autumn 2011 after the shutdown to install the ITER-like wall (hereafter called ILW). In 2020, the study of the pedestal properties in JET with the ILW has continued. The work has been developed along three key topics: (1) comparison of the pedestal stability in JET-C and JET-ILW, (2) creation of a large database with JET pedestal parameters and its exploitation.

Comparison of the pedestal stability in JET-C and JET-ILW.

The work has investigated the differences in pedestal structure and stability of JET-C and JET-ILW Peeling-Ballooning (PB) limited plasmas with similar engineering parameters. The work has experimentally characterized the pedestal structure and performed simulations with the pedestal predictive code Europed. The goal has been to study the effect of the parameters that affect the pedestal stability ($n^\text{pos}$- $T^\text{pos}$, $n^\text{ped}$, $Z_{\text{eff}}$, $w_{\text{pe}}$, $\beta_N$), and to quantify their contributions to the change in pedestal height and pressure gradient from JET-C to JET-ILW in PB limited plasmas. The analysis has shown that the relative shift and the pedestal density play a major role in affecting the stability, while relative shift, pedestal width and $Z_{\text{eff}}$ have a major impact on the pedestal pressure height.

To quantify the total effect on the critical pressure gradient and on the predicted pedestal pressure height, a set of simulations has investigated the combined effect of all the parameters discussed above. To quantify the effect of each single parameter, the scan of each new parameter has started where the previous specific scan was finished. At the very end, a full ‘path’ in the $\alpha_{\text{crit}}$-$p_{\text{ped}}$ space of the gradual change from JET-C parameters to JET-ILW parameters has been created. This is shown in figure 2.6.1(a). Considering the change in all the five parameters, the reduction of $p_{\text{ped}}$ is $\sim$18% and the reduction of $\alpha_{\text{crit}}$ is $\sim$25%. The final estimated pedestal height and normalized pressure gradient are in reasonable agreement with experimental values of pedestal of the JET-ILW.

![Figure 2.6.1. Results of the Europed simulations of the JET-C discharge 78672. (a) The figure shows a gradual change in critical normalized pressure gradient and predicted pressure height, when parameters $n^\text{pos}$, $T^\text{pos}$, $n^\text{ped}$, $Z_{\text{eff}}$, $w_{\text{pe}}$ from similar JET-ILW discharge were used as an input. (b) Critical $p_e$ profiles and (c) critical $p_e$ gradient for experimental values of parameters for discharge 78672 (green) and for discharge 78672 with ILW parameters from discharge 83583 (red).]
pulse. This is also shown in figure 2.6.1(b) and in figure 2.6.1(c) using profiles of the pressure and of the pressure gradient respectively. The green continuous line highlights the critical profiles obtained using the JET-C pulse input parameters while the red continuous line highlights those using the JET-ILW input parameters. The dashed lines show the fit to the experimental data.

Figure 2.6.2. Schematic diagram of parameters influencing the PB stability and pedestal pressure height of JET-C and JET-ILW plasmas and their possible interconnections.

* due to operational constraints

The work proposes a mechanism that could explain the degradation of the pedestal pressure of PB limited plasmas from JET-C to JET-ILW. It must be emphasized that this proposed mechanism is an explanation that can be applied only to the PB limited plasmas. Therefore, it is only a partial explanation for the lower JET-ILW pedestal performance of baseline plasmas. Further mechanisms will need to be invoked to explain the difference in non-PB limited plasmas.

A schematic diagram of the proposed mechanism is shown in figure 2.6.2. The change of the wall from carbon to metal has led to four key differences. First of all, this has changed the main plasma impurity from carbon to beryllium and has significantly reduced $Z_{\text{eff}}$, from $Z_{\text{eff}} \approx 2.0$-2.5 to $Z_{\text{eff}} \approx 1.0$-1.5. Then, to operate safely, metal wall discharges are characterized by a relatively high gas fuelling rate, while JET-C operation has no fuelling rate. Finally, the change of the wall might have also led to a different recycling. The change of the main impurity, and of $Z_{\text{eff}}$, has not led to a significant change in the pedestal stability, while it has reduced the electron pedestal pressure via the dilution effect. The contribution of this term to the degradation of the electron pedestal pressure from JET-C to JET-ILW is approximately 78%. The effect of higher fuelling and different recycling is instead more complicated. Mainly, this might have led to a higher neutral pressure. It is reasonable to assume that, in turn, the increase of the neutral pressure has led to the increase of the pedestal density and of the separatrix density. This has also led to the outward shift of the density and hence to the increase of the relative shift. These links are currently under investigation in JET with the EDGE2D-EIRENE code while in AUG have been already verified using SOLPS. Note that density position and separatrix density are strongly correlated and the increase of $n_{e}^{\text{sep}}$ has led to the increase of $n_{e}^{\text{sep}}$ due to a simple geometrical effect. The increase in $n_{e}^{\text{sep}}$ and the outward shift of the density (modeled with the increased relative shift in section 4 and 5) and the increase in the pedestal density has led to a reduction of the pedestal stability. The overall contribution of these terms to the $\alpha_{\text{crit}}$ degradation from JET-C to JET-ILW is approximately 95%.
The EUROfusion JET pedestal database.

The work has completed the EUROfusion JET pedestal database and has started its exploitation, emphasizing on two main topics. First, a description of the JET-ILW pedestal structure and stability. In particular, the work describes the links between the engineering parameters (power, gas and divertor configuration) and the disagreement with the peeling-ballooning (PB) model implemented with ideal MHD equations. Specifically, the work clarifies why the JET-ILW pedestal tends to be far from the PB boundary at high gas fueling and high power, showing that a universal threshold in power and gas fueling cannot be found but that that the relative shift (the distance between the position of the pedestal density and of the pedestal temperature) plays a key role. These links are then used to achieve an empirical explanation of the behavior of the JET-ILW pedestal pressure with gas fueling, power and divertor configuration.

Figure 2.6.3. (a) $T_{\text{ped}}$ versus $n_{\text{ped}}$. The colors highlight the plasma current (see the color bar on the right). The dashed lines show the isobars at constant $p_{\text{ped}}$. (b) Correlation between the pressure pedestal width and the poloidal beta at the pedestal for the entire database. The continuous line shows the EPED1 assumption with $D=0.076$. The dashed lines highlight the trends $D(\beta_\theta^{\text{ped}})^{1/2}$, with the corresponding $D$ specified near the end of each line. The colors highlight the ratio $\alpha_{\text{crit}}/\alpha_{\text{exp}}$.

Figure 2.6.3 shows an example of the results, with frame (a) that highlights the link between the height of the pedestal and the plasma current. The increasing plasma current lead to the increase of both the temperature and the density at the pedestal top. As a consequence, also the pedestal pressure increases with increasing $I_p$. Frame (b) show the correlation between the width of the pedestal pressure and the poloidal beta. The continuous line shows the EPED1 assumption $w_{\text{ped}}^{\text{EPED}}=0.076(\beta_\theta^{\text{ped}})^{1/2}$. It is clear that the JET-ILW pedestal width does not follow the $0.076(\beta_\theta^{\text{ped}})^{1/2}$ dependence assumed in EPED1. Moreover, the $D$ value (defined as $w_{\text{ped}}/(\beta_\theta^{\text{ped}})^{1/2}$) is not constant and varies in the range 0.04-0.16. Therefore, we can conclude that, in general, the EPED1 assumption on the pedestal width is not satisfied in JET-ILW. It is still possible that such a large variation in $D$ is consistent with EPED1.6 model.

Two further important results can be extracted from figure 2.6.3(b), both related to the distance of the pre-ELM pedestal from the PB boundary. This has been quantified as the ratio $\alpha_{\text{crit}}/\alpha_{\text{exp}}$. When the ratio is close to 1, the ELM is triggered when the pedestal reaches the PB boundary. With $\alpha_{\text{crit}}/\alpha_{\text{exp}} >>1$, the ELM is triggered when the pedestal is still PB stable. The colors in figure 2.6.3(b) highlight the ratio $\alpha_{\text{crit}}/\alpha_{\text{exp}}$, with a light blue color showing the PB limited pedestal ($\alpha_{\text{crit}}/\alpha_{\text{exp}} \approx 1$) and green/yellow/red colors a PB stable pedestal ($\alpha_{\text{crit}}/\alpha_{\text{exp}} > 1.5$).
Figure 2.6.3(b) shows a clear systematic pattern, in which, at constant $\beta_\theta^{\text{ped}}$, the distance from the PB boundary increases with increasing pedestal width. Basically, a wide pedestal tends to be far from the PB boundary. The second interesting result is that PB limited pedestals tend to be consistent with the EPED1 assumption. As shown in figure 2.6.3(b), the data with $\alpha_{\text{crit}} / \alpha_{\text{exp}} \approx 1$ (light blue data) have a pedestal width consistent with the expression $w_{\text{ped}}^{\text{EPED}} = 0.076(\beta_\theta^{\text{ped}})^{1/2}$ (the continuous black line). This is an important result because it shows that the JET-ILW pedestals that are PB limited can be correctly predicted by EPED1, both in terms of pedestal height and pedestal width.

Second, the pedestal database is used to revise the scaling law of the pedestal stored energy. The work shows a reasonable agreement with the earlier Cordey scaling in terms of plasma current and triangularity dependence, but highlights some differences in terms of power and isotope mass dependence.

Figure 2.6.4(a) compares the experimental $W_90$ and the $W_{\text{ped}}$ from the Cordey scaling for D plasmas. Colors highlight the plasma current. (b) Distribution of $W_90 / W_{\text{ped}}^{\text{Cordey}}$. The colors highlight the distributions for subsets at different $I_p$. (c) Correlation between experimental $W_90$ and $W_{90}^{\text{fit_0}}$ for D plasmas. (d) Distribution of $W_90 / W_{90}^{\text{fit_0}}$.

Figure 2.6.4(a) compares the experimental $W_90$ (stored energy at $\psi_{N=0.90}$) with the pedestal stored energy expected from the Cordey scaling. The colours highlight the plasma current. $W_90$ and $W_{\text{ped}}^{\text{Cordey}}$ are well correlated. 67% of pulses have $W_90 / W_{\text{ped}}^{\text{Cordey}}$ in the range 0.8-1.2 (i.e. the Cordey scaling is consistent within ±20% with 67% of the data), see figure 2.6.4(b). However, $W_90$ is slightly lower than $W_{\text{ped}}^{\text{Cordey}}$ at high plasma current. Note that the database used in this work has excluded, for simplicity, pulses with N seeding and with pellets. From
earlier JET-ILW studies, it is known that the pedestal pressure can be increased by seeding nitrogen and that the highest plasma performances have been achieved in pulses with pellets. These subsets are shown figure 2.6.4(a) as empty symbols. The N seeded pulses (mainly at 2.5MA, high triangularity) significantly exceed the Cordey scaling. The pulses with pellets (mainly at 3.0MA and 3.5MA) achieve a $W_{90}$ roughly consistent with the Cordey scaling.

The next step in the comparison with the Cordey scaling is to determine the scaling law of $W_{90}$ and compare the exponents. Ideally, the same power law used by Cordey should be implemented. However, due to (i) the correlation between $I_p$ and $B_t$ and between $I_p$ and density and (ii) the negligible variation of $R$, $F_q$ and $k_a$, only $I_p$, $P$, and $M$ (where $M$ is the isotope mass) can be considered. On the other hand, since also triangularity and fueling rate affect the pedestal, the new scaling law includes also $\delta$ and $I_{D2}$. The dataset used to determine the scaling law is the same as used throughout this work, i.e. excluding plasmas with pellets, seeding, RMPs and kicks and considering only type I ELMs. However, due to the importance of the isotope mass, hydrogen plasmas have been included. The majority of the hydrogen plasmas is characterized by low $I_p$, so their inclusion might adversely influence the fit. To bypass the problem, two scaling laws have been derived, the first one including only deuterium plasmas (hereafter labelled “fit 0”), the second one including both deuterium and hydrogen plasmas (hereafter labelled “fit 1”). In both cases, a robust Bayesian regression technique has been used, allowing errors in all variables. The results are summarized in expressions (2.6.1) and (2.6.2):

$$W_{90}^{fit0} = (0.31 \pm 0.07) I_p^{1.27\pm0.15} P^{0.30\pm0.08} \delta^{0.29\pm0.12} \Gamma^{-0.07\pm0.04}, \quad (2.6.1)$$

$$W_{90}^{fit1} = (0.23 \pm 0.07) I_p^{1.26\pm0.15} P^{0.31\pm0.08} \delta^{0.29\pm0.12} \Gamma^{-0.07\pm0.04} M^{0.5\pm0.2}, \quad (2.6.2)$$

where $W_{90}$ is expressed in MW, $I_p$ in MA, $P$ is the loss power in MW and $\Gamma$ is in $10^{22}$e/s. The inclusion of hydrogen plasmas does not significantly affect the exponents, showing that the result is robust. The error bars on the estimated exponents (posterior means) reflect the range over which the exponents can be varied, taking into account the posterior correlations, such that the median absolute percentage error of the fit increases by a 1-2 percent at most. The error bars defined in this way are typically larger than the usual error estimates based on the standard error (ordinary least squares, OLS) or the posterior standard deviation (Bayesian).

By comparing expression 2.6.1 with the Cordey scaling, it is possible to note that the $I_p$ exponent and the $P$ exponent are slightly lower than those obtained by Cordey ($\alpha_I=1.41\pm0.06$ and $\alpha_P=0.5\pm0.04$). The lower $\alpha_I$ is due to the fact that the density is not considered in expressions 2.6.1. The exponent of the isotope mass, $\alpha_M$, is similar to the one recently determined for the energy confinement in JET-ILW ($\alpha_M=0.4$) but is significantly larger than that in the IPB98(y,2) scaling ($\alpha_M=0.19$). For JET, this difference has been discussed in a recent work on the JET global confinement and it appears to be related to the inclusion of T plasmas. The positive dependence on the triangularity is very reasonable, as at low collisionality the high-$\delta$ plasmas tends to have higher pedestal performance than low-$\delta$. The extremely weak dependence on $I_{D2}$ is likely due to the fact that the gas fueling rate is not the optimal parameter to estimate the neutral pressure and the actual fueling.
2.6.2. Pedestal performance in TCV

In 2020, the work related to TCV has continued, in order to shed light on the role of the divertor baffles on the pedestal performance. The activity has been developed both from the experimental side and the data analysis side. From the experimental side, a set of new discharges without baffles have performed in order to have reference shots for comparison with the baffled plasmas obtained in 2019. The work has shown a clear difference in the pedestal between operations with and without baffles. Figure 2.6.5 shows the correlation of the divertor neutral pressure with the pedestal height electron temperature (a) and density (b).

![Figure 2.6.5](image)

Figure 2.6.5. (a) Correlation between pedestal temperature and divertor neutral pressure. (b) Correlation between pedestal density and divertor neutral pressure. Full symbols highlight pulses without baffles, empty symbols pulses with baffles.

The pedestal temperature decreases with increasing divertor neutral pressure in plasma without baffles. The same trend is less clear in plasmas with baffles. In particular, for the same divertor neutral pressure, plasma with baffles tend to have higher pedestal temperature than without. The pedestal density does not show any clear difference between operations with and without baffles. At present, further investigation is ongoing. In particular peeling-balloonning stability analysis should be able to clarify the origin of the difference in the pedestal temperature between the two operations.

This work will contribute to understand the role of the divertor closure in the plasma performance and is therefore high relevance for the optimization of the design of future machine and future fusion power plant.
3. Education and research training

3.1. Basic and advanced level education

The following courses were given in 2020 by the Division of Fusion Plasma Physics:

**Basic level courses**

ED1100 Engineering Science

ED1110 Vector Analysis
Learning oriented course in vector calculus. The course is useful for further studies of electromagnetic theory, wave propagation, fluid mechanics, plasma physics, gas dynamics and the theory of relativity.

**Advanced level courses**

Fusion Plasma Physics provides Advanced level courses for the KTH Master Programme in Electromagnetics, Fusion and Space Engineering. The programme is given in collaboration with the Electromagnetic Engineering, Space and Plasma Physics Divisions at the EECS School. The programme focuses on the foundations of electrical engineering such as electromagnetic fields and their interaction with matter. Physical principles, mathematical methods and numerical models make up the core of the programme, providing the tools and skills needed to describe electro technical processes and analyse complex systems and problems in the field.

ED2200 Energy and Fusion Research
An introduction to fusion oriented plasma physics is given. The central areas of fusion research are emphasised. The progress of fusion research and its present state are discussed in the perspective of future power generation.

ED2210 Electromagnetic waves in Dispersive Media
The course introduces students to methods of treating electromagnetic waves. The electromagnetic theory is described by Fourier transforms in space and time which is advantageous when treating propagation and emission of waves in dispersive, anisotropic media.

ED2235 Atomic Physics for Fusion
The purpose of this course is to make the student familiar with those aspects of atomic physics that are most important in fusion research. The focus of the course is on basic understanding of atomic collisions and applications in plasma modelling, plasma diagnostics and plasma surface interactions. Much of the course content is applicable also in other contexts in plasma processing and technology, ion implantation and radiation effects.
**ED2240 Introduction Course to Fusion Technology**  
The course is designed for physicists and engineers who want to learn about technology and properties of wall materials applied under extreme conditions in operation of a reactor-class fusion device. The completion of the course should provide background for future work at fusion physics laboratories and fusion-related industry.

**ED2246 Project in Fusion Physics**  
The student will learn about practical experimental research work by carrying out a small research project. The projects are performed in a real research laboratory environment; utilizing the EXTRAP T2R fusion plasma experiment at the Division of Fusion Plasma Physics. The student will engage in a project that also leads to a more in-depth understanding of some common fusion plasma diagnostics methods.

**Degree projects**

The following degree project in fusion plasma physics were completed in 2020:

**Master Degree (30 credits)**

Léo Belleil  
_Implementation of Collisions in FEMIC for Modelling of the RF Heating with a Lower Hybrid Resonance_

Vilhelm Dinevik  
_Comparative Analysis of Adaptive Domain Decomposition Algorithms for a Time-Spectral Method_

Amit Kharwandikar  
_Optimization of Heat Exhaust in the Edge of Tokamaks via Controlled Magnetic Stochastization_

**Bachelor Degree (15 credits)**

Alexandr Djadkin and Emrah Tortumlu  
_Upgrade of the Analytical System for Studies of Plasma-Facing Components from a Tokamak_

Erik Lindström and Christos Tegkelidis  
_Modeling of Radio Frequency Heating in JET_
3.2. Research training

The research training in Fusion plasma physics and technology at the Division is part of the Plasma Physics track of the Electrical Engineering Doctoral Programme (E2DOC) of the School of Electrical Engineering and Computational Science (EECS). During 2020, there were altogether nine PhD students active at the Division.

The following PhD thesis defences took place during 2020:

**Respondent:** Pablo Vallejos  
**Thesis title:** Modeling RF waves in hot plasmas using the finite element method and wavelet decomposition  
**Date:** 17 January 2020  
**Opponent:** Dr. Philippe Lamalle, ITER Organisation

**Abstract:** Fusion energy has the potential to provide a sustainable solution for generating large quantities of clean energy for human societies. The tokamak fusion reactor is a toroidal device where the hot ionized fuel (plasma) is confined by magnetic fields. Several heating systems are used in order to reach fusion relevant temperatures. Ion cyclotron resonance heating (ICRH) is one of these systems, where the plasma is heated by injecting radio frequency (RF) waves from an antenna located outside the plasma. This thesis concerns modeling of RF wave propagation and damping in hot tokamak plasmas. However, solving the wave equation is complicated because of spatial dispersion. This effect makes the wave equation an integro-differential equation that is difficult to solve using common numerical tools. The objective of this thesis is to develop numerical methods that can handle spatial dispersion and account for the geometric complexity outside the core plasma, such as the antenna and low-density regions (or SOL). The main results of this work is the development of the FEMIC code and the so-called iterative wavelet finite element scheme. FEMIC is a 2D axisymmetric code based on the finite element method. Its main feature is the integration of the core plasma with the SOL and antenna regions, where arbitrary geometric complexity is allowed. Moreover, FEMIC can apply a dielectric response in the SOL and in the region between the SOL and the core plasma (i.e. the pedestal). The code can account for perpendicular spatial dispersion (or FLR effects) for the fast wave only, which is sufficient for modeling harmonic cyclotron damping and transit time magnetic pumping. FEMIC was used for studying the effect of poloidal phasing on the ICRH power deposition on JET and ITER, and was benchmarked against other ICRH modeling codes in the fusion community successfully. The iterative wavelet finite element scheme was developed in order to account for spatial dispersion in a rigorous way. The method adds spatial dispersion effects to the wave equation by using a fixed point iteration scheme. Spatial dispersion effects are evaluated using a novel method based on Morlet wavelet decomposition. The method has been tested successfully for parallel and perpendicular spatial dispersion in one-dimensional models. The FEMIC1D code was developed in order to model ICRH and to study the properties of the numerical scheme. FEMIC1D was used to study second harmonic heating and mode conversion to ion-Bernstein waves (IBW), including a model for the SOL and pedestal. By studying the propagation and damping of the IBW, we verified that the scheme can account for FLR effects.
Respondent: Estera Stefániková

Thesis title: Pedestal structure and stability in JET-ILW and comparison with JET-C

Date: 9 June 2020

Opponent: Dr. Joelle Mailloux, CCFE

Abstract: Controlled thermonuclear fusion offers a promising concept for safe and sustainable production of electrical energy. However, there are still many issues to be investigated on the way to a commercial fusion reactor. An important point for detailed studies is connected to wall materials surrounding hot thermonuclear plasma. The JET tokamak (the largest fusion experiment in the world) in the United Kingdom has completed a major upgrade in 2011 in which the materials of the vessel surrounding the fusion fuel have been changed from a carbon-fibre-composite (or JET-C wall) to Beryllium and Tungsten. These new materials are the same as those that will be used in a next step fusion device International Thermonuclear Experimental Reactor ITER (hence the name ITER-like wall or JET-ILW), designed to demonstrate the feasibility of fusion reactor based on the tokamak concept. One of the goals of JET with the ILW is to act as a test bed for ITER technologies and for ITER operating scenarios. The overall purpose of the thesis work is to characterise the effect of the ILW on the structure and stability of edge plasma phenomenon called the pedestal, a steep pressure gradient associated with the H-mode, an operational regime with improved confinement. The aim is to contribute to the understanding of the difference in the pedestal performance between JET-C and JET-ILW. The work is focused on experimental characterisation of the pedestal structure in deuterium discharges by analysing the experimental data (radial profiles of electron temperature and density measured in H-mode plasmas) from Thomson scattering diagnostics at JET and on investigating the differences in pedestal stability between JET-ILW and JET-C plasmas in terms of the pedestal modelling. The pedestal structure is determined using a modified hyperbolic tangent fit to the experimental Thomson scattering profiles. The modelling is performed with the pedestal predictive code Europed, based on the EPED model commonly used to predict the pedestal height in JET. The experimental analysis has shown several differences in the pedestal structure of comparable JET-ILW and JET-C discharges. One of the key differences introduced in this work is the pedestal relative shift (a separation between the middle of the pedestals of the electron density and temperature) that plays a major role in the difference in the pedestal performance between JET-C and JET-ILW. The work shows that the relative shift can vary significantly from pulse to pulse and that, on average, JET-C plasmas have lower relative shift than JET-ILW plasmas. The pedestal relative shift tends to increase with increase in the gas fuelling and the heating power. Furthermore, the increase in the relative shift has been empirically correlated with the degradation of the experimental normalized pressure gradient $\alpha_{\text{exp}}$. To understand the differences in the JET-C and JET-ILW pedestal stability, parameters that affect the pedestal stability and that tend to vary between comparable JET-C and JET-ILW discharges have been identified. These parameters are the pedestal relative shift, pedestal density $n_{\text{ped}}$, effective charge number $Z_{\text{eff}}$, pedestal pressure width $w_{\text{ped}}$, and normalized pressure $\beta_N$. The modelling performed with the predictive Europed code has shown that these five parameters are sufficient to explain the difference in the pedestal performance between JET-C and JET-ILW. Furthermore, the modelling has shown that the relative shift and $n_{\text{ped}}$ play a major role in affecting the critical normalized pressure gradient $\alpha_{\text{crit}}$ (normalized pressure gradient expected by the model comparable to $\alpha_{\text{exp}}$), while the relative shift, $w_{\text{ped}}$ and $Z_{\text{eff}}$ have a major impact on the pedestal pressure height. Finally, a possible mechanism that has led to the degradation of the pedestal pressure from JET-C to JET-ILW is proposed.
Doctoral programme development and reorganisation

The E2DOC doctoral programme, to which the PhD students at the Division of Fusion Plasma Physics belongs, is the largest at KTH, encompassing some 250 doctoral students. The program director (Jan Scheffel) from January 2018 and onwards is employed at the Division. Among the events within the program in 2020 the following may be highlighted.

• Further discussions on the required course credit points in the program have been carried out. The present requirement, 75 hp, is regarded as too high by the vast majority of the PhD students, by six out of the eight associated divisions, by the programme council and by the programme director. The FA (director of doctoral education), however, does not support the suggested change to the standard KTH level of 60 hp. A change to 60 hp was further supported by an alumni follow-up in 2020, where the respondents expressed that some of the courses taken during their studies were less relevant both for the thesis and the career.

• Judging from the comments by the PhD representatives in the programme council, which gathers monthly, the PhD students are quite satisfied with the E2DOC programme. Rarely, apart from the 60 hp course credit discussion, seriously critical issues are raised. This is a major reason to that the program, for example, does not arrange events like annual joint workshops with the supervisors or the PhD students.

• In 2019, a working group within the EECS school suggested that a mentoring programme for recruitment of female doctoral students will be developed. This entails offering female master students mentors (PhD students or faculty), closer contacts with the divisions and laboratories, 3-6 months of pay for work as research engineers when performing their Master theses as well as club rooms, both at Campus and in Kista. The EECS school management has declared that it will support the project financially, and preliminary work is now carried out in terms of focus group interviews of master students, PhD students and supervisors. A project leader was assigned in 2020.

• A LaTeX template for PhD theses was assembled by two PhD students at the Division, free for anyone at KTH to use.

FuseNet

The Division of Fusion Plasma Physics is a member of FuseNet since a few years. This is an organization for increasing, enhancing and broadening fusion science and technology training in Europe. In particular, our PhD students are taking part in the FuseNet PhD Events, the aims of which are to enable students to disseminate their research, develop a network of contacts and learn from each other's experiences.

Graduate level courses

In E2DOC, courses for a PhD Degree should cover 75-120 credits whereas thesis work should cover 120-165 credits, altogether 240 credits. For a Licentiate degree, courses should cover 45-60 credits and thesis work 60-75 credits, adding up to 120 credits. Courses at the advanced undergraduate level may be included, insofar as they are not requirements for admission. For licentiate and PhD degrees at most 15 credits or 30 credits from undergraduate courses may be included, respectively. The institutional work carried out by the PhD students, typically lasting for a period of one year in total, is mainly within undergraduate teaching (ED1100 Engineering Science and ED1110 Vector Analysis).
All students enrolled into the new E2DOC Doctoral Programme should take four general skills courses, amounting to 10 credit points, in the topics of oral and written communication, pedagogics, theory of science and research methodology, as well as research ethics, sustainability and gender/diversity equality:

- FLH3000 Basic Communication and Teaching
- FAK3014 The Theory and Methodology of Science – Minor Course
- FAK3127 The Sustainable Scientist
- FDS3103 Introduction to Scientific Writing for Doctoral Students

The following graduate level courses in the subject area of fusion plasma physics are recommended, depending on the doctoral student’s research direction (course responsible teacher is given in parentheses):

- FED3220 Motion of Charged Particles, Collision Processes and Basis of Transport Theory, 8 credits (T. Jonsson)
- FJD3300 Kinetic Plasma Theory (J. Scheffel)
- FED3230 Magnetohydrodynamics, 8 credits (J. Scheffel)
- FED3240 Plasma waves I, 8 credits (T. Jonsson)
- FED3260 Fusion Plasma diagnostics, 8 credits (P. Brunsell)
- FED3305 Magnetohydrodynamics, advanced course, 6 credits (J. Scheffel)

The two courses (or courses with similar content) listed below are compulsory. They both include activities and learning goals related to sustainable development topics.

- ED2200 Energy and Fusion Research, 6 credits (J. Scheffel, P. Brunsell, master level)
- FED3320 Fusion research, 8 credits (L. Frassinetti)
4. Personnel

**Professor**
- Per Brunsell (Division Head)
- Marek Rubel
- Jan Scheffel

**Professor, emeritus**
- James Drake
- Torbjörn Hellsten
- Bo Lehnert
- Michael Tendler

**Associate Professor**
- Henric Bergsåker
- Lorenzo Frassinetti
- Thomas Johnson

**Researcher**
- Per Petersson

**Engineer**
- Håkan Ferm

**PhD student**
- Laura Dittrich
- Kristoffer Lindvall
- Björn Ljungberg
- Sunwoo Moon
- Hampus Nyström
- Erik Saad
- Stefan Schmuck
- Estera Stefániková
- Pablo Vallejos Olivares
5. Income & Expenditure

The accounts for the Division of Fusion Plasma Physics for the year 2020 are summarized in the table below. Note that from the year 2019, all income and expenditure related to education is allocated at the Department of Electrical Engineering, and therefore removed from the Division accounts.

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Bibliography


