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Tritium Experience in Large Tokamaks: Application to ITER

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ABSTRACT:

Recent experience with the use of tritium fuel in the Tokamak Fusion Test Reactor and the Joint European Torus, together with progress in developing the technical design of the International Thermonuclear Experimental Reactor has expanded the technical knowledge base for tritium issues in fusion. This paper reports on an IEA workshop that brought together scientists and engineers to share experience and expertise on all fusion related tritium issues. Extensive discussion periods were devoted to exploring outstanding issues and identifying potential R&D avenues to address them. This paper summarizes the presentations, discussions and recommendations.

I: INTRODUCTION

"Tritium Experience in Large Tokamaks; application to ITER" was the subject of an International Energy Agency workshop held at the Princeton Plasma Physics Laboratory on March 16-18th, 1998. Practical fusion power has come one step closer in recent years with the introduction of tritium fuel into the Tokamak Fusion Test Reactor (TFTR) and the Joint European Torus (JET) plasmas and the generation of high levels of fusion power, 10.7 MW and 16 MW respectively, in these tokamaks. At the same time, the International Thermonuclear Experimental Reactor (ITER) design activity is completing an integrated design of a machine with the mission to demonstrate the scientific and technological feasibility of fusion energy for electric power generation. ITER will integrate the experience of present day machines in a tokamak designed to demonstrate technologies required for the development of fusion as a practical source of energy. A key element in the success of the design process is the thorough communication and debate of the first hand, present day experience of tritium issues and tritium technology in fusion facilities within the wider fusion science community.

The IEA workshop brought together 90 scientists and engineers from: ITER, TFTR, JET, Asdex, DIII-D, Textor, Tore Supra; from tritium laboratories: AEC(Canada), JAERI(Japan), INEEL, LANL, SNL, SRS Westinghouse (US), Karlsruhe (Germany), Russian Nuclear Center, and from many other institutions to share experience and expertise in tritium issues on large tokamaks with a view to providing the optimal knowledge base for all aspects of tritium use in

ITER. The workshop incorporated the IEA Tritium Plant Operation workshop (W31). A total of 58 presentations were given. Each day's session concluded with an open discussion period devoted to exploring outstanding issues and identifying potential R&D avenues to address them.

The management and accounting of tritium will be crucial to the acceptance of fusion as an environmentally benign power source. Fortunately, both TFTR and JET have demonstrated safe tritium handling in a fusion machine. A 1,000 s ITER pulse will use three orders of magnitude more tritium than a JET pulse and, to minimize the on-site tritium inventory, tritium will be recycled from the plasma exhaust through the tritium plant and back into the same discharge. The TFTR and JET tritium fuel cycle experience and the ITER design were described (see talks by Hosea, Bell and Haange), together with a report on the Tritium Systems Test Assembly at Los Alamos (Carlson). Related work in tritium processing technology in the Japan Atomic Energy Research Institute (JAERI), Research Center Karlsruhe and the All-Russia Scientific and Research Institute was described (see talks by Nishi, Penzhorn and Milechkine).

Erosion on a scale of cm is predicted for the long ITER pulse length. This cm-scale compares to microns in present tokamaks and represents a 3-4 orders of magnitude change that is much larger than the change in any of the core physics parameters from present tokamaks. Tritium retention and dust generation are direct consequences of erosion and impact the design, operation and safety of the ITER. The issues include tritium accumulating in codeposited films and implanted in the armor surrounding the core and divertor plasmas, dust accumulation from eroded armor and the monitoring and removal of both in-vessel tritium and dust. As part of the R&D program supporting the engineering design phase of ITER, laboratory experiments designed to probe the fundamental processes, together with tokamak investigations and modeling are continuing to establish the needed database and gain insight on how to make high confidence predictions of the in-vessel tritium and dust inventory.

In addition to the tritium retention data from JET and TFTR, new important data particularly for beryllium and for tungsten alloys, are now available from several laboratories, for conditions of direct relevance to ITER, where previous data were either missing or largely scattered. Based on these data and on advancement of related modeling it is now possible to conclude that for ITER, the dominant mechanism for hydrogenic retention is codeposition of carbon and possibly beryllium (the latter only if a large amount of oxygen is present) with deuterium and tritium (see talk by Federici). Retention by other mechanisms such as implantation and surface absorption, which may be significant for small short-pulse machines, are expected to rapidly reach saturation in ITER. However, there are still significant uncertainties associated with the use of mixed-materials. Moreover, there are some uncertainties associated with the plasma edge physics parameters (i.e., heat and particle fluxes, erosion/redeposition rates and patterns) which are seen to strongly affect the deposition patterns and rates.

Tritium codeposition in ITER is predicted to be in the range from 1-20g/pulse, depending mainly on plasma edge physics parameters. Operation scenarios with high local redeposition probability have lower tritium retention, whereas detached plasma conditions are more transparent for the hydrocarbons formed at the target and are predicted by the modeling to retain large amount of stored tritium in thick codeposits on side areas. The above prediction of tritium codeposition is based on modeling studies (see talk by Brooks). The studies incorporate a detailed model of the plasma edge parameters at the strike zones, impurity release from the target (mainly chemical erosion) and near wall transport and molecular break-up of methane followed by the redeposition of eroded material.

While the amount of tritium retained is predicted to be high, the tritium retention fraction in ITER is projected to be lower than that observed in present-day tokamaks (see for example talks by Wampler and Mueller for TFTR, and Andrew and Coad for JET). In general, the fraction of tritium retained in a tokamak is a complex function of the machine and plasma geometry and the discharge history. The retention fractions experienced in existing machines cannot be simply applied to ITER because of the many significant differences between ITER and the existing machines. Retention models need to be benchmarked against the experience of existing machines to identify the contribution of transient effects such as impurity generation and transport during the startup and shutdown periods and hence better quantify the retention in the quasi-steady state 1,000 s pulse of ITER. In addition, the ITER edge density is some ten times higher than JET or TFTR and the walls are hotter.

The prospect of ITER reaching its in-vessel tritium inventory limit after the order of one hundred pulses highlights the critical need to develop and test in-situ efficient cleaning techniques that are much more efficient than those applied in current tokamaks. As long as carbon is used in ITER, the efficient control and removal of the codeposited tritium will be essential. High temperature baking of the divertor system in an oxygen atmosphere (see talks by Haasz and Philipps) and low-pressure plasma discharges with oxygen, e.g., ECR, ICR, (see talks by Cowgill and Landkammer), show good potential, but further R&D is urgently required to determine their ability to detritiate and/or remove thick codeposited layers. The introduction of oxygen also raises potential complications in its interaction with in-vessel components and the oxygen needs to be subsequently removed to restore good wall conditions. The exhaust of significant quantities of highly tritiated water, DTO, also raises concerns in the plasma exhaust handling systems (see sec.4). Techniques for tritium removal that do not use oxygen are also being investigated (see talks by Nygren and Skinner).

Dust generation is another consequence of erosion and will be an important safety constraint in long pulse machines. Research into dust production mechanisms, and its biological interactions is just beginning (see talks by McCarthy and Richardson). Just measuring the in-vessel dust inventory is a challenge in existing machines.

In summary, there has been significant progress in the last few years on the understanding of the complex interrelated subjects of erosion, codeposition and tritium retention. However, the large scale up predicted for both these issues in ITER will require challenging physics-engineering solutions to be developed.

2. TRITIUM FUEL CYCLE IN TFTR, JET AND ITER.

The tritium fuel cycle operational experience on TFTR was presented by J Hosea and related topics of tritium delivery and exhaust, including tritium purification were presented by C Gentile. Over 3 years of tritium operation has enabled TFTR to explore many important issues of transport, alpha physics and MHD stability in a reactor core.[1] Approximately 100 g of tritium were safely processed through the TFTR fueling system. Of this, 5g of tritium were injected into the torus during TFTR D-T operations: ~3 g by neutral beam injection and ~2 g with direct gas injection. Extensive cleanup campaigns over periods of months resulted in ~4 g of the tritium being removed from the torus i.e. ~ 1 g (20%) of tritium is presently retained in co-deposits with carbon on the limiter and wall.

Tritium gas was brought on-site in approved shipping canisters and stored in uranium beds. These were heated to transfer the gas to the neutral beams or torus gas-injection system in a batch mode. Tritium exhaust was pumped by the liquid helium cooled cryopanels in the beam boxes and/or by turbopumps in the torus vacuum pumping system. Most of the tritium along with deuterium and other effluent gases were collected, oxidized, and shipped off site for reprocessing or burial in disposable molecular sieve beds. A rigorous chain of command for batch mode tritium transfers and detailed accounting enabled TFTR to successfully track the in-vessel inventory and stay well within stringent site limits (5 g facility, 2 g vessel, 0.05 g stack release / year, 0.0001 g tritium in surface water release / year).[2] Calibration of tritium detectors in the variety of gases in the tokamak exhaust was found to show significant differences to calibration factors based on tritium admixtures with pure gases. Partial correction of this effect was made by weighting the contribution of each gas by its collision frequency and the energy per ion pair.

A tritium purification system was commissioned and successfully employed to recycle about 8 g of tritium back for use by TFTR during the final months of operation.[3] The tritium systems at TFTR continue to be employed in support of on going tritium removal activities, analysis of TFTR graphite tiles, and tests of tritium technology for fusion applications. Based on TFTR experience, recommendations for ITER included central management of tritium transfers, in-situ calibrations, effective tritium control valves, provisions for maintenance of the entire tritium fueling and exhaust system, and development of tritium removal techniques to control the tritium inventory. In particular, redundant systems should be used to permit repairs/calibrations on one tritium fuel/exhaust system while the other is supporting operations. Aggressive tritium removal techniques will be needed because of the high ITER duty cycle. Tokamak exhaust contains exotic and sometimes corrosive impurities and it is important to test tritium processing technology in a fusion environment.

W. West reported on the Savannah River Site's (SRS) support of the PPPL tokamak operations. SRS supplied to PPPL approximately 100g of 99.7% pure gaseous tritium enclosed in either LP-50 or UC609 type shipping containers, each containing up to 2.3 g. After use at PPPL, the torus exhaust gas of mainly deuterium with up to about 2% tritium was oxidized and stored on disposable molecular sieve beds and shipped back to SRS for recovery of the tritium. C. H. Myler discussed the material control, accounting and security systems used to safeguard the nuclear materials at SRS. It is clear that careful attention to these topics will be important issues in future long-pulse DT fusion facilities that plan to have several kg of tritium on site.

The JET tritium handling experience was summarized by A. C. Bell. Preparations for the deuterium-tritium experiment (DTE1) included regulatory approval of safety analyses and waste discharges, installation of tritium compatible systems e.g. double windows, training and emergency planning and exercises. During DT operations there were 70 interventions into potentially contaminated systems. Pre-planning was found to be essential to limit the personnel exposure in maintenance activities and interventions into tritiated vacuum systems. Interventions which mainly involved exposure to HT were relatively easy to clean-up to a level which permitted non-suited access. Interventions in systems which generally involved exposure to HTO or hydrocarbons were significantly more difficult to decontaminate. Ventilation was used to provide an effective barrier to the spread of HTO contamination to personnel. All interventions, apart from those involving the exhaust detritiation system, resulted in very low workplace tritium levels (100 DAC transiently, <<1 DAC during task). R. Lasser described the Active Gas Handling System (AGHS) at JET. Key functions of the AGHS are to pump exhaust gases, to separate helium, protium, deuterium, tritium and impurities so that deuterium and tritium could be reused and tritium was not exhausted to the environment. The AGHS performed as required and successfully supplied and recycled 99 g of tritium for use in JET within a 20 g site limit of tritium.

A status report of the design of the ITER tritium plant was given by R. Haange. The design of the ITER tritium plant is mature in many respects and is capable of supplying ~200 Pa*m^3/s of deuterium/tritium fuel with bursts of up to 500 Pa*m^3/s for 10 s. The plasma exhaust of ~250 Pa*m^3/s will be detritiated using a Pd/Ag front end permeator to remove impurities. Several options to recover tritium from the impurity gases after the front end permeator are being investigated. The tritium and deuterium will be stored in ZrCo metal hydride beds. The design will be further improved when R&D results are available. Recovery of tritium from codeposited layers, and from test blanket modules are outstanding issues that require further R&D.

M. J. Gouge reviewed the potential of tritium pellet injection for core fueling of ITER.[4] The fueling efficiency of pellets measured in tokamak experiments is higher than for gas puffing: typically 30% to 100% for pellets versus 2% to 20 % for gas puffing. The higher efficiency of pellet fueling allows lower tritium gas flows into and out of the vessel and a potential reduction of tritium wall inventory of a factor of two or more. The extrusion of solid tritium has been demonstrated at the Tritium Systems Test Assembly and full scale ITER D_2, DT and T_2 pellets have been produced. The recent demonstration of higher pellet fueling efficiency for pellet launch from the high magnetic field side versus the usual low field side has initiated work to develop a high field side injection system for ITER. L. Garzotti discussed how the recent experimental evidence of an outwardly directed drift of the material deposited by the pellets influences the fueling profile. Modeling simulations showed that, due to this effect, only high field side injection and/or high velocity pellets (≥2 km/s from the high field side to 5 km/s from the low field side) allow a significant improvement with respect to pure gas puff fueling. Fueling of the plasma core by recycling in TFTR was briefly discussed by C Skinner. In plasmas fueled by deuterium recycled from the limiter and tritium-only neutral beam injection, the DT neutron rate provided a measure of the deuterium influx into the core plasma.[5] The analysis showed that influx from the limiter could contribute a large fraction of the TFTR core density,

W. Kuan described a simplified model for use in the ITER integrated dynamic fuel cycle model CFTSIM.[6] This describes the retention dynamics in the ITER torus, taking into account all relevant processes. The model was validated on experimental fueling data from JET, Tore Supra, and JT60. The model predicts that, for a 50:50 DT discharge with 200 Pa*m^3/s fueling, the ITER in-vessel tritium inventory will increase by about 10 g per 1000 s pulse. Major issues to be addressed are that codeposition of tritium with carbon is expected to dominate, periodic tritium removal from the surface will be very important, and disruption effects are very uncertain and need to be taken into account.

3. TRITIUM RELATED TECHNOLOGY.

M. Nishi summarized tritium processing technology development for fusion reactors at the Tritium Process Laboratory (TPL) of JAERI. Work has been carried out using gram level tritium since 1987. A modified fuel cleanup unit (FCU), composed of a Pd diffuser and an electrolytic reactor, has been developed for the fuel cycle of ITER. The performance of the system has been demonstrated with H-D-T in an integrated loop of the FCU and isotope separation system (ISS) at TPL. The study of the ISS was carried out for the cryogenic distillation and thermal diffusion columns. A chemical exchange column for the water detritiation system has also been studied. A system of cryogenic molecular sieve beds was proposed and successfully demonstrated for the blanket tritium recovery and He GDC systems. Some basic experiments have also been carried out for the tritium release behavior from the blanket materials. Y. Iwai reported on tritium accountancy and safety studies at the Tritium Process Laboratory of JAERI. Gas flowing calorimetry and multi-channel laser Raman spectroscopy system for process gas analysis has been successfully developed for tritium accounting. A cassion has been built to study tritium behavior in confined spaces and/or in the environment and to test tritium confinement equipment. S O'Hira described studies of tritium-materials interactions at TPL. It was found that thermal desorption of deuterium implanted in bulk tungsten was strongly correlated to changes in the microstructure. Studies of tritium retention in the carbon fiber reinforced carbon material, CX-2002, showed retention proportional to the half power of the total incident fluence at 293 K and 593 K, implying significant diffusion of hydrogen in the structure. At 893 K, retention was proportional to about the one third power of the total incident fluence, suggesting that release by recombination becomes the more important process at higher temperatures.

R. Carlson described the work at the Tritium Systems Test Assembly (TSTA) at Los Alamos. This system was built to test the tritium systems required for a tokamak fusion reactor. The primary TSTA fuel processing systems are hydrogen isotope separation, transfer pumping, cryopumps and a fuel cleanup system. The method chosen to separate hydrogen isotopes was cryogenic separation. The system produces streams of HD, D_2, DT and T_2; the T_2 is 99.99% pure. The current system being tested at TSTA to remove impurities is a Pd membrane reactor that produces a stream of pure hydrogen isotopes and a tritium free waste stream. TSTA has tested two cryopumps for tokamak evacuation. TSTA is an international program with an active participation from the JAERI (Japan Atomic Energy Research Institute). Other experimental activities conducted at TSTA in support of the fusion program are: tritium plasma experiments, tritium pellet injectors, tritium decontamination and decommissioning work and numerous basic experimental measurements on tritium properties and interactions.

M. Glugla presented results from plasma exhaust clean-up experiments and outlined the design of the endurance test facility Caper at the Tritium Laboratory, Research Center Karlsruhe.[7, 8] Caper is a three stage concept for fuel recovery from torus exhaust: a front end permeator, next catalytic processing, followed by counter current isotope swamping in a permeator/catalyst combination called permcat. The continuously operating system produces a waste gas stream with less than 0.1 Ci/m3. Endurance tests are planned to investigate the dynamic behavior of the process under transient conditions, to validate control regimes and stability of control loops, to demonstrate automatic plant control, to test for possible plant degradation with time, and to accumulate operational experience relevant to a continuously working facility. The infrastructure at the Research Center is currently being improved to accommodate further tests with Caper.

W. T. Shmayda discussed tritium recovery from tritiated liquids such as pump oils, HPLC effluent mixed waste and synthesis mixed waste commonly generated by the pharmaceutical industry. The recovery system comprises a catalytic oxidation stage to convert the organic liquids to water, an electrolysis cell to reduce the water to hydrogen, and an isotopic separation system to recover the tritium from deuterium and hydrogen. Process rates on the order of 10 liters per day appear viable. Destruction of the organic liquids within the catalytic combustion system is typically better than 6 nines (99.9999%) and have been shown to be below limits prescribed by the Environmental Protection Act. Tritium releases are about 1×10^{-6} of the tritium inventory processed.

W. T. Shmayda also described a novel device to measure tritium activity on surfaces non destructively. This portable device can be used on metallic and insulating surfaces in the lab or in the field and has been calibrated against traceable standards. Comparisons against smearing, leaching and combustion measurements demonstrate that it accurately measures the total tritium inventory within the first 100 monolayers of the surface. A broad range of tests have been completed to support the hypothesis that the current produced by beta decay of tritium is linearly proportional to the areal concentration of tritium atoms. On insulating surfaces, the device needs to be fitted with a screened boot to reduce noise induced by capacitive coupling between the sensing plate and the surroundings.

R Hsu described a glovebox design aimed at minimizing tritium release, moisture and oxygen ingress and protecting workers at the proposed Tritium Separation Facility at Savannah River Site. M Nishikawa discussed the surface chemistry underlying decontamination of surface tritium. While tritium removal by isotope exchange at the surface can be rapid, diffusion through e.g. 10 microns of graphite can limit tritium release from deeper layers. J Koonce presented studies of the form and radioactivity of ITER waste and decontamination techniques. Solid wastes have a wide range of shapes and sizes ranging from high level multi-ton plasma facing components to low level hydride beds, molecular sieve beds, piping, tanks, and gloveboxes. Tritium contents vary from 10^6 to 10^{14} Bq/ton.

D. L. Jassby presented the results of tritium breeding experiments using ^{3}He cells placed near TFTR. The experiments were intended to develop a diagnostic of long-term neutron fluence as well as to investigate the use of ^{3}He as a potential reactor blanket medium. The asymptotic value of the local tritium breeding ratio, defined as the number of tritons produced per incident neutron on the cell assembly, was 0.32. Possible reasons for this low value were neutron losses in the CH_2 neutron moderator, insufficient energy moderation and incomplete tritium recovery.

Yu. Zouev presented measurements of the deuterium and tritium permeability through steels that are candidates for fusion reactor materials. Permeation was shown to be limited by diffusion of D and T atoms. V Andryushin presented electron microscope autoradiography measurements of residual tritium in first wall steel alloys. Tritium solubility in austenitic steel was twice that of martensitic steel. Residual tritium was associated with regions having increased concentrations of dislocations.

G. L. Saksagansky described a specialized (reflectron) time-of-flight mass spectrometer. The reflectron holds promise as an effective and comparatively inexpensive instrument for the detection of tritium without interference from the noise current from the beta decay of tritium. Small-scale static mass spectrometers are also being developed. Aging tests of cryopanels exposed to tritium were satisfactory. A large-scale tritium test stand, the Russian Experimental Close Loop, is under construction. This is intended for dynamic simulation and lifetime testing as well as investigation of safety issues. Yu. Milechkine presented an investigation in progress at the All-Russia Research Institute of Inorganic Materials on the degradation of palladium diffusers, which will have prolonged exposure to tritium in the ITER tritium clean-up system.

The measurement of in-vessel tritium is important for a complete tritium accounting, but is technically challenging. An in-situ measurement of tritium in TFTR was discussed by S. J. Zweben.[9] The technique was based on measurement of the secondary electron current created by the tritium Betas when the vacuum vessel was filled with N_2 gas. The current was collected by a probe inside the chamber biased at 15-30 VDC, and peaked at an N_2 fill pressure of about 4 Torr. This technique is limited by the range of Beta particles in carbon and is only sensitive to tritium that resides within about 1 micron of the surface. Further, the complicated geometry of the vacuum vessel surface and the limited range of the betas in the N_2 gas means that tritium on surfaces far away from the probe will not be detected. An in-situ calibration performed by adding tritium gas to the N_2 fill suggests that about 10% of the TFTR in-vessel tritium could reside within about 1 micron of the surface.

4. TRITIUM PLANT DESIGN ISSUES. (Facilitator R. Haange).

The design of the ITER Tritium Plant was discussed in a special session at the end of the first day of the workshop. There are three issues that will need addressing in the near future and may require dedicated R&D in support of the design: (i) tritium recovery from co-deposited layers in the ITER vacuum vessel, (ii) tritium measurement in an area with gamma-radiation background and (iii) tritium removal/decontamination in areas that require hands-on maintenance access.

4.1. Tritium recovery from co-deposited layers in the vacuum vessel.

To avoid the potential for public evacuation in case of the worst credible accident, tritium recovery will be administratively required when up to 1000 g of tritium have accumulated in the vessel. Various methods have been proposed for tritium recovery from carbon/hydrogen (tritium) co-deposited layers produced during DT plasma discharges. Experimental results indicate (see Sec. 6) that the dominant chemical species produced by oxygen baking are DTO, CO, and CO_2. Complete removal of 1000 g tritium would produce 7000 g DTO assuming that the co-deposited hydrogen species would be DT. Recent R&D results indicate that a chemical erosion approach by using low pressure O_2 gas at temperatures as low as 250°C (the presently envisaged maximum baking temperature of the in-vessel components is 240°C) may provide an acceptable co-deposit removal rate.

From a tritium plant design point of view, it is highly desirable to develop recovery methods of tritium from co-deposited layers that avoid formation of tritium oxide. Formation and accumulation of highly tritiated water is, where practical, to be avoided in tritium processing systems as it raises a safety concern due to the very high specific activity of DTO (~1400Ci/g DTO) and the four orders of magnitude higher radio-toxicity of tritium oxide in comparison with the elemental form, at the same level of radioactivity. A method that may have a sufficiently high recovery rate could be in-situ rapid heating of co-deposited layers in vacuum or inert gas using an impinging laser beam or other form of surface heating. One such method using laser heating was proposed in the workshop (Skinner, Sect. 6). A preliminary design study is underway to investigate in more detail the problems associated with the routine processing of

DTO recovered from co-deposited layers and R&D is strongly encouraged to develop tritium recovery from co-deposited layers that avoids the formation of DTO.

4.2. Tritium measurement in an area with gamma-radiation background

ITER plans to inject gases such as Ar and Ne into the divertor area at flow rates up to 10% of total fueling gas flow rate in order to reduce the heat load onto the high heat flux components. Neutronic calculations indicate activities of Ar-41 (half life 1.83h) and Ne-23 (half life 37s) of 7.72E-02TBq/kg-Ar and 1.42E-07TBq/kg-Ne respectively under normal DT discharge conditions (1000s burn- 1200s dwell, 200Pa m^3/s of 50/50 DT fuel). Due to a relatively large tritium throughput (270g-T /1000s fueling), delayed bed type process technology is not applicable to the tokamak exhaust gas processing system, which is instead composed of a front-end permeator system (HDT separation from impurities for direct recirculation) and an impurity processing system (recovery of DT and minimization of environmental tritium exhaust). Due to the gamma-radiation background resulting from Ar-41 and Ne-23, commonly used real-time beta counters, such as ionization chambers have to be replaced by other analytical detectors. Plastic scintillation detector techniques may provide a solution to this problem. Because of lack of tritium-qualified data, R&D of plastic scintillation detectors with a wide tritium dynamic range (up to 10^7) is required.

4.3. Tritium removal/decontamination in areas that require hands-on maintenance access.

Decades of experience in tritium facilities world-wide indicates that extensive decontamination of tritium contaminated areas requiring decontamination factors of three orders of magnitude or more in a short time (days - week) by using volumetric gas replacement methods involving either air purge or recirculation of inert gas through atmosphere detritiation systems, will be difficult to achieve.

Literature data of experiments with stainless steel samples exposed to elemental tritium (13.3 kPa-HT, up to 35 days) at ambient temperature, show that the stainless steel surface becomes contaminated with HTO (>90% of tritium retention in the surface layer and bulk). Recent experience at JET (reported in this workshop) involved the decontamination of a neutral beam injection box after a minor coolant leakage, to allow workers access in protected suits. This experience showed that the tritium concentration in the atmosphere could be reduced from 100,000 DAC to the required ~20 DAC level (1 DAC = $3.1.10^5$ Bq/m^3) using air purging, but at the cost of producing large quantities of (low-level) tritiated water in an air detritiation system.

To allow hands-on maintenance access to the ITER tokamak port interspace (~45 m^3/each port), atmospheric tritium levels resulting from surface contamination, will be required to be reduced from levels expected to as high as 10^5 DAC by three to four orders of magnitude. The discussion showed that the only viable method available would be as per the JET experience. This, in the case of ITER, may lead to unacceptably large amounts of tritiated water which would require processing , and hence require concomitant increased tritium plant investment cost, to allow discharge into the environment. There is therefore a strong incentive to avoid altogether or limit to very low levels the contamination of the port interspace during maintenance operations. Hence it is recommended to develop corresponding maintenance procedures.

5. TRITIUM RETENTION AND REMOVAL IN TOKAMAKS.

W R Wampler presented results from analysis of in-vessel components from TFTR prior to DT operations and from DIII-D. The deuterium retained inside TFTR was estimated for each year of operation between 1987 and 1991 by ion-beam analysis of limiter tiles and wall coupons[10, 11]. Deuterium accumulation rates were 4 to 11 grams/year. Compared with the quantity of deuterium used to fuel the plasmas, the average fraction of deuterium retained in the vessel was 44% ± 17%. Of this, 41% was on the wall and 44% on the plasma-facing surface of the limiter and 15% in the gaps between limiter tiles. In DIII-D about 1 gram of D was retained on the lower divertor after 2000 plasmas, 40% of which was in the gaps between tiles[12]. The DIII-D DiMES probe showed net carbon deposition was highest on surfaces shadowed from direct ion flux, suggesting a high flux of carbon or hydrocarbon neutrals from the plasma boundary onto the divertor surface. In both TFTR and DIII-D co-deposition of D with C, with D/C ~ 0.2-0.4, was found to be the main mechanism for D retention. For this process the D accumulation is proportional to the net C deposition which equals the net C erosion. Potential ways to reduce retention would be to reduce the net erosion rate by using a high Z material, or a material which does not retain D when it redeposits. It is also advantageous to keep all surfaces exposed to flux from the plasma at a high temperature.

D Mueller gave a summary of the TFTR retention experience in the DT phase. Over 3.5 years of DT operations, 5 g tritium fuel was supplied to the plasma: ~3 g by neutral beam injection and ~2 g by direct gas puffs. During the three periods of DT plasma operations (excluding periods of active tritium removal), approximately 53% of the tritium fuel was retained in the torus[13-15], a retention fraction similar to that experienced with deuterium. Active tritium removal techniques of D or He-O glow discharges, pulse discharge cleaning and air ventilation were successful in removing substantial amounts of tritium in between periods of plasma operations. Of these, only

He-O glow had a removal rate that did not decrease with time. D and He soaks were found to be ineffective in removing tritium. The current in-vessel inventory, according to in-situ calibrated measurements of tritium delivery and exhaust from the vessel, is less than 1 g. Eleven bumper limiter tiles exposed to TFTR DT operations and dust samples have been removed for analysis to further quantify the retention due to co-deposition.

Y Hirooka reported analysis data on a C-C composite probe exposed to TFTR discharges with lithium pellet injection. The ion side surface exhibits a sign of codeposition of Li, O, C, D whereas the electron side indicates high-temperature erosion. Lithium is found to be incorporated in these codeposits in the form of oxide. On the electron side, the penetration of lithium is as deep as 12 microns, indicating rapid diffusion between parallel planes of the graphite crystalline. D Ruzic presented modeling of some lithium conditioned TFTR discharges and discussed the implications for tritium retention. It was found that Li lowers the reflectivity of graphite to energetic ions and atoms, absorbs molecular D and T but still allows a weak sputtered flux of D and T to be returned to the plasma.[16]

The tritium fueling of JET per discharge in the recent DTE1 campaign was an order of magnitude higher than TFTR due to the longer pulse length, larger size and extensive use of tritium gas puffing. P. Andrew reported that 35 g of tritium were introduced to the torus: 30 g by gas injection and 5 g by neutral beam delivery. The fraction of tritium retained was between 40% and 20%. While the isotopic composition of the plasma followed a model derived from the 1991 preliminary tritium experiment (PTE) [17], the tritium inventory was 3x higher than extrapolated from the PTE. The new divertor configuration has large relatively cold areas with line of sight to the plasma contact surfaces and it appears tritium is retained in saturated, co-deposited films on these surfaces. The in-vessel inventory was reduced by tokamak pulses by a factor of two, and by air ventilation at 150 C and at room temperature (over 1 gram recovered after 3 months). Deuterium glow, ECRH discharges or simple deuterium 'soaks' were ineffective.

Wall materials effects on JET were further explored by P. Coad. The changeover from an 'all-carbon' plasma facing surface to beryllium caused an immediate increase in fueling rate, but the number of D atoms retained per pulse was relatively unchanged. Most of the D redeposition was on the inner divertor leg. In 1995, the divertor geometry was changed with much larger carbon fiber reinforced carbon, (CFC) tiles and no significant poloidal shadowing effects. A 40 micron thick deposit saturated with D was found on the support structure and the shadowed edges of tiles adjacent to the inner corner of the divertor. Similar deposits, generated during the DTE1 experiment, are believed to be responsible for a higher than expected tritium in-vessel inventory . An analysis program for samples of tiles and flakes from the DTE1 experiment is planned.

D. Hillis reported results from a novel Penning Gauge diagnostic which measured the neutral tritium concentration in the subdivertor region of JET. This was found to be a sensitive measure of the tritium saturation in nearby surfaces, an important factor in determining the dynamics of tritium recycling throughout a discharge. Analysis of T uptake in the first discharge of the D to T changeover sequence, which began the DTE campaign in JET, related the T recycling coefficient to the subdivertor T fraction. The previous JET PTE model, with the addition of particle-induced detrapping processes was found to give reasonable agreement. These results for a single discharge were not sensitive to the presence of tritium in codeposited layers.

G. Federici reviewed tritium inventory and control in the ITER torus. New measurements of H-isotope retention in Be and W at ITER relevant conditions showed that tritium retention in these materials was less than previously anticipated.[18,19] However co-deposition of tritium with carbon remains a serious issue and under worst conditions the ITER administrative T inventory limit could be reached in a few weeks operation. This tritium represents a hazard as it can be easily mobilized in the event of an accident. Several options to remove tritium have been identified and R&D is in progress to determine their applicability to ITER. (see Sec. 6)

J. Brooks presented key modeling results on erosion and co-deposition in ITER. Encouraging agreement of code erosion predictions with results from the DiMES probe on DIII-D has been obtained. The analysis shows significant net erosion (50cm/burn-yr.) and tritium codeposition (≈ 10g/1000s pulse) for the carbon divertor target in detached and semidetached plasma regimes. Continuous operation at high (3-10%) duty cycles will require reliable and efficient tritium removal methods. The dominating process is the chemical erosion of carbon atoms and subsequent transport and deposition elsewhere due to the low electron temperature in detached plasmas. Beryllium erosion is similar to carbon but beryllium has 5x less tritium co-deposition. Tungsten is far superior. More analysis and data is needed to narrow the uncertainties for these critical processes in ITER.

J. Roth discussed the present understanding of the details of chemical erosion. Previous data showed large scatter and did not allow the extraction of firm extrapolations, especially of a flux dependence. New experimental data provides an improved understanding of the threshold energies of ion induced processes in chemical erosion. It could be shown that the discrepancies between weight loss and mass spectrometry lie in the emission of non-volatile radicals at low ion energies. At high fluxes, data from ASDEX Upgrade and the Berlin plasma generator support the trend of a decrease of the yield with increasing flux above $1\times10^{22}/m^2s$. The improved model now describes, within a factor two, all aspects of the available data.

M. Mayer reported on D retention in graphite doped with silicon or titanium to reduce its sputtering yield. The atomic fraction of D per target atom was 0.6 with SiC , and 0.15 with TiC compared with 0.4 in C alone. The presence of Si also tended to increase the temperature needed to remove D. The potential advantages of these materials needs to be balanced against their additional manufacturing cost.

The increase in plasma duration will be the largest change in operational conditions of ITER compared to present tokamaks. Plasma wall interactions during long pulse ($t > 15$ s) operation in Tore-Supra was reported by C. Grisolia. An uncontrolled rise in the electron density after 20-60 s was correlated with a rise in oxygen emission and hydrogen (protium) concentration. The increase was ascribed to photon induced water desorption from surfaces far from the plasma. A positive feedback loop resulted in which the desorbed H_2O raised the O concentration in the plasma and hence the electron density resulting in higher plasma radiation and in turn still more H_2O released. This phenomenon will clearly have to be controlled in ITER.

J. Hogan discussed modeling of wall saturation with the fuel gas in Tore Supra. In a series of shots with an initially depleted (conditioned) wall, wall saturation resulted in a disruption after 15 shots. When particle induced detrapping was added to the JET PTE model, predictions were in reasonable agreement with the observed wall saturation timescale. Further work was presented relating typical impurity production processes (physical and chemical sputtering, radiation-enhanced sublimation) in inner-wall limited tokamaks (TFTR) with likely co-deposition patterns. The very localized nature of redeposition from radiation-enhanced sublimation and chemical sputtering sources was found to suggest that long term co-deposition patterns should be due to the longer spatial co-deposition range arising from physically re-sputtered material.

P. Franzen presented measurements of deuterium in the ASDEX Upgrade divertor tiles by thermal desorption spectroscopy, nuclear reaction analysis, secondary ion mass spectroscopy and accelerator mass spectroscopy. Deuterium retention in graphite divertor tiles was found to be dominated by diffusion out of the co-deposited/implanted layer with a total inventory exceeding the near surface (< 25 micron) inventory by factors of 10 to 100.[20] This is in contrast to the results presented by Wampler and Penzhorn from TFTR and JET tiles and may be related to the higher target tile temperatures in ASDEX. Retention in the outer divertor tungsten tiles was a factor 10 lower than in graphite; here the near surface inventory has the same order of magnitude as the total inventory. In contrast, the inner divertor tungsten tiles are covered with a several micrometer thick co-deposited C-H layer.

Erosion, co-deposition and redeposition in DIII-D was reported by D. Whyte. A materials sample probe "DiMES" enables single shot measurement to be made of a plasma facing surface and experiments were performed benchmarking erosion/deposition codes used to predict retention in ITER (see also Brooks presentation above). Modeling of low recycling attached divertor plasmas was successful in reproducing the measured erosion in DIII-D. The hydrogenic inventory buildup rate is determined by erosion in the outer divertor and subsequent redeposition in the inner leg. Leading edges eroded rapidly and formed difficult to clean co-deposited layers in adjacent shadowed regions. Preliminary results showed reduced erosion/redeposition rates in detached plasmas and the elimination of erosion for tungsten.

V. Philipps discussed the use of oxygen to remove thick carbon deposits on plasma facing components of TEXTOR. The plasma facing surface in TEXTOR is carbon, typically heated to 430°- 620° K. At lower filling pressure (0.01 mbar) most of the oxygen was absorbed on the wall, the remaining 10-20% oxidized the deposits to CO and CO_2 which was then released. At higher filling pressures (0.3 mbar) the fraction of oxygen adsorbed decreased whereas the fraction of CO formation was about constant and CO_2 increased. The oxygen treatment contaminated subsequent plasmas and caused them to disrupt. However, a 10 minute deuterium glow discharge was successful in reducing the oxygen level and restoring normal plasma operation.

6. LABORATORY EXPERIMENTS ON TRITIUM REMOVAL.

Efficient tritium removal is critical to ITER's program since under worst retention conditions, the ITER administrative tritium inventory limit could be reached in a few weeks of operation. Significant progress on R&D aimed at removing tritium from tokamaks by oxidation and other methods was reported at the workshop. T. Haasz described the removal by oxidation of co-deposited carbon/deuterium films on TFTR and JET tiles on exposure to oxygen, air, and water, at elevated temperatures [250-350° C][21, 22]. At a temperature of 250° C, similar to that planned for the ITER divertor, D removal rates due to oxygen were 2-20 $\times 10^{21}$ D/m^2/hour. The oxygen exposure method of T removal would also be effective on surfaces far from the plasma. The erosion rate is much lower on bare graphite. This is a potentially attractive removal technique for ITER; however the collateral effects of the oxygen on in-vessel components and the tritium plant need to be investigated (see Sec. 4.1).

R.-D. Penzhorn presented measurements of the surface and bulk tritium concentrations in graphite tiles from JET and TFTR performed at the Karlsruhe Research Center. The tritium originated from D(d,p)T nuclear reactions during deuterium operations. More than 98% of the tritium was found to be in a 50 micron surface layer with significant amounts on the sides of the tiles (similar to the results of Wampler [Sec. 5]). An effective detritiation technique consisted of moist oxygen or air treatment at 350-400° C. The detritiation factor depended mostly on temperature and not on exposure time. Very high detritiation was accomplished by treating the tiles with an open flame for a few seconds.

A paper by S. Alberici was presented describing studies of the thermal reaction of oxygen with co-deposited layers from TEXTOR. Sufficient D release was obtained at temperatures above 573 K. The oxygen partial pressure was found to be an important influence on the D-release rate.

Work by K. Maruyama et al. [23] on a comparison of the erosion behavior of plasma-deposited soft and hard a-C:D films by heat treatment in air was reported. The oxidation behavior depends strongly on the microscopic film structure which is mainly determined by the initial hydrogen isotope content of the films. In hard a-C:D films (H/C = 0.4) erosion is limited by oxygen diffusion [24]. Soft films are permeable to oxygen and oxidation proceeds in the whole film thickness. Different erosion processes were identified leading to a preferential release of hydrogen isotopes at temperatures about 100 K lower than that necessary for complete removal of the layers.

A significant constraint in tritium removal techniques for ITER is the presence of the toroidal field. The superconducting coils generating the field are rated for 1,000 on/off cycles in their lifetime so cleaning techniques performed (e.g. weekly) will need to be done with the toroidal field on. D Cowgill presented an experimental study of C-H co-deposit removal by O containing plasmas in the presence of a magnetic field. Preliminary measurements showed atomic oxygen from a low pressure He-O discharge (50kW/m^3) crossing field lines and eroding carbon at rates in excess of 0.05 microns/h. This erosion rate was believed to be limited by the thermal (photon-assisted) COx desorption rate. There were also indications that atomic oxygen can migrate from oxidized surfaces, allowing it to reach around corners and penetrate into confined regions.

A paper by B. Landkammer et al.[25] on the removal of co-deposited layers by ECR discharge cleaning was presented. Oxygen, water, a oxygen/hydrogen mixture (O_2 : H_2 = 1:2), deuterium, and hydrogen were used as working gases. Erosion rates increase in all cases with increasing substrate temperature (factor of 3-4, if T is increased from 300 to 650 K) and with increasing ion energy (up to a factor of 10, if bias potential is increased from 0 to 260 V). Pure oxygen turned out to be most effective. The erosion rates depend critically on the film structure. Soft films (H/C about 1.2) were eroded a factor of two faster than hard films (H/C about 0.5).

C. Skinner presented a novel tritium removal concept using laser surface heating.[26] Substantial amounts of co-deposited tritium might be removed by laser surface heating in an overnight cleanup in ITER. Modeling showed that exposure to a multi-kw/cm2 laser flux for approximately 10 ms raised the temperature of a 50 micron co-deposited layer to 1,000-2,000 K, a temperature at which tritium would be released. Improved wall conditioning may be a significant side benefit. This method also avoids potential collateral effects associated with the use of oxygen. Experimental tests are needed.

A tritium removal technique using CO_2 pellet cleaning was reported by R. Nygren. With CO_2 pellet blast cleaning, widely used in the semiconductor and nuclear industries, pellets sublime on impact leaving no chemical or pellet residue. Cleaning can occur by particle abrasion, thermal shock, and, according to Hills[27], by chemical solvation in a transient liquid CO_2 film that forms upon pellet impact. The procedure works best for soft films that are soluble in CO_2 and lie on hard substrates. CO_2 pellet cleaning has successfully cleaned Be films from JET[28]. Preliminary tests using existing equipment at Sandia did remove the surface layer from a graphite tile from the DIII-D tokamak but also severely eroded the tile. Better control of the removal rate by using softer pellets is being explored.

7. DUST, SAFETY ISSUES.

As the design of ITER has matured, increased attention has been placed on its engineering aspects, including safety and disruptions[29-31]. The environmental impact of DT fusion reactors will come under greater scrutiny as ITER moves toward the site selection stage. Erosion and consequent dust generation, will become, for the first time, a significant operational factor in ITER, due to its much longer pulse length. However a basic understanding of dust generation mechanisms and quantitative in-vessel diagnostics remain to be developed.

K McCarthy discussed the radiological and chemical hazards associated with tokamak dust and analyses in progress of the particle size and chemical reactivity of tokamak dust from C-Mod, DIII-D, TFTR and the SIRENS disruption simulator. The statistical distribution of dust particles centered typically around one micron. Much work remains to be done on correlating dust samples with tokamak dust generation mechanisms.

Monte Carlo modeling of the biological and radiological interactions of tritiated particles in the lung was reported by R. B. Richardson. The dose to alveoli from particulates below one micron diameter is the most significant radiation hazard to the lung. Further *in-vitro* and *in-vivo* characterization of tritiated materials is needed to develop dosimetric models. Monte Carlo methods were also employed in another study which estimated the dose to the cell nucleus from tritiated macromolecules. Organically bound tritium in the environment is another potential hazard of concern. A model was described for estimating the dose from organically bound tritium derived from diet.

A. Hassanein reported results from models of erosion and dust production during transient plasma events such as disruptions and ELMs. Severe erosion from melt layer losses of metallic materials occurs during plasma instabilities[32] in good agreement with recent simulation experiments.[33,34] Carbon based materials may also suffer significant erosion due to brittle destruction. Models are being developed to predict tritium behavior in redeposited and splashed plasma facing materials. More reactor relevant experiments, however, are needed before a final selection of materials can be made.

R. Little discussed analysis of the consequences of various potential accident events in ITER. None of the credible accidents resulted in public radiation exposure greater than regulatory limits but increased margins were desirable in a few cases. There are no adequate predictions at present on dust production rates. Safety limits have been established on the in-vessel dust and tritium inventories but, at present, there are no proven in-vessel dust or tritium diagnostics. More emphasis was recommended on developing diagnostics and methods for dust removal, as well as modeling and experimental studies of dust production. A related administrative issue is that at present no group has responsibility for this thorny area.

8. TRITIUM RETENTION, REMOVAL AND DUST ISSUES.

(Facilitators G. Federici and C. Skinner).

It was clear at the workshop that recent results from fusion facilities around the world have greatly expanded our knowledge of tritium behavior in a fusion environment. Advances span large scale tritium handling in the tritium plant to details of atomic processes at the plasma wall. Non-the-less serious issues remain to be resolved for ITER, issues that may appear intimidating when considered in the context of a DT reactor operating at a commercially relevant duty cycle. Two afternoon discussion sections were held during the Workshop to identify the unresolved in-vessel tritium and dust issues and to stimulate some 'brainstorming' on potential R&D avenues for their resolution. We first give a summary of the background of the choice of plasma facing materials for ITER and the fundamental reasons why the issues are much more serious for ITER than existing plasma devices.

In the present ITER design, Carbon Fiber reinforced Carbon (CFC) is used only for the divertor plate near the separatrix strike points, tungsten is used elsewhere in the divertor, and beryllium is used on the first wall. The utilization of CFC at the divertor target is viewed as indispensable to withstand the very high thermal loads expected during attached-plasma transients and disruptions and because it can tolerate high surface temperatures and sublimes rather than melts during the thermal quench. Tungsten has been chosen for the divertor chamber walls and lower baffles surfaces to minimize the sputtering from charge exchange neutrals because the sputtering threshold energy is much higher than the threshold for graphite or beryllium. Be has been chosen for the remainder of the wall surface where sputtering is not as large a concern. Be minimizes the area of the vessel covered with a high Z material such as tungsten which could be deleterious if it contaminated the plasma. Beryllium is a low Z metal and has the potential for in-situ repair with plasma spray. The experience with Be in JET indicates that it may have operational advantages as a getter for O as well.

It was clear from the beginning that the use of C-based materials would exacerbate the problem of tritium codeposition. Additionally, the simultaneous use of Be, C and W introduces significant uncertainties into the operation of a tokamak like ITER. Erosion and redeposition phenomena are expected to generate complex mixed-material layers whose behavior in terms of erosion, tritium retention and removal cannot be predicted from present laboratory experiments. While the development of models for simulating these phenomena has only just begun, current laboratory data do indicate that these effects can be significant and that they are strongly influenced by surface chemistry.

The fundamental change that is driving these concerns is the longer pulse length and cumulative run-time of ITER, together with the higher edge density and more intense disruptions. The ITER plasma duration is two to three orders of magnitude longer than present machines and the resulting erosion is projected to be on a cm scale. This erosion generates dust and high levels of tritium in co-deposited layers. In present experiments, by contrast, the net erosion is barely measurable. For the first time in fusion research, the ITER experimental program must address the physics of the erosion mechanisms and the physics of how the eroded material is transported and redeposited. Focused R&D is needed to resolve the issues.

8.1. ITER relevant modeling

It is necessary and urgent to develop a quantitative understanding of tritium retention and develop models of retention that can be validated by comparing the predicted retention with that experienced over the wide range of tokamak conditions obtained in TFTR and JET. These models would form the best basis for predicting the retention in ITER. To date there have not been global models of retention that have predicted the observed levels of retention in tokamaks. Such benchmarking is common in other areas of plasma physics, and helps build confidence in expectations of tokamak performance in future devices. There is a coupled need to develop more detailed tokamak edge diagnostics (see sec. 8.3). Continued and increased communication/ collaboration among plasma experimentalists, modellers and the materials community is necessary.

8.1.1 A primary need is to provide a physically realistic explanation of the amount of tritium and deuterium retained in codeposited layers of tokamaks, in particular in JET and TFTR, their location and the original source of C, so that reliable extrapolations to ITER can be made. An attempt to model, to the best extent possible, the recent results of the overall retention JET and TFTR and especially retention in the "flakes" observed at the bottom of the inner leg of JET are seen as a high leverage items. If we cannot explain these results, the confidence in ITER predictions will remain highly questionable.

8.1.2 Plasma linear simulators devices such as PSI-1 (IPP Berlin) and PISCES (UCSD) are capable of delivering high particle fluxes and are being used to investigate chemical erosion of CFC. It was recommended that, in conjunction with further development of experimental plasma controls and diagnostics, plasma modeling be undertaken to guide the direction of future experiments. The interpretation of the experiments designed to reveal flux variations is presently complicated by changes in the plasma conditions with flux. A comprehensive mapping of the plasma, coupled with particle transport and redeposition calculations, will give net erosion predictions which, when compared with mass loss measurements, could be used to derive erosion yields.

8.1.3 Modeling of mixed-material effects occurring under ITER relevant conditions is very complex. Experiments require careful considerations and planning before definitive conclusions can be made. There should be more attention to co-ordinate specific mixed-material experiments, including tokamak results, and complementary modeling to interpret the results and to gain insight on this complex subject. Most of the existing models only include surface collisional effects. Diffusion as well as chemistry effects such as segregation, mixing, etc., are not included in the models so that high temperature bombardments which are relevant for ITER cannot be properly handled at the moment.

8.1.4 Better modeling is needed to estimate the dust inventory in ITER. There is a fundamental lack of understanding of mechanisms which lead to the production of dust in tokamaks. Although models for particle formation (different forms of nucleation) have been studied extensively in the aerosol science literature, there has been no effort yet to determine the applicability of these models to the production of dust in tokamaks. Models for erosion/sputtering and disruption effects exist and have been partially validated versus experimental data. There is a need to couple these tools with models for film and dust formation (e.g., physical adsorption, nucleation physics/condensation). Models should be validated against existing tokamak experience and disruption simulators to the largest extent possible.

8.2. Experiments with linear plasmas and disruption simulators

8.2.1. The current chemical erosion database for C at divertor-relevant particle fluxes is insufficient to draw confident conclusions, as to the yields and the nature of the flux dependence (if any) of the chemical erosion yield of carbon for conditions of direct relevance to ITER (low energies, high surface temperature and high particle flux). Experiments to determine the mechanism and the magnitude of chemical sputtering and its dependence on plasma parameters (energy and flux - up to few times 10^{23} m^2/s) are urgently needed.

8.2.2. In present-day tokamaks, erosion redeposition processes yield only very shallow erosion and thin erosion deposits originating from specific areas. In ITER, for the most severe assumptions, the erosion lifetime of the plasma-facing components is anticipated to be sufficiently short that several replacements will be required during the ITER lifetime. The assumption that redeposited carbon films behave identically to as-fabricated materials needs to be further investigated. Prompt redeposition plays a dominant role in minimizing the net erosion (net-erosion = gross-erosion *minus* prompt redeposition) of regions subject to high-particle fluxes (i.e., divertor target near the strike points); however, the surfaces which have been sputter-eroded may develop features that change the sputtering yield. Therefore, more measurements are clearly warranted for the properties of tokamak exposed materials and of laboratory redeposited films created at high flux rates and/ or at elevated substrate temperatures.

8.2.3. Effort should be continued in disruption simulation facilities to better characterize particulates generated from the plasma gun for ITER-relevant materials and mixtures thereof. Plasma simulators offer an attractive location to study the issue of dust production resulting strictly from the erosion of wall materials. Work performed in plasma simulators complements investigations of dust production in disruption devices. A co-ordinated approach in these complementary facilities could help unravel the importance of the various contributors to dust production in tokamaks.

8.2.4. D(T) retention experiments should be conducted in neutron-irradiated Be samples together with permeation engineering tests of duplex structures mimicking the actual armor material, the interface joint and the copper-alloy heat-sink substrate as designed for ITER.

8.2.5. There is a need to continue experiments for "basic" understanding of mixed-material effects. The experimental database remains very meager in this area and specific experiments and theoretical simulations need to be urgently planned and carried out to investigate effects on erosion rates, D and T uptake and release for ITER relevant conditions (see also remark in Sec. 8.1).

8.2.6. Vigorous investigations should continue in bench-top experiments on promising identified options for the in-situ removal of the codeposited layers (e.g., baking in O_2 atmosphere, low-pressure plasma discharges of ECR, ICR type, with some O etc.). It is clear, from laboratory experiments that tritium can be removed from thick codeposits of carbon and tritium by heating the deposits in air or oxygen or using plasma discharges with O. However the efficient removal of tritium from the surface of in-vessel components of ITER remains speculative. Several questions concerning the required removal efficiency remain and the suitable techniques need to be demonstrated in tokamaks (see Sec. 8.3).

8.2.7 Encouragingly, TEXTOR did not experience any long term consequences after the use of oxygen to remove deuterium. However, there is a need to investigate collateral damage and implications of using oxygen on surrounding materials, bolts, joints welds, etc.

8.2.8 Techniques for oxygen free tritium removal that avoid the potential complications of oxygen in the torus and tokamak exhaust detritiation plant should continue to be pursued. These include laser surface heating and CO_2 pellet ablation.

8.3. Tokamak experiments

Bench-top experiments, although very useful to probe fundamental processes, cannot provide alone the level of understanding required to predict all the complex interactions in tokamaks. Therefore, tokamak experiments are urgently needed to provide relevant integrated data in the areas of tritium retention and removal, gross and net erosion, wall conditioning, and dust production, accumulation and removal. To make such experiments useful there must of course be appropriate measurements and diagnostics. There is in particular a need for *dedicated* operation time, diagnostics and instrumentation, and in-vessel reconfigurations in tokamaks dedicated to addressing specific T issues of high relevance for ITER such as net erosion rates and redistribution of eroded material, mixed-materials effects, etc. The complex and varied discharge history in tokamaks makes 'archeological' comparisons of post-campaign in-vessel components with retention models of limited utility with regard to transient behavior during a single long pulse discharge as envisioned for ITER. In-vessel, real-time diagnostics are necessary to better relate tokamak experience to the models. Such techniques exist (see Whyte [sec. 5] and refs: [35-39]) but are not in routine use. Development of novel diagnostic techniques would be welcome. Real time measurements of erosion rates and retention under different operational conditions (startup/shutdown, disruptions, attached/detached divertor, high power operations) are needed. To be able to use the results to benchmark predictions it will be necessary to improve the modeling, do good bench-marking experiments, and improve communication/ collaboration among plasma experimentalists, modellers and the materials community.

ITER is unique in that it is designed to have a carbon divertor plate with a metallic wall. Present divertor tokamaks (except for Alcator C-Mod) have C divertor plates and walls. They are also operated with a variety of plasma wall interaction and plasma conditions that make it difficult to relate the results of a post-mortem analysis of samples to specific edge processes.

8.3.1 There is an urgent need to conduct experiments under controlled plasma conditions and perform pulse by pulse measurements installing film thickness diagnostics wherever possible. Measurements should be repeated as a function of divertor density, power to the target, and other plasma parameters of interest.

8.3.2 As long as we do not have experience on tokamaks with wall/divertor configurations similar to ITER, we are forced to place a great deal of trust in models and analysis to predict erosion/codeposition and to hope that they will be borne out in the future. The development of an understanding of plasma surface interactions in this complex environment requires tests on tokamaks with impurities and wall materials that would closely mirror the ITER conditions (i.e. Be walls with C and W in the divertor). Such experiments would help answer some of the relevant questions and address synergistic effects that cannot be easily determined in lab simulation experiments and one-material lined tokamaks, i.e., (i) the quantitative nature, magnitude and location of tritium codeposition and dust to be expected in ITER; (ii) the origin of carbon impurities in the plasma; (iii) the importance of chemical sputtering; (iv) effects on hydrogen (tritium) retention in the wall; (v) mixed-materials effects; (vi) operational aspects of using beryllium on the first wall. Such experiments would provide a realistic test bed for model development and validation and would lead to there being a much smaller step between present understanding of these key subjects and the expected experience on ITER.

8.3.3 Present tokamak operation with tritium indicates that present, zero-magnetic field conditioning methods (glow discharge cleaning) are marginal in controlling inventory within allowable limits and would not be adequate for the planned ITER duty cycle. However, baking in oxygen atmosphere or ECR/ ICR plasma discharges with O shows good potential. Questions remain regarding (i) the cleaning duration required (ii) the conditioning required for subsequent high performance plasma operation, (iii) the access to hidden regions and (iv) the exhaust handling by the tritium plant. Tritium removal rates measured in small lab experiments may not scale to the ITER situation. There is an urgent need to test these removal methods in a real tokamak environment with a realistically large complicated vacuum chamber with instrumented trap sites.

8.3.4 Experiments designed to separate transient tritium retention and erosion effects in start-up & shut-down from those of steady-state plasma operation are also needed, so we can more reliably scale to the ITER pulse length. Variable pulse length experiments with proper instrumentation would be useful.

8.3.5 There is a need to test (see Sec. 8.7) erosion, tritium and dust diagnostics for in-situ measurements. Here one should start from the tokamak experience available to date.

8.3.6 There is a need to test critical ITER component configurations whose codeposition properties have not been characterized. An area of emerging concern, for example, is the new divertor liner protecting the pumping duct in the divertor private region, where, non-line-of sight deposition processes may lead to thick deposits saturated with D and T in cold regions. Modeling is also required to establish a quantitative understanding of this issue.

8.3.7 As far as dust is concerned, there is a strong need to improve the knowledge and understanding on mechanisms leading to dust formation and the determination of dust location and its survivability on the plasma facing surface in tokamaks. This requires experiments to determine dust production rates associated with well characterized plasma conditions (e.g., power density, number of disruptions, disruption energy, etc.) and target conditions (surface temperature, outgoing eroded or vaporized flux). Dedicated dust collection experiments during normal discharges and disruption conditions should be continued [see for example talk by Whyte (Sec. 5) where the DiMES probe in DIII-D was used to make interesting studies related to dust and debris]. It is important that rates be related to the conditions by which the dust has been generated. Additionally, collection and analysis of dust and flakes during major tokamak openings should be planned and continued. It may be useful to check dust production and levels in existing stellarators that operate almost completely disruption free. This may also help separate the erosion vs. disruption source terms. Demonstrations of efficient dust removal from tokamaks without in-vessel personnel intervention are critically needed to develop dust mitigation strategies for ITER.

8.4. Laboratory studies

8.4.1 Post-mortem analysis of JET tiles and flakes and TFTR tiles to map/ characterize T, D, H, chemical composition etc. is planned. Potential dust generation by the spalling and breakup of flakes should be quantified.

8.4.2 Work should be carried out to continue characterization of dust (size, effective surface area (BET), chemical composition, D and T content). A combination of BET and chemical reactivity experiments must be performed, instead of only BET measurements and calculations. A large amount of dust (~ grams) would be needed to perform chemical reactivity tests relative to the mg amounts normally used for BET measurements. A uniform measurement protocol should be used to make the data from different tokamaks more comparable and to ensure the validity of the results.

8.4.3 Safety related studies of the time and temperature dependence of tritium mobilization from co-deposited layers and dust would be useful. These should include the effect of steam as in potential accident scenarios involving loss of coolant.

8.5. Design improvements

Some new important experimental information presented at this workshop from the JET MK-II divertor and other divertor tokamaks would impact the design of the ITER divertor, in particular the unequivocal existence of a large redepository of D(T) codeposited films and flakes localized in regions of the inner divertor target. Based on these data, new design solutions should be explored to mitigate the consequences of codeposition. Dedicated cold "catchers" which could be periodically heated at high temperature with helium auxiliary lines to recover the tritium, or hot removable "chimneys" are just two examples of possible mitigating solutions.

There was also consensus that if carbon were eliminated from the ITER divertor the situation as far as tritium inventory is concerned could be radically different and the control of the tritium inventory would be much more manageable. However, dust from Be and W might still be an issue. In any case, it would be prudent to consider design options without C. The primary candidate in lieu of C for high heat flux regions is W. Development of a W target is being considered. Specific requirements on disruption loads and frequency need to be defined. It is important that the plasma physics community recognize that the projected levels of thermal loads expected during attached-plasma transients and disruptions lead directly to the use of carbon plasma facing components which in turn will require the allocation of a significant fraction of ITER's operational schedule for detritiation. Efforts to reduce transients and mitigate disruptions must continue at the highest priority.

8.6. Dust Removal

While dust is not an issue for operation and safety of present-day tokamaks, it represents a potential safety hazard for ITER. It can cause steam induced hydrogen explosions, and can increase the spread of radioactivity during an accident involving a vent. The study of dust production and the formation and erosion of mixed materials has just begun (see for example Ref. 29). The fraction of dust particles generated that are burned up the plasma needs to be modeled.

A variety of dust capture and removal devices will be required in conjunction with various in-vessel entry devices, to satisfy the anticipated requirements. The only device that has shown promise for removal of dust under vacuum and magnetic fields is the electro-static transport device being developed by the Japanese Home Team, and it has been successfully demonstrated on a laboratory scale. Based on present available experience at least three specific techniques need to be considered for in-torus removal:

8.6.1 "Gas agitation and vent (GAAV)". This needs practical laboratory tests which can be scaled to ITER to determine how much turbulence is needed to pump out dust before it settles. The chemical activity and hence, allowed dust inventory in ITER depends on whether the dust is on a 'hot' or 'cold' surface. Given a definition of hot surfaces and cold surfaces, can the dust be moved preferentially from hot to cold?

8.6.2 Baking at 350°C in O_2, or O^+ combined with ICR or ECR has the potential to passivate dust and/or remove it, and is strongly coupled to T removal. Initial R&D is in progress.

8.6.3 The electrostatic conveyor needs to have a more specific design study to find out how it could be incorporated in the existing overall design.

8.7. Tritium and Dust Diagnostics

For ITER, the major diagnostic needs related to tritium retention and erosion considerations, are to measure the total amount of tritium inside the vacuum vessel to a small fraction of a kg, and to measure the quantities of dust (Be, C, W) in various locations to small fractions of 10 kg. There is an additional need to measure the amounts of H and D to comply with the 11 kg limit, set to avoid the possibility of a steam induced hydrogen explosion. Since these measurements are needed for operational protection and safety it might not be necessary to measure more frequently than daily. To understand and track the processes, measurements during a pulse or after a disruption will be desirable. At present it is expected that no single measurement will be adequate and several different partial and indirect techniques will need to be correlated and cross-calibrated. An obvious example is that there will be vents of the vacuum vessel to repair or change components, At that time many samples can be collected, erosion inspected, dust removed etc. In addition, removal techniques which will have been developed can be used as diagnostics if calibrations are available. Several classes of diagnostics were qualitatively discussed during the meeting.

8.7.1 The measurement of erosion and redeposition is basic though it does not directly measure tritium retained or dust produced. There need to be multiple devices to measure the location of the first wall to a few mm accuracy[40], to look at damage, and to measure wall temperatures. Such devices have the advantage of being able to look at large fractions of the plasma-facing surfaces but the disadvantage of not being able to see details or non-plasma facing components, especially in the divertor where much of the co-deposition will occur.

8.7.2 Local measurements by sampling are possible. It is straightforward in principle to remove samples exposed to plasma flux and analyze them for tritium or dust. The DIII-D DiMES probe is a good example of this class. Speculative possibilities include sending a robot inside to collect and analyze samples.

8.7.3 Specific local in-situ measurements are possible. Examples are :

a) collect dust in a small container and measure the level

b) heat a surface by laser and measure the amount of tritium released

c) measure laser scattering by suspended dust particles

d) measure the increase in a film thickness by change in resistivity.

There is a strong need for some innovative R&D, testing and design for diagnostics relevant to these topics. The technology bases for plasma coating technology, aerosols, clean rooms etc. may offer some useful concepts and need to be investigated. An significant parameter is the threshold size of particles that will be suspended in vacuum. An important issue is the relation between flaking of deposits (as found in the JET divertor area) and the generation of dust.

Various options should be identified and proof of principle and proof of performance should be demonstrated in present-day tokamaks and during the hydrogen phase of ITER. It was agreed that a strategy should be defined that is based on using the hydrogen operation phase of ITER to gather as much information as possible on the problem of redeposition. Information is needed both on the extent and rate of the redeposition, as well as the locations where redeposition occurs. There is a need to articulate an experimental campaign during ITER protium plasmas to determine location and extent of codeposited layers and dust and to demonstrate 95% recovery of trace D (or T). Diagnostic requirements for this campaign must be set. It should be noted though that the process of redeposition and dust production occuring during the hydrogen phase of ITER is expected to be quantitatively different from that during D-T operation, basically due to the differences in plasma parameters, power crossing the separatrix, possible isotopic dependence of chemical erosion, etc.. Nonetheless, by understanding both during hydrogen operation, considerably more confidence will be gained for extrapolating the amount of tritium inside ITER during D-T operation, based on localized measurements.

9. IMPLICATIONS FOR FUTURE COMMERCIAL REACTORS.

The issues discussed above are generic to any DT fusion machine and must be taken seriously in any conceptual path to a commercial reactor. For example, current experience is that weeks of tritium clean up are necessary in JET and TFTR after a cumulative 10-30 minutes of DT operations. In contrast, a commercial reactor would need to operate almost continuously with little time available for cleanup. Retention fractions less than 0.1% are required. Carbon, because of its high co-deposition rate appears to be impractical in a DT machine with a commercially relevant duty cycle. High Z materials, while attractive because of their low sputtering coefficients, must be prevented from contaminating the plasma core, and pose activation issues. Clearly it is critical to minimize the heat load on plasma facing components to allow greater flexibility in material choice. This implies drastic reductions in disruption frequency and high power transients. These considerations should be an important element in research on innovative fusion concepts. Often the economic advantage of higher core power densities tends to increase wall loadings, however innovative initiatives on plasma facing materials such as the Advanced Liquid Plasma Surface (ALPS) and Advanced Power Extraction System (APEX) efforts are underway in US.

10. CONCLUSIONS AND SUMMARY

The realization of practical fusion presents one of the grand engineering challenges of the 21st century and tritium issues are among the most challenging as they demand an integrated solution to erosion and co-deposition of tritium in the radiological environment of the first wall. High confidence resolution of these issues is critical to meeting the anticipated schedule for ITER plasma operations, public acceptance of the ITER site and ultimately public support of magnetic fusion as a practical, environmentally attractive power source. For fusion to realize its promise of an inexhaustible energy source that will sustain future civilization, R&D in fusion technology must be maintained to guide the development of ITER and make progress toward future commercial reactors.

Fusion is a collaborative endeavor crossing both national and disciplinary boundaries. An integrated solution to all physics and engineering issues is required for success. Although the issues at hand may, at times, seem intimidating, it is in fact a sign of the progress and maturity of fusion science that engineering concerns are coming to the fore. After the demonstration of fusion power on TFTR and JET there is no longer a serious debate on the scientific feasibility of fusion. The history of human creativity and ingenuity in overcoming technical challenges provides reassurance that with sufficient will and resources, the task of providing for the energy needs of future society can be achieved with fusion energy.

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