Considerations towards the Fuel Cycle of a steady-state DT Fusion Device

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Abstract. Control and management of the fuel and fusion product streams is one of the most difficult issues for fusion energy development. The closed fusion fuel cycle comprises an inner and an outer part. The inner part includes the fuelling systems (gas puffing, pellet injection, etc.), the vacuum pumping systems and the tritium plant systems (fuel cleanup, isotope separation, storage and delivery). The outer part includes the breeding blankets and their tritium extraction and recovery systems. The challenges of these components are given by the need to handle large flows of tritium resulting in strong requirements on containment and accountancy. ITER will in some areas serve as a good basis for scale-up to a fusion reactor, in many other areas not. In all latter cases, new technologies have to be developed. This paper presents the current state of technology and identifies the areas which are considered to require essential supporting R&D towards a functional and technically ready system for a tokamak based reactor. This paper also recommends topical areas which need focussed efforts and illustrate the needs to provide better and well validated modelling tools. Efforts under EFDA are currently on the way to move in this direction.

1. Introduction and description of the present situation (ITER)

ITER is the first large fusion machine which is rigorously based and designed for a deuterium–tritium plasma and, hence, involves a dedicated fuel cycle, which has been developed over the last 20 years [1]. It represents an essential step towards the systems needed for future power plant fusion devices. Control of the gas throughput via the fuel cycle (unburnt fuel, impurities plus the helium ash produced by D–T fusion reactions) is one of the key issues affecting the performance and achievable burn time of a fusion reactor. The fuel cycle does not only provide the technical functions of processing, cleaning, separating, storage etc, but also the multiple barriers needed for proper confinement of tritium. The main sub-systems of the fuel cycle are illustrated in FIG. 1.

The requirements of ITER can be satisfied with the inner fuel cycle [2]. Tritium transported to ITER will be transferred to the storage and delivery system, which supplies all the gases necessary for machine operation to the fuelling systems (gas puffing, pellet injection, disruption mitigation etc.) and to neutral beam heating (NBI). An analytical system assists in the characterization of the gases and in accountancy procedures. The gases leaving the torus are processed via cryopumps and moved by roughing pumps into the tokamak exhaust processing system, which separates the hydrogen isotopologues that are transferred to the isotope separation system, while the remaining waste gas is sent to a detritiation system for decontamination purposes before release into the environment. The inner loop is closed by the return of deuterium and tritium to the storage.

Tritium for ITER will be supplied from external sources, but various concepts of tritium breeding test blanket modules will also be investigated. This will happen on very small scale only (approximately 3 orders of magnitude lower than what is expected for a steady-state device) but shall demonstrate the principle feasibility of the concepts [3]. For a next step machine, the challenges associated with the development of a self-sufficient blanket technology, the corresponding tritium extraction system, and the accountancy system needed as interface between blanket loops and the inner cycle [4] are huge.

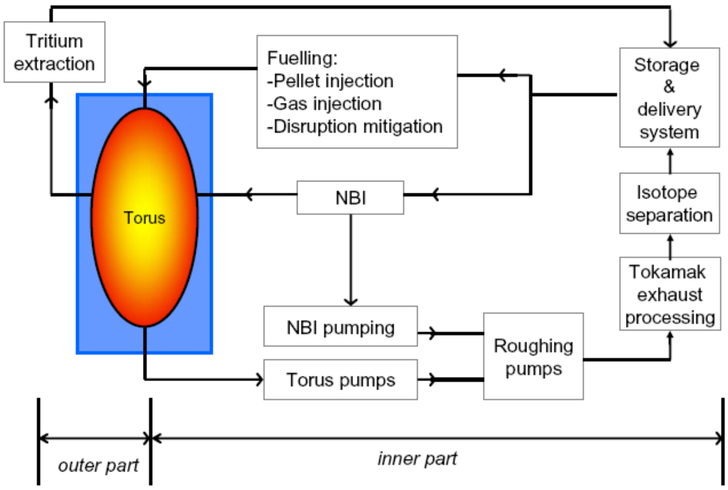

FIG. 1. The fuel cycle of a fusion device.

2. Power plant concepts

ITER will in some areas serve as a good basis for scale-up to a fusion reactor and power plant, in many other areas not. For comparison with ITER, the following section briefly describes the concepts of power plants developed in EU in the last decade (models A to D) [5], see TABLE I. Models A and B adopt the well establish H-mode regime assuming monotonic q profile with q_{95} = 3 within the limits for the global plasma performance HH < 1.2, n/n_{Gr}<1.2, β_N < 3.5 and first stability.

TABLE I: COMPARISON OF DIFFERENT CONCEPTS FOR POWER PLANTS.

	ITER	DEMO	REACTOR
Fusion power (GW)	0.5	2-2.5	3-4
Heat flux (first wall) (MW/m²)	0.1-0.3	0.5	0.5
Neutron wall load (first wall) (MW/m²)	0.78	<2	~2
Integrated wall load (first wall)	0.07 MW/m^2 (3 yrs inductive operation)	5-8 MW·year/m^2	10-15 MW·year/m^2

	ITER	A	B	C	D
Unit size (GWe)		1.55	1.33	1.45	1.53
Fusion power (GW)	0.5	5.00	3.60	3.41	2.53
Major radius (m)	6.2	9.55	8.6	7.5	6.1
Net efficiency		0.31-0.33	0.36	0.42	0.6
Plasma current (MA)	15	30.5	28.0	20.1	14.1
Bootstrap fraction	>0.71 Advanced steady state	0.45	0.43	0.63	0.76
P_{add} (MW)	73 (110)	246	270	112	71
Divertor peak load (MW/m²)		15	10	10	5
Av. neutron wall load (MW/m²)	0.57	2.2	2.0	2.2	2.4

Model A is based on the physics models employed in the ITER design, on water cooled lead-lithium blanket while the divertor is similar as in ITER. Both blanket and divertor are based on material and technologies needing little additional research. Model B uses a slightly more advanced physics model to compensate for the lower load capabilities of the divertor target as compared to model A. The model uses helium as coolant of the ceramic breeder blanket and divertor targets with refractory metals as structural material. An exit temperature of 480 °C allows for the use of a superheated steam cycle in the power conversion system.
Model C employs technologies and plasma physics models largely extrapolated from models A and B with consequently a more efficient power conversion system. It is based on a self-cooled lead-lithium breeding zone and a helium cooled structure made of reduced-activation ferritic steel. MHD problems are eliminated and high coolant exit temperatures are obtained using inserts made of SiC composite as thermal and electrical insulators in the large coolant channels. He-cooled divertor targets are designed for high coolant exit temperatures for efficient closed-cycle helium gas turbine. Finally, model D is based on high superconducting magnets and uses self-cooled lead-lithium blanket with SiC composite as structural material. The divertor targets made of refractory metals combined with SiC composites are lead-lithium cooled allowing coolant exit temperature of up to 1100 °C leading to high efficiency (>55 %) of the closed cycle helium turbine power conversion system. Heat fluxes on the divertor targets are reduced to less than 10 MW/m^2 assuming advanced plasma physics models.
To better characterize the various concepts, a programme on new technologies, parallel to the ITER programme has to be developed, devoted to the following fuel cycle relevant issues:

- validate breeding blanket concepts able to ensure the tritium self-sufficiency of a steady-state power plant and to operate with high reliability and availability. This includes the prior qualification of the technology necessary for tritium production, extraction and control and the qualification of plasma-facing components (PFCs);
- validate divertor concept able to operate with high reliability and availability;
- demonstrate tritium and vacuum pumping technology to provide the fuel cycle process and operational functions;
- develop high efficiency core fuelling systems to fully decouple the fuelling and ELM control functions and to allow for a large flexibility for controlling the divertor detachment through gas injection.

3. Functional requirements on a fuel cycle of a steady-state device

It must be noted that simple direct extrapolation of the ITER fuel cycle is not suitable to provide efficient solutions for the fuel cycle of a future reactor. Principle considerations should therefore start from the elementary functions, a fusion fuel cycle has to provide:
(a) provision of the fuel to the plasma; (b) provision of fuel-type gases to NBI; (c) provision of additional plasma control (torus exhaust pumping, ELM pacing); (d) tritium accountancy and gas analysis measurement for tritium inventory determination; (e) fusion ash and power exhaust via divertor; (f) vacuum pumping of exhaust gas from torus and NBI; (g) Exhaust gas cleaning and processing as well as fuel recovery; (h) removal and recovery of tritium from the breeding blanket extraction system.
It is noted that some functions which are integrated in the ITER fuel cycle (such as provision of non fuel-type gases for impurity seeding or disruption mitigation) are not listed above, because they are not considered as primary fuel cycle functions, i.e. they do not have necessarily to be combined with the fuel cycle.
In the following sections, this paper reviews the current state of technology and identifies the areas considered to require essential supporting R&D towards a functional and technically ready fuel cycle system for a reactor.

4. Critical fuel cycle areas

4.1. Integrated modeling of fuelling and pumping systems

The design of today´s fuel cycle systems was developed in separate individual efforts and, up to now, the design process lacks integral modelling tools [6]. It is one of the major goals to develop these science-based predictive modelling capabilities for a next-stage device.
Divertor operation is conditioned by plasma physics dynamics through the divertor configuration and divertor capabilities in terms of power handling. Impurity seeding is required to obtain partial detachment in the outer divertor to keep peak power loads in ITER to values < 10 MW/m^2 but the spatial radiation distribution depends on the ratio of the loss power flow into the scrape-off layer to the L-mode power threshold $P_{Loss}/P_{H\text{-}L}$ (which is around 2 in ITER). Thus, we have predominantly edge and divertor radiation but around a power ratio of 4 in a reactor with significant core radiation. In AUG, full detachment from the inner divertor leg is observed in between ELMs, however D puff, degrading confinement, is required for detachment from the outer leg. Impurity seeding is therefore considered which for core radiation could induce a H-L transition, density peaking and impurity accumulation. Good fuelling and seeding capabilities for core, edge, and divertor are needed but related to that is a strong pumping capability necessary not only for He ash removal but for fuel and seeded particles density control. For vacuum systems, flows of mixtures of particles in ranges from viscous to free molecular (Knudsen numbers from 10^{-2} to 10^{+2}) require a consistent description [7, 8] to get a reliable unified flow dynamic solution. These calculations are important to validate the vacuum pump calculation tools such as MOVAK3D, ProVac3D or ITERVAC which have been developed for the non-commercial vacuum pumps of ITER [9-11]. Further efforts in flow modelling inside complex structures will ease the reactor grade pump development.

4.2. Tritium plant systems

The ITER tritium plant is basically a chemical factory used as a service unit. The operational data are essential to the demonstration reactor tritium plant (factor 4-5 in throughput), but the database will not be parametric and therefore of limited use with regard to scale-up in terms of higher throughputs and higher duty factors. Moreover, the basic batchwise operation of the tritium plant processes seems to be problematic in view of a steady-state operated tokamak (problem of inventory minimisation). An additional problem with a simply scaled up ITER fuel cycle would be the very large response times in the basically batchwise operated tritium plant sub-systems.
Above all that, as outlined below, it is suggested to develop a new tritium plant concept which is much better integrated in the overall fuel cycle.

4.3. Blanket extraction systems

A demonstration fusion reactor differs from ITER by the condition of tritium self-sufficiency, adding constraints on the tokamak wall structure to accommodate blanket modules, which may affect the plasma edge physics (e.g. test blanket module ripple effect) and therefore also the plasma performance. Integration of the test blanket modules in scenarios is also important in terms of tritium control as they participate to tritium migration and containment. It is obvious that with extrapolation factors of the order of 10^3 to 10^5 from ITER to a demonstration reactor, significant complementary R&D and design efforts are needed [12]. The tritium extraction system is a most challenging core issue for a power reactor.

4.4. Fuelling systems

The fuelling systems of ITER take over functions of fuel supply and plasma control (disruption mitigation and ELM pacing). Fuelling in ITER relies on pellets at relatively low velocities, injected from the high field side, and on the ∇B-drift to increase core fuelling efficiency [13], whereas for a larger fusion chamber, deep fuelling is under discussion. Expected gains are (1) a better discharge control through a full decoupling between the fuelling and ELM pacing functions, (2) a lower DT throughput due to the direct feeding of the plasma core which would reduce the part of the injected material expelled out of the discharge by the ELMs and (3) the possibility to partly monitor the isotopic ratio profile in the plasma by using preferentially T-pellets and D-puffs. From the physical point of view, experiments on present day tokamaks are mandatory because the design of ITER does not allow high speed pellet injection. Points to investigate are – among others - the maximum acceptable pellet particle content, the optimum deposition radius and the possible triggering of NTMs. From an operational point of view, the plasma control function is of key importance in view of the requested availability for a commercial power plant. For massive gas injection for disruption mitigation there results a direct interlink to the torus exhaust vacuum system, which has to be properly considered.

Consequently, there remains the issue of proper scale-up. Injection velocities up to ~10÷15 km/s are required for a deep deposition (ρ/a ~0.7) in a reactor grade machine and a thorough optimization of the injection line is required since the resultant deposition profile is expected to be strongly dependent on the injection point. An example is displayed in FIG. 2, which shows the ablation and deposition profiles calculated with the HPI2 code [14] for two L = D = 5 mm cylindrical pellets injected at 5 km/s (HFS injection) and 10 km/s (VHFS injection) in an ITER reference scenario plasma: for a given deposition radius, each displacement of the injection point from the HFS to the top of the chamber is paid by a significant increase of the required injection velocity. In spite of a new acceleration concept recently proposed by Parks and Perkins [15], only little efforts were spent for R&D of high velocity pellet injectors (see, e.g. [16] in what concerns the possibility offered by electromagnetic acceleration). On the long term, the possibility of fuelling through the injection of compact toroids will be discussed [17].

4.5. Vacuum pumping systems

Primary Pumping Systems: The vacuum technology for ITER is very much customized and a direct scale-up towards DEMO is problematic [18]. A future torus exhaust pumping technology also has to include an embedded control function for gas throughputs (especially He ash recycle flows) to assist the plasma control system with improved density control.

The ITER baseline technology for NBI and torus pumping is cryosorption on activated charcoal, which is working at higher temperatures than cryocondensation, but still provides a strong load to the cryoplant (on the order of ~ 100 W @ 4.5 K and ~ 1 kW @ 80 K per torus pump) [19]. If helium has to be pumped (which is the case for the torus system), the temperature may not exceed 5 K; for NBI pumping, where only hydrogen gas species appear, a higher temperature (up to 15 K) is sufficient. The primary reason to use cryopumping was given by the need to pump very large gas flows during burn, necessitating very high pumping speeds in the few 100 m^3/s range, which can only be provided by cropumping. The total neutral gas pressures at the divertor of the order 1 to 10 Pa are below what is currently achievable with mechanical pumping. It must be noted that the divertor cassettes and their openings through which the gas has to be pumped out establish significant conductance losses, so that only 25 to 50% (depending on the machine operation scenario) of the connected pumping speed is effectively available at the divertor system.

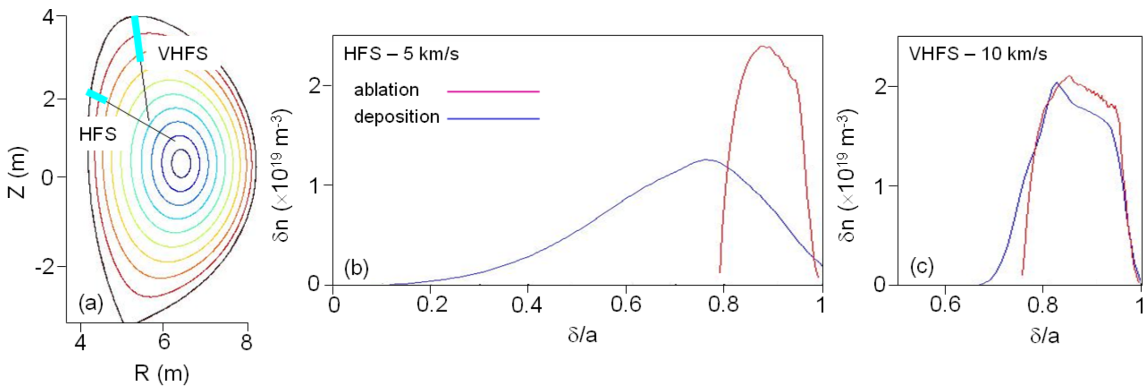

FIG. 2. (a) Geometry of injection for two L =D = 5 mm pellets injected in a reference scenario plasma in ITER; (b) ablation and deposition profile for a 5 km/s injection from the HFS; (c) identical to (b) for a 10 km/s VHFS injection.

The cryosorption technology development was focused on co-pumping of all gases and the achievement of stable operation at 4.5 K that is the temperature level at ITER for the magnets. With regard to a steady-state device, the continuous energy consumption associated with cryopump regeneration becomes a non-negligible point. All the more, if the reactor will be based on high temperature superconductor technology for the magnets, which might leave the torus cryopumps as only clients for a dedicated cryoplant at 5 K. Hence, several alternative concepts are currently discussed:

a) Development of a cold turbopump, which provides increased pumping speeds because it is able to accept cold and therewith correspondingly denser gas at the inlet [20]. The problem of eddy currents in the magnetic fields and associated self heating effects as well as the maintenance requirements have to be addressed by magnetic shielding and new bearing concepts.
b) Development of a cryosorbent for cryogenic pumping that is able to pump helium (and hydrogens) at temperatures above 5 K. The currently preferred high-performance charcoal used for ITER (see FIG. 3, left) would have to be replaced by novel materials, carbon nano structures are a promising candidate. A good temperature level would be 20 K as this would also allow to get rid of helium as cryogen, which will be very limited and become very costly in the next half century [21].
c) Separation of the pumping functions for burn and dwell, which asks to develop a mechanical pump that is able to provide the requested ultimate pressures at the high pumping speeds needed for burn at divertor pressures between 1 and 10 Pa, and limit the use of a cryopump to the dwell period, at which ultrahigh vacuum at pressures of the order of 10^{-5} Pa has to be provided in the torus chamber. Due to the low gas flow during dwell (only coming from outgassing of the walls), the cryopumps could be considerably smaller.
d) Re-visit the use of metallic superpermeable membranes and associated atomizer units for pumping, a promising option is Nb membranes with Ta atomizers [22].

The concepts above were delineated primarily for the torus pumping system, as it is likely that the NBI will be based on completely new technologies (laser-based) which may not require the handling of large gas flows anymore.

Roughing pump systems: A demonstration reactor will need a continuous, tritium-compatible roughing pump system. This area will not benefit from lessons learned at ITER which will have a very specific solution based on a viscous cryogenic compressor [23]. Although it is an absolute pre-requisite for any operation beyond ITER, the development of tritium compatible roughing pumps with minimized ultimate pressures at full speed has not really started.

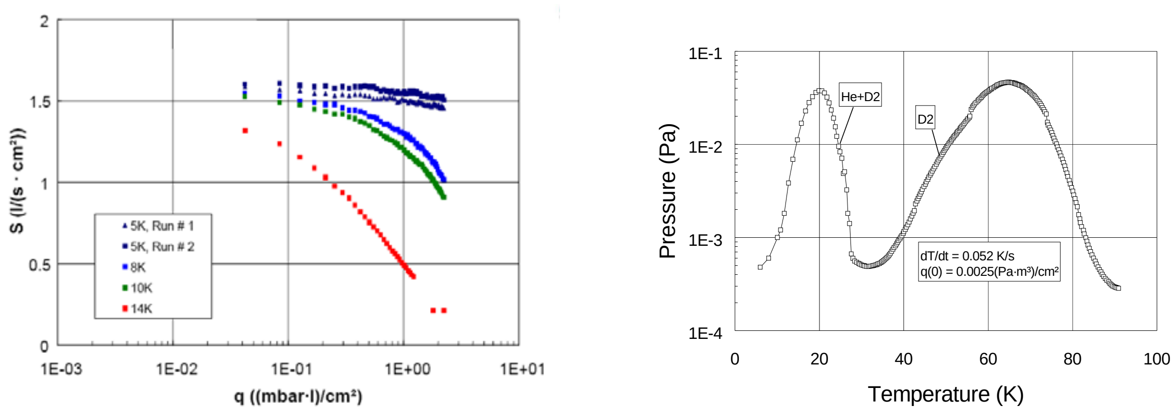

FIG. 3. Examples showing the best performances possible with existing cryopump technology [25, 26]. Sorbents have to be found for which the pumping speed curves (left, mixture 90%D_2+10%He) are shifted to higher temperatures, and the helium separation during regeneration (right) to become sharper.

For the tritium operation, every element of the pump has to meet the special requirements, in particular concerning leak tightness and the double containment to the environment, the structural materials should have low hydrogen permeability, the gas must not be contaminated with oil and it cannot contact with any organic/polymer material. As a rotary seal between pumping and lubricant containing compartments of the pump, a ferrofluidic seal has been investigated in the past [24] but this European R&D programme has been stopped due to the fact that the roughing pumps have become a US procurement package for ITER.

5. Aspects of an advanced inner fuel cycle scheme

One primary design driver for the ITER tritium plant systems has always been the throughput coming from the torus and NBI cryopump regeneration. The major species in this gas flow are hydrogen isotopologues, but the processing of the total flowrate was deemed necessary because ITER requires very high purity of the injected gas. If this requirement could be relaxed, one could think of an advanced fuel cycle scheme, in which the hydrogen fractions are directly recycled (in ´uncleaned´ form) from below the divertor up to the fuelling systems. This would drastically reduce the total throughput sent to the tritium systems.
A pre-requisite for such a system would be the installation of a simple separation stage, which would separate the hydrogens (for recycle) from the other gases, especially helium. This would be possible by adopting existing membrane technology for torus exhaust processing. But for this, the gas to be separated must be present at pressures about ambient to have a good efficiency, so that a powerful mechanical pump is needed in any case upstream the membrane (which unfortunately means to include all problems associated with handling the high total throughput). However, this new requirement could also be dealt with in conjunction with the alternative pumping concepts discussed above. In this context, a cryopump becomes again very attractive, as it is known that it can be perfectly used for separation of the pumped gases if the heating during regeneration is being performed properly, see FIG. 3 (right). Also, the superpermeable membranes can provide simultaneous compression and separation. The minimum pressures at which the so produced hydrogen streams have to be made available depends on the input requirements of the fuelling systems, but should be fine at least for any cryogenic pellet injection systems.

6. Conclusions, need for future developments

For the tritium technologies, the present concepts for tritium extraction have to be improved towards reliability and simplicity of the process, minimised tritium inventory, and facilitated accountancy. In parallel, additional efforts to develop and demonstrate advanced spectroscopic methods for on-line tritium measurements are mandatory to ensure safe and reliable operation of the machine.

For pumping purposes, a viable alternative to the cryopump for burn operation with reduced operating costs and equivalent availability should be demonstrated using a multi-stage tritium-compatible mechanical pump with reduced ultimate pressures. For dwell operation, a sorbent must be identified and validated which works at the expected temperatures of high temperature superconductors.

An integral approach which interlinks fuelling and pumping systems with tritium systems and plasma physics must be developed. The elaboration of predictive tools for neutral gas flows including recent computational algorithms for vacuum gas dynamics is mandatory to end up with a sound design for all fuel cycle sub-systems impacting the plasma operation. This would reduce the needs for future large scale validation experiments.

On a longer-term perspective it may be necessary to develop a novel, much better interlinked pumping and tritium system as is the case for ITER; for example based on internal recycling of the unburnt fuel fractions upstream of the torus vacuum pumps, thus resulting in significantly reduced throughputs for the tritium plant and vacuum pumping systems.

Acknowledgements

This work, supported by the European Communities under the contract of Association EURATOM/KIT and EURATOM/CEA was carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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