The Fuel Cycle of Fusion Power Plants and Experimental Fusion Reactors

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Abstract

The tritium fusion fuel cycle has to be based upon well proven technology to assure the safe handling of tritium along with credible accountancy, low tritium inventory, low generation of wastes and a high reliability of all components throughout the lifetime of the reactor. A coherent understanding of all the equipment and process details covering conceptual studies to technical scale experiments together with mathematical modeling is therefore indispensable. Corresponding research is therefore on-going in various laboratories around the world. The cryo-pumping systems for Torus exhaust and neutral beam injectors have been tested to confirm fast regeneration scenarios and to reduce the accumulation of semi-permanent inventories of tritiated impurities. A comprehensive program to test the integrated three step (permeation, chemical impurity processing and isotopic exchange) process for Torus Exhaust Processing was carried out in the last decade. An integrated concept for the Isotope Separation (ISS) and Water Detritiation systems (WDS) has been developed and is under investigation on relevant scale. A fast delivery, self-assay storage bed has been developed and is under testing.

Key words: Fuel Cycle, Tritium, ITER

1. Introduction

The fuel cycle of fusion power plants mainly consists of three major systems: fuelling, vacuum pumping and tritium processing, the latter referred to as Tritium Plant.

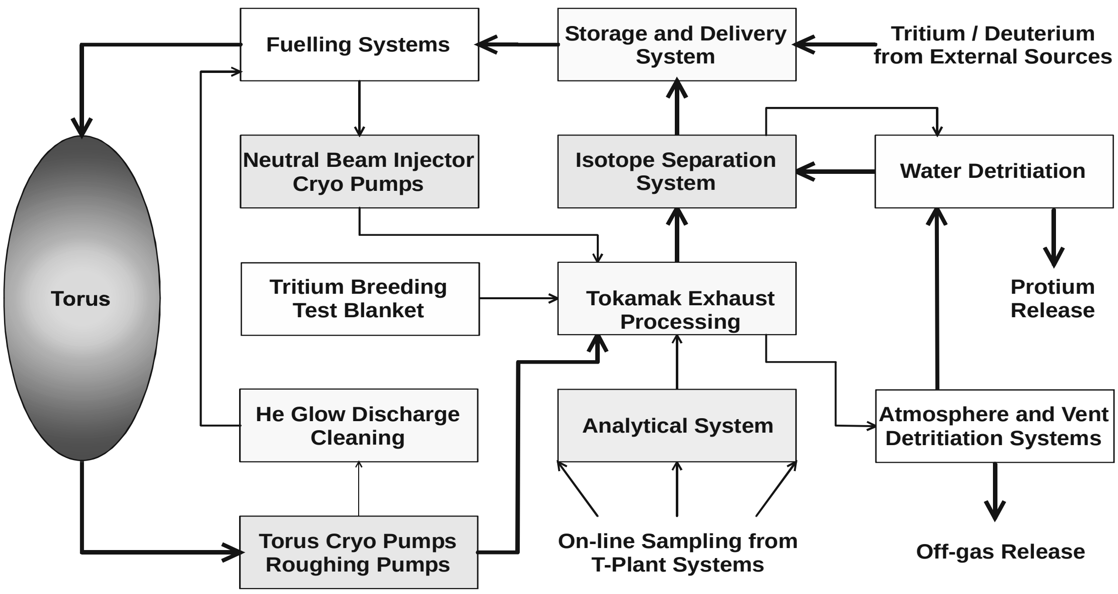

Figure 1: Block diagram of the ITER deuterium/tritium fuel cycle

The task of the fuelling systems is to supply DT fuel and other gases into the main plasma chamber and divertor at the required fuelling rates and response times. The vacuum pumping system provides high vacuum pumping of the Torus during all phases of operation. The Tritium Plant has the major functions of processing the Torus exhaust for recycling of tritium and deuterium, detritiation of impurities and water for controlled release into environment, and extraction of tritium from breeding blanket or processing of tritium from test blanket modules.

The major criteria for the development of fuel cycle systems are a low tritium inventory (along with measures for tritium accountancy), safe handling and confinement of tritium, low effluents and releases during normal and off-normal conditions or accidents, high reliability, low waste production (particularly solid waste) and an effective control of costs for both the procurement and the operation phase.

The ultimate goal of the Fuel Cycle from ITER (Figure 1) is to demonstrate the system technologies at reactor scale which allows consistent and trustworthy scale-up for DEMO. A coherent understanding of all the equipment and process details covering conceptual studies, technical scale experiments (with hydrogen and deuterium) and tritium tracer experiments together with mathematical modeling is therefore indispensable.

2. Fuelling and Neutral Beam System

Three fuelling scenarios are envisaged for ITER, i.e. short pulses with 450s fuelling, followed by 1350s dwell, intermediate pulses with 1000s fuelling followed by 3000s dwell and long pulses with 3000s fuelling and 9000s dwell.

The fuelling systems supply DT fuel and other gases into the vacuum vessel at the necessary flow rate and response time to maintain fusion power at the required level. Gas puffing provides a source near the edge of the plasma; a certain isotope profile can be achieved by firing high speed pellets of frozen hydrogen/deuterium/tritium into the plasma. When such pellets enter the plasma they ablate by the bombardment from plasma particles. Presently pellet injector systems have been successfully tested in several fusion reactors [1] and the technology is under development at Oak Ridge National Laboratory.

In combination with other heating systems (ion cyclotron, electron cyclotron, lower hybrid) the Neutral Beam (NB) system supports the plasma heating. High energy neutral particles penetrate the plasma and their energy is transferred by collisions to the plasma while the beam particles are ionized, slowed down and join the plasma. ITER comprises two heating NB injectors (with provision to add a third one), and one diagnostic NB injector system. The ITER NB system is fuelled with deuterium or protium. Each injector incorporates a large cryo-pump in order to maintain the vacuum in the ion source, accelerator and neutralizer chamber [2]. The NB cryopumps are based on cryo-panels forced-cooled by 5K supercritical helium. To guarantee sufficient pumping, the cryo-pumps are based on cryo-sorption (sorption for light and re-sublimation for heavier gases) by using cryo-panels coated with activated charcoal as adsorbent material. As the cryopumps are accumulation pumps, a regeneration procedure for the NB injector cryopumps is implemented in ITER; a staggered regeneration pattern to minimize hydrogen inventories for the NB injectors have been defined for ITER [3].

3. Vacuum pumping system

The Vacuum pumping system tasks for a fusion reactor are to evacuate the Torus prior to plasma pulses and to provide high vacuum pumping during all modes of operation (burn, dwell, cleaning). These requirements impose a high pumping speed and throughput at low

pressure, hence the pumps should be close to the Torus. At the same time, due to high magnetic fields and the presence of highly mobile dust particles, it is desirable to have pumps without moving parts. For ITER a system consisting of cryo-pumps backed by roughing pumps away from the magnetic field was chosen. A broad spectrum of gases, all the molecular hydrogen species, noble gases and low- and high-molecular impurities (air-likes, water and hydrocarbons) will have to be pumped at variable throughputs. 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.
During plasma operation, the ITER torus is pumped by eight torus cryo-pumps which are located in four divertor ducts, two pumps per duct, one branched and one in-line. The hydrogen inventory of all torus cryo-pumps open to the torus volume is subject to an administrative safety limit. During long pulse D-T plasma discharges the cryo-pumps have to be regenerated on-line to keep the hydrogen gases within these limits. This is done using a staggered operation pattern [4] in which at any instant four of the pumps (two branched and two direct) are pumping and the remaining four are in regular regeneration at 100K to release the hydrogenic species. A second regeneration stage is taking place at ambient temperatures to release air-likes and light hydrocarbons (when the plasma operation are ceased), and a high temperature reactivation stage at 450 - 470K to release any strongly adsorbed substances such as water and heavy hydrocarbons [5]. It has to be mentioned that the above mentioned vacuum pumping systems for ITER are conceived for unlimited burn time, therefore applicable to future fusion reactors, due to the philosophy of staggered pumping and regeneration operation
The ITER roughing pump system serves for initial pump down from atmospheric pressure of the torus volumes and for vacuum cryo-pumps regeneration. The system comprises four pumping units; each backing unit is a multistage arrangement of Roots pumps backed by a screw or scroll pump [6]. The different types of tritium compatible mechanical pumps currently available on the market are too small for the pumping duties of fusion power plants (ITER requires pumping speeds higher than 1 m^3s^{-1}). Development of fully tritium compatible Roots pumps is undertaken in co-operation with industry. To prevent tritium transport into the lubricant of the bearings and to avoid process gas contamination by oil or purge gas, a pump equipped with ferrofluidic shaft seal was developed and is under tests. The ferrofluidic seal has principally demonstrated its feasibility and required performances [6].

4. Tritium processing system

The Tritium Plant has the functions of receiving tritium shipments and loading into fuel cycle, storage of tritium and deuterium, processing of Torus exhaust and NB exhaust for recycling of tritium and deuterium (consisting mainly in separation of hydrogen from all other exhaust gases, separation of hydrogen into specific isotopic species for refuelling and detritiation of impurities for controlled release into environment), detritiation of water and extraction of tritium from breeding blanket modules (if required) and processing of tritium from blanket modules.

4.1. Tokamak Exhaust Processing

The Tokamak Exhaust Processing (TEP) system receives exhaust gas from the Torus during normal operation, later stages of pumpdown, wall conditioning and codeposited tritium recovery gas and tritiated process gases, purge gases and process vacuum pump exhausts from various tritium systems. Its main task is to separate hydrogenic species from all other exhaust gases and to detritiate impurities for controlled release into environment through vent

detritiation. The TEP concept for ITER [7] is shown in Figure 2. The process employs a palladium/silver permeator as a first step ('impurity separation') to separate the bulk of un-burnt D-T fuel from the tritiated and non-tritiated impurities. The second step ('impurity processing') is carried out in a closed loop involving heterogeneously catalyzed cracking or conversion reactions combined with the permeation of hydrogen isotopes through another palladium/silver permeator, to liberate and to recover tritium from tritiated hydrocarbons or tritiated water. The third step ('final clean-up') removes almost all of the residual tritium by counter current isotopic swamping and is based on a so called permeator catalyst (PERMCAT) reactor. The PERMCAT reactor is a direct combination of a palladium/silver permeation membrane and a catalyst bed and has been specifically developed for final clean-up of gases containing up to about 1% of tritium in different chemical forms such as water, hydrocarbons or molecular hydrogen isotopes [8].
To investigate the individual steps and to demonstrate the overall process in an integral manner a facility with a typical tritium inventory of about 3 - 5 g is operated at the Tritium Laboratory Karlsruhe (TLK). Gases of different compositions have been fed into the PERMCAT, thereby varying major parameters such as impurity feed flow rate and impurity feed pressure, hydrogen purge flow rate and hydrogen outlet pressure. The main focus of these parametric studies has been set on achieving low tritium concentrations at the impurity outlet of the PERMCAT. Even under conditions considerably beyond the design limits (tritium at higher concentrations and flow rates), the tritium concentration at the outlet of PERMCAT could be kept below the design target, i.e. 1 Cim^{-3} proving the decontamination of tritiated gases at levels required by ITER.
Thus not only the high decontamination factor as required by ITER, but also the operational mode of TEP as a waste dump for gases from diverse sources has been experimentally validated.

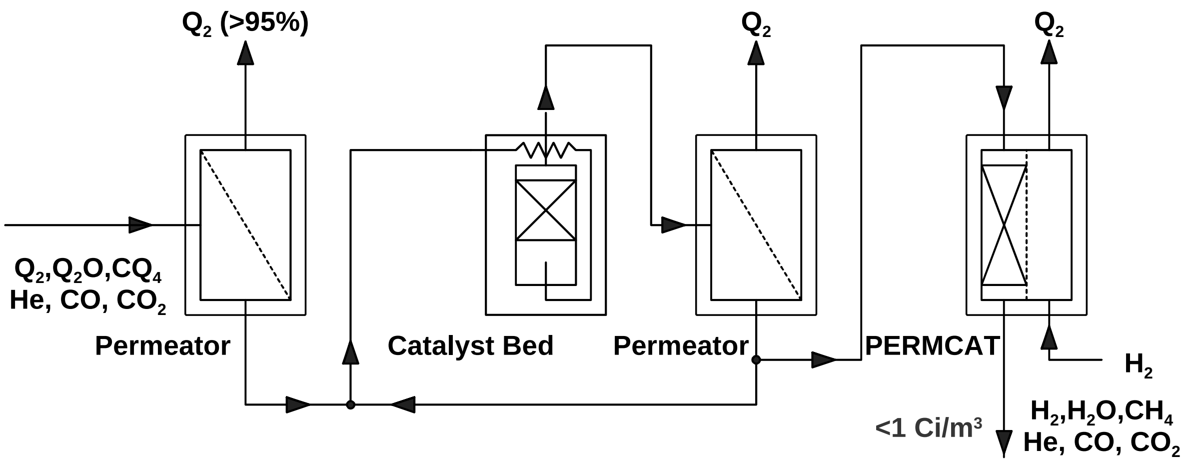

Figure 2 Block diagram of Tokamak Exhaust Processing for ITER

4.2. Water Detritiation System (WDS)

For ITER, the WDS is one of the key systems to control the tritium content in the effluents streams, to recover as much tritium as possible and consequently to minimize the impact to the environment. The most suitable process for water detritiation is the well known CECE process, a combination of electrolysis and catalytic exchange of hydrogen isotopes in a Liquid Phase Catalytic Exchange (LPCE) column.
In the electrolysis unit tritiated water (with a possible some deuterium) is decomposed into tritiated and deuteriated hydrogen; a part of this gaseous stream is sent to the Isotope

Separation System for tritium recovery and the remaining is fed in at the bottom of the LPCE column. On the catalyst/packing mixture that fills the LPCE column the heavier isotopes are transferred in counter current from gaseous phase to water. It should be noted that the hydrogen depleted in tritium shall be released through the stack into the environment and not returned into the plant as for the heavy water detritiation processes for CANDU reactors [9]; therefore the separation duty of a CECE process for fusion reactors is very high (tritium concentration in H_2 rejection from the WDS for ITER shall be < 700 Bqm^{-3} air at a typical activity of tritium in the water feed of 370 $GBqm^{-3}$). As the separation performances are given by the efficiency of the catalyst/packing mixture from the LPCE column, a sustained R&D activity is devoted to develop and characterize high-performance catalyst/packing mixtures for a wide range of operating parameters (i.e. concentration, gas to liquid relative ratio) [10].

4.3. Isotope Separation System (ISS)

The isotope separation concept relies on the slight differences in boiling points (within the 20-25K range at atmospheric pressure) for the six combinations of H, D and T isotopes. These differences in volatility of the species permit separation of isotopes by distillation (usually in a cascade of columns) with the heavier species - tritium at the bottom and most volatile species - protium at the top of the distillation cascade.
The Isotope Separation System (ISS) envisaged to be used in ITER consists of a cascade of four cryogenic distillation columns (Figure 3) with the aim to process two gas streams, one from Torus exhaust and another from Water Detritiation System mixed with the return stream from NBI [7]. The ISS have to supply streams with well-determined tritium, deuterium and hydrogen compositions. The purpose of the first column CD1 is to remove tritium from predominantly hydrogen rich streams and to produce a tritium depleted hydrogen stream as the overhead product. A D_2 stream from CD2 is extracted, a part of it is used for fuelling, and a part is fed into the CD3 where a final purification of D_2 takes place and NB quality deuterium being extracted at the top of CD3. CD4 has as main feed stream the exhaust from TEP.

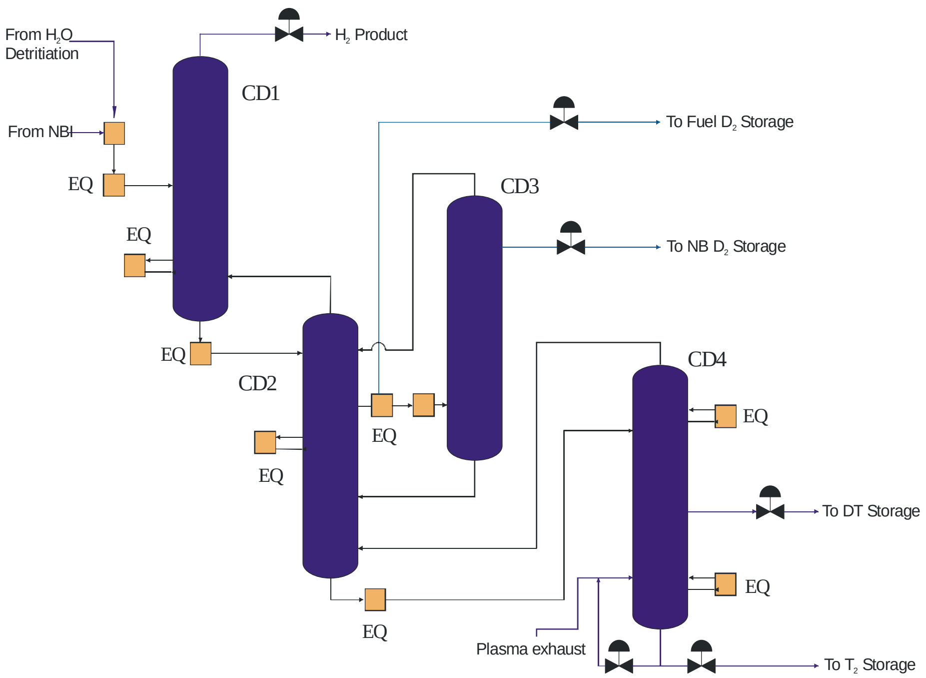

Figure 3 The Isotope Separation System of ITER

From CD4 two products are extracted, DT, approximately at the middle of the column and 90%T - 10%D at the bottom of the column. Several equilibrators are envisaged to be used in ISS to help the separation process, by promoting exchange reactions between the molecules isotopomers. Since equilibration at cryogenic temperature is too slow, isotope equilibrators can be effective only at ambient temperatures. For this reason all equilibrated streams must be warmed up to ambient temperature by exchanging heat with the returning equilibrated stream.
During normal operation of the Torus, the flow-rates and composition of feeding streams into the ISS are fluctuating along with the Torus pulses. Therefore, the behavior of the CD cascade has to be characterized with high accuracy with respect to thermal and isotopic fluctuations. Moreover, the top product of one of the CD1 column has to be almost tritium free to be discharged into the environment, as this is one of the main concerns from the licensing point of view. Based on these considerations, the top product of the CD1 column is proposed to be returned into the LPCE column of the Water Detritiation system.
Several cryogenic distillation based isotope separation systems have been developed and tested with the aim to develop the technology for fusion. The JET Active Gas Handling cryogenic distillation system was designed to separate a mixture of hydrogen isotopes into individual isotopic components, discharging low activity protium directly to stack and storing deuterium and tritium for re-use. The CD at JET was originally designed to operate a daily throughput of 30g tritium, 60g deuterium and 300 g protium. Three interconnected columns have been considered, with feeds from the multi pellet injector (high protium/low tritium stream) and from the DT torus exhaust respectively. Since the operating parameters have been significantly changed from the design values, (high deuterium content instead of protium) the overall tritium enrichment was only up to 35% [11].
A system of four interlinked cryogenic distillation columns has been tested at Tritium System Test Assembly (TSTA) at Los Alamos National Laboratory. The capacity of the system was 360 mol/day of a 50/50 D-T product with 1% H. The system was designed to separate four streams i.e. a tritium-free stream for waste disposal, a high-purity D_2 stream, a D-T stream and a high-purity tritium stream. The column separation characteristics were satisfactory, however the system was highly unstable due to fluctuations of the system flows and liquid level at the reboiler and possibly to dynamic behavior of a multicolumn cascade system [12].
A cryogenic distillation facility with the aim to investigate the trade-off between CECE–CD, to validate different components and mathematical modelling software is current under development at the Tritium Laboratory Karlsruhe (TLK) as an extension of the existing CECE facility [13]. The main goals are to investigate the decontamination factor along the LPCE column against thermal and isotopic fluctuation in the CD column and to improve the control philosophy of the ITER-ISS.

4.4. Storage and Delivery System (SDS)

Uranium beds have been used successfully for decades to store tritium. Uranium tritide has a very low dissociation pressure at room temperature. This reversible liberation and uptake of tritium from uranium beds can be performed many times under appropriate conditions without loss of efficiency. However as it is non-radioactive the intermetallic compound ZrCo is the reference getter material for ITER, thereby compromising with a number of drawbacks. The main requirements of a Storage and Delivery System for a fusion reactor are fast delivery and recovery of fuel and in-bed calorimetry for rapid tritium quantity measurements (± 1% in 24 h) during DT plasma campaigns. In the present design of the SDS for ITER tritium and deuterium with certain tritium content is stored in storage beds with a nominal storage capacity of 100 g tritium.

A 1:1 prototype of as ITER storage bed with a capacity of 100 g tritium and a target supply rate of up to 200 Pam^3s^{-1} was designed and manufactured at TLK. The getter bed is filled with ZrCo and is under testing for different operation conditions and to characterize the possible isotope effects. The disproportionation of the ZrCo is of serious concern since even though conditions for disproportionation have been carefully avoided in the fast delivery experiments, the zirconium-cobalt getter bed was disproportionated to more than 30% [14].
Tritium inventory measurements by in-bed calorimetry on a 25 g tritium storage bed have been performed at Tritium Processing Laboratory (TPL), Japan. The accuracy was about 0.15g at full tritium storage, therefore the ITER target value of 1% for 100g tritium storage bed appear to be confidently achievable [15].

5. Extraction/removal of tritium from breeding blanket

A Helium Cooled Lithium Lead (HCLL) and a Helium Cooled Pebble Bed (HCPB) Blanket are currently considered to be the most promising blanket concepts and are expected to undergo extensive testing within the frame of the operation of ITER. To simplify design activities it has been agreed to use the same coolant, i.e. gaseous helium, and the same coolant circuit technologies. Because a non-negligible amount of tritium permeates into the helium primary cooling circuit, a Coolant Purification System (CPS), designed to allow continuous removal of tritium from the helium cooling gas has to be considered.
In view of the large differences in the feed streams of the DEMO and ITER blankets there will be a considerably divergence in the layout of the respective tritium processing systems. The tritium generation rate during pulses for DEMO is approx 400 times higher than the expected values for ITER, while the tritium permeation rate from purge gas stream for DEMO is expected to be 6 order of magnitude higher than in ITER.
To extract tritium from the blanket and from the coolant gas stream, two major separate loops are planned:

- Tritium Extraction System (TES), designed to accomplish the extraction of tritium from the blanket purge gas and the removal of tritium from this purge gas. For TES several process have been proposed, the most promising one being a Cold trap combined with Thermal Swing Adsorption using a cryogenic molecular sieve bed [16]. After its development for ITER it could be scaled up without much conceptual modification to the requirements of DEMO. The final tritium recovery in ITER will be done via existing TEP and ISS as the contribution of the blanket stream is very small. Due to the differences in the purge gas flow rates for DEMO (~600 times higher in comparison with ITER), a dedicated and conceptually different ISS will have to be considered for DEMO.
- Coolant Purification System (CPS) designed to process the tritium permeated from the blanket modules into the primary helium coolant. According to the present concept for ITER [17] a fraction of only 0.01-1% of the helium coolant stream is fed into CPS. The reference CPS process proposed by FzK is a combination of an bed for oxidising Q_2 to Q_2O, followed by a cold trap at 200K where water freezes out and finally a cryoadsorber bed containing molecular sieve at 77K where the impurities and all remaining non-oxidised Q_2 is retained. For this system, the feed flowrate for DEMO will probably be $5{\cdot}10^5$ higher than for ITER and very likely a different concept of the CPS will be needed for DEMO.

On account of the above, a direct extrapolation of the operational results expected from ITER will most probably not suffice for the design of the processes needed to satisfy all requirements of DEMO. Significant complementary experimental and design activities are therefore needed.

6. Safe handling of Tritium

Within the Tritium Plant of ITER a total inventory of about 2 to 3 kg will be necessary to operate the machine in the DT phase at a throughput of about 1 kg tritium per hour. During plasma operation, tritium will be distributed in the different subsystems of the fuel cycle, as shown in Figure 4, computed with the tritium inventory mathematical model for ITER, TRIMO [18]. About 1 kg tritium will be located in the Long Term storage; a significant amount of tritium will also be in the vacuum vessel and in the Hot Cells.
The confinement of tritium within the respective fusion plant processing systems is one of the most important safety objectives. The design of the deuterium / tritium fuel cycle of ITER includes a multiple barrier concept for the confinement of tritium [19]. The concept follows the experience and practice of other tritium facilities such as TSTA in Los Alamos (US), TLK in Germany or AGHS of JET in UK [20]-[22] and has two major features, namely a two (primary and secondary) barrier design and secondary containment atmosphere and gaseous waste treatment systems. Hard shells or glove boxes are employed as a second containment, the latter allowing for an easy access in case of maintenance or repair. Ultimately the building is equipped with a vent detritiation system, required as an emergency tritium confinement barrier during possible accidental events. Following occurrence of such accidental event, the normal ventilation system is isolated and the vent detritiation system is actuated to maintain negative room pressure.

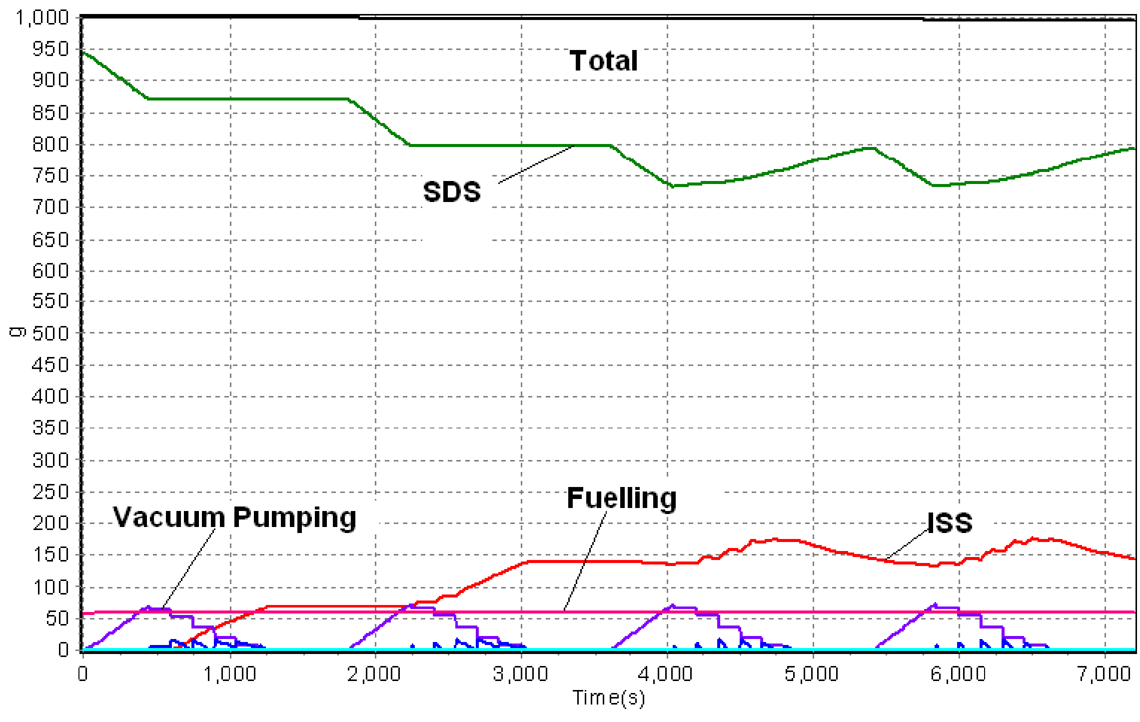

Figure 4 Tritium inventory in fuel cycle of ITER during short-burn pulses

Although recommended and discussed, the applicability of the international standard IEC 61508 for the management of functional safety to the fuel cycle of ITER will probably be decided after the ITER site will be known and in agreement with the local authorities. To follow the philosophy of IEC 61508 a general description of the fuel cycle concept and a definition of the overall scope is required. Such a description is compiled in the Final Design Report of ITER (2001) [23] and is detailed in Process Flow Diagrams (PFDs) and Pipe and Instrumentation Diagrams (P&IDs). The risk evaluation by a HAZOP study is typically carried out by analyzing each loop as described in the P&IDs and their associated list of components, heaters, instruments, pipes and valves. The documented results of the study may have consequences leading to design modifications. A further step is the functional safety analysis with the identification of Safety Instrumented Functions (SIF). A specific Safety Integrity Level (SIL) is assigned to each SIF, and software solutions are accepted for low risks, hardware solutions are required for medium risk levels and none of the loops within the inner deuterium / tritium fuel cycle should involve the highest risk level.

7. Conclusion

Experience and practice in tritium technologies relevant for fuel cycle do exist in several civil tritium facilities such as the TLK in Germany, the AGHS of JET in UK and the Tritium Plasma Laboratory in Japan. For all physical processes employed in a fuel cycle for fusion reactor, experiments at relevant scale have been and are carried out and sometimes even 1:1 prototype components are under testing. The detailed design of the inner D-T cycle of ITER has been developed [24], [25]. Not only outline flow diagrams, but process flow diagrams including chemical flow sheets and even pipe and instrumentation diagrams along with the associated lists for components, heaters, instruments, pipes and valves have been prepared for all the fuel cycle systems of the ITER Tritium Plant. This degree of detail together with mathematical models allows determining the tritium inventory and the tritium distribution within the Fuel Cycle of ITER. As the inputs are not yet defined, the system for tritium extraction/removal from breeding blanket is still in a conceptual phase and R&D is still needed until this system will be developed until the same degree of detail as the other sub-systems of the fuel cycle. The commissioning of the tritium plant at ITER will be the ultimate test related to the performances of the fuel cycle as a whole.

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References

[1] C. Andelfinger, E. Buchelt, P. Cierpka, H. Kollotzek, P. T. Lang, R. S. Lang, G. Prausner, F. X. Söldner, M. Ulrich, and G. Weber, *A new centrifuge pellet injector for fusion experiments*, Rev. of Sci. Instr. **64,** 983 (1993).

[2] M. Dremel, C. Day, A. Mack, H. Jensen, E. Speth, H.D. Falter, R. Riedl, J.-J. Cordier, B. Gravil, D. Henry, P. Zaccaria, *Design of Cryo-sorption Pumps for Test Beds of ITER Relevant Neutral Beam Injectors*, Proc. 22nd SOFT, Venice, Italy, Sept. 2004 (to be published in Fusion Eng. Design 2005).

[3] M. Wykes, *Minimization of the Hydrogenic Inventory of the ITER Neutral Beam Line and Torus Cryo-sorption Pumps*, accepted for publication in Fusion Sci. Technol. 2005.

[4] ITER Baseline Documentation, PID 1.0 (9 Sep 2004).

[5] C. Day, A. Mack, M. Glugla, D.K. Murdoch, *Tritium Inventories in the High Vacuum Pumps of ITER*, Fusion Sci. Technol. **41**, 602 (2002).

[6] A. Antipenkov, Ch. Day, R. Lässer, A. Mack, R.Wagner, *Tritium Pumps for ITER Roughing System*, accepted for publication in Fusion Sci. Technol. 2005.

[7] M. Glugla, L. Doerr, R. Laesser, D.K. Murdoch, H. Yoshida, *Recovery of Tritium from Different Sources by the ITER Tokamak Exhaust Processing System*, Fusion Eng. Design **61-62**, 537 (2002).

[8] B. Bornschein, M. Glugla, K. Günther, T.L. Le, K.H. Simon, S. Welte, *Successful Experimental Verification of the Tokamak Exhaust Processing Concept of ITER with the CAPER Facility*, accepted for publication in Fusion Sci. Technol. 2005.

[9] J. Miller, S. Celovski, A. Everatt, W. Graham, J. Tremblay, *Design and operational experience with a pilot scale CECE Detritiation process*, Fusion Sci. Technol. **41**, 1077 (2002).

[10] I. Cristescu, U. Tamm, Ioana-R. Cristescu, M. Glugla, C.J. Caldwell-Nichols, *Investigation of simultaneous tritium and deuterium transfer in a catalytic isotope exchange column for water detritiation*, Fus. Eng. Design **61-62**, 537 (2002).

[11] N. Bainbridge, A.Bell, P. Brennan, S. Knipe, R. Lässer, R. Stagg, *Operational experience with the JET AGHS cryodistillation system during and after DTE1*, Fusion Eng. Design **47**, 321 (1999).

[12] M. Inoue, S. Konishi, T. Yamanishi, S. Ohira, T. Watanabe, K. Okuno, Y. Naruse, R. Shermann, W. Barnes, J. Bartlitt, J. Anderson, *Isotope Separation System Experiments at the TSTA*, Fusion Technol., **21**, 293 (1992).

[13] I. Cristescu, I-R. Cristescu, L. Doerr, M. Glugla, G. Hellriegel, O. Kveton, D.K. Murdoch, P. Schaefer, S. Welte, *TRENTA facility for trade-off studies between Combined Electrolysis Catalytic Exchange and Cryogenic Distillation processes*, accepted for publication in Fusion Sci. Technol. 2005.

[14] M. Glugla, D. K. Murdoch, A. Antipenkov, S. Beloglazov, I. Cristescu, I.-R. Cristescu, Ch. Day, R. Laesser, A. Mack, *ITER Fuel Cycle R&D: Consequences For The Design*, ISFNT-7, 22-28May, Tokyo, Japan (2005), (to be published in Fusion Sci. Technol. 2005).

[15] T. Hayashi, T. Suzuki, M. Yamada, M. Nishi, *Tritium accounting stability of a ZrCo bed with in-bed gas flowing calorimetry*, accepted for publication in Fusion Sci. Technol. 2005.

[16] H. Albrecht, E. Hutter, *Tritium recovery from an ITER ceramic test blanket module – process options and critical R&D issues*, Fusion Eng. Design **49-50**, 769 (2000).

[17] L. Boccaccini, H. Albrecht, L. Berardinucci, M. Dalle Donne, U. Fischer, S. Gordeev, S. Hermsmeyer, E. Hutter, K. Kleefeldt, T. Lechler, M. Lenartz, P. Norajitra, G. Piazza, G. Reimann, P. Ruatto, F. Scaffidi-Argentina, K. Schleisiek, I. Schmuck, H. Schnauder, H. Tsige-Tamirat, *The European HCPB Test Blanket Module to be irradiated in ITER*, Fusion Technology, Proc. 20th Symp. Fusion Technology, Marseille, France, 7-11 September, 1255 (1998).

[18] I.R. Cristescu, L. Doerr, A. Busigin, D.K. Murdoch, *ITER Dynamic Tritium Inventory Modeling Code*, accepted for publication in Fusion Sci. Technol. 2005.

[19] M. Glugla, R. Laesser, L. Doerr, D.K. Murdoch, R. Haange, H. Yoshida, *The inner deuterium/tritium fuel cycle of ITER*, Fusion Eng. Design **69**, 39 (2003).

[20] R.V. Carlson, K. Okuno, *Twelve Years of Tritium Operating Experience at the Tritium Systems Test Assembly*, Fusion Technol. **30**, 900 (1996).

[21] T. Vollmer, U. Besserer, K. Borcherding, J. Dehne, H. Dilger, L. Doerr, M. Glugla, W. Hellriegel, E. Hutter, R. Kraemer, R.-D. Penzhorn, B. Reinhardt, D. Roehrig, K. Schubert, *Tritium Laboratory Karlsruhe: Safety Concept*, Fusion Technol. **28**, 988 (1995).

[22] A.C. Bell, P. Ballantyne, C. Gordon, M.A. Wright, *The Safety Case for JET D-T Operation*, Fusion Eng. Design **47**, 115 (1999).

[23] ITER Final Design Report (2001).

[24] M. Glugla, A. Busigin, L. Doerr, R. Haange, T. Hayashi, O. Kveton, R. Laesser, D.K. Murdoch, M. Nishi, R.-D. Penzhorn, H. Yoshida, *The Tritium Fuel Cycle of ITER-FEAT*, Fusion Eng. Design **58-59**, 349 (2001).

[25] M. Glugla, R. Laesser, L. Doerr, D.K. Murdoch, R. Haange, H. Yoshida, *The Inner Deuterium/Tritium Fuel Cycle of ITER*, Fusion Eng. Design **69**, 39 (2003).