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Key Issues in Plasma-Wall Interactions for ITER: A European Approach

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ABSTRACT.
The first burning fusion plasma experiment, ITER, based on the tokamak principle, is now ready for construction. Based on the continuous progress of many years of fusion research, the design relies on a large and robust set of experimental data. The focus of present day fusion research is therefore shifting towards questions of the ITER plasma operation and the availability of ITER. The latter is governed mainly by plasma wall interaction issues, in particular the life time of plasma facing components and long term tritium retention. To coordinate the research activities in this area a task force for plasma wall interaction (EU-PWI-TF) has been initiated by the European fusion research programme under EFDA. This contribution describes the experimental data base in these areas and outlines the task force strategies and research needs to address the critical issues.

1. INTRODUCTION

The International Tokamak Experimental Reactor (ITER) has been finally designed and can be constructed. This is the collaborative result of decades of fusion research from many experimental magnetic fusion devices world wide which result in a large and robust database. The design meets the scientific and technological objectives within appropriate margins. These are:

- a long-pulse (about 8min) burning fusion plasma at an energy amplification factor Q of at least 10 for the a plasma current of 15MA,
- the capability to investigate steady-state plasma operation, with parameters which allow Q = 5 and 'hybrid' scenarios with pulse lengths of the order of 30 min,
- integration and investigation of fusion technology relevant for the DEMO reactor

However, various questions remain open for which the confidence is less robust and/or which cannot be addressed sufficiently in present experiments. They are the key areas of ongoing research but final answers might only be given by ITER itself as a physics experiment. These issues are largely related to with plasma-wall interaction processes mainly connected with the extended duty cycle and increase of the plasma stored energy. Constraints are

- to achieve technically acceptable conditions for the heat and power exhaust (10MW)
- to achieve a sufficient lifetime of the plasma facing components (>3000 ITER shots). This goal is largely related to the control of transient heat loads during edge localised modes (ELMs) and to disruptions. Sublimation or melt layer limit of the target materials has to be avoided.
- to stay below the long term tritium inventory limit which is set by safety consideration to 350g T.

In the past, the efforts in divertor and plasma edge physics have been emphasised strongly to develop scenarios for power and particle exhaust. They are available now and based on high density divertor operation which leads to detachment reducing the peak power loads by radiation, charge exchange and recombination processes [1,2]. This will not be discussed here but any plasma scenario or new wall material choices has to be compatible with these conditions.

The lifetime of plasma facing components (PFC) and the long-term tritium retention are priority subjects of the EU PWI-TF. These processes are controlled by material erosion and the succeeding short and long range migration. The extrapolation from present-day devices having a full carbon wall clearly shows that the long term tritium retention in ITER would soon exceed the T-retention limit (350g). This means only an unacceptably limited number of pulses in a D-T mixture would be allowed in a full graphitic wall environment. However ITER will have different wall materials with Be as the material in the main chamber ($700m^2$), tungsten in the upper regions of the divertor and the dome region ($70m^2$) and carbon fiber composit (CFC) tiles on a relatively small area ($50m^2$) on the high flux regions in the lower divertor. This will change the carbon erosion and redeposition behaviour and most probably relax the T-retention problem compared to a full carbon device. The challenge is to predict the lifetime and T- retention under these conditions quantitatively without any tokamak experiment having an ITER like material composition.

Thus physics understanding of the processes is needed to predict the fuel retention and lifetime. This is also the basis to development of techniques to mitigate or control the T retention, e.g. by means of carbon traps, temperature tailoring or sophisticated geometry. In parallel more work has to be done to develop ITER compatible methods to remove the T from PFC or co-deposited layers. Finally one has to consider the situation that the long term T retention in ITER will be unacceptably high such that the use of any carbonaceous wall material must be avoided. In order to be prepared, a full metal first wall scenario must be developed. As a consequence, a co-ordinated European research activity has been established (EU Task Force on Plasma-Wall Interaction) to provide an improved information concerning the lifetime of the ITER target plates, the rate of tritium inventory build-up and to suggest improvements, including material changes, which could be implemented at an appropriate stage. This contribution summarises the present experiences and data on those subjects and outlines the strategies of the EU-PWI TF.

2. OBSERVATIONS ON FUEL (TRITIUM) RETENTION IN PRESENT DEVICES

The retention of hydrogen in graphite by implantation and the formation of amorphous hydrogen rich carbon layers upon impact of carbon atoms or ions together with hydrogen species is well known since long and has also been observed in fusion devices [2, 3,]. However, this issue has been realised only after the first Tritium experiences in TFTR [5] and JET [6]. About 3 and 36g of T were injected in TFTR and JET, respectively, from which large amounts (30-40%) were retained on short term scale in the machine. Similarly large amounts of fuel stored in the walls are also observed just at the end of single discharges, e.g. in JET, Tore Supra and other machines [e.g. 4]. Despite various cleaning mechanism (not discussed here), about 13 and 10% of the T remained in TFTR and JET respectively. These observations are well in line with findings devices operating in H/ D, such as TEXTOR. The amount of long term D retention has been assessed by adding up the amounts of D found in deposited carbon layers on various locations related to the cumulative D input over the plasma operation time (about half a year) yield ing a very similar retained fraction of 8% [7]. Extrapolating such values to ITER (fuelling rate $200Pam^3/s$) the T safety limit of 350g would be reached in less then 50 shots.

Thereafter cleaning procedures must be applied to recover the fuel for further Tritium plasma operation.

It is found that the majority of the long term retained fuel is stored in carbon co-deposits formed in different locations in the machine [7, 9, 10, 11,]. On erosion dominated graphite areas, the fuel retention is restricted to implantation in a shallow surface layer, enhanced due adsorption onto rough surfaces of CFC tiles, saturating at few 10^{17} H/cm^2 [8]. This would result into 5g T for the first wall area of ITER and is tolerable. Co-deposited C-layers, however contain hydrogen fractions up to 1:1 H/C depending on temperature and impact particle energy and the layer growth continues as long as erosion occurs. Layers are formed on many locations of the plasma facing wall tiles but also on areas with no direct plasma ion impact (“remote areas”). It turns out that the understanding of the Tritium retention requires the understanding of the erosion of carbon and local and global transport inside the device or to remote areas. The first goal of the EU PWI-TF work is to initiate co-ordinated analysis and experiments to understand these processes in detail.

3. LOCATION AND STRENGTH OF IMPURITY SOURCES

In the past a priority attention was to measure the divertor impurity sources and their screening while emphasis on wall sources was less intense. Post mortem analysis of wall tiles revealed, however, that the material deposition found in the divertor is predominantly by main chamber erosion. Strong evidence of this came from AUG [9] and JET. In JET the inner divertor shows strong deposition while erosion/ deposition in the outer is balanced pointing towards the wall as main source of material [10]. In addition, Be is evaporated routinely in the main chamber with no line of sight to the divertor while Be is found mainly on the plasma facing sides of the inner divertor [13]. In AUG both divertor legs are deposition dominated requiring a net material source in the main chamber [11]. Also, the previous AUG W-divertor experiment showed little change of the main plasma carbon content and a strong C deposition in the inner divertor, indicating that erosion of carbon in the main chamber is the source of the carbon depositio n in the divertor and the impurity content of the main plasma [14]. In DIII, recent spectroscopic analysis of the divertor and main chamber C-sources revealed a major contribution of carbon from main chamber to the carbon content. [15].

The quantification of the main chamber impurity sources is difficult due to the large areas and the restricted number of spectroscopic lines of sight. In JET, a combination of main chamber spectroscopy, methane injection screening experiments and edge modelling yielded a carbon source strength of about 3×10^{20} C/s and 3.2×10^{20} C/s in the MKIIA and MKGB operational campaigns respectively. This amounts to a total carbon wall source of about 390 and 440g C, respectively [16]. The averaged Be wall source in JET is about ten times smaller, reflecting the inhomogeneous Be-coverage of the wall and the fast erosion of the thin Be layer [16]. These C- sources have to be compared with the carbon deposition in the JET divertors where a total amount of 1000g in MKIIA and 400-500g in MKBG were found. Although analysis to consolidate the latter value is still going on, the values are in surprisingly fair agreement with the carbon sources from the main chamber . The carbon influx from the main chamber in AUG has been evaluated by spectroscopy and is about 1×10^{20} C/s [17] . Thus the

C influx in JET and AUG scales roughly with the first wall area (≈ 50 and ≈ 200m^2 respectively).

Present data give only indirect indication to discriminate the contribution of ion and neutral induced erosion of main chamber graphite tiles and of the fraction of chemical and physical erosion. Lowering the wall temperature in JET from 320 to 220°C did not affect the carbon flux from the wall (based on CIII spectroscopy) indicating either a significant contribution of physical sputtering or a weak dependence of chemical erosion in this temperature range [18]. Comparison of discharges in D and He showed a strongly reduced Carbon influx in He indicating that ion induced physical sputtering cannot play the dominant role. These questions need to be addressed further in order to better predict the main chamber Be-erosion in ITER. It is also important to analyse the locations from where the impurities are released in the main chamber. In AUG the coverage of the inner wall with tungsten did not decrease significantly the C-content of the plasma . It resulted in a partial coverage of the inner wall with carbon. This strongly points to the outer wall (graphite guard limiters) as the dominant carbon source [19]. Zeemann resolved spectroscopy revealed, however, a larger C-recycling source at the inner wall, coated with the tungsten layer. Obviously the C recycles many times on the inner wall while it is initially released on the outer wall [17]. In JET, Zeemann analysis showed no clear preference of carbon released from inner or outer wall. This is in line with Ha tomography of inner and outer recycling fluxes, indicating about equal contributions from inner and outer wall [20]. Recently, an enhanced transport mechanism in the outer SOL is discussed , based on ballooning- like blobby transport which predicts plasma interaction mainly on the outer wall areas [21]. This is confirmed qualitatively by camera observations and probe measurements A quantification of the strength of this contribution to the main chamber plasma interaction is still open.

Connected with this is the question of the contribution of main chamber interaction during and in between ELMS. The fraction the ELM power deposited at the divertor target plates is about 30%, but this fraction may inc rease in the main chamber. Hints for this are obtained in JET and elsewhere [22,23] but again a quantification is still missing. It should be mentioned that the target temperature in the JET divertor approaches already the carbon sublimation limit in high performance type I ELM discharges (low ELM frequency) but the main chamber observations, which are restricted to CCD observations, do not indicate such high temperatures. This indicates that ELMs in the divertor are more severe than in the main chamber. If this is confirmed it would ease the problem of the narrow power margins for the target plates.

However, the knowledge about plasma wall interaction processes in the main chamber is insufficient and effort should focus on

- identifying locations and strengths of main chamber impurity sources
- identifying the contribution of ion and neutral induced erosion and , for C, the fraction of physical and chemically induced erosion
- identifying the fraction of erosion in between ELMs and during ELMs and analysing the dependence of ELM induced erosion on the ELM type and parameters (type I,III, ELM size)

4. LONG RANGE MATERIAL MIGRATION

Material eroded in the main chamber preferentially flows into the inner divertor and turn it into a deposition dominated area. This is observed in all major divertor tokamaks (JET, DIII, AUG, JT60). This behaviour has been confirmed in JET by ^{13}C marked methane injection from the top of the machine. The results from subsequent surface analysis of wall and divertor tiles were: the majority of ^{13}C is found in the inner divertor and, to a lesser degree, on the main chamber tiles, whereas no ^{13}C was detected in the outer divertor [24]. The behaviour in the outer divertor differs from machine to machine and can depend on divertor geometry (JET, AUG) or plasma conditions (DIII).

The larger area at the outer torus periphery which favours the power flow to the outer divertor can not explain the in-out asymmetry observed in present devices. It is largely determined by preferential SOL flows towards the inner divertor (with $\nabla B \times B$ towards the divertor). These flows reach Mach numbers as large as 0.5 in JET at the top of the machine and is by far larger than the classical flow speed predicted from $E \times B$ and $\nabla B \times B$ drifts (Mach number of about 0.1) [25]. In JET it has been shown that localised main chamber recycling cannot produce such flows and an explanation of their nature is still missing. Promising attempts have been made to explain the flows by radial fluctuations [26]. Observations in JT-60U show a material flow through the private flux region from the outer to the inner divertor. This is confirmed by modelling and of importance for the material transport discussed here [27]. Two issues have to be addressed, therefore:

- explaining the flows in the existing devices. This is the basis to predict the SOL flows in ITER which influence the Be-deposition of the inner and outer graphite divertor tiles and thus the physical and chemical erosion of the underlying graphite .
- analysing possible flows in the private flux region. Such flows may short-circuit the outer and inner divertor which would strongly affect the material transport.

5. SHORT RANGE MATERIAL TRANSPORT INSIDE THE DIVERTOR

Experimental data on material transport in the inner divertor are available mainly for carbon. JET provides important data on transport of Be which is evaporated in the main chamber and additional data will be available soon on tungsten transport in the AUG divertor [28]. In devices which apply boronization , boron transport can also be analysed and compared with that of carbon. This has not been used extensively so far but is strongly recommended. In JET MKIIA divertor the influx ratio of Carbon to Beryllium into the divertor is measured to be about 12[16] but the layers formed on the plasma facing side of the tiles are Be-rich with a ratio C/Be of typically 0.3-1 [13]. This shows that carbon does not remain in the layers and undergoes further erosion induced transport while Be adheres. The majority (>90%) of the carbon C flowing into the divertor is found on shadowed areas of the horizontal tile 4 (Fig.1) and on the water cooled louvers at the entrance of the pump duct. Large parts have fallen down in form of flakes to the divertor floor [10]. In the JET gas box divertor (MKGB) the carbon deposition in the divertor is very similar (≈ 500g in total compared with 1000g in MKIIA) but the majority of the carbon is now on tile 4 (fig.1) and on the septum whereas the Be is mostly deposited on tile 1 and 3. The amount of carbon that enters the louver entrance area has been determined by in situ Quartz micro-balance techniques (QMB) [29] and sticking monitors [30]. Both diagnostic show that the material flow to this area is largely reduced (at least a factor of ten) compared with the previous MKIIA . In addition the QMB data show that the amount of carbon deposited on the location of the QMB increases strongly with decreasing distance of the strike point to the louver entrance. It increases even more with the strike point on the horizontal tile. This was the favoured configuration in the previous MKIIA divertor configuration and may explain partly the large carbon deposition on the louvers. In AUG and JT 60 divertor tile analysis is ongoing but the present data indicate the majority of carbon is deposited on the plasma facing side of the divertor tiles, similar as in JET MKIIGB. In AUG layer growth is also observed in the subdivertor region [11] but the carbon deposition on the plasma facing sides is stronger. Sticking monitors mounted in the inner divertor region of the JET gas boy divertor [30] show that the majority (>90%) of particles have high sticking and only a small fraction has low sticking. A similar conclusion has to be drawn from observations made in TEXTOR where carbon is deposited into shadowed areas of the limiters (drill holes) [7] . Also boron is not transported in TEXTOR towards remote areas, e.g. the TEXTOR limiter pump ducts, while carbon films are formed on those areas, albeit representing only a small part of the total carbon transport in TEXTOR [31]. The high sticking probability of the majority of the carbon species allows only a minority of the carbon species to reach the pump ducts before being deposited Elements which show no chemical erosion like Be in JET, B in TEXTOR or W in AUG do not show long range transport at all.

The outer divertor in JET shows almost no clear erosion nor deposition (except on a narrow band on the horizontal tile near the edge of the lower vertical tile) and behaves thus in a favourable manner: carbon is eroded but locally re-deposited and not transported long ways. The particle fluence towards the outer divertor in JET MKIIGB campaign (1.5×10^{27} D^+) would produced a total erosion of about 600mm suggesting effective local re-deposition of the eroded carbon. Actually, in AUG inner and outer divertors are now deposition dominated indicating influx from sources in the main chamber [32]. Important questions remains for further clarification

- clarify further the physics of carbon migration to remote areas: by preferential chemical erosion of C or by thermal decomposition of “soft” mixed C/Be films. Is transient heating by ELMs important for the carbon transport?
- is the transport of the carbon into the inner divertor from the main chamber different from the transport of carbon which is eroded from the divertor target itself. This questio n is important for ITER where the C flux from the main chamber is absent.
- what is the reason for the difference in transport behaviour of carbon in the JET outer and inner divertor: is this due to differences in plasma parameters (higher temperatures) or to the reduced carbon flow from the main chamber.

6. MODELLING OF IMPURITY TRANSPORT

Development of impurity transport codes validated thoroughly against experimental results is the final tool to predict material transport and fuel retention in ITER. Such codes are Monte Carlo codes following up the pathways of impurities in a limited volume near the target plates (WBC-code[33], ERO-code, [34]). Other edge codes provide the input of particles arriving into this volume and the local plasma parameters (EDGE2D, B2-Eirene). Comparison of the transport codes with various experimental observations in TEXTOR [35] and JET [36] revealed that the modelled re-deposition and transport pattern of C does not agree with experimental observations. In JET the modelled amount of carbon migrating to the louvers is about a factor of ten less compared with observations and the local re-deposition of $^{13}CH_4$ on the TEXTOR limiters is about a factor of 50 too small compared to experiments. "Standard" assumptions on erosion yields (physical sputtering according to Yamamura sputter formulas, 2% chemical erosion and sticking probabilities of re-deposited species according to the TRIM kinetic reflection model) were used. Much better agreement can be achieved if the erosion yields are enhanced and the sticking probabilities decreased. However high sticking probabilities [7,30] are observed in experiments and the present assumption is thus that the re-erosion of a re-deposited carbon by the background plasma is much larger compared with a carbon atom in a graphitic surface plane. Also the sticking of low energy hydrocarbon radicals returning to the surface has been lowered in accordance with new molecular dynamic calculations [37] and experimental investigations using thermal radical beams. In this area we need to

- improve the input data base, such as chemical erosion yields, atomic data for the break-up of hydrocarbon species or sticking probabilities of hydrocarbon radicals and carbon atoms.
- benchmark transport models in dedicated experiments of impurity transport under different plasma conditions.

7. OTHER KEY PWI QUESTIONS FOR ITER

It is a challenge to predict the effect of Be deposition on the erosion behaviour of the graphite target tiles in ITER, in particular in the presence of transient heat loads by ELMS or disruptions. A large Be impurity flow towards the inner divertor is predicted (a better quantification is needed) which will turn the inner divertor in a deposition dominated zone. The influence of Beryllium deposition on the carbon erosion was investigated in PISCES in co-operation with the EU- PWI programme [38]. A complete suppression of the carbon chemical erosion due to Be-coverage was observed confirming the hope to reduce the carbon transport in the inner ITER divertor significantly. However the ELM power deposition in ITER will only be marginally below the sublimation limit of graphite. This can ablate the protecting Be layer regularly and expose the carbon surface again. On the other hand the formation of Be-C compounds may also occur under these conditions which may survive these temperature excursion. Predictions for the outer target are more difficult since the Be flux to the divertor is unclear (see chapter 5) and physical re-sputtering of Be is higher. The T- retention in the Be layers must be assessed also. In [16] a Be deposition of about 60g/discharge is predicted. This requires a T/Be ratio < 10^{-2} in order to stay below the T- limit of 350g for 1000 discharges. Pure Be retains only minor amounts of T while incorporation of carbon or oxygen will enhance the retention.

Another key area of research is to explore all possible techniques to remove the fuel from fusion devices. This is a priori easier on areas directly facing the plasma (where we expect the Be- layers) compared with shadowed or remote areas without direct line of sight. On plasma facings surfaces the plasma impact itself can be used to remove the retained T by isotope exchange or plasma induced surface heating. This would require flexibility of the plasma configuration in the divertor. On those areas external heating by lamps or lasers might also be possible. These techniques can not be used on remote areas for which no clear scenario presently exist except mechanical tools or the use gaseous chemically treatment by active oxygen or ozone [39,40]. These techniques have to be compatible with plasma operation which needs research in present tokamaks. The main important issues which should be addressed here are :

- to evaluate the influence of Be deposition on carbon physical and chemical erosion
- to evaluate the thermal stability of Be/C layers during transient heat loads
- to explore all possib le methods to remove fuel from plasma facing components and from co-deposits and to demonstrate their tokamak applicability and plasma compatibility

8. DEVELOPMENT OF A SCENARIO WITHOUT GRAPHITE WALL COMPONENTS

Graphite is no option for DEMO due to the neutron induced material damage. In addition, the T-retention limit with graphitic components in ITER might not be fulfilled. Thus a coherent metal wall scenario has to be developed in parallel with the present first wall material option. This implies to demonstrate

- the compatibility of a tungsten divertor with the plasma operation scenarios foreseen for ITER. Here the constrains are in particular the non inductive current drive and the improved confinement scenarios
- the compatibility of a tungsten divertor with the power exhaust requirement allowing for a maximum power flux of $10MW/m^2$. In particular the detachment physics in the absence of carbon as a main radiator has to be investigated. Be radiation will replace this partly but additional impurity seeding might be necessary which may trigger tungsten impurity release and the tungsten contamination of the plasma.
- an acceptable lifetime of the tungsten target in presence of possible melt layer erosion by transient heat loads during ELMs or disruptions.

The main task of the EU-PWI TF is to concentrate on the latter issue while for a coherent approach also the first two issues are necessary. For ELMs the tolerable energy loss limit is slightly higher for tungsten then for graphite . The carbon sublimation limit for graphite is $40MW/m^2s^{0.5}$, while for tungsten the melt layer limit is $60MW/mm^2s^{0.5}$. The main advantage of CFC is that it does not melt if few ELMs may exceed this limit while tungsten will show melt layer erosion. Tungsten surfaces may also develop hot spots on the edges of the molten zones resulting in a reduced power handling capability.

The constrains on plasma operation are similar for W and C to keep the ELM power loss below about 2%-3% of stored plasma energy, but graphite can tolerate larger deviations of the ELM power loss from the averaged value [38]. The main driver for CFC in the strike-point divertor region is the possible melt layer loss of W in disruptions. In the ITER physics basis report [1] the melt layer erosion of W in disruptions is estimated based on the assumptions that

i) the plasma stored energy is lost towards the divertor,

ii) the power decay length of a disruption is 3 times broader than for normal plasma operation

iii) half the melt layer is lost by spla shing.

These assumptions result in loss of tungsten of about 50μm per disruption which reduced the lifetime of a W target to unacceptable low values. However, recent observations in AUG, DIII and JET [42,43,44] show that only a fraction of the thermal energy is deposited in the divertor and the power is spread over much larger areas as previously assumed. It is evident that the power loss by disruptions is a key issue of the EU-PWI TF. Research focus on

- the energy flux to the divertor and main chamber and the spatial and time evolution during thermal quench and dependence on disruption type
- the development disruption mitigation techniques relevant for ITER.

SUMMARY

The long term tritium retention is the most critical issue for ITER. Predictions for ITER need improvements based on dedicated understanding in present devices. An integrated approach and understanding is necessary of

- where and how impurities are produced in the main chamber
- how they are transported towards the divertor
- how the material is transported inside the divertor, in particular to remote areas

Transport models on long and short range impurity migration have to be further developed and benchmarked against dedicated experiments under various conditions.

Techniques to control in situ Tritium retention and to remove tritium from plasma facing components and co-deposited layers on remote areas have to be developed. Their tokamak applicability and compatibility must be addressed in present tokamak research.

It might turn out that the use of graphite in ITER has to be avoided since the T retention limit can not be kept. This requires a co-ordinated work to develop a tungsten divertor scenario and the use of a metal in the main chamber. The scenario has to be compatible with the main requirements of plasma operation, power exhaust and target lifetime. The EU-TF PWI work will focus on the characterisation of the disruption power deposition and the development of disruption mitigatio n techniques.

More tokamak experiences are needed to investigate the issues discussed in this contribution under ITER like wall materials conditions or under conditions avoiding the use of graphite in the main chamber.

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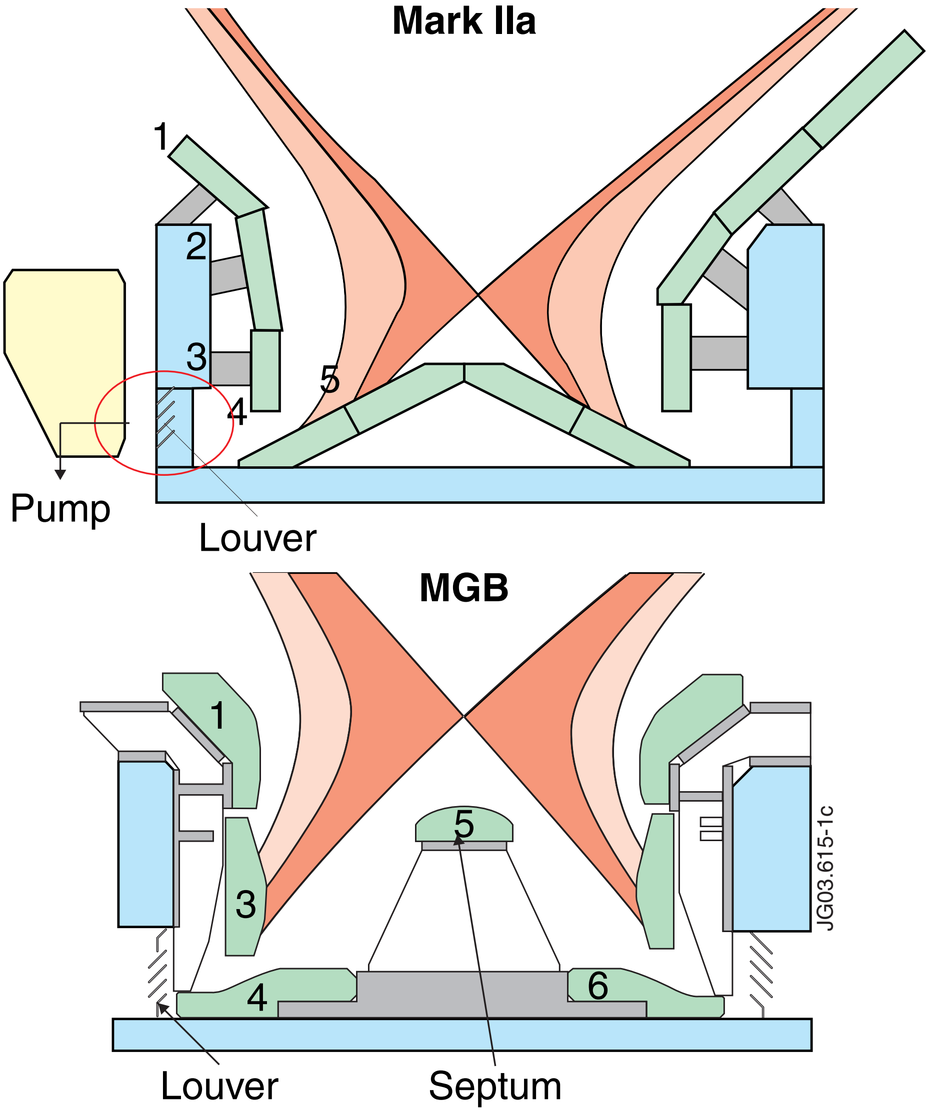

Figure 1: Schematic view of the JET MKIIA and MKGB configuration