EUROfusion

EUROFUSION CP(15)04/22

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(18th May 2015 – 22nd May 2015)
Aix-en-Provence, France

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Experience of handling beryllium, tritium and activated components from JET ITER-Like Wall

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Abstract. JET components are removed periodically for surface analysis to assess material migration and fuel retention. This paper describes issues related to handling JET components and procedures for preparing samples for analysis; in particular a newly developed procedure for cutting beryllium tiles is presented. Consideration is also given to the hazards likely due to increased tritium inventory and material activation from 14 MeV neutrons following the planned TT and DT operations (DTE2) in 2017. Conclusions are drawn as to the feasibility of handling components from JET post DTE2.

Keywords: Beryllium, Tritium, JET

PACS: 28.52.Fa, 28.52.Nh

* See the Appendix of F. Romanelli et al., Proceedings of the 25th IAEA Fusion Energy Conference 2014, Saint Petersburg, Russia

1. Introduction

Samples from JET are periodically removed from the vessel using the remote handling MASCOT to study fuel retention and material migration. At the point of removal many components are manually handled from the remote handling equipment into a storage location before moving to the Beryllium Handling Facility (BeHF) at JET. At the BeHF components are unloaded from the shipping container and worked on by operatives to remove, refurbish, repair and replace tiles and other diagnostics before being returned to the vessel. Once tiles and diagnostics are removed, preparation for shipping components and subsequent processing and analysis requires further handling.

When handling samples from JET there are three main hazards, Beryllium (Be), Tritium (T) and activated materials. A significant understanding of dealing with these hazards was obtained after the Deuterium Tritium Experiment 1 (DTE1) with the JET carbon wall (JET-C) [1]. T inventory and Be contamination of components and 14 MeV neutron activation of the vessel were important factors taken into consideration when removing components. In terms of manual handling of components, the more significant hazard was T contamination of carbon fibre composite (CFC) components and dust containing T and Be. The role of activation as a hazard was minimal as carrier structures were largely constructed from CFC with few metallic Inconel components – the source of ^{60}Co activation. This remains the same for the JET-ITER Like Wall (JET-ILW) tungsten coated CFC tiles (W-CFC) with relatively few Inconel fasteners in a largely CFC structure [2]. However, for Be tiles [2] in the main chamber and the bulk tungsten load bearing septum replacement plate (W-LBSRP) [3] in the divertor, Inconel 625 is used extensively as a carrier material. In addition, other materials with potential for high activation such as Inconel 718, Nimonic 80A and Nimonic 90, are used in these assemblies. This brings the need to assess neutron activation following the DT Experimental campaign (DTE2) planned in JET in 2017.

2. Sample preparation of JET-ILW tiles

The analysis of samples within the EUROFusion JET 2 Work Programme *Investigation of Plasma Facing Components for ITER* (JET2WP) and JET 3 Work Programme *Technological Exploitation of DT Operation for the ITER Preparation* (JET3WP) requires that tiles removed from JET are processed to a manageable size for analysis. This is driven by two requirements: (i) many instruments require smaller samples than the whole tile pieces and (ii) the overall T inventory and Be contamination is reduced thus minimizing the risk of contamination to analysis equipment.

The cutting of JET CFC tiles by means of coring has been ongoing since 2002 at the VTT Technical Research Centre of Finland following the removal of the first CFC tiles from JET-C in 2001 and continues for the latest W-coated CFC tiles removed from JET. In this method tiles are handled in an isolator with glove ports. Cores are cut from tiles to give both a poloidal and a toroidal distribution of

samples from across the tile surface. The exact number of samples is governed by analysis requirements. The cores cut are generally 17 mm diameter, although smaller cores are also possible. Discs ~ 10 mm thick are sliced from the surface of the cores, and from the bulk material when required. From these discs, smaller samples may be prepared, for example cross sections for metallurgical characterization and micro-beam analysis are mounted in cold epoxy and polished using a virtually water free diamond suspension to minimize reduction in D and T content by isotopic exchange [5][6]. In addition samples with a surface area ~ 12×12 mm^2 and ≤ 3 mm thick are prepared for Thermal Desorption Spectroscopy (TDS) using a back heated stage [7].

A new capability for beryllium sample preparation has been recently developed at the National Institute for Laser, Plasma and Radiation Physics, Romania in conjunction with JET2WP. This involves cutting the individual castellations from Be tiles removed from the JET main chamber. The tiles cut are from the Inner Wall Guard Limiter (IWGL) and Outer Poloidal Limiter (OPL) beams and the upper dump plate region. An example of an IWGL assembly is shown in Figure 1(a) which illustrates the complex shape and curved surfaces involved. The IWGL tiles split into 5 pieces, in Figure 1(b) the cutting requirements for one section, a right hand wing, is indicated: this is generally 2 - 4 rows of castellations taken in the toroidal direction from the centre of the tile.

To ensure a coherent analysis programme it is necessary to keep track of the location and orientation of every castellation removed from the tile. To achieve this, a systematic numbering system for each castellation is implemented and each castellation is marked on one side to give a unique identifier and define orientation. The position of the unique number is recorded against the row-column position (Rn-Cn) shown in Figure 1(b). The cutting procedure is controlled such that the sides of each castellation and the top surface are not touched. The cutting and sample labelling are documented and photographed at each stage to ensure traceability.

There are eight basic designs of the Be tile pieces from the IWGL, OPL and dump plates of JET. The cutting procedure was developed using mock-ups of each design made from aluminium alloy. Using these mock-ups the mounting jigs were designed and manufactured. The jigs were designed to minimize vibration and cross contamination to tile surfaces.

During cutting it is necessary to keep the temperature of the samples low to minimize desorption of D and T from the surfaces. However it is also a requirement to cut the tiles dry, i.e., without coolant or lubricant, as water cooling would decrease D and T content by isotopic exchange and other coolants would contaminate the surface. By optimizing the band saw speed and applied load it is possible to keep the temperature of the samples < 55ºC. At every cutting stage the temperature of the tile is monitored using a Thermovision FLIR ThermaCAM E45 camera and at the end of each cutting stage the maximum temperature recorded.

An additional cutting requirement is to achieve a flat surface finish such that the castellation sample sits flat onto an instrument stage. That is to say that the cut surface should have no burrs and should be flat, i.e., not bowed, on a sub millimetre scale. Following cutting trial a Morse Achiever bi-metal band saw blade 27 mm width and 10/14 teeth per inch was selected for cutting with cutting speed and load of 75 mm/s and ~25 N respectively.

Aspiration of the swarf and dust particles produced during cutting is also implemented to reduce cross contamination of surfaces and to remove respirable Be and tritiated particles which are extremely hazardous to health. Operatives are required to wear respiratory protective equipment to minimize exposure. Figure 2 shows an image of particulates from cutting. The morphology of the particulates indicates that they are produced by brittle fracture. After sieving 1 g of swarf material the size distribution of the particulates was 1 mg with size < 36 μm, 136 mg in the range 36 μm ≤ 90 μm and ~ 800 mg > 90 μm. Of this the first fraction may contain respirable dust, i.e., particulates < 10 μm.

The procedure for cutting a tile consists of many steps which are photographed and documented. Prior to cutting the unique number of the whole tile is photographed and the temperature monitoring and the vacuum for aspirating particulates implemented. The tile is mounted onto the first jig and cut from the back side, as indicated in Figure 3, to expose the castellations in the centre of the tile which are of most interest. After recording the maximum temperature the tile is demounted from the jig and the castellations to be cut are numbered. The tile piece is re-mounted to a second jig which clamps the castellations together to ensure that they are not ejected during cutting. The depth of the cut from the tile surface is set according to cutting schedule shown in Figure 1(b); a depth of 10 – 12 mm is suitable for a majority of analysis techniques, however for some techniques, in particular back heated TDS [7], a sample thickness 2 - 3 mm is needed. It is not possible to cut thinner slices as internal stresses can results in bowing of the samples. The maximum temperature during cutting is recorded. Each castellation is removed from the jig and secured into individual boxes to ensure they are protected during transportation. The cutting procedure is repeated for each row of castellations.

Using these methods > 150 Be wall tile samples and > 50 W-CFC samples from 2011 - 2012 JET tiles have been produced so far for analysis. For the purposes of distributing samples to laboratories the T inventory of samples is estimated. From TDS the T inventories for W-CFC divertor samples and Be wall samples are ~ 10^{14} T atoms/cm^2 and ~ 10^{13} T atoms/cm^2 respectively. Taking into account the surface area of the samples ~ 1cm^2 and assuming a specific activity of T of 357×10^{12} Bq/g, the T inventory for samples is $10^4 – 10^5$ Bq. The mass of the samples is ~ 1 g, giving a specific T inventory of $10^4 – 10^5$ Bq/g.

During 2015 a cutting procedure for W lamellae from the W-LBSRP assembly will be developed at the National Institute for Laser, Plasma and Radiation Physics, Romania in conjunction with JET2WP.

3. Issues related to handling with DT operations

The JET baseline scenario plans for DT experiments (DTE2) in 2017. The total operating period will involve a TT campaign of 4 weeks, DT campaign of 16 weeks followed by a DD campaign of 12 weeks. During the TT and DT phases the T inventory in the machine will increase and during the DT phases a significant increase in activation due to 14 MeV neutron interactions will take place. There is a need to understand the increase in T inventory and activation of the machine on several levels: prior to operations estimates of T inventory and activation are vital for the safety case; during operations the increase in T inventory and activation needs to be understood in case of unforeseen breakdowns requiring remote handling entry to the vessel; after operations the T inventory and activation of any components removed from the vessel would need to be understood and an assessment of the liability to be transferred to the Nuclear Decommissioning Agency at the end of JET should be made.

3.1. Tritium Inventory

An upper limit on the number of T atoms to be injected into JET in the baseline DTE2 campaign scenario can be made. The maximum mass of T that can be stored on the cryopanels in JET and the NIB columns is 11 g. This defines the maximum permissible mass of T that can be used per day before cryopanels are regenerated. Of this approximately 9.9 g is injected into the vessel via the Gas Injection Modules (GIMs), the remaining 1.1 g is injected to the NIB column and does not make a significant contribution to the main vessel (also known as the *pumped divertor*) accountancy. To calculate the total injected inventory a total of 96 days of TT and DT operation is assumed (8 weeks TT and 16 weeks DT with 4 days of gas injection per week). Based on this scenario an upper limit for the throughput of T in the vessel is 950 g.

The T inventory can be estimated from the deuterium (D) retention measured on tiles exposed in JET from 2011 – 2012, the first JET-ILW campaign. In this DD programme the total amount of D puffed into JET was 1.67×10^{26} D atoms and the total amount retained in the vessel was 3.92×10^{23} D giving an overall long term retention of 0.24 % [8].

During the DTE1 in 1997 35 g of T was injected into JET. T accountancy from the Active Gas Handling System immediately after operations indicated that ~ 40% (14 g) of the total amount of T injected was retained in the vessel. This value was reduced by H and D fuelling to give a final amount of T left in the vessel prior to opening of ~ 6 g, i.e., 17% of the T injected [9]. As a percentage of injected gas 6% (2 g) T was released on purging the vessel, 4% (1.5 g) T was held in deposits mainly on divertor tiles, 0.03% (0.3 g) T was trapped in tiles and 6% (2.2 g) T was attributed to flakes in remote areas, particularly in the inaccessible sub-divertor. For the JET-ILW the retention in deposits is 0.24% with no additional contribution from flakes in the sub-divertor region as long

range migration is almost totally absent and hence deposits are not spalling. Therefore the distribution of T retained after purging is expected to be 0.24% (2.28 g) in deposits, from JET-ILW retention [8], and 0.03% (0.29 g) in tiles, as seen in DTE1 [9][10], of 950 g T injected, giving a total of 2.57 g T. The main error is in deriving the retention value from extrapolating surface analysis results for a selection of tiles to the whole vessel. Therefore the retention value could be 30% higher. Taking this into account the long term retention in tiles could be up to 3.33 g T from TT and DT operations (not including a DD clean-up phase) and after purge. This is equivalent to $\sim 10^{17}$ T atoms/cm^2 assuming a vessel area of the order of 100 m^2.

From isotopic exchange experiments the accessible reservoir of hydrogen isotopes for exchange is shown to be 2.3×10^{22} atoms [11]. This is $< 0.02\%$ (0.2 g) of injected T and would reduce the retained T inventory by ~ 10%.

The contribution of T from previous sources is now reduced as any remaining T held in deposits and tiles was removed when all tiles were replaced in 2010, leaving the main T legacy in flakes in the sub-divertor. Half-life calculations based on 17 years since analysis of the DTE1 tiles in 1998 reduce this to ~38% of the original inventory, i.e., 0.85 g remaining. This is an upper limit as off-gassing of T would reduce this value still further.

3.2. Neutron activation due to JET DT operations

Neutron activation of JET in-vessel components is currently evaluated using neutron transport and activation codes and data libraries [12][13][14]. The neutron flux in in-vessel components is calculated [13][14] using the emission source from a typical DT plasma and the resulting neutron spectra provide the input for activation calculations [12]. For each irradiation scenario, the time behaviour of the contact dose rate is determined. These calculations are useful for determining the relative activity and cooling times of different materials found in in-vessel assemblies. However the contact dose rate calculations do not provide a good method for the evaluation of potential effective dose to operatives when handling samples as they do not consider the decay of gamma radiation, the geometry of the component and the neutron flux gradient in sub-components. To make a more realistic assessment for handling, shutdown dose rate calculations are required. These take into account the dimensions, geometry, mass and materials of components [17][18] and neutron and decay gamma transport [16] to give dose rates for individual assemblies. Results of both approaches are presented in the following sections.

Uncertainties in the calculations are related to defining the exact DT pulse schedule and neutron flux, location of the component in the vessel, and modeling of the assembly dimensions and materials, in particular the presence of trace elements. Variations in the pulse strategy and the inclusion of DD

operations following TT and DT pulses are only likely to affect results by 10% [18]. In this work a total neutron yield of 1.7×10^{21} DT n [19] is considered.

A range of components have been studied, these include assemblies from the upper dump plate region, mid-plane IWGL, mid-plane OPL, W-LBSRP [15][16][17] and foils for activation studies and associated caddie [18]. Within each of these components a range of materials have been considered, e.g., Inconel 600, Inconel 625, Inconel 718, stainless steel 316 (SS316), W, Be, Nimonic 80A, Nimonic 90. CFC components such as divertor tiles and carriers are not considered: although these assemblies contain fasteners manufactured from Inconel and Nimonic alloys they do not represent the worst case in terms of activation due to location in the vessel.

The following results are discussed in terms of days after the last DT pulse where the last DT pulse is at 0 days. The main point of interest is at 150 days when the machine is opened. These 150 days include 90 days of DD operation and 60 days to deploy remote handling equipment to remove samples from the vessel. No waiting time between the end of the JET operations and deployment of RH equipment is assumed. It is likely that if parts are removed from JET post-DTE2 (for example the caddie containing activation foil samples) they may need to be stored before they can be handled for disassembly, therefore a storage time of 515 days, i.e., 1 year after removal from the vessel, is also considered. The results discussed are summarised in Figure 4.

3.2.1. Contact dose rate

The materials in the mid-plane OPL tile assembly and W-LBSRP tile assembly in the divertor are found to have the highest activation levels for contact does rate [15]. Therefore the contact dose rates of materials in these assemblies are considered as a worst case scenario.

In terms of contact dose rate the specialist alloys Nimonic 80A and Nimonic 90 give the highest calculated values due to very high Co content. Nimonic 80A is used in the OPL tile assembly and both alloys are found in the W-LBSRP assembly as specialist fasteners/screws. For Nimonic 80A at the mid-plane OPL the contact dose rate at 150 days is < 35 mSv/h. For Nimonic 90 in the W-LBSRP this is < 80mSv/h. If the materials are left to cool, contact dose rates fall to < 12 mSv/h and < 53 mSv/h for Nimonic 80A and Nimonic 90 respectively at 547 days (1year and 1 month after removal) [16].

The alloy Inconel 718 is also used for specialist fasteners/screws on OPL and W-LBSRP assemblies. The results reported are for the outer mid-plane OPL location as these return the highest values. The higher activation of Inconel 718 is due to a relatively higher ^{182}Ta content than other Inconel alloys which dominates the short term activation, i.e. < 1 year. At 150 days the contact dose rate for Inconel 718 < 160 mSv/h and at longer cooling times this drops to < 20 mSv/h at 547 days [15]. An updated

composition with lower ^{182}Ta is reported in [16] giving contact dose rates of < 24 mSv/h at 150 days falling to < 7 mSv/h at 547 days. The variation in these results highlights the importance of understanding impurity content in materials.

The carriers of the OPL and W-LBSRP assemblies are manufactured from Inconel 625. This is the largest individual component on each. In [15] the highest Inconel 625 activation values are found in the W-LBSRP; <90 mSv/h at 150 days and <15 mSv/h at 515 days. With updated composition data [16] the contact dose rates for the OPL assembly are <22 mSv/h at 150 days and <3 mSv/h at 547 days.

3.2.2. Shutdown dose rate

Shutdown dose rates for mid-plane OPL [16][17] and SS316 caddie [18] are presented here. For the OPL tile assembly the shutdown dose rate is dominated by Inconel 625 carrier and these are the result shown in Figure 4.

The study of a 3D model of the mid-plane OPL assembly [16] gives an shutdown dose rate at 0 cm for the Inconel 625 carrier of the assembly of < 6.3 mSv/h at 150 days falling to < 0.9 mSv/h at 547 days. The Be tile activation is at least 2 orders of magnitude lower than the Inconel 625 component.

Calculations for the mid-plane OPL assembly at 30 cm [17] show that shutdown dose at 30 cm after 120 days of < 6.6 mSv/hr falling to < 4.3 mSv/h at 180 days and < 1.2 mSv/h at 455 days (10 months after removal).

The shutdown dose date at 30cm for the stainless steel caddie with tungsten cover for housing foils for activation studies [18] gives the shutdown dose rate at 30 cm as <1.6 mSv/h at 120 days (30 days before removal). The shutdown dose rates of the activated foils, e.g., W, within the caddie are more than two orders of magnitude lower at < 0.0035 mSv/h.

3.3. Tungsten and beryllium activation

Both shutdown and contact dose rates for W and Be have been assessed however contact dose rate information is presented here as this represents the worst case and demonstrates the significantly lower dose rates involved compared with Inconel and Nimonic alloys.

Contact dose rates for W are initially very high, decaying rapidly to < 0.1 mSv/h in less than 30 days after the last pulse of DT operations [16] and would not be an issue when samples are removed from JET at 150 days, when components will have cooled to 0.032 mSv/h [16].

For beryllium at the outer mid-plane the contact dose rate 1 s after the last DT pulse is < 2 mSv/h, falling to < 0.017 mSv/h at 150 days [16]. Contact dose rates for Be in [17] are < 0.010 mSv/h at 120 days.

One of the issues with all calculations is the knowledge of trace elements in the tiles. Some experience has been gained from exposing three pieces of Be JET tiles at the Lilleström fission reactor, Norway as part of the ^{10}Be tracer experiment installed in JET [20]. The three pieces were exposed to a total neutron fluence of 200.1×10^{15} n/cm^2. With an approximate surface area of 200 cm^2, this gives a flux of 40×10^{18} n, lower than the expected n flux for DTE2. The main aim was to increase the ^{10}Be isotope concentration; however activation of impurities ^{60}Co and ^{46}Sc resulted in a measured dose rates of one of the samples of 0.060 mSv/h. The measurement was taken approximately 100 days after removal from the reactor and at a distance of a few centimeters. The presence of these impurities already gives a dose rate exceeding calculated contact dose rates for Be.

4. Discussion

With the estimation of T retention and calculations for activation of materials an assessment of requirements for handling components post-DTE2 can be made. T inventory is assessed in terms of the levels set for notification of radioactive materials given in the UK Ionising Radiation Regulations (IRR-99 (UK)) [21]. Exposure due to T and activation is assessed in terms of occupational dose limits set in the IRR-99 (UK) and IEAE safety specifications (IAEA-SS) [22]. The contact and shutdown dose rates discussed sections 3.2.1 and 3.2.2 are used to consider whether effective dose rates to operatives fall within defined exposure limits. Whilst this discussion considers the worst case for exposure, in reality all work handling radioactive materials would require measures to minimize exposure to operatives to a level as low as reasonably practicable as mandated by the IRR-99 (UK).

UK occupational ionising radiation exposure limits set out in the IRR-99 (UK) are 20 mSv/year to the body and 500 mSv/year to extremities (hands, forearms, feet, and ankles). CCFE occupation exposure limits are set an order of magnitude lower at 3 mSv/year for the body.

To assess handling at the point of removal from vessel (150 days) the worst case values are considered, i.e., the shutdown dose rates for Inconel 625 at 30 cm [17]. Data points are available at 120 day and 180 days, interpolating between these points the shutdown dose is 5.5 mSv/h at 150 days. In terms of the current CCFE exposure limits, handling this component at removal would not be possible as the potential effective dose to an operative would limit handling to half an hour. In terms of the UK limits for annual body dose, limited handling could be considered. These results, however do not give data for evaluating extremity exposure. In the case of extremity exposure data the worst case contact dose rate is < 160 mSv/h for Inconel 718 [16] on W-LBSRP components. These values would limit handling by operatives due to the potential effective dose to extremities. Based on these

results the removal of samples from vessel and placement in appropriately shielded storage would ideally be completed remotely.

If components are stored from 1 year significant cooling occurs. Shutdown dose rate at 30 cm from the mid-plane OPL component falls to < 1.2 mSv/h for Inconel 625 [17] and contact dose rate < 15 mSv/h [15] and < 6 mSv/h [16] depending on material composition. These effective and contact dose rates are far more manageable in terms of body and extremity exposure levels respectively. The contact dose rates for the Nimonic alloys are still high, < 53mSv/h [16], however the components are relatively small therefore effective dose rate at 30 cm would be lower. In contrast Be and W present a significantly lower hazard as their contact dose rates are of the order of a few micro Sieverts at the time they would be removed from the vessel. Taking this into account it is conceivable that the Be and W tiles could be separated from their more active carriers and fasteners provided that sufficient planning and shielding is implemented to minimize the dose to operatives. This would allow the Be and W tiles to be processed for analysis provided T levels could be handled.

Some data has been gained on handling the active Be tiles exposed at Lilleström. Operatives have handled these tiles to assemble them in 2011 and to take surface samples from them in 2013. During this work operatives wore extremity dosimeters located on the fingers. During the original assembly an operative received an extremity dose of 0.13 mSv. During this procedure handling of the tile was expected to be < 0.5 hours giving an actual contact dose rate > 0.26 mSv/h. The results indicate a significantly higher dose rate than the measured value 0.060 mSv/h and is even higher than calculated contact dose rates. This highlights the important role that impurities play in activation and shows that higher values than expected could arise if material specifications are not known in detail.

From section 3.1 the areal concentration of T is estimated at ~10^{17} T atoms/cm^2. By scaling TDS results for D concentrations for 2011-2012 [8] the distribution of T between the divertor and wall tiles is expected to be 10^{18} - 10^{19} T atoms/cm^2 for divertor and 10^{17} - 10^{18} T atoms/cm^2 for Be wall tiles for 1.91×10^{26} T atoms injected during DTE2. This gives T inventories for divertor and wall tiles of 10^9 - 10^{10} Bq/cm^2 and 10^8 - 10^9 Bq/cm^2. For whole tiles the total T inventory will be of the order $0.5 - 5 \times 10^{12}$ Bq and $0.020 – 0.20 \times 10^{12}$ Bq, assuming surface areas ~ 500 cm^2 for divertor tiles and ~ 200 cm^2 for wall tile pieces. Masses are of the order 2000 g for divertor tiles and 200 g for Be wall tiles giving specific T inventories > 10^8 Bq/g. The surface area of laboratory samples post processing is ~ 1 – 2.5 cm^2 giving a total inventory > 10^8 Bq. Small samples would weigh a few grammes giving specific T inventory of > 10^8 Bq/g. At these levels the handling of whole tiles and laboratory samples would require notification under the IRR-99 (UK) as the following T levels are exceeded: 1×10^9 Bq or > 1×10^6 Bq/g. In addition, transport regulations will apply [23]. Based on this assessment, only laboratories with an appropriate T licence would be able to accept tiles or samples from DTE2. A similar assessment of T inventory of samples from 2011-2012 tiles is shown in section 2. When

compared with the requirements of the IRR-99 (UK) these samples may be handled by a wider range of laboratories as the values for notification are not exceeded.

Be tiles from 2011-2012 are found to off-gas at a rate ~ 50 Bq/h for a tile containing ~ 10^{15} T atoms. If the T levels are scaled for post DTE2 T inventories of 10^{17} - 10^{19} T atoms, the off-gas rates could increase by two to four orders of magnitude to 0.05 - 5 × 10^{6} Bq/h for a Be tile. Off-gas from smaller laboratory samples will be two orders of magnitude lower. The derived air concentration (DAC) for T exposure (derived from annual exposure limits) is 0.2 × 10^{6} Bq/m^{3} [22]. Based on these maximum exposure limits, it will be necessary for laboratories handling samples to take precautions to mitigate operator exposure. In particular for whole tiles with higher T inventories, an operator could be exposed to the DAC level in less than an hour without sufficient protection. The required protection can be achieved by working in isolators with glove ports or slit boxes with sufficient air velocities across openings and wearing appropriate respiratory and personal protection to minimize inhalation, ingestion and contamination.

Be exposure from handling tiles will remain largely unchanged from DD to DTE2 operations. However exposure should also be limited. In the UK the workplace exposure limit in the Control of Substances Hazardous to Health (COSHH) Regulations 2002 is 0.002 mg/m^{3} of air averaged over an 8 hour period [24]. Samples cut from JET tiles are not routinely monitored for Be contamination as this involves sampling from surfaces to be analysed. In principle a laboratory taking precautions for T exposure will also provide sufficient control for Be exposure.

5. Conclusions

The ability to handle samples from JET-ILW and prepare laboratory samples for surface analysis and characterisation has been shown. In particular a new process for cutting Be tiles has been developed and information about the swarf produced presented. T inventories and activation calculations indicate that handling samples after DTE2 will pose additional hazards. Activation of carrier and fastener materials makes the handing of assemblies containing tiles more challenging. Dose rates at the point of removal from vessel are sufficiently high that remote handling is the preferred method of handling. After a cooling period of 1 year it may be possible to handle these components in appropriate laboratories where shielding is possible to remove the Be and W samples of interest. T inventory levels of on removal from the vessel will be high and therefore only laboratories with appropriate T handling capabilities and appropriate licencing will be able to handle samples.

6. Acknowledgements

This work has been carried out within the framework of the Contract for the Operation of the JET Facilities and has received funding from the European Union's Horizon 2020 research and innovation

programme. The views and opinions expressed herein and do not necessarily reflect those of the European Commission.

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8. Figure captions

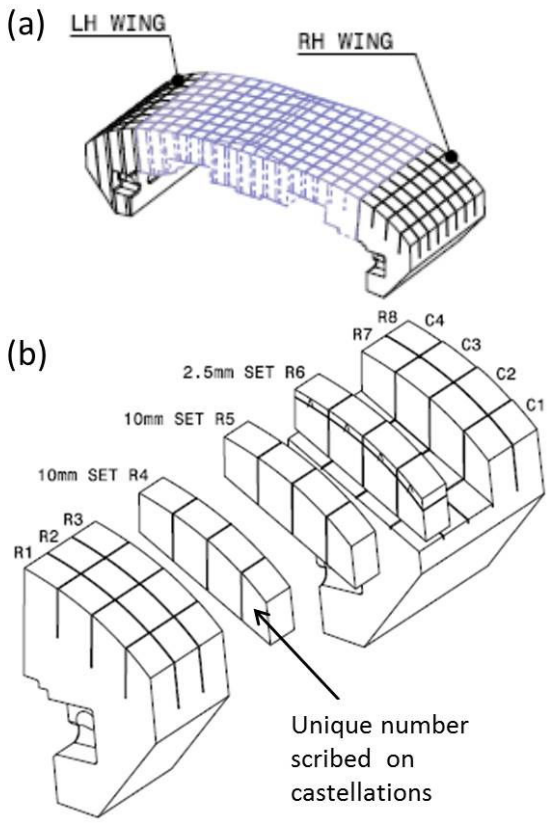
Figure 1. Example of cutting schedule for right hand wing tile of an inner wall guard limiter.

Figure 2. Be particulates produced during cutting.

Figure 3. Photographs from cutting of Be tiles. Curved wing tile from an Inner Wall Guard limiter is shown. (a) cutting the tile piece in half from the back side, (b) cutting rows of castellations.

Figure 4. Summary of contact dose rates (closed symbols) and shutdown dose rate (open symbols) data from [15][16][17][18].

9. Figures



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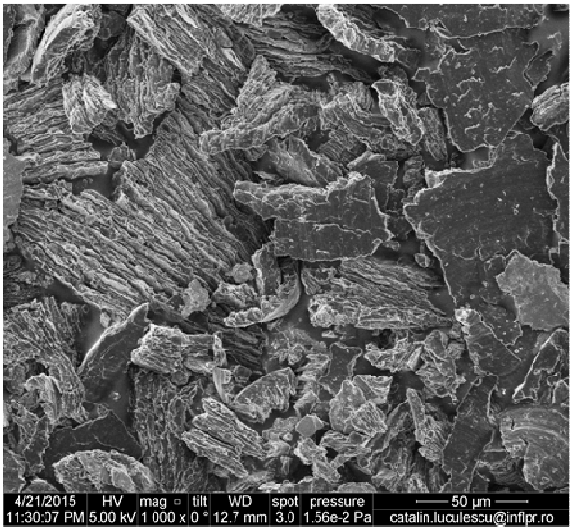

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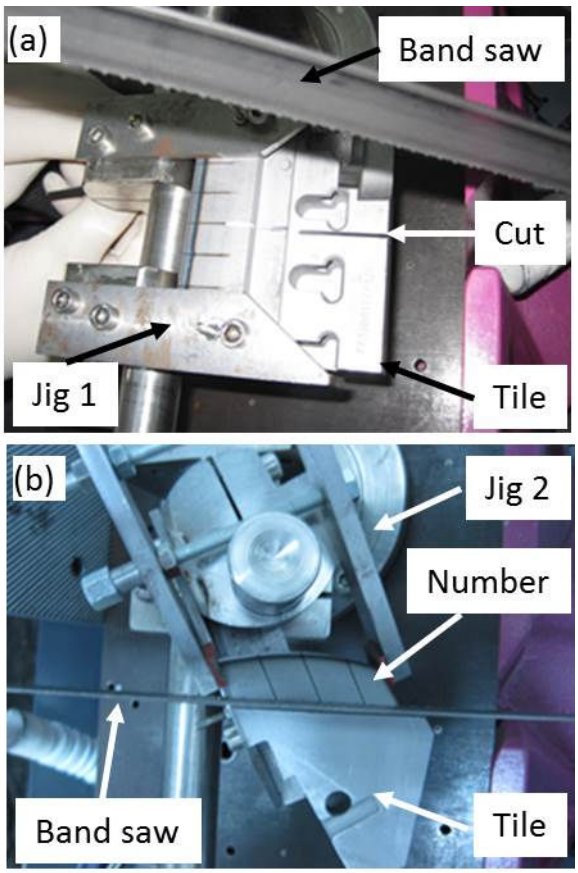

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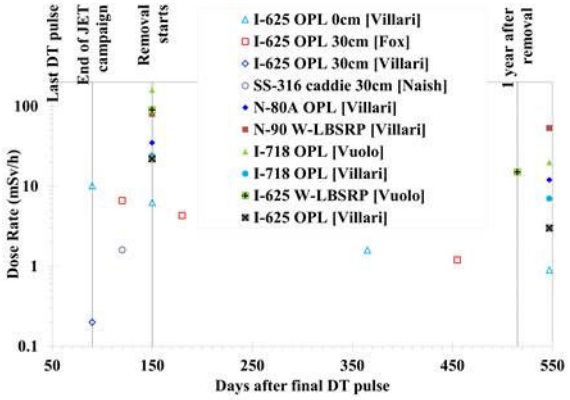

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