The ITER Vacuum Systems

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Abstract. ITER is a large vacuum facility which comprises many service, diagnostic and monitoring vacuum sub-systems as well as three large cryogenic pumping systems for evacuation and maintenance of the required pressure levels. Control of the gas throughput is one of the key issues affecting the performance and achievable burn time of a fusion reactor. The main pumping systems are the torus exhaust pumping, the cryopumps for the neutral beam injection systems for plasma heating, and the cryopumps for the ITER cryostat. All customized cryosorption pumps are force-cooled with supercritical helium and share a similar modular design of cryosorption pumping panels. For regeneration of the cryopumps as well as for roughing down the system volumes prior to operation, four identical sets of forepump trains are used. This paper will focus on the areas of the ITER vacuum systems which require customized developments and cannot rely on commercial solutions. The complex pumps have been tailored for the very specific applications and requirements at ITER, especially characterised by the need to be tritium compatible. An outline of the development path which was needed to come up with a sound design for the ITER cryopumps is given. The way of development is culminating in the manufacturing of 1:1 scale prototypes, which will be extensively tested in dedicated test facilities to ensure compatibility with all design requirements.

1.Introduction

ITER is the next generation thermonuclear fusion device of tokamak type, and represents the experimental step between today's studies of plasma physics and tomorrow's electricity-producing fusion power plants [1]. It is a key effort to demonstrate that nuclear fusion power is a viable energy source on a scale unprecedented on present day fusion devices. ITER is a joint international research and development; the partners in the project - the ITER Parties - are the European Union (represented by EURATOM), Japan, the People´s Republic of China, India, the Republic of Korea, the Russian Federation and the USA. ITER will be constructed in Europe, at Cadarache in the South of France and the first 1 MA plasma shot is scheduled for 2016.

It is based on a deuterium–tritium (DT) plasma operating at several million K, and will produce 500 MW of fusion power during pulses lasting up to one hour. This will put massive gas loads on its vacuum systems which will also be exposed to radiation and magnetic fields. Compared with the size

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of the JET fusion device in Culham, UK [2] which is the only one of the existing machines having been routinely operated with tritium during dedicated campaigns, the torus volume of ITER is one order of magnitude larger. The DT throughput of one day of ITER operation is approximately equivalent to the amount of gas processed during the whole main DT campaign at JET. While the base pressures of the order of $\sim 10^{-5}$ Pa are similar to those of existing fusion facilities, the encountered gas loads are several orders of magnitude higher, making vacuum gas flows a crucial problem. The use of tritium excludes almost all commercially available pumping techniques and the special ones being compatible with the requirement to pump tritium can only be manufactured at about two orders of magnitude lower pumping speeds than actually needed. As a result, new vacuum technologies are being developed for ITER.

Control of the gas throughput, especially 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. This is implemented via a closed fuel cycle, see figure 1 [3]. 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 and pellet injection) and to neutral beam heating. The 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 isotopomers that are transferred to the isotope separation system, while the remaining detritiated waste gas is sent to a detritiation system for decontamination purposes before release into the environment. The inner fuel loop is closed by the return of deuterium and tritium to the storage.
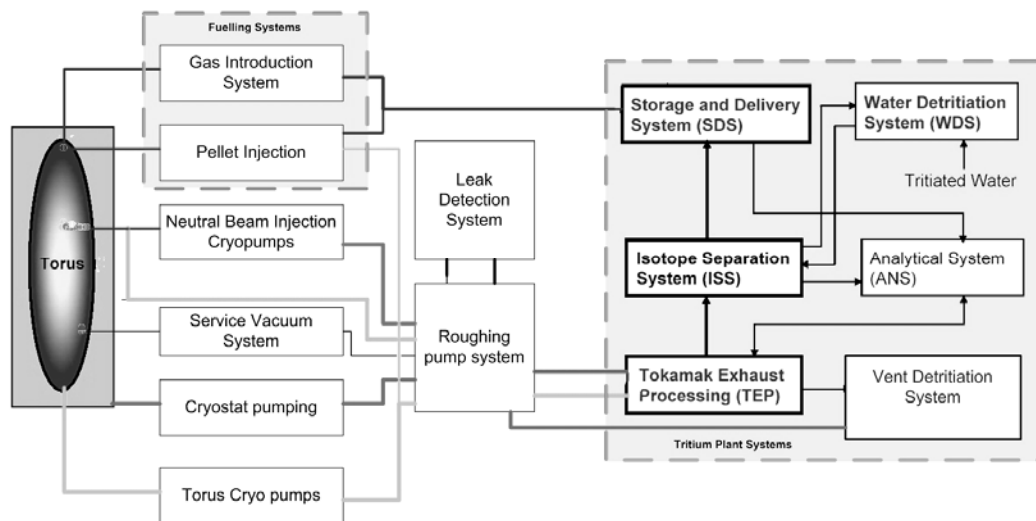

Figure 1. Schematic of the ITER fuel cycle.

To maintain the essential vacuum conditions and to provide a good confinement, the plasma chamber is equipped with a pumped divertor, through which the exhaust gas is extracted. As the burn-up fraction is relatively small (of the order of 2%), very high gas throughputs have to be processed, considerably more than actually needed from the pure fusion reaction point of view. This duty is fulfilled via a powerful, high-speed cryopumping system, which also provides for the necessary base vacuum in between the pulses. Other cryopumping systems can be found in the neutral beam injectors, which are high power auxiliary heating systems to heat the gas up to plasma conditions. The third very large cryopumping system is the pumping of the cryostat which houses the plasma chamber and the superconducting coil system in between. Finally, there are numerous sets of cryocooler refrigerator cryopumps used for the service and auxiliary vacuum systems.

Finally, there are roughing pump systems to rough the above mentioned volumes down to the cross-over pressure of the order of 10 Pa at which the cryogenic pumps can be switched on. For this purpose, they will be connected via a ringline directly to the chamber. Once the cryopump systems are under operation, the roughing pumps also have to take over the function as forepumps needed for the regeneration of the cryopumps. For this operation, they will be connected directly to the exhaust side of the cryopumps, via a second ringline system. Consequently, the cryogenic and mechanical vacuum pumping systems are closely interconnected. The same type of mechanical pumps is also used for the pellet injection system, a fuelling system which introduces frozen fuel pellets at high speeds into the plasma.

Last not least, due to its unique complexity, ITER requires the development of very challenging leak detection and localisation systems.

This paper gives an overview of the different systems, it summarizes the requirements and shows how the specific fusion relevant issues are being solved. The presence - actual or potential - of tritiated gas species exerts a strong influence on the technologies which can be deployed in ITER vacuum systems. The outgassing rates from plasma facing components and the gas species evolved are strongly dependent on the selection of plasma facing materials; the present design of ITER includes beryllium, tungsten and carbon fibre composites. However, to mitigate the operational issues associated with carbon plasma interaction, the carbon tiles are planned to be exchanged prior to the start of tritium operation.

2. Cryopump development

2.1. Common features of all the large cryopump systems

The typical vacua required for a fusion device during burn conditions are quite moderate (in the Pa range). Thus, the rationale to use cryogenic pumping is not given by the need to establish lowest pressures, but by the need to process very high gas throughputs necessitating highest pumping speeds. Furthermore, cryopumping does not introduce any problem associated with moving parts, it can be built without organic materials and is therefore inherently tritium-compatible, and can benefit from the cryosupply which is anyway existing to cool the superconducting magnets.

The cryopump programme ongoing at Forschungszentrum Karlsruhe (FZK) for the last 20 years covers all aspects of the development, from investigation of candidate pumping concepts through characterization of small scale cryopanel coupons to characterising full scale pumps. Designs of the cryopumps for the ITER vacuum systems have been standardized to the greatest extent practical. A focussed programme to develop a robust, tritium-compatible cryosorption panel concept has been successfully accomplished and used in the production of cryopanels for ITER and other fusion cryopumping applications.

The ITER machine includes three large cryogenic high vacuum pumping systems. One is for evacuation and maintenance of the required pressure levels in the torus (~ 1350 m³); the second is for generation of the required vacuum conditions in the neutral beam injectors (NBI) (~ 160 m³/heating injector), which are used to heat up the plasma by injection of highly energetic accelerated hydrogen; and the third is for providing the insulation vacuum in the cryostat (~ 8400 m³), which houses the superconducting coil system.

Design concepts for these have been developed according to the following guidelines:

- Cryopumps are force-cooled from a supply which is integrated with the ITER superconducting magnets; this is not based on liquid cryogens but provides one-phase supercritical helium at 4.5 K (at 4 bar) and gaseous helium at 80 K (at 18 bar);
- Cryopumping system regeneration schemes (at different temperatures in the range 80-475 K for the different gas species) are based on sequential regeneration of individual pumps to minimise gas inventories (all hydrogen isotopomers for explosion safety, additionally tritium for administrative inventory limitation) and smooth demands on the cryodistribution system;

- Torus exhaust and cryostat pumps shall have the same configuration (to be confirmed in an ongoing study);
- The similar cryosorption concept is used in all systems;
- Most ITER vacuum pumping systems have to be designed for tritium compatibility.

For the torus exhaust and the cryostat cryopumps, the supercritical helium loop return manifold temperature is specified to be maximum 4.7 K, which corresponds to solid-vapour equilibrium pressures of about 10^{-3} Pa (for H_2) and 10^{-9} Pa (for T_2) [4]. As a rule of thumb, an oversaturation by two decades in pressure is normally chosen to provide high pumping speeds. This results in the fact that temperatures below the available 4.5 K would be needed to pump H_2, HD and HT by re-sublimation; helium cannot be pumped at all.

Consequently, the cryopumps are based on cryosorption (sorption at 4.5 K for light gases and re-sublimation for other gases). The pumping surfaces are characterised by a modular design of identical hydroformed cryopanels, each of which is coated with a cryosorbent. The attainable equilibrium pressure of adsorbed gas particles is significantly lower than the corresponding vapour-liquid or vapour-solid saturation pressure. Hence, gas can be retained by adsorption even in a subsaturated state, i.e. at considerable higher temperatures than would be required for condensation. Consequently, by using a cryosorbent, the pumping characteristic of the cryopump becomes much less dependent on any oscillation of the cryogen supply temperature.

Extensive preparatory tests were run in the past to find an optimum combination of sorbent type and bonding cement [5]. More than 400 cryosorption specimens (50 mm diam.) were prepared using various sorbents (activated carbons, molecular sieves, sintered metal, porous ceramics, metal fibre fleece), bondings (solders, plasma sprayed layers, inorganic cements, mechanical techniques), substrates (copper, aluminium, stainless steel) and combinatuions thereof. A spraying technique was developed for the coating of large surfaces with cryosorption layers. As a result of these tests, activated charcoals bonded by cement and braze and molecular sieves bonded by cement were chosen for the fabrication of 430 mm diameter panels for the next stage of development in which approximately ten liquid helium cooled cryosorption panels (including one based on argon frost cryotrapping) were tested in a vertical vacuum vessel [6]. Resulting from these tests the reference cryosorption panel set-up was found and finally adopted for all main cryopumping systems in ITER. It is a ~ 1 mm thick layer of specific granular activated charcoal, bonded by an inorganic tritium-compatible cement. Based on this reference panel set-up, an essential qualification programme has been performed at FZK to determine the performance characteristics and to assess the suitability and effectiveness of the quilted cryosorption panel design [7, 8]. A facility for application of calibrated layers of bonding agent and sorbent to panel substrate has been constructed at FZK. For reasons of design simplification by maximum commonality, this concept is also used for the NBI cryopumps [9]; it would not be absolutely needed as no helium has to be pumped, but provides additional operational advantages to a cryocondensation pump.

2.2. The torus exhaust cryopumping system

2.2.1. Requirements. Being of tokamak type, ITER is a priori a pulsed machine, so that the main duty of the torus exhaust pumping system is to pump the exhaust gases during plasma burn time through the divertor slots (at divertor pressures between 1 and 10 Pa) and to prepare the requested base vacuum in the order of 10^{-4} Pa in the dwell time in between the plasma shots. Additionally, it is also needed to provide high vacuum during fine leak-testing of the torus, for wall conditioning and bake-out and to provide ultimate vacuum in the torus. The main parameters for these operations are given in table 1 .

Table 1. Torus pumping system parameters.	
Parameters	Value
Vacuum vessel free volume	~ 1350 m³
Ultimate base pressure for hydrogen isotopes/ impurities	10^{-5} / 10^{-7} Pa
Base pressure in the dwell phase	10^{-4} Pa
Available effective molecular pumping speed	> 30 m³/s
Typical divertor pressure during plasma operation	1-10 Pa
Maximum throughput during plasma operation	200 Pam³/s

This means that a broad spectrum of gases, such as all the six hydrogen isotopomers, noble gases and low- and high-molecular impurities (air-likes, water, hydrocarbons) will have to be pumped at variable throughputs, ranging from very small to very high. Due to this wide range of requirements, the torus exhaust pumping system was used as the reference pumping system on which the design efforts were concentrated. The other systems were conceptuated to follow the torus pump design as much as possible and to benefit from the underlying R&D work.

The reference design of the ITER torus exhaust pumping system is based on 8 cryopumps, connected via 4 ducts to the torus divertor ring where the gas to be pumped comes from, each of them containing a pump in direct line of sight and a branched pump, see figure 2. To provide a quasi constant pumping speed for long pulse conditions (pulse lengths longer than 600 s), at each moment of time half of the pumps are in pumping mode, the other half are in various stages of intra-pulse regeneration. The cycle time is 600 s. The staggering pattern of the cryopumps (at an interval of 600/4=150 s) is determined by the need to limit the complete hydrogen inventory for explosion safety reasons in general, and to stay within an administrative tritium limit of 120 g-T in the complement of pumps open to the torus [10]. The staggering pattern also helps to reduce the coolant consumption through an initial enthalpy recovery process at which the pump to be cooled down is done so by the cold gas leaving another pump to be regenerated at the initial phase of regeneration.
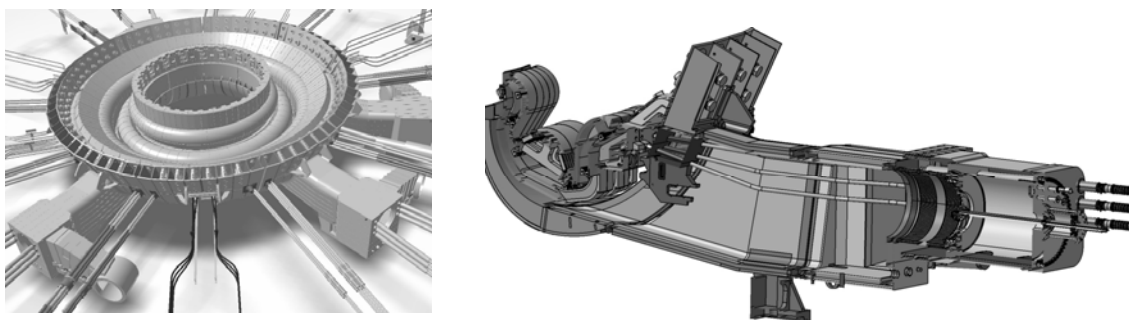

Figure 2. *Left:* Top view on the lower port plane showing the torus exhaust pumping ducts (which feature a direct and a branched pump). *Right:* Axial cut through one pumping port, highlighting the flow path from the divertor cassette (left) to the pump (right).

2.2.2. Vacuum flow analysis. To achieve a sound and balanced design for the complete vacuum system and to study the influential parameters, simulation calculations were performed for the gas flow through the divertor slots and along the pump ducts into the cryopumps. A significant obstacle in providing sufficient pumping speed at the torus is the low conductance of the divertor and pumping

duct system. Modelling of the flow through the complex divertor geometry is complicated by the fact that the flow conditions are in transition regime where comparison of existing models exhibit large discrepancies.

To investigate this critical regime, the ITERVAC code has been developed [11] which covers all flow regimes from laminar flow (Knudsen numbers of about 0.05) at the divertor region, transitional in the duct to molecular flow (Knudsen numbers of about 200) in the cryopump. The results of analyses of the system conductance are currently benchmarked against experimental data to be obtained from the ITERVAC test loop operated at FZK. The ITERVAC calculations identified a clear bottleneck in the vacuum system given by the limited pumping paths in the connection to the divertor, resulting in a pronounced throughput limitation [12]. Therefore, there exists an optimum pumping speed of the individual cryopump from which a further increase does not add significantly to the overall performance. This resulted the target design pumping speed for the individual cryopump to be in the order of 80 m³/s. Based on this number, the detailed design of the torus cryopump was developed. A further speciality of the ITER cryopumps is the inclusion of an inlet valve, to control the throughput and to isolate the pump from the torus during regeneration. The inlet valve adds an additional degree of freedom to the pump control. Several design variants have been studied in detail [12].

2.2.3. The full size prototype torus cryopump. Following the panel development programme, a model pump with 4 m² of cryosorption surface, approximately 50% of the (then) ITER torus exhaust pump area, was designed, fabricated and installed in the dedicated TIMO test facility (Test of ITER model pump) at FZK. The tests, which extended over several years included performance tests (pumping speed, capacity, impurity (poisoning effects) and regeneration tests), safety tests (loss of vacuum, sudden venting) and mechanical tests (cycling tests and post operational inspection following dismantling of the pump [13, 14].

Tritium compatibility was demonstrated in separate tests of a representative cryopanel arrangement at JET, which was also used at the last tritium campaign 2003 [15]. Pumping of tokamak exhaust gas was successfully completed; post-service inspection, including residual tritium measurements, are in progress.

EFDA, the European Fusion Development Agreement for collective activities in fusion, in collaboration with FZK and EU industry is presently building a full scale Prototype Torus Cryopump (PTC) [16] under consideration of ITER codes and standards as well as the full tritium design rule catalogue. The schematic design of the PTC is shown in figure 3. It differs from the ITER series torus exhaust cryopumps only in a few minor details to facilitate its accommodation in the existing TIMO test vessel, but these differences will not compromise the relevance of the test results.
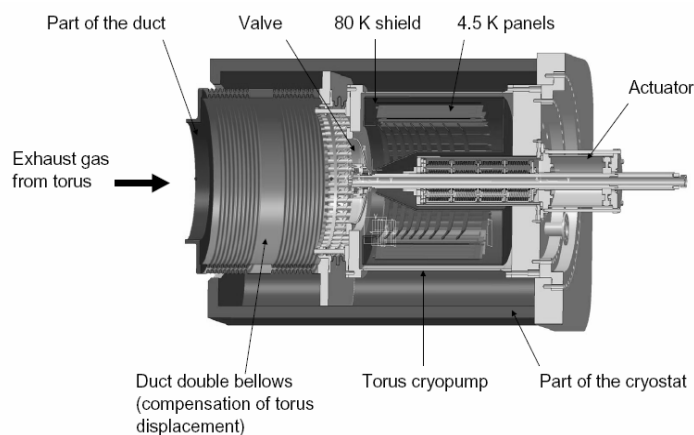

Figure 3. Cross-section of the PTC (1.8 m long, 1.6 m diameter).

2.3. The neutral beam cryopumping system

ITER will be provided with two (optionally three) heating and one diagnostic Neutral Beam Injectors located on the equatorial plane of the tokamak, fuelled by D_2 or H_2. The task of the NBI cryopumps is to maintain low pressure in the injector outside the ion source and the neutralizer, see figure 4. The required integral pumping speeds for the heating injectors are 3800 m³/s for H_2, and 2600 m³/s for D_2, respectively.
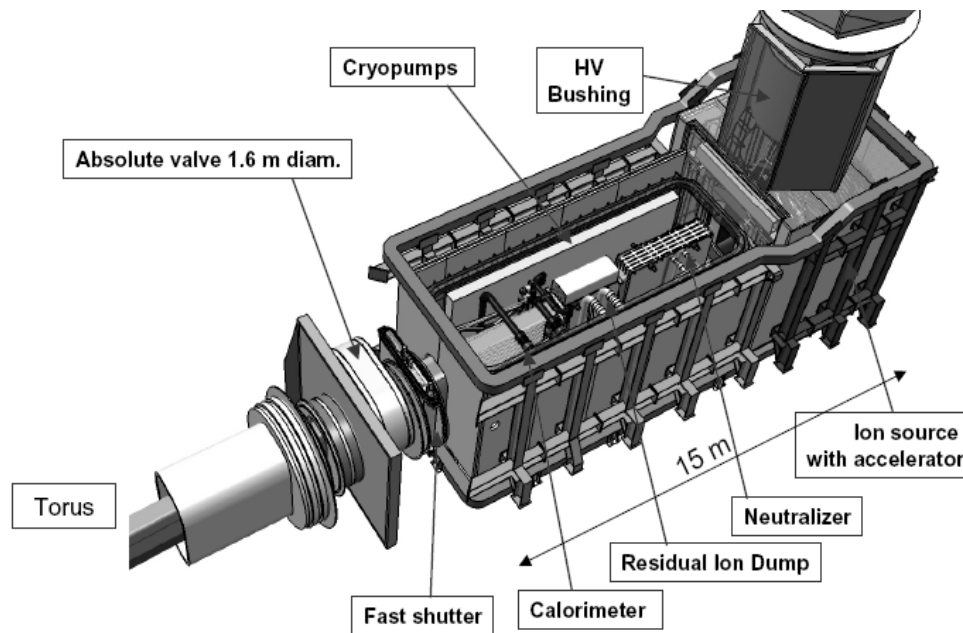

Figure 4. Overview of the heating NBI injection line (cryopumps are 8 m long, 2.8 m high).

The cryopumps are of rectangular shape, the current design has flat, vertical panel arrays arranged on the sides of the NBI boxes, to permit vertical removal of components for maintenance. The beamline includes distributed gas sources, which, together with the cryopumps leads to strong density profiles inside the NBI vessel. The pressure between the accelerator and the neutralizer must be better than 0.02 Pa to reduce heat loads. To establish the needed density profile along the beamline of the neutral beam components two gas baffles will have to be integrated to separate the chambers. The NBI system is most challenging to design with regard to the limited space available and the pumping speed needed; moreover, it is characterized by strong temperature gradients (cold cryopump surfaces vs. hot (100 °C) beamline components). For density profile calculations, a dedicated software code has been developed at FZK, named ProVac3D. The theoretical basis and another application example are given in [17].

Finally, an optimisation had to be performed to come up with a pumping system with highest capture coefficients. The final design is still under work, but it is obvious that it results as a compromise between the need for open structures to have high capture coefficients and the need for closed structures to minimize heat loads on the 4.5K system. A typical example for this situation is the choice of a chevron-like design for the baffles and shields to significantly reduce thermal transmission, which has to be paid for by a distinct loss in the molecular transmission probability. Figure 5 is exemplifying a candidate NBI cryopump design. The details result from comprehensive test particle Monte Carlo optimisation calculations. The illustrated design provides a capture coefficient of 35% (which is 60% higher than for a ´classical´ cryopump design, as for example in [18]) and heat loads (related to the projected pump opening cross-section) of 140 W/m² on the 80 K system and 4 W/m² on the 4.5 K system. The latter value depends significantly on the choice of which surfaces are to be coated with

activated charcoal, which, by itself is given by the requested integrated gas amounts to be pumped (integrated throughputs over time). The detailed design of the NBI cryopumps is under way. They will be manufactured, installed and operated in the planned Neutral Beam Test Facility, a side facility of ITER; details are given in [19].
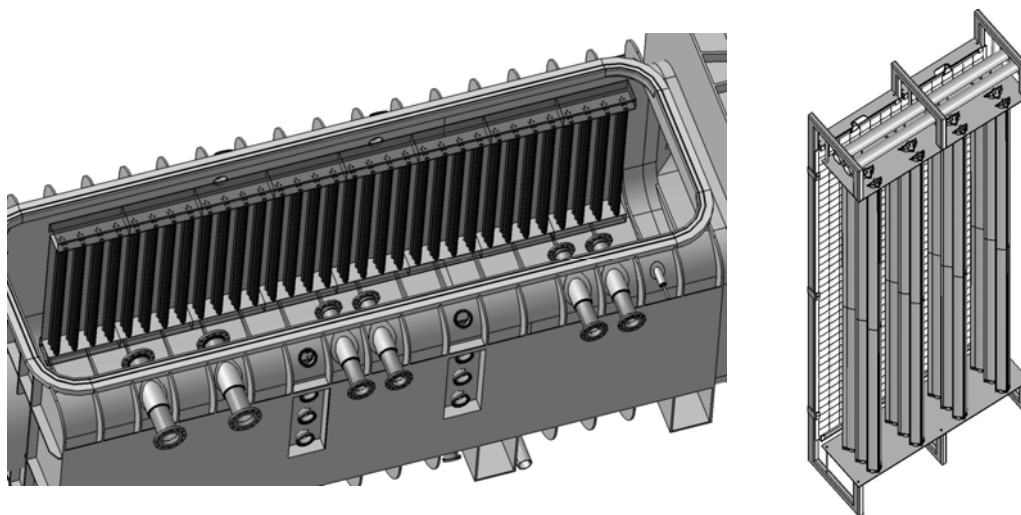

Figure 5. A candidate optimised NBI cryopump design, featuring 8 identical pump modules (as shown on the right) per NBI vessel side.

2.4. The cryostat cryopumps
The cryostat pumps fulfil the following functions:

- Dehydration of the cryostat internals prior to magnet cool-down;
- Evacuation of cryostat to high vacuum (10^{-4} Pa range) prior to magnet cool-down;
- Pumping of helium leaks from magnet cooling circuits;
- Pumping of H_2 from long term outgassing from warm in-cryostat metallic surfaces;
- Pumping and detection of external air leaks to reduce ozone hazard;
- Pumping of gases generated by irradiation of exposed magnet coil epoxy.

The panel arrays and inlet valve/actuators of the two cryostat cryopumps are expected to be identical to the torus cryopump, if confirmed so in an ongoing design study. However, as these pumps are installed in the cryostat volume rather than ducts (as in the case of the torus pumps) they will embody a casing to provide inlet valve seat & private regeneration volume, which can be pumped out by the roughing system.

2.5. The cryorefrigerator cryopumps
ITER features many service vacuum systems which are normally equipped with cryorefrigerator pumps. These will be based on commercial devices, but the ITER reference cryosorption panel set-up will have to be included to make them tritium compatible to account for leaks/permeation from adjacent tritium bearing components. There are different cryocooler types under discussion, but they will all be distributed (i.e. installed in a ring manifold to which individual clients are connected), see figure 6. Their main functions are:
- Providing vacuum roughing, leak detection and guard vacuum where needed by clients which are not tritium bearing during normal operation (including diagnostic systems);
- Pumping interspaces between feedthroughs, bellows and windows, deemed sufficiently fragile to need a second vacuum barrier to enable differential pumping, to allow leak mitigation and

continued plasma operation if the reliability of a single feed-through would otherwise not be adequate to meet the overall availability requirements of the tokamak or would result in excessive individual and collective maintenance worker dose;

- Provision of controlled venting of the torus, cryostat and all other vacuum systems;
- Conveying all gases from the service vacuum system back to the tritium plant.
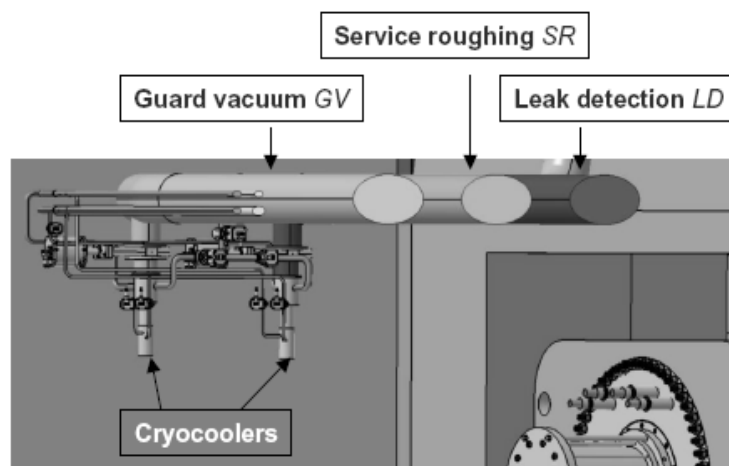

Figure 6. A typical cryocooler vacuum distributor to take over different functions.

3. Mechanical pumping systems

3.1. Roughing pumps

The first main function of the mechanical roughing pump system is to perform the initial pump-down of the vacuum vessel, NBI and cryostat volumes from ambient pressure. The second main function of the rough pumping system is the regeneration of the various cryopump systems, which are tritiated to a different degree (torus cryopumps highly tritiated; NBI cryopumps moderately tritiated; cryostat cryopumps non tritiated, service vacuum system slightly tritiated). Each system has its own forepump train to fulfil this task. The cryopump regeneration represents a cyclic load to the mechanical pumps. The regeneration pump-down of the cryopump volume starts after all cryopumped hydrogen and helium is released (at 90 to 100K) at pressures typically in the several kPa range. The cross-over pressure which has to be provided in the cryopump volume is generally defined as 10 Pa.

All four trains are located in the vacuum pump room at a distance of about 40 m away from the torus, connected via 300 mm diameter pipes to reduce conductance limitation effects, especially at the high vacuum side of the pumping process. They are identical for maximum flexibility. Complex manifolding and changeover valves are foreseen to facilitate these operations, and procedures for purging sections of the system to prevent cross-contamination between campaigns when they have handled gases with different tritium concentrations. The complete forepumping system has to be designed for the operation with tritium as tritium presence cannot be excluded. There are tritium compatible pumps on the market, but they are too small for the duties mentioned above. The mechanical pumps for ITER therefore have to be especially developed, this work has started on the basis of roots pumps, which are known to provide highest pumping speeds. In order to achieve a full tritium compatibility the roughing pump design for ITER has to satisfy two major requirements:

1. The leak rate to the environment must not exceed 10^{-9} Pam³/s. This requires the use of metal flange seals to the outside and stainless steel material (with low hydrogen diffusion) for the housing. There should not be shaft seals to the environment, the pump has to be equipped with a magnetic coupling or a canned motor.

2. Any contact of the process gas with organic material must be excluded (tritium substitutes the hydrogen and will lead to material embrittlement and cracking). For this purpose all elastomer seals have to be replaced by metal seals. The gas exchange between the oil containing volumes and the process volume has to be excluded. Oil traces in the tritium gas are also not allowed, because they affect the efficiency of the process units in the tritium plant. As a design target we ask for a maximum permitted leak rate along the rotor shaft of 10^{-7} Pam³/s. To satisfy this requirement, it has been proposed to replace the conventional shaft seal between the process and the lubricant containing volumes, based on labyrinth and piston rings, with a ferrofluidic shaft seal. The ferrofluid is a liquid (typically a synthetic hydrocarbon) carrying fine ferromagnetic particles. Attracted by a magnetic field gradient the liquid fills a gap around the shaft and closes it under stationary and dynamic (rotating shaft) conditions. Several ferrofluidic seals can be placed in order to sustain a pressure difference of 0.1 MPa.

The first fully tritium compatible 250 m³/h roots pump, however without the required compression ratio, has been manufactured and delivered to FZK. The ferrofluidic seal (in two different design options) performance was tested under non active gases [20] and is currently being confirmed with a tritium test loop inside a glove-box, see figure 7. If the validation is positive, a prototype of ITER size (e.g. 1000 m³/h) will be manufactured in collaboration with industry. It has been proposed to go for parallel and serial arrangements of such reasonably sized pumps equipped with variable frequency drives to achieve the full range of pumping speeds needed in the ITER roughing system, which is likely to be in the range of 4000 m³/h per train.
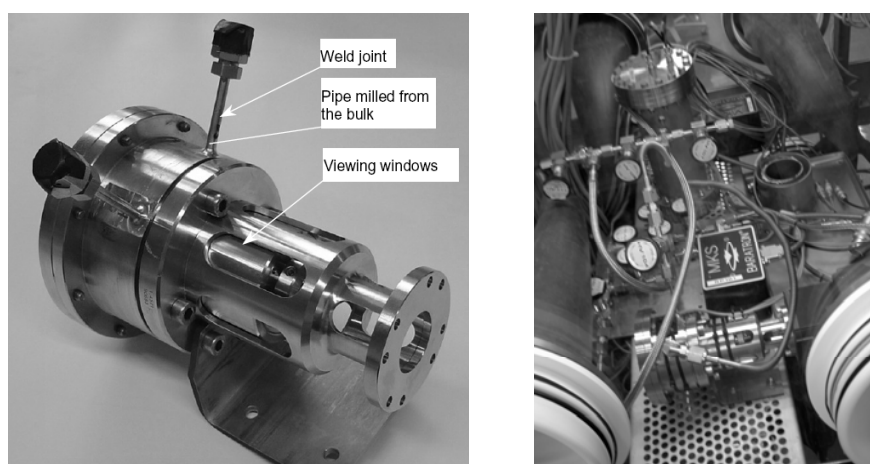

Figure 7. The ferrofluidic seal rotary unit for tritium testing.
Left: Before installation (without motor); *right:* As installed in the glove box.

3.2. Pellet injector mechanical pumping system
Pellet injection systems using frozen 3-5 mm sized pellets (D_2/DT/T_2 at a temperature of about 10 K) is the primary technique for core fueling of the ITER plasma [21]. The system will use pneumatic (gas gun) acceleration techniques to achieve the requested pellets speeds in the order of 300 m/s. The pellets are typically transported to the plasma through metal guide tubes that end at the inner plasma chamber wall for deep fueling. There they transport across the confining magnetic fields and enter the plasma where they ablate and add fuel particles to the plasma. In the pellet injection system, the scheme for recovery of propellant gas includes roots and screw pumps, with capacities broadly in line with roughing pumps. As the gases handled will be tritiated, similar upgrading of standard pumps to that proposed for the main roughing pumps is envisaged.

4. Leak detection and localization

Early leak detection and precise leak localization are essential to maximization of machine availability. Localization of water leaks to the level of individual in-vessel components is challenging due to the large number of first wall modules and divertor cassettes [22]. The detection and localization of air leaks, which will in many cases be associated with vacuum vessel ports or penetrations for plasma heating and diagnostic devices is complicated by the diversity of these systems. A sequence of procedures for leak localization is under development. This step will be followed by the deployment of an in-vessel remotely operated detection device to traverse the plasma-facing components and pinpoint the location of the leak. Strategies for leak detection and localization have been outlined and work on validation of the proposed techniques is under way.

5. Conclusions and outlook

The development programme for cryopumps has extensively qualified the design of all ITER cryopumping applications by validating a broad range of representative operating conditions.

The torus (and cryostat) vacuum pumping system is in a very advanced stage, currently manufacturing a prototype and preparing the serial pump manufacturing following the prototype results. The NBI cryopumps are in the conceptual design phase, but the test bed is already being set up to become operational for testing the 1:1 scale NBI cryopump prototype.

Principle solutions for the service and mechanical pumping have been developed. Adaptation of the roots pump concept for tritium compatibility was demonstrated with a small scale pump. Larger scale roots pumps and other types need to be investigated, but it is confidently expected that scaling will be feasible within the range of pumping speeds required.

6. Acknowledgements

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