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ABSTRACT

The JET machine was built from the beginning to perform DT studies and is for the next decade the only machine with DT capability. JET is thus the tokamak best suited for studies of various nuclear technology issues that will be of importance for future DT fuelled machines. The Active Gas Handling System (AGHS) built for the supply and reprocessing of tritium is still in operation to detritiate the exhaust gases and to provide the necessary ventilation during vessel openings because >1 g of tritium remain in the machine even four years after the DTE1 campaign. The tritium is bound in co-deposited layers and flakes formed mainly on the shadowed inner part of the divertor, and in the bulk of tiles. Areas of erosion and deposition are being identified and preliminary models for the material movement developed. Characterisation of the properties of tiles and flakes is very important to develop the best handling techniques. Due to the activation of the machine and the use of Be and T entry to the torus is limited by the need to minimise worker doses and is currently only possible in full pressurised suits. Work inside it is therefore performed with remote handling tools where practicable, e.g. the removal of the divertor septum and the collection of dust and flakes from areas below the divertor to achieve a more accurate tritium balance. Techniques need to be developed to reduce the tritium content in waste to permit disposal as Low Level Waste (<12kBq/g) and to recover tritium for re-use. Validation of activation models, understanding of the risks due to inhalation of tritiated particles and the local environmental impact of tritium are further important aspects. The paper presents various nuclear issues studied at JET due to activation and tritium.

1. INTRODUCTION

The JET Tokamak was from the very beginning designed to handle large amounts of tritium, to cope with the activation produced by DT neutrons and to make modifications of the vessel interior with remote handling tools. JET will be at least for the next decade the only machine with true DT capability and is the best suited test facility to study nuclear technology issues for ITER. JET has successfully and safely carried out an extended DT experiment, the first fully remote exchange of a divertor in a hostile environment, and operated safely with beryllium and tritium in the machine for many years. This has been due to the staff and specialised equipment developed by the Active Gas Handling, Remote Handling, Waste Management, Health Physic and other groups.

This paper will present details of various nuclear aspects due to tritium retention and materials activation that had to be managed during and after the operations with tritium.

2. THE DEUTERIUM-TRITIUM EXPERIMENT (DTE)

In 1997 the Deuterium-Tritium Experiment [1, 2, 3] known as DTE1 was performed at JET which achieved world records in peak power of 16.1 MW ($5.7\ 10^{18}$ n/s) and fusion energy of 21.7 MJ ($7.7\ 10^{18}$ n). The first ever DT fusion experiment (PTE) [4] performed at JET in 1991 produced significant fusion energy (2MJ) in a hot-ion plasma containing 11% tritium in deuterium.

2.1. THE JET ACTIVE GAS HANDLING SYSTEM (AGHS)

Fig. 1 shows the tritium recycling scheme at JET. The tokamak exhaust gases, mainly tritium and deuterium with impurities (hydrocarbons, N_2, CO, CO_2, water, He, etc.), are pumped by the Cryogenic Forevacuum (CF). Controlled regeneration of the CF cryo-pumps releases first He, then the hydrogen isotopes Q_2 (Q stands for protium (H), deuterium (D) and tritium (T)) and finally the impurities which are detritiated in the Impurity Processing (IP) system. Q_2 is stored in the Intermediate Storage (IS) system and subsequently moved into one of the two isotope separation systems: Gas Chromatography (GC) or Cryo-Distillation (CD). The isotopically pure products D_2 and T_2 are transferred to the Product Storage (PS) system from where they are supplied via the Gas Introduction (GI) and the Gas Distribution (GD) systems to the users (the Neutral Injection Beam Systems NIB4 and NIB8 and the Gas Puffing system GIM15). The Analytical System (AN), which characterises the mixtures to be processed, accepts

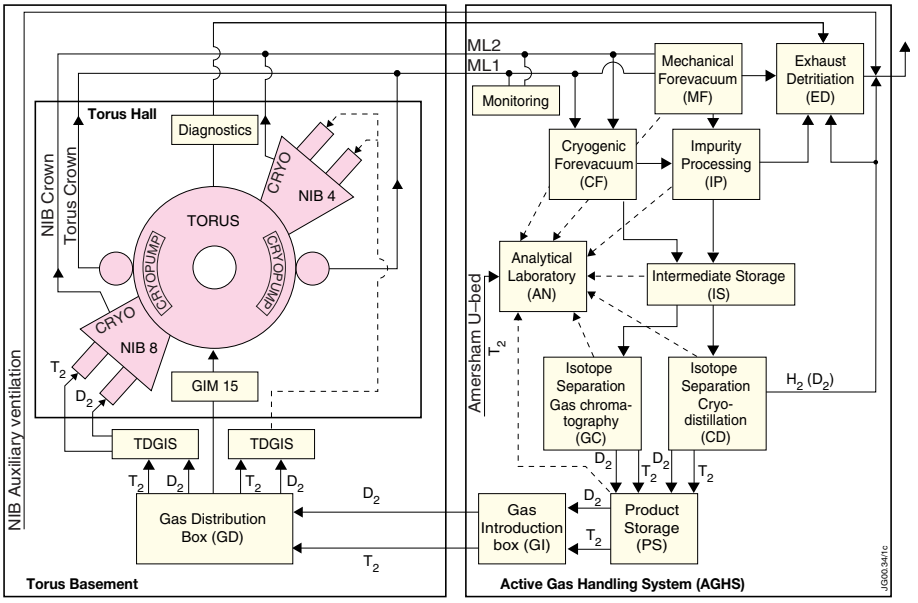

Fig. 1: Tritium recycling at JET

samples via interconnecting lines from other subsystems. The Mechanical Forevacuum (MF) system is mainly used to evacuate the torus or other systems after ventilation. Almost all gases have to pass through the Exhaust Detritiation System (EDS) before their release into the environment. EDS provides also local ventilation needed for repair, maintenance and enhancements.

2.2. TRITIUM PROCESSING IN AGHS DURING DTE1 AND THE SUBSEQUENT CLEAN-UP PROCESS

Immediately before the commencement of DTE1 a "closed gas loop" was created by connecting the torus, the NIBs and the AGHS with the intention of recycling the 20 g tritium on site between the AGHS and the users. During DTE1 34.4 g and 65.6 g of tritium were supplied from the AGHS [5] to the gas puffing and Neutral Injection Beam Systems, respectively. Of the 65.6 g sent to NIB8 only 1% (0.6 g) was injected into the torus. In total 35 g tritium were introduced into the tokamak. The 100 g

tritium could only be supplied by multiple reprocessing the tritium in AGHS. For most of the time the tritium available in AGHS was less than 20 g. This was the first recycling of large tritium amounts in a fusion device. The gas loop mentioned above was only "closed" with respect to tritium, because NIB4 used continuously its own gas supply and NIB8 was reconnected to its own gas supply again shortly after the end of DTE1. AGHS had to discharge these gases because of limited storage capacities. In the case of hydrogen these gases were released from top of column-1 of CD [6], in all other cases via ED [7]. During DTE1 all subsystems of AGHS [8] were in intensive use. During the clean-up phase GC [9] and PS were rarely used because of the very small tritium concentrations. GI and GD were physically isolated for safety. The main task of AGHS during the clean-up phase was to recover as much tritium as possible from the 11300 bar litre received from the users.

The tritium and deuterium supplied by AGHS to the users was determined accurately by pressure (p), volume (V), temperature (T) and composition (c) measurements, but the daily accountancy of the returned gases and their tritium content was inaccurate due to large volumes, pressure dependent ionisation chamber readings, etc. The tritium inventory of the whole AGHS was accurately determined on an approximately weekly basis by isotopic enrichment, collection of the tritium present in AGHS in one U-bed and subsequent (pVT-c) measurement. The tritium not accounted for in the AGHS was retained in the torus and the NIB boxes. Repair work performed in NIB8 revealed clearly that the tritium external to the AGHS was mainly trapped inside the vessel.

2.3. TRITIUM INVENTORY IN THE VESSEL DURING AND AFTER DTE1

Major changes were made to the vessel interior after PTE. Toroidal cryo-pumps and a divertor were installed. Carbon-carbon fibre composite (CFC) tiles were mounted on the water-cooled (40°C) support structure of the divertor and graphite tiles protected the vessel walls. Periodically Be was evaporated on these surfaces. During plasma discharges active pumping was achieved by the cryo-pumps. The whole vessel was heated to 320°C, but the base temperature of the divertor tiles was only 200°C due to the cooled structure.

The tritium clean-up experiments following DTE1 revealed that a T/(T+D) ratio in the plasma of about 1% was achieved after 4 days of pulsing in deuterium. This decrease followed the predictions from PTE, but the PTE extrapolation underestimated the actual tritium inventory in the vessel by the factor 3 at the end of the campaign. The tritium retained in the vessel increased at 40% of the total tritium injected, but with clean-up shots this reduced to 17% [10]. Various clean-up methods such as deuterium soaks, glow discharges, ECRH discharges, pulsing in deuterium, warm up of the water-cooled in-vessel support structure after water removal from 40°C to 135°C were tried. When the tritium release was finally only 10 mg/d and the tritium inventory in the vessel still 6 g, the end of the clean-up phase of DTE1 was declared.

The difference in the tritium clean-up efficiency between PTE and DTE1 was attributed to codeposition observed in the relatively cold, shadowed regions in the inner corner of the MkIIA divertor. See Section 5.

2.4 USE OF AGHS DURING REMOTE TILE EXCHANGE (RTE)

Three months after DTE1 the vessel was vented twice with N_2 to 350 Pa, the gases transferred with the MF pumps to IP [11], their tritium content determined and then discharged via EDS using again the MF pumps. Afterwards air was introduced into the 120°C hot vessel at pressures up to 10, 25 and 90 kPa and after periods of 7, 12 and 4 hours, respectively, discharged via MF and EDS. 0.06 g and 0.54 g of tritium were released by the N_2 and air exposure, respectively. The vessel was then cooled to room temperature and a constant air stream of around 330 m^3/h was provided by the EDS blowers. This flow rate produced the required depression in the vessel and allowed enough time for full regeneration of the EDS molecular sieve beds. The first few regenerations resulted in high tritium concentrations of up to 1.3 TBq/litre in the water, lead to tritium concentrations of 13 DAC next to the water collection tank and to the highest daily discharge of 85 GBq from the AGHS building. The total amount of water collected by EDS during RTE was 6.921 m^3 with a tritium content of 1.97 g. The water was finally transferred into 200 litres Briggs drums and with no water detritiation system existing at JET shipped to Canada for tritium recovery.

2.5. TRITIUM PROCESSING IN AGHS DURING PLASMA OPERATIONS AND SHUTDOWNS IN THE YEARS 1998-2001

Two further shutdowns were performed after RTE: Pellet Injection Track (PIT) Shutdown from May to July 1999 and Septum Replacement (SRP) Shutdown from April to November 2001. At the end of each shutdown the vessel and the NIB boxes were evacuated with the MF pumps and then the vessel heated. Restart after each shutdown was continued with deuterium and helium glow discharge cleaning until operations could commence. During all these campaigns the gases used for operations were pumped by CF and processed in AGHS. Hydrogen was sent to CD to recover the very diluted tritium and to release the detritiated excess hydrogen. Gases from NIB4 and the pellet centrifuge are normally directly discharged via EDS. Table 1 lists the gas and tritium amounts recovered from pump divertor regenerations as well as the total gas and water amounts processed and collected in AGHS from 1998 to 2001. In total, AGHS detritiated more than 80 bar m^3 of gas. Typical tritium release rates of the torus, NIB8 and NIB4 in the year 2000 were 60, 5 and 1.5 GBq/d, respectively.

During shutdowns and repairs EDS provided the required torus air streams and detritiation as well as the purge streams from ISO containers holding the MkIIA divertor tiles, from drums containing the flakes and from the glove box built for handling the flakes. Table 2 gives the averaged tritium concentrations in the vessel and the tritium amounts collected in the form of water during RTE, PIT and SRP. Table 1 and 2 clearly show that the tritium release rate of the components inside the vessel during operations and the tritium concentration in the vessel decreased, but the high emanation rates, especially of the torus, and the large tritium inventory in the vessel still require the continuous operation of the AGHS for the foreseeable future.

Year	No. of regenerations	Gas from PD bar litre	Tritium from PD bar litre	T (mg) recovered per day (d)	AGHS gases bar litre	AGHS water litre
1998	40	1566	0.31	0.843	19530	10720
1999	64	1936	0.19	0.296	31670	12780
2000	53	2709	0.12	0.198	9290	7490
2001	42	2045	0.04	0.162	19860	13970

Table 1: Number of regenerations, amounts of gas, tritium pumped and recovered per day by the pump divertor (PD), total amount of gases and water handled in AGHS from 1998 - 2001

Shutdown	Averaged tritium concentration in vessel (MBq / m^3)	Tritium (g) collected as tritiated water
Remote Tile Exchange (1998)	117	2
Pellet Injection Track (1999)	55	0.7
Septum Replacement	23	to be verified

Table 2: Averaged tritium concentrations in the vessel and the tritium collected in the water during different shutdowns

2.6. REMARKS ON THE USE OF AGHS

The AGHS has now been continuously in operation since DTE1 for more than 4 years and has performed all duties in an excellent and safe way. These goals could only be achieved with the high flexibility and redundancy built into the AGHS design. This does not mean that the equipment in AGHS never failed, but that always routes were available to bypass failed equipment and to keep AGHS in service for the operation of the torus.

2.7. FAILURES OF EQUIPMENT IN AGHS AND POSSIBLE IMPLICATIONS FOR ITER

From the various equipment failures [8, 12] in AGHS only the failure of the pumps shall be discussed briefly: 8 Normetex pumps (4x150 m^3/h, 1x600 m^3/h, 3x30 m^3/h), 2 metal bellows pumps (single bellows type) and 1 turbo molecular pump were in operation in AGHS. Three of the large Normetex pumps, one metal bellows- and one turbo pump failed. The failure of the three large Normetex pumps of MF meant that MF could not continue to transfer gases to EDS. By re-routing pipe work one of the two IP Normetex pumps still operational could be operated as a MF pump. The other Normetex pumps were replaced during the SRP shutdown when EDS provided the required air stream through the vessel. The failures are considered to be caused by pumping significant quantities of atmospheric moisture during JET maintenance and repair phases and further investigation is underway. In contrast, getter beds used also for pumping at JET have performed faultlessly.

In the ITER Tritium Plant pumps are foreseen for the processing of gases through Pd/Ag permeators and to avoid disproportionation of the alloy ZrCo. Considering the JET experience of the relative reliability of mechanical and getter pumps, the use of special getter beds with high pumping speeds on the permeate side could be reconsidered and the advantages/disadvantages of

ZrCo again compared with storage materials capable of supplying tritium by heating alone. In addition, possible improvements in redundancy and flexibility to permit continued processing in the event of component failure could be considered.

2.8. AGHS: AN IDEAL TEST BED FOR THE ITER TRITIUM PLANT

AGHS can be the ideal test facility for ITER components. Within the Task Force Fusion Technology the following ITER relevant components will be tested in AGHS: i) a PERMCAT reactor [13] which is one of the main components of the ITER Tokamak Exhaust Processing System [14] and ii) cryo-panels of the same design as proposed for ITER. Further proposals are tests of iii) a tritium compatible Roots pump equipped with ferrofluidic seals and of iv) micro-gas chromatographs with very short retention times.

3. THE REMOTE TILE EXCHANGE (RTE)

The main objective of the Remote Tile Exchange (RTE) was the entirely remote exchange of 144 MkIIA divertor carriers with 192 Gas Box (MkIIGB) divertor carriers [15]. The MkIIA divertor carriers were removed with the help of a two arm servo-manipulator positioned inside the torus by an articulated boom. A further boom equipped with an end effector and tine interfaced with the articulated boom, accepted the divertor carriers, moved from the torus back into the so called Tile Carrier Transfer Facility (TCTF) and transferred them into an ISO container attached to the TCTF [16]. Furthermore, dust and flakes on tiles, the support structure and especially on the inner louvers were collected using an adapted vacuum cleaner. The new MkIIGB divertor carriers were installed in reverse order. This remote handling work was performed in a very hostile environment. Typical in-vessel gamma dose rates were 5 mSv/h, with contact dose rates of 0.6 mSv/h [17] on the machine exterior. Additionally, in the vessel there was significant Be contamination, highly tritiated dust and an initial tritium concentration in air of about 1 GBq/m^3 [16]. Tritium and beryllium were contained by a slight depression of the atmosphere inside the torus relative to the torus hall. EDS of AGHS was used to supply the depression for the vessel and to purge the 200 m^3 volume of the torus, but could not provide purge streams to other enclosures connected to the machine such as TCTF, Boom tent, ISO containers. These had their own ventilation systems and discharged through a filtered stack. Control of all these ventilation systems was critical. Stack discharges reached 200 GBq/day in the first days, but with time reduced to less than 10 GBq/d. Off-gassing values up to 0.6 GBq/h were observed from carriers with flakes. Higher contamination levels of up to 2 MBq/cm^2 were observed on hot spots, but the general contamination was in the order of 100-1000 Bq/cm^2.
The two port openings for the booms have been so far the largest breach of the vessel at JET. Furthermore, this was the first full remote exchange of a whole divertor in a hostile environment and a great technological achievement because the remote handling tools and procedures in place functioned very well. Since RTE the remote handling tools have been used in two further activities, removal of the divertor septum and collection of flakes from sub-divertor regions.

4. AUTHORISED DISCHARGE LIMITS AND ACTUAL DISPOSALS

Table 3 presents the authorised discharge limits for tritium before and after the handover to UKAEA. The authorised discharge limits were reduced after the handover to UKAEA. The new values have been justified on the basis of the maximum operational requirement and of insignificant environmental impact. Table 4 gives the actual tritium discharges of the JET site for 1997 to 2001 and the inventory of tritiated water sent for reprocessing. In 1997 the HTO discharges were mainly due to the NIB8 intervention [18] and tritium permeated through the torus walls and released via the torus hall depression stack [17] during DTE1 and the clean-up phase. The HT discharge was mainly caused by AGHS releases of excess hydrogen. The HTO releases in 1998 were mainly due to diffusion from the torus into the remote handling enclosures in the torus hall during RTE and partly due to AGHS discharges.

Tritium	Annual gaseous discharges (TBq)		Monthly liquid discharges (TBq)
	HTO	Other tritiated species	
1991 - 1999	90	110	2
2000 - now	50	50	0.5

Table 3: Authorised radioactive discharges for the JET machine

Tritium	Atmosphere		River	T-Water
	HTO (TBq)	Others (TBq)	(TBq)	(TBq)
1997	1.5	2.0	0.16	0
1998	4.1	0.5	0.19	0
1999	1.2	0.4	0.02	900
2000	0.5	0.2	0.01	0
2001	2.5	0.1	0.03	217

Table 4: Actual tritium discharges of the JET site

5. TRITIUM IN PLASMA FACING COMPONENTS

5.1. HANDLING OF FLAKES

Due to the high specific activity of the collected flakes a special glove box was installed in AGHS to limit the spread of contamination. During handling of the flakes large tritium releases of up to 550 GBq were observed and purged to EDS. 154.3 g of flakes were collected during RTE and their averaged specific tritium activity calorimetrically determined to be 1.2 TBq/g [19]. Change of glove box atmosphere from air to argon and use of a PVC isolator with inflatable limbs inside the glove box were important steps in reducing tritium off-gassing and contamination of the box.

5.2. TRITIUM IN TILES, CODEPOSITS AND FLAKES

Codeposition of hydrogen with carbon has been observed in various fusion devices. Carbon eroded from plasma exposed surfaces is redeposited on surrounding areas. During deposition hydrogen

and C_xQ_z molecules are incorporated in the newly built layers as they are continuously exposed to the plasma flux. At JET on surfaces heated to 320°C ratios of hydrogen to carbon below about 0.4 were found. After the MkIIA installation and shortly before DTE1 a new codeposition region was observed on the inner leg of the divertor with almost no deposition in the outer region. Ion Beam Analysis of the layers revealed very high hydrogen to carbon ratios of at least 0.7 to 0.8 [18]. As these layers were created in areas protected from plasma erosion and on structures with temperatures below 200°C these layers are quite stable, will grow and may finally flake. In plasmas fuelled with tritium these layers will take up large amounts of tritium which cannot be easily recovered by plasma operations. Part of the brittle flakes might have detached themselves from the chevrons (40°C) and other structures and found their way into regions below the divertor from where they could not be recovered during RTE.

The tritium content on the surface and in the bulk of selected tiles removed during RTE was determined with different methods [20]. The main qualitative results were as follows: More than 99% of the tritium content in graphite tiles was found in codeposited surface layers of a thickness less than 50 _m, whereas the tritium content in the bulk of a few CFC tiles was found to be up to three times that in the surface layer. The tritium concentrations in the surface layer were inhomogeneous, but in the bulk essentially homogenous, strongly bound and not easy to recover. The tritium concentration near the rear side of CFC tiles increased slightly and the tritium surface content in the inner divertor wall tiles was substantially higher than in the outer ones. Over 100 determinations of the tritium content in carbon tiles allowed calculation of the tritium inventories for divertor, inner wall and outer poloidal wall tiles to be 60, 20.4 and 50.5 mg, respectively [21]. This clearly demonstrated that only a small part of the tritium in the vessel is trapped in the tiles. At the end of RTE the tritium inventory in the vessel was 3.4 g. Assuming that this tritium is incorporated in flakes of the same specific activity, which were not collected during RTE, about 1 kg of flakes must still be in the vessel. During the recent SRP shutdown an attempt was made to collect also the flakes from the sub-divertor region of one Octant with specially built long extension tools mounted on the vacuum cleaner.

5.3. RESEARCH ACTIVITIES FOR TILES AND FLAKES

At the moment the MkIIA tiles with their carriers are stored in purged ISO containers. Two future scenarios are discussed for these tiles: reuse after removal or detritiation of the codeposited layers, or detritiation to the specific activity < 12 kBq/g so that they can be treated as Low Level Waste. Within the JET Fusion Technology Task Force the following research activities have been performed, or are planned, for tiles and flakes: a) determination of surface and bulk tritium concentrations in carbon tiles, b) determination of most efficient detritiation techniques for carbon tiles, c) detritiation trials of whole carbon tiles, d) surface analysis using various ion beam techniques, e) collection and determination of the amounts of flakes in the sub-divertor regions, f) characterisation of tritium and other impurities in the flakes, g) study of the tritium release behaviour of flakes as a function of

temperature, h) full combustion of flakes to recover the tritium bound in the flakes, etc.

5.4 GENERATION OF FLAKES

At JET and in other tokamaks it was observed that the outer divertor is mainly an area of erosion (the tiles are "clean" with very little deposits), whereas the inner divertor is mainly an area for deposition with thick codeposited layers, especially in plasma shadowed regions. This behaviour could not be explained until recently. Only neutrals are expected to reach the deposition areas so far away from the plasma. The present qualitative understanding of the erosion/deposition behaviour is as follows [21]: Erosion which leads to deposition in the divertor occurs predominantly on the tiles of the main chamber. If an asymmetrical drift in the Scrape Off Layer (SOL) is included the eroded particles are moved to the inner leg of the divertor. At JET due to repeated Be evaporation the ratio of Be to carbon at surfaces of the main chamber is approximately 10%. The same ratio of Be to C impurities is also observed in the plasmas. Assuming that the ratio of Be to C transported in the SOL is also 10%, this leads to a similar deposition ratio at the inner divertor, but the experimentally observed ratio on MkIIA and MkIIGB tiles is about 1:1 [21]. If the assumptions so far are correct, an erosion process must exist which removes the deposited carbon preferentially. C is unusual in that it can also be chemically sputtered by low energy ions whereas other species such as Be can only be physically sputtered. Thus it is assumed that 90% of the deposited carbon is re-sputtered chemically. During this process various neutral fragments of higher hydrocarbons are created which are able to fly in any direction independent of the magnetic fields. Some of these species have low sticking coefficient at the temperature of the divertor surface and may migrate to regions far away from where they were created. If they hit cooler surfaces they are more likely to be deposited. The amount of carbon found in the layers in the shadowed regions is of the order of 10 times the amount in the Be/C layers on the plasma exposed tiles and the layers contain no Be. Thus the experimental observations agree with the model assumptions made [21].

5.5. ISSUES FOR THE FIRST WALL MATERIAL FOR ITER

In ITER the vessel walls of the main chamber are protected with Be tiles, carbon is used for the vertical tiles in the divertor, whereas the other divertor tiles are made of tungsten. If the model discussed above is correct, one should expect that due to the erosion of Be in the main chamber and the transport of Be into the divertor the carbon tiles are almost fully covered with Be. As a consequence the chemical sputtering of carbon, the creation of deposited carbon layers and the resulting uptake of tritium should be greatly reduced. Although Be has an extremely limited solubility for hydrogen, it can when deposited together with impurities such as oxygen, carbon, etc. still incorporate large amounts of hydrogen with Q to Be ratios up to 0.30 [22]. For ITER cold traps are now proposed in front of the cryo-pumps to trap the hydrocarbon fragments (produced by chemical sputtering) and to decompose them by heating in certain intervals so that they cannot reach the cryo-pumps.

In summary use of carbon in ITER should be avoided. Be is an improvement, but codeposited Be or BeO_x layers may still absorb significant tritium. The JET tokamak with its Be and T handling capabilities is best suited to check this. The exchange of carbon tiles might not be necessary because of the capability of Be evaporation onto the main chamber walls and of the amounts of Be transported onto the carbon tiles in the divertor from the main chamber.

6. MONITORING AND CONTROL OF RADIATION HAZARDS

6.1 ENVIRONMENTAL MONITORING

At JET radiological environmental monitoring has been in place for a number of years. Four monitoring points are located approximately 300 m from the JET buildings and agricultural products collected on-site and from local farms are monitored for HT and HTO. Slightly elevated on-site tritium levels could just be correlated with discharges. Chronic discharges of 1 TBq HTO and 1 TBq HT per year lead to doses of 0.12 and 0.0005 _Sv/a [23] from inhalation, skin up-take and ingestion. For the highest annual discharge to date in 1998, an averaged chronic annual dose of <0.5 _Sv has been calculated. This demonstrates that the JET tritium discharges had negligible radiological off-site effects.

6.2. CONTROL MEASURES FOR MITIGATION OF RADIOLOGICAL & CHEMICAL HAZARDS

Engineering control measures are required to minimise possible exposure of workers to radiation and hazardous chemical materials (Be). Control measures in use at JET [24] are: a) containments in the form of glove boxes or of flexible membrane isolators; b) confinements provided with local ventilation systems; c) contamination controls such as regular cleaning, use of sacrificial layers, etc.; d) respiratory protective equipment (RPE) such as full pressurised suits; and e) procedures detailing the potential hazards and countermeasures.

The dose due to incorporation of tritiated dust, flakes or tritiated hydrocarbons, the intake probability by inhalation as a function of the particle size, the dissolution rate of the activity from the tritiated materials, etc. are more or less unknown factors, but of importance for future work in similar environments. Simple estimates show that the intake of only 1 mg of tritiated flakes can give rise to doses above 20 mSv [25].

Tritium exposure of workers has been very well controlled throughout the operations of JET. The maximum individual dose has been less than 170 _Sv, the collective dose for the years 1997 to 2001 were 1.46, 2.01, 0.70, 0.80 and 0.55 mSv, respectively.

7. WASTE MANAGEMENT ISSUES AT JET

At JET the collection, processing and radiological clearance of radioactive and beryllium contaminated waste are the duties of the waste management group. Several supporting facilities (e.g. the JET Radioactive Drain System, the JET Waste Handling Facility, beryllium-controlled areas, etc.), and provision of respiratory protection, are also provided by this Group. The ISO containers which house the MKIIA

divertor carriers have 2 cm steel cladding are stored in a special concrete building for shielding purposes. Due to the high off-gassing rates of tiles with codeposited layers the ISO containers and the concrete building are purged by air vented to AGHS. The concrete building serves also as a secondary containment.

During RTE more than 600 suited support entries were conducted. Without them massive delays in the programme would have occurred. These activities contributed about 60% to the soft housekeeping waste. The plastic suits when in contact with highly tritiated materials pick up large amounts of tritium activity and could often not be detritiated by conventional methods. With the large amounts of highly tritiated flakes in the vessel contamination control was especially important during breaches. Regular change of protective clothes, the use of sacrificial layers and of peelable coatings helped to keep the contamination levels low, but increased the volume of soft housekeeping waste.

In conclusion, the waste handling facilities and the procedures in place allowed the large variety of waste handling activities to be performed in a professional and safe way. Further aspects, such as characterisation, detritiation, categorisation, radiological clearance, the establishment of clear, cost efficient waste disposal routes, etc. are either in the process of being established or need immediate attention.

8. DETRITIATION ISSUES FOR JET

The activation of the vessel and the injection of 35 g into the vessel during DTE1 has still today a large influence on the operations of JET. A large work effort is required to control the tritium and beryllium contamination and the transfers of components out of the vessel and on the JET site. At JET the tritiated flakes, dust, codeposited layers, and Be are the main causes of the contamination hazards.

The most effective method against spread of contamination is the removal or even better the detritiation of the contamination sources. The development of effective and cost efficient detritiation methods is of fundamental importance for JET and even more for ITER because decommissioning has already to be addressed during the licensing process. JET is again the best suited test bed for ITER because detritiation techniques can best be tested with samples created at JET under tritium conditions most similar to ITER.

Within the JET Fusion Technology Task Force (FT TF) the following detritiation activities have been, or are being, addressed: a) development of a facility to study tritium retention and tritium recovery from dust and flakes with the purpose to recover the tritium and reduce waste; b) development of equipment and procedures for detritiation of stainless steel and graphite; c) determination of most effective methods for detritiation of whole JET tiles; d) detritiation of samples taken from an ITER equivalent cryo-panel to measure the residual tritium inventories; e) detritiation and deactivation of ZrCo and U storage beds; f) experimental tests and evaluation of a catalyst most suitable for Liquid Phase Catalytic exchange (LPCE); g) optimisation of codes for the simulation of LPCE; h) design study of a water detritiation system for JET; and i) experimental assessment of electrolysers for the JET water detritiation system.

At JET large amounts of tritiated water were created or collected in EDS. Furthermore, many of

the detritiation methods developed so far will create tritiated water. Until now tritiated water has been shipped to Canada for reprocessing, but this disposal route may not always be available. Therefore, a water detritiation plant for JET needs to be designed and is the main justification for the last four FT TF activities listed above.

Detritiation methods for materials in use at JET such as Be, Cu, ST707, molecular sieve, etc. need to be found as well as for other categories of waste such as soft house-keeping waste, organic liquids (oils from pumps, liquid scintillation cocktails) and building materials (structural steels, cladding, painted materials, concrete) [26].

A further issue important for detritiation and for radiological clearance is the determination of the radioactive nuclides in the materials which need to be transferred between various areas (controlled, restricted, un-restricted) or disposed as waste. In the case of activated materials high resolution gamma spectroscopy allows the determination of levels as low as 0.01 Bq/g of typical radionuclides such as ^{51}Cr, ^{60}Co, ^{58}Co, ^{54}Mn and ^{56}Ni. However, the detection of soft beta emitters ^{55}Fe or ^{63}Ni which do not emit gammas, is more problematic. Similarly the detection of tritium in the bulk is difficult and cost intensive. The usual technique is full combustion of representative samples. Other possible techniques measure the off-gassing rate of tritiated material with ionisation chambers and try to determine the tritium content by means of a pre-determined relationship between off-gassing and tritium inventory. This technique can give reasonable results of the specific tritium activity for soft housekeeping waste exposed to tritiated gases, but the technique underestimates the tritium content when tritiated particulates are involved.

Limits such as the maximum specific tritium activity permitted for waste disposal directly influence the cost for disposal. The Drigg facility in the UK can only accept tritiated materials with a specific activity < 12 kBq/g. As a large fraction of the waste collected at JET is above this limit, detritiation to a level below is necessary to minimise disposal costs. The development of cost effective detritiation techniques, the unambiguous determination of the radioactive inventories and the clear specification of waste disposal routes is of fundamental importance for JET and ITER. Its proper solution will finally be an important step for the public acceptance of fusion power.

9. VALIDATION OF ACTIVATION MODELS

Neutronics calculations are essential for safety, decommissioning and diagnostic design studies. During the DTE1 phase of operation, activation measurements were made to provide a dataset for the development and benchmarking of neutron transport calculations throughout the JET facility (i.e. the machine and all diagnostic and heating systems in the torus hall and the torus hall walls). The measurements consisted of the levels of activation in fifty “Activation Foil Packs” (AFP) which were placed at 32 locations on the torus hall walls, the vertical sections of the transformer limbs, and above the machine, and left there for the duration of the DTE1 campaign. Each foil pack contained either seven or nine foils allowing a set of activation reactions with a range of energy thresholds. Eighteen of the packs had cadmium liners to absorb thermal neutrons and each was positioned alongisde an unlined pack.

A detailed MCNP model of JET has been built up over many years. It is a large model consisting

of 1267 cells described by 934 surfaces using 32 transformations and 260 material definitions describing a volume in excess of 50000 m^3. To perform the calculations described here, version 4C of MCNP [27] was run on a parallel cluster of 16 dual processor linux boxes using ENDF/B-VI nuclear data libraries. MCNP was used to calculate the neutron flux and spectrum at each foils location. These were then used to at input to the inventory code FISPACT [28] to determine the activation level of each foil taking into account the neutron production history on JET. These results were compared to measurements. The mean C/E is 0.87 and the ±1_ range is _/÷ 2.1 (0.42 to 1.83). The validated MCNP model can be used to predict activation of components during operation and for planning decommissioning.

10. CONCLUSIONS

Consequences of the successful DTE1 operation such as activation and tritium contamination still play an important role four years later during the present JET operations. Due to the generation of highly tritiated codeposited layers and flakes in relatively cold regions shadowed from the plasma the tritium inventory in the vessel did not decrease as expected from PTE. This required further operation of the AGHS, use of remote handling tools and necessary control measures to avoid spread of contamination. Man entry into the vessel could only be done in full pressurised suits. The safe operation of the JET machine was possible because of the systems created and managed by AGHS, remote handling, waste management and health physic groups. New important issues are cost effective detritiation techniques, unambiguous determination of radioactive inventories, first wall materials with negligible affinity to tritium. Carbon in a DT fuelled machine should be avoided as far as possible and the appropriateness of Be with respect to tritium demonstrated in dedicated experiments before its use as first wall material in ITER.

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