PAPER

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IOP Publishing | Royal Swedish Academy of Sciences Physica Scripta

Phys. Scr. **T167** (2016) 014057 (10pp) doi:10.1088/0031-8949/T167/1/014057

Experience of handling beryllium, tritium and activated components from JET ITER like wall

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Received 8 June 2015, revised 4 November 2015
Accepted for publication 11 December 2015
Published 25 January 2016

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: JET, beryllium, tritium, dose rate

(Some figures may appear in colour only in the online journal)

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

[8] See the appendix of F Romanelli *et al Proc. 25th IAEA Fusion Energy Conf. 2014 (Saint Petersburg, Russia).*

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 minimising 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 characterisation and micro-beam analysis are mounted in cold epoxy and polished using a virtually water free diamond suspension to minimise reduction in D and T content by isotopic exchange [4, 5]. In addition samples with a surface area $\sim 12 \times 12\ \text{mm}^2$ and $\leqslant 3$ mm thick are prepared for thermal desorption spectroscopy (TDS) using a back heated stage.

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 five pieces, in figure 1(b) the cutting requirements for one section, a right

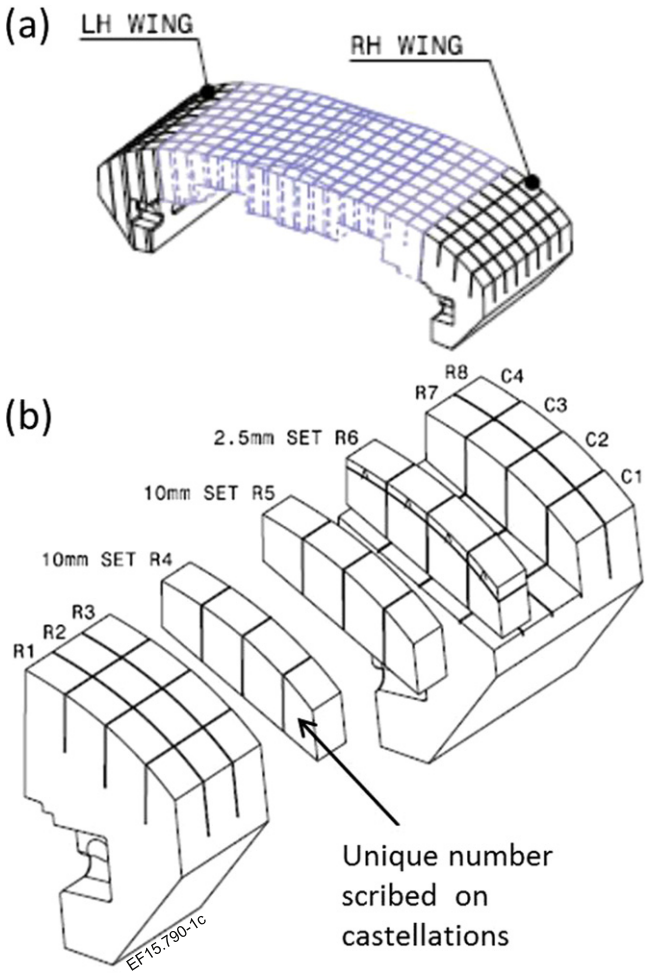

Figure 1. Example of cutting schedule for right hand wing tile of an inner wall guard limiter.

hand wing, is indicated: this is generally 2–4 rows of castellations taken in the toroidal direction from the centre of the tile. The depth of the cut from the tile surface is set according to the 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, a sample thickness 2–3 mm is needed. It is not possible to cut thinner slices as internal stresses results in bowing of the samples.

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

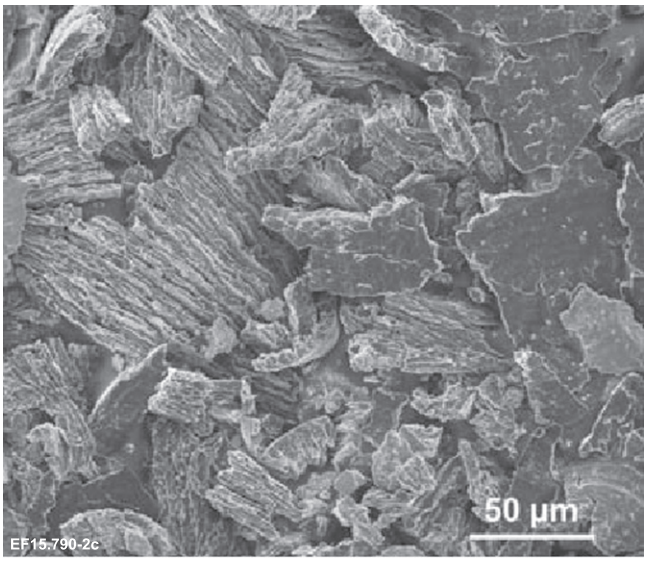

Figure 2. Be particulates produced during cutting.

from aluminium alloy. Using these mock-ups the mounting jigs were designed and manufactured. The jigs were designed to minimise vibration and cross contamination to tile surfaces.

During cutting it is necessary to keep the temperature of the samples low to minimise 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 optimising the band saw speed and applied load it is possible to keep the temperature of the samples <55 °C. 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 trials 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^{-1} 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 minimise exposure. Figure 2 shows an image of particulates from cutting. The morphology of the particulates is indicative of production by brittle fracture in that they appear as flakes with well-defined edges and surfaces and do not resemble ductile failure whereby elongation of particles would be visible. After cutting, 1 g of swarf material was sieved using different sized meshes to give an indication of the size distribution in the sample. It was found that 1 mg of the particles were $<36\ \mu$m. In this fraction respirable dust, i.e., particulates $<10\ \mu$m, may be present although no additional separation was performed to confirm this. The remaining particle distributions were 136 mg in the range $36\ \mu\text{m} \leqslant 90\ \mu$m and 863 mg $> 90\ \mu$m.

Using these cutting 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 $\sim 10^{14}$ T atoms cm^{-2} and $\sim 10^{13}$ T atoms cm^{-2} respectively. Taking into account the surface area of the samples ~1 cm^2 and assuming a specific activity of T of 357×10^{12} Bq g^{-1}, 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^{-1}.

During 2015 a cutting procedure for bulk 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 8 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. An understanding of the increase in T inventory and activation of the machine is needed for several reasons. Firstly, before operations begin estimates of T inventory and activation are vital for the safety case which must be approved before TT and DT operations may proceed. Secondly, during TT and DT operations it is essential to understand how the T inventory and activation of the vessel increases with operation time in case of unforeseen breakdowns requiring remote handling entry to the vessel. Thirdly, at the end of TT and DT operations the T inventory and activation of components is required to assess the liability to be transferred to the Nuclear Decommissioning Agency.

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, 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.

Table 1. Composition of NIMONIC alloys [17, 19] and stainless steel 316 [16, 20] (wt%).

Material	Ni	Cr	Mo	Mn	Cu	C	Si	B	P	S	Fe	Co	Al	Ti	Zr	Pb
NIMONIC 80A	69.875	19.0	—	1.0	0.2	0.1	1.0	0.008	—	0.015	3.0	2.0	1.4	2.25	0.15	0.003
NIMONIC 90A	55.333	19.0	—	1.0	0.2	0.13	1.0	0.0	—	0.015	1.15	18.0	1.5	2.5	0.15	0.0020
Stainless Steel 316	12.00	17.00	2.500	1.014	—	0.041	0.507	—	0.023	0.015	66.9	—	—	—	—	—

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 3.7×10^{23} D giving a global long term retention of ~0.2% [6]. Taking into account the operating time for 2011–2012 of 6.8×10^{2} s the global retention rate is 5.5×10^{18} D s^{-1} [6].

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 D and H 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 [7]. 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 0.2% [6] 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 T retained after purging is expected to be 1.90 g in deposits, i.e., 0.2% of 950 g T injected. Whilst the error in the surface analysis of individual tiles is <6% [6] the error in deriving the global retention value is dominated by the extrapolation from individual tiles to cover the whole vessel surface. Therefore the retention value could be 30% higher. Taking this into account the long term retention in tiles could be up to 2.47 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. The contribution of 0.03% retention in the bulk material of the JET-C tiles is not considered as the value may be subject to cross contamination during cutting [8] and evaluation.

From isotopic exchange experiments the accessible reservoir of hydrogen isotopes for exchange is shown to be 2.3×10^{22} atoms [9]. This is <0.02% (0.2 g) of injected T and would reduce the T inventory by ~10% of the total retained T, if a clean-up phase is considered.

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.

The discussion so far estimates the final T inventory on opening (i.e. purging) the JET vessel to air, assuming the full 950 g throughput of T fuel in the vessel. Estimates due to unforeseen openings or breaches of the JET vacuum vessel during TT or DT operations may be determined from the long term T retention rate; 5.5×10^{18} T s^{-1} for TT operations and 2.8×10^{18} T s^{-1} for DT operations. Alternatively the value may be estimated from the T accountancy from operations and the global retention value. By either method the value will not exceed the overall T retention.

Enhanced T retention due to 14 MeV neutron damage during DT operations is not be a significant issue in JET materials. Typical dpa calculated for W, Be, C and Inconel are <40 μdpa for a total neutron yield of 1.7×10^{21} DT neutrons over a surface area of ~100 m^2. This is three orders of magnitude lower than materials considered to be irradiated at low levels [10].

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, FISPACT-II [11], and data libraries, EAF-2007 [12] and MCNP5 [13]. The neutron flux in in-vessel components is calculated [12, 13] using the emission source from typical DT plasmas. An example neutron energy spectrum for the outer poloidal limiter in the main chamber of JET using MCNP5 is shown in [14]. Such a spectrum provides the input for activation calculations using FISPACT-II [11], also described in [14]. For each irradiation scenario, the contact dose rate, i.e., at the surface of the element under analysis, is calculated with increasing time after irradiation. 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 [15, 16] and neutron and decay gamma transport [17] (details of the method are given in [18]) 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 modelling 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% [16]. In this work a total neutron yield of 1.7×10^{21} DT neutrons [21] 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 [14, 15, 17] and foils for activation studies and associated caddie [16]. Within each of these components a range of materials have been considered, e.g., Inconel 625, Inconel 718, stainless steel 316 (SS316), Nimonic 80A, Nimonic 90, W, Be. The composition of these materials used in the activation calculations are given in tables 1–3. CFC components such as divertor tiles and carriers are not considered: although these assemblies contain fasteners manufactured from Inconel and Nimonic alloys they

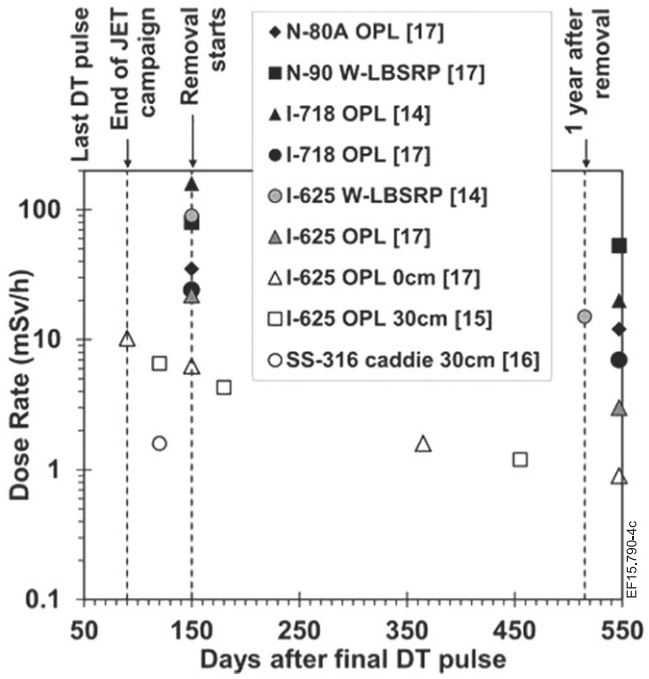

Figure 3. Summary of contact dose rates (closed black and grey symbols) and shutdown dose rate (open symbols) data from [14–17].

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 3.

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 dose rate [14]. 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 (wt%), i.e., at least an order of magnitude higher compared with Inconel 625, resulting in ^{60}Co nuclides, see tables 1 and 2. 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 $<35\,\mathrm{mSv\,h^{-1}}$. For Nimonic 90 in the W-LBSRP this is $<80\,\mathrm{mSv\,h^{-1}}$. If the materials are left to cool down, contact dose rates fall to $<12\,\mathrm{mSv\,h^{-1}}$ and $<53\,\mathrm{mSv\,h^{-1}}$ for Nimonic 80A and Nimonic 90 respectively at 547 days (1year and 1 month after removal) [17].

The alloy Inconel 718 is also used for specialist fasteners/screws on OPL and W-LBSRP assemblies. Results from [14] for the outer mid-plane OPL location return the highest activation values. The high activation values are due to high ^{182}Ta content (3.125 wt%) listed in the material impurities which dominates the short term activation, i.e. <1 year. At 150 days the contact dose rate for Inconel 718 $< 160\,\mathrm{mSv\,h^{-1}}$ and at longer cooling times this drops to $<20\,\mathrm{mSv\,h^{-1}}$ at 547 days [14]. An updated composition with lower ^{182}Ta content (0.076 wt%) [17] gives contact dose rates of $<24\,\mathrm{mSv\,h^{-1}}$ at 150 days falling to $<7\,\mathrm{mSv\,h^{-1}}$ 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 [14] the highest Inconel 625 activation values are found in the W-LBSRP; $<90\,\mathrm{mSv\,h^{-1}}$ at 150 days and $<15\,\mathrm{mSv\,h^{-1}}$ at 515 days. With updated composition data [17] the contact dose rates for the OPL assembly are $<22\,\mathrm{mSv\,h^{-1}}$ at 150 days and $<3\,\mathrm{mSv\,h^{-1}}$ at 547 days.

3.2.2. Shutdown dose rate. Shutdown dose rates for mid-plane OPL [15, 17] and SS316 caddie [16] are presented here. For the OPL tile assembly the shutdown dose rate is dominated by Inconel 625 carrier, these results are shown in figure 3.

The study of a 3D model of the mid-plane OPL assembly [17] gives a shutdown dose rate at 0 cm for the Inconel 625 carrier of the assembly of $<6.3\,\mathrm{mSv\,h^{-1}}$ at 150 days falling to $<0.9\,\mathrm{mSv\,h^{-1}}$ 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 [15] show that shutdown dose at 30 cm after 120 days of $<6.6\,\mathrm{mSv\,hr^{-1}}$ falling to $<4.3\,\mathrm{mSv\,h^{-1}}$ at 180 days and $<1.2\,\mathrm{mSv\,h^{-1}}$ at 455 days (10 months after removal).

The shutdown dose rate at 30 cm for the stainless steel caddie with tungsten cover for housing foils for activation studies [16] are $<1.6\,\mathrm{mSv\,h^{-1}}$ 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 $<3.5\,\mu\mathrm{Sv\,h^{-1}}$.

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 $<0.1\,\mathrm{mSv\,h^{-1}}$ in less than 30 days after the last pulse of DT operations [17] and would not be an issue when samples are removed from JET at 150 days, when components will have cooled to $0.032\,\mathrm{mSv\,h^{-1}}$ [17].

For beryllium at the outer mid-plane the contact dose rate immediately after the last DT pulse is $<2\,\mathrm{mSv\,h^{-1}}$, falling to

Table 2. Composition of Inconel alloys (wt%). Compositions from [14] not reproduced.

Material	Ni	Cr	Mo	Mn	Cu	C	Si	B	P	S	Fe	Co	Al	Ti	Nb	Ta
Inconel 625[a] [17]	60.317	22.323	8.349	0.376	0.010	0.071	0.201	—	0.009	0.006	4.254	0.20	0.236	0.193	3.379	0.076
Inconel 625 [15]	75.001	15.7	—	0.41	—	—	0.16	—	—	—	7.9	0.05	—	—	—	—
Inconel 718 [17, 19]	52.50	19.00	6.1	0.35	0.3	0.08	0.35	0.006	0.015	0.015	13.784	1	0.50	0.875	5.075	0.05

[a] Composition based on manufacturer certificate supplied with material (not published) with the addition of 0.2% Cobalt.

Table 3. Impurities (wt%) used in activation calculations for tungsten and beryllium.

wt% 0.9995	1e-2	5e-3	3e-3	2e-3	1.5e-3	1e-3		5e-4	
Tungsten	Mo	P	C, Fe, O	Ni, Si	Al	Ba, Ca, Cd, Co, Cr, Cu, K, N, Na, Nb. Pb, Ta, Ti, Zr		Ag, As, H, Mg, Mn, S, Zn	
wt% 1	4.3e-4	1.14e-4	1.06e-4	1e-4	6e-5	3.7e-5	3.2e-5	3.1e-5	2e-5
Beryllium	Zr	N	Ni	W	Ti	Cr	U	Cu	Ca, Mo, Pb
1.5e-5	1e-5	9.7e-6	6e-6	5e-6	3e-6	2e-6	4.6e-7	3e-7	1.2e-7
Mn, Na	Cl, S, Zn	Ta	Co	F, Sc	Ag, Li	B, Cd	P	Hf	Nb

$<0.017\,\mathrm{mSv\,h^{-1}}$ at 150 days [17]. Contact dose rates for Be in [15] are $<0.010\,\mathrm{mSv\,h^{-1}}$ 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 [22]. The three pieces were exposed to a total neutron fluence of $200 \times 10^{15}\,\mathrm{n\,cm^{-2}}$. With an approximate surface area of $200\,\mathrm{cm^2}$, this gives a flux of 40×10^{18} neutrons, lower than the expected neutron flux for DTE2. The main aim was to increase the ^{10}Be isotope concentration; however activation of impurities giving ^{60}Co and ^{46}Sc isotopes resulted in a measured dose rate of one of the samples of $0.060\,\mathrm{mSv\,h^{-1}}$. The measurement was taken approximately 100 days after removal from the reactor and at a distance of a few centimeters. The Be tiles were exposed to thermal neutron energies in the range 10^{-4} eV–10^{6} eV, with the peak flux at 1 eV. This will result in different activation efficiencies compared to 14 MeV neutrons from DT operations, however the data suggests that the initial concentration of Co and Sc in the Be tiles is higher than the concentrations used here (table 3) to calculate contact dose rates.

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)) [23]. Exposure due to T and activation is assessed in terms of occupational dose limits set in the IRR-99 (UK) and IAEA safety specifications (IAEA-SS) [24]. The contact and shutdown dose rates discussed in 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 minimise exposure to operatives to a level as low as reasonably practicable (ALARP) as mandated by the IRR-99 (UK).

UK occupational ionising radiation exposure limits set out in the IRR-99 (UK) are $20\,\mathrm{mSv\,yr^{-1}}$ to the body and $500\,\mathrm{mSv\,yr^{-1}}$ to extremities (hands, forearms, feet, and ankles). CCFE occupation exposure limits are set an order of magnitude lower at $3\,\mathrm{mSv\,yr^{-1}}$ for the body.

To assess handling at the point of removal from the vessel (150 days), the worst case values are considered, i.e., the shutdown dose rates for Inconel 625 at 30 cm [15]. Data points are available at 120 day and 180 days, interpolating between these points the shutdown dose is $5.5\,\mathrm{mSv\,h^{-1}}$ 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 $<160\,\mathrm{mSv\,h^{-1}}$ for Inconel 718 [17] on OPL 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 for 1 year significant cooling occurs. Shutdown dose rate at 30 cm from the mid-plane OPL component falls to $<1.2\,\mathrm{mSv\,h^{-1}}$ for Inconel 625 [15] and contact dose rate $<15\,\mathrm{mSv\,h^{-1}}$ [14] and $<6\,\mathrm{mSv\,h^{-1}}$ [17] 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 $90 < 53\,\mathrm{mSv\,h^{-1}}$ [17] and Inconel $718 < 20\,\mathrm{mSv\,h^{-1}}$ [14], however the components are relatively small, i.e. fasteners and washers, 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 minimise 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 h giving an actual contact dose rate >0.26 mSv h^{-1}. The results indicate a significantly higher dose rate than the measured value 0.060 mSv h^{-1} and are 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 [6] 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 × 10^{12} Bq and 0.020–0.20 × 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^{-1}. 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 grams giving specific T inventory of >10^{8} Bq g^{-1}. 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^{-1}. In addition, transport regulations will apply [25]. 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 of ~50 Bq h^{-1} 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^{-1} 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} [24]. 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 minimise 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 Regulations 2002 is 0.002 mg m^{-3} of air averaged over an 8 h period [26]. 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 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.

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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