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Radiological protection considerations for fusion reactors

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Working Party on Research Implications on Health and Safety Standards of the Article 31 Group of Experts

Directorate-General for Energy

Directorate D — Nuclear Energy, Safety and ITER

Unit D3 — Radiation Protection and Nuclear Safety

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Foreword

Luxembourg, December 2023

The European Commission organises every year, in cooperation with the Group of Experts referred to in Article 31 of the Euratom Treaty, a Scientific Seminar on emerging issues in Radiation Protection – generally addressing new research findings with potential policy and/or regulatory implications. Leading scientists are invited to present the status of scientific knowledge in the selected topic. Based on the outcome of the Scientific Seminar, the Group of Experts referred to in Article 31 of the Euratom Treaty may recommend research, regulatory or legislative initiatives. The European Commission takes into account the conclusions of the Experts when setting up its radiation protection programme. The Experts' conclusions are valuable input to the process of reviewing and potentially revising European radiation protection legislation and may assist in the implementation of Council Directive 2013/59/Euratom (Basic Safety Standards Directive).

In November 2022, the EU Scientific Seminar covered the issue *Radiological protection considerations for fusion reactors*. Internationally renowned scientists presented the following topics:

- An introduction to basics of fusion technologies
- Basic radiation protection issues in future fusion reactors
- Radiological inventories and source terms
- Occupational Radiation Exposure
- Environmental Releases and public exposure in normal and accidental situations

The presentations were followed by a round table discussion, in which the speakers and additional invited experts discussed potential *policy implications and research needs* under the key areas of:

- Issues with tritium
- Dosimetry
- Radioactive waste issues

The Group of Experts discussed this information and drew conclusions that are relevant for consideration by the European Commission and other international bodies.

Contents

An introduction to basics of fusion technologies

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Thermonuclear fusion is supposed to harness the energy of stars on earth. It consists in fusing two light nuclei which then produce a very large quantity of energy, i.e. of the order of 100 times per unit of mass than the nuclear fission reactions.

In order for fusion reactions to occur, the medium must be very hot (in order to overcome the Coulomb repulsion). In ITER, the temperature at which the machine will operate will be of the order of 20 keV or ~200 million degrees. At these temperatures, the plasma is an electrically neutral medium composed of ions and electrons.

Furthermore, the plasma must be as concentrated (and pure) as possible. To achieve a stable plasma its concentration must be low compared to that corresponding to the atmospheric pressure: it operates in a vacuum chamber.

Finally, in order to produce energy, it is necessary to confine the plasma: one of the solutions considered is a confinement by magnetic field.

The most used and advanced configuration worldwide is the tokamak configuration. A tokamak is a vacuum chamber containing a mixture of low atomic mass ions heated to high temperature and confined by a strong magnetic field.

The reaction that has the largest cross section (the easiest to obtain) at the temperature considered in ITER involves deuterium and tritium, producing an alpha particle and a neutron. The energy of the neutron is very high (14 MeV). The interaction of the neutron with the materials surrounding the plasma and constituting the vacuum vessel produces their activation.

Deuterium is a stable isotope of hydrogen that is abundant on earth (~150 ppm in water).

Tritium is a radioactive compound with a half-life of 12.5 years. It is currently produced in quantity in CANDU fission reactors. For the future of the fusion at an industrial level, tritium will have to be produced during the operation of the reactor by means of the reaction of fusion neutron with an isotope of lithium (Lithium-6) in the tritium breeding system surrounding the plasma.

Many tokamaks are currently in operation in the world.

Today, JET in Culham, England is the only one able to perform fusion reactions and to use tritium. Its radius is 3 m, its magnetic field is 3.45 T and its plasma volume is about 50

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m^3. The experiments are short (some seconds to a few tens of seconds). The fusion power produced over short periods (5 to 10 s) is 5 MW maximum.

ITER, under construction in the south of France, is a global project involving Europe, China, Japan, Korea, India, the United States and Russia. ITER will not produce electricity: ITER must demonstrate the scientific and technical feasibility of fusion as a source of energy and test the breeding blanket for tritium production. Its radius is 6 m, its magnetic field is 5.3 T and its plasma volume is about 830 m^3. The fusion power is supposed to be 500 MW for long duration of plasma (300 s).

Beyond ITER, an electricity-producing reactor coupled to the grid is being considered. It is currently called DEMO. Its characteristics are still under discussion, but it will be larger than ITER. Its radius could be of 8 to 9 m, its plasma volume of 1000 to 3500 m^3 with a magnetic field identical to that of ITER.

Regarding tritium issue and its management in a fusion machine such as a tokamak, a few very important points should be noted:

- The amount of tritium in plasma is very low (0.5 g).
- The circulation of tritium in the reactor (the vacuum vessel but also the tritium plant and all the auxiliaries) is very large due to the low efficiency of tritium injection in the confined plasma. For example, in the case of a full power fusion reaction of 500 MW in ITER, 2 10^{-4} g/s are burned by the reaction while the quantity of tritium injected into the vacuum chamber is 4 10^{-2} g/s, more than two orders of magnitude difference.

Finally, the quantity of tritium present on the site is high about 4 kg. This implies that tritium interacts with many materials: those of the vacuum chamber such as tungsten but also those of the tritium plant and auxiliary such as 316L steel.

Tritium is extremely mobile in all these materials and it can permeate very easily. This can lead to tritium releases that must be controlled.

Moreover, at the end of the life of a fusion reactor, tritiated waste that has been generated must be particularly well managed in order to avoid any contamination of the environment.

The fusion community is aware of all these tritium issues: many R&D activities are underway to ensure the lowest possible tritium releases during the operation of the machine and during its dismantling.

As a conclusion, in a tokamak:

1. **There is no possibility of uncontrolled fusion reaction (no chain reaction)**. This is mainly due to a very small quantity of tritium in the plasma. Moreover, the fuel of the reaction e.g. the deuterium/tritium mixture is injected continuously in the vacuum vessel. Shutting down the fuel injection will stop fusion reactions. Finally, a massive gas injection would suffocate fusion reactions inducing a very fast actuator stopping the reactor.
2. **A very large re-circulation of tritium in the installation (tokamak, tritium plant, ...) is observed.** This leads to tritium implantation in materials that can then degas inducing waste management issues. Tritium is very mobile in materials and tritium permeation related to possible contamination of the cooling loop or

environment can occur. Large R&D activities are currently ongoing to limit tritium contamination or induced releases.

3. **A priori, no long-lived wastes are produced (if materials are well chosen) but particular materials have to be managed.** Such materials include massive beryllium and beryllium dust in the ITER vacuum vessel (in the vacuum vessel of future fusion reactors beryllium will not be used), breeding system (as PbLi, Be) for which R&D is necessary, and waste with a high tritium content leading to confinement issues (releases control).

Basic radiation protection issues in future fusion reactors

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Abstract

The goal of this paper is to give an overview of general safety and radiation protection issues to be expected in future nuclear fusion reactors, especially in the European DEMO. Based on the ITER facility, DEMO[3] will be the first fusion commercial reactor in Europe which will produce electricity.

A quick outline of radioactive materials which may be found in fusion facilities is given. Starting from differences in scale and concept between ITER and DEMO, and technological challenges to overcome in order to make fusion power industrializable and sustainable, the paper concludes with a comparison of the main safety issues and issues with worker and public exposure between ITER and DEMO. These issues will need to be dealt with since the design-phase of DEMO reactors.

1 Introduction

IRSN is the Technical Safety Organization (TSO) and a major research institution in the nuclear and radiation safety in France. It is in charge, on behalf of the French nuclear safety authority (ASN), of evaluating safety and radiation protection for all nuclear installations in France, including ITER. ITER, currently under construction in southern France, is an international nuclear fusion research and engineering project aimed to "demonstrate the scientific and technological feasibility of fusion power as a large-scale, carbon-free source of energy".

To reproduce the conditions for nuclear fusion on Earth, high temperatures (150 M°C) and intense pressure are needed. Moreover, enough confinement is needed to hold the plasma and maintain the fusion reaction long enough for a net power gain.

Magnetic confinement fusion is an approach to generate thermonuclear fusion reactions in a plasma of ions and electrons at high temperature, confined by magnetic fields. Magnetic fields are made by coils. On the other hand, inertial confinement fusion generates thermonuclear fusion reactions by compressing and heating targets filled with

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[3] From here on out, "DEMO" refers to the "EU-DEMO".

thermonuclear fuel. Fuel is confined in a tiny capsule, surrounded by powerful lasers which compress and heat up targets until reaching the fusion conditions.

A Tokamak (literally "toroidal chamber with magnetic coils") is a vacuum chamber surrounded by strong magnetic fields which uses magnetic confinement to generate thermonuclear fusion reactions. It is the most widely studied magnetic confinement fusion device at this time.

2 Types of radioactive materials to be found in fusion facilities (including waste)

2.1 Tritium

Tritium is used to fuel the fusion reaction and for this reason it is one of the most abundant radioactive materials that can be found in fusion facilities. It has a half-life of about 12.5 years and decays into ^3He by beta decay.

In a tokamak, tritium is present mainly in the tritium building (several kg), which is the equivalent of the "fuel building" of fission reactors, and in the vacuum vessel (VV), which is the plasma chamber. To a lesser extent, it can be found in the hot cell building. In this building, maintenance operations are performed on the vacuum vessel internal components and wastes are stored.

In fusion reactors, tritium can be found as a gas (HT) or as tritiated water (HTO).

HT is high volatile and easily diffuses across materials and seals. Without intervention, the quantity of tritium absorbed onto the first wall of the vacuum vessel internal components could become very large. In order to avoid losing too much fuel and to avoid excessive consequences in the event of an accident involving tritium, thermal desorption is considered in order to remove the tritium from the first wall of VV internal components.

HTO is produced when tritium atoms exchange with ordinary hydrogen atoms in water molecules (H_2O). It is often corrosive for the equipment involved. Tritium is most dangerous as tritiated water. Therefore, the IRSN recommend that operators consider all tritium as tritiated water when accounting for worker exposure and exposures of members of the public.

2.2 Activated materials or products

Interactions between neutrons from fusion reactions in the plasma and the nearby environment produce:

- ACTIVATED STRUCTURAL MATERIAL (blankets, divertor, VV sectors, field coils, and all support systems such as heating, diagnostics, cooling, vacuum…). Those elements are the main wastes;
- ACTIVATED DUST resulting from the erosion of the first wall of plasma-facing components mostly during plasma quench. Dust must be limited as it disturbs the plasma and may be a source of accident (e.g. a tritium/dust explosion in the VV). For this reason, dust is regularly removed from the VV;

- WATER ACTIVATION PRODUCTS in primary cooling systems (^{14}C, ^{16}N, ^{17}N, ^{19}O and tritium[4]). ^{16}N, an isotope of nitrogen generated by neutron activation of oxygen contained in the water, is a high-energy γ-emitter[5] with a half-life of 7.1 s;
- ACTIVATED CORROSION PRODUCTS IN WATER in primary cooling system (ions in solution and non-fixed deposits on the walls), mobilizable in the event of water leak;
- ACTIVATED GASES: the air present between the cryostat and bioshield, and to a lesser extent, the air in the tokamak building rooms, is activated. The main isotopes entailing irradiation risks are ^{14}C and ^{41}Ar.

3 ITER

ITER is an international nuclear fusion research and engineering project aimed to demonstrate the scientific and technological feasibility of fusion power, in the path to build commercial fusion reactors. Being built next to the Cadarache facility in southern France since 2012, it will be the world's largest experimental tokamak nuclear fusion reactor.

The ITER construction license decree passed in 2012 based on the in-depth assessment of the safety and radiation protection measures adopted by ITER Organization[6] conducted by the IRSN. The PSAR (Preliminary Safety Analysis Report) is one of the regulatory documents which must be submitted by the operator when asking for the construction license of a Basic Nuclear Installation (Installation nucléaire de base). For ITER, it has been deposed by ITER Organization (IO) in 2010 [1].

3.1 What are ITER main safety issues?

3.1.1 Tritium confinement

Because of its large radioactive inventory, especially of tritium, ITER is the first fusion reactor to require a construction license decree under the French nuclear facilities regulation. In fact, for the French law, about 27 g of tritium are enough to have an installation classified as NBI. ITER will host up to 4 kg of tritium.

Tritium releases as a gas from ITER are mainly associated with transfers of internal components to the hot cells and their processing in the cells during years of heavy maintenance, and with treatment of tritiated waste in other years.

In a fusion facility, detritiation systems are key elements for the containment of gaseous tritium. In ITER, a "scrubber column" is used to capture T2 in water. Tritiated water is then processed by a water detritiation system, which uses components such as electrolyzers and permeators.

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[4] Tritium production by activation of the water is negligible compared with the quantities of tritium which diffuse from the VV into the cooling systems.

[5] 6.1 MeV principally.

[6] ITER Organization acts as the overall integrator of the project and nuclear operator of the ITER facility.

Even if the scrubber column technology is well known, ITER will be the first facility to use detritiation system in such a large scale. Qualification of this system to normal, incident and accidental conditions[7] is still ongoing.

In the PSAR of ITER it is stated that other gaseous radioactive elements, mainly resulting from neutron activation of uranium impurities in beryllium metal[8], are negligible. However, this statement has been brought back into discussion recently (2021) by ITER Organization. This may result in important changes in the radiological consequences of accidents, ventilation and detritiation system design and dust production. IRSN is waiting for elements to start a technical instruction on this topic.

3.1.2 Radiation protection of public and workers

In a fusion reactor, radiation exposure risk is mainly driven by 14 MeV neutrons. Neutron energy is very high, and more important in terms of risk than in fission reactors. For this reason, access to the tokamak building is forbidden during plasma pulses.

Interactions between neutrons from fusion reactions in the plasma and the materials in the nearby environment produce high-energy γ-emitting radionuclides by neutron activation. Due to high dose rates, incompatible with human life, associated with ITER components, robotic means are used to transfer and treat those components that need maintenance.

4 Differences in scale between experimental facilities (ITER) and future reactors (DEMO)

Based on IRSN technical expertise acquired on ITER since 2006 and other scientific publications, the IRSN first published in November 2017 the "Nuclear fusion reactors" report [2]. The purpose of this work is to present preliminary safety and radiation protection issues that should be examined in DEMO, since early design stage.

The conceptual design of the tokamak will be the same but future demonstration reactors will differ from ITER reactor mainly in that they will be seeking to attain tritium self-sufficiency and achieve significantly longer operating times.

These differences, among others, are expected to have a significant impact on design and a direct influence on safety and radiation protection.

4.1 Self-sufficiency in tritium

DEMO will have a significantly bigger tritium inventory (7.5 kg vs 4 kg for ITER) and consumption (60 kg/y vs 1 kg/y for ITER) than ITER (see *Table 1*). For ITER, tritium will come from the detritiation of heavy water of Canadian CANDU reactors. For DEMO and

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[7] ITER Organization claims a 99 % efficiency of detritiation system during normal operation and 90 % during accidental condition including in an event of fire.

[8] Beryllium has been chosen as one the elements covering the first wall due to its physical properties (low plasma contamination, low fuel retention).

other future commercial fusion reactors, this will not be sufficient because of the large quantity of tritium involved and consumed.

As naturally occurring tritium is extremely rare on Earth and very expensive to produce, self-sufficiency in tritium is essential for the industrialization of fusion reactors. In order to recover tritium losses[9], future reactors will produce more T than consumed via TBM (Tritium Breeding Modules), lithium-based modules which produce tritium under neutron bombardment.

A few parameters must be considered when dealing with self-sufficiency in tritium (see *Table 1*):

- TBR (Tritium Breeding Ratio) is the ratio between tritium production by TBM and consumption. The lower the minimum TBR to be achieved, the easier the self-sufficiency in tritium;
- Burn-up fraction: percentage of the tritium introduced that undergoes nuclear fusion reaction. The greater the burn up fraction, the less tritium is involved and the minimum TBR is lower;
- Time for recycling: the shorter, the lower the minimum TBR.

Table 1: *Differences between ITER and DEMO in tritium consumption*

	ITER	**DEMO**
Fusion power [MW]	500	2000
Tritium inventory [kg]	4	Up to 7.5
Tritium consumption [kg/y]	1	60
TBR (minimum)	Not applicable	1.1 to 1.2
Burn-up fraction [%]	0.3	1 to 5
Time for recycling	off plasma pulses	during plasma pulses btw 1 and 24 h

The choice of TBM technology, number of modules and emplacement, as well as the burn-up fraction, is essential to attain self-sufficiency in tritium. Tritium produced by TBM must be maximized taking into account a limited surface of the VV covered by TBM. Numerous publications covering the ability of a nuclear fusion facility to be self-sufficient in tritium show that this objective is difficult to achieve.

[9] Losses: radioactive decay and permeation.

As tritium releases into the environment strongly depend on the type of TBM adopted (see §4.4) and, to a lesser extent, on the target burn-up fraction, IRSN considers that the safety impact of the choices to attain self-sufficiency in tritium must be taken into account since early design stage.

4.2 Operating times

For ITER, the average operating time with plasma will be only 1 % (see Table 2). The operating time is needed to be extended for the industrialization of fusion reactors.

As DEMO is expected to have longer operating time, ionizing radiation exposure will be (public/workers) more severe than ITER. Furthermore, the DPA (number of Displacements Per Atom) is expected to be bigger in DEMO (see Table 2).

Table 2: *Differences between ITER and DEMO in operating time with plasma*

	Average operating time with plasma [%]	**DPA**
ITER	1	2 to 3
DEMO	30 to 70	Up to 150

The DPA estimate the average displacements of each atom under irradiation and is one of the key parameters to evaluate the irradiation damage. In fact, materials may experience loss of ductility under irradiation, for example.

A longer operating time will induce a faster deterioration of the initial properties of the materials of the structures surrounding the plasma. Therefore, the IRSN considers that the choice of materials to withstand intense neutron bombardment is essential to ensure the safety of future fusion reactors.

Furthermore, the time needed to extract blanket modules and divertor cassettes of ITER will be probably incompatible with the operating time target of DEMO[10]. The maintenance operation time is then needed to be reduced for the industrialization of fusion reactors. This may lead to major design impacts[11]. Therefore, the IRSN considers that the safety impact of the choices made in order to reduce maintenance operation time must be evaluated since early design stage.

4.3 Residual heat removal

Residual heat removal is not a safety function in ITER since the temperature rise slowly in case of loss of cooling systems. For DEMO, residual heat is much higher due to longer operating time and greater fusion power (see Table 3).

[10] It is estimated that the replacement of all plasma facing components (blankets and divertor) will take up to 2 years for ITER.

[11] It is estimated that the surface area of DEMO Hot Cell Complex will be up to 6 times the ITER one

***Table 3:** Differences between ITER and DEMO in residual heat after plasma operation*

	Fusion power [MW]	Residual heat at shutdown [MW]	Residual heat after one day [MW]
ITER	500	11	0.6
DEMO	2000	80	30

In DEMO, a new accident scenario must be considered in case of a loss of tokamak's cooling system combined with a loss of vacuum of the VV: a rise in the temperature of the first wall combined with an air ingress may produce other radioactive gases to be confined, like tungsten trioxide (WO_3), a high volatile radioactive aerosol.

Therefore, since the design of tokamak's cooling systems is heavily dependent on the assessment of issues concerning residual heat removal, the IRSN considers that it is essential to cover these issues from the safety options stage.

4.4 Accidents to be considered in DEMO since early design stage

Preliminary studies show that most of accidents to be considered for DEMO as design basis and Design Extended Conditions are identical to ITER. However, the likelihood and consequences may be very different because of the greater radioactive inventory and energy involved in DEMO. Also, other types of accidents may need to be considered due to the increased complexity of DEMO vs ITER (like the production of WO_3).

In ITER, plasma discharges are short, because the plasma current is produced by gradually increasing the current in the central solenoid. In DEMO, in order to increase operating time with plasma, permanent discharges without any limitation of duration are planned. Consequently, a more sophisticated plasma control system (PCS) is necessary in order to obtain a good control of plasma stability, and therefore malfunctions of the PCS may be more frequent/consequences may be different/more severe. These malfunctions of the PCS must be considered in DEMO.

In DEMO, the magnetic energy of coils will be significantly higher than in ITER (10 GJ for DEMO toroidal coil (TC) vs 2.28 GJ for ITER field coil). Hence, an electric arc on a DEMO field coil may lead to a loss of integrity of the VV (first confinement barrier) with a consequent ingress of vacuum vessel cooling water into both the vessel itself and the cryostat. This accident has been excluded for ITER (its VV has been designed to maintain its integrity after a TC electrical arc).

Helium is used to cool the superconducting magnetic fields coils of ITER. A leak of liquid He may increase the room/equipment/VV pressure and affect its integrity. For example, for ITER, the quantity of He that could be accidentally split into the VV is limited by design to 25kg, in order to preserve the integrity of the VV and the VVPSS[12]. In DEMO, He inventory in the magnetic field coils cooling system will be significantly larger. He quantity

[12] Vacuum-Vessel Pressure Suppression System.

will be even greater in DEMO reactors using He-cooled TMB. Several publications stress the need to provide expansion volumes to limit pressure on confinement barriers in the event of helium leakage. Therefore, the IRSN considers that DEMO reactors designers must take into account the risks of a helium leakage since the preliminary design stage.

As already mentioned, Tritium releases from ITER are mainly associated with transfers of internal components to the hot cells and their processing in the cells during years of heavy maintenance, and with treatment of tritiated waste in other years. It is difficult to estimate tritium releases for DEMO as the design of their blankets/sectors is different from ITER ones.

Also, part of the tritium produced in the TBM will diffuse via the cooling system, which pass inside the blankets near the areas where tritium is produced. Purification systems are planned to permanently extract tritium from the cooling system of DEMO TBM but some of the tritium present in these systems will diffuse via the cooling system heat exchangers into the environment. This release path doesn't exist in ITER. Initial estimates of releases from planned DEMO reactors vary greatly depending on the type of tritium breeding blanket adopted. Hence, for DEMO, tritium releases into the environment via TBM cooling system has to be taken into account when choosing the type of TBM to achieve self-sufficiency in tritium.

As mentioned above, the increased tritium inventory in DEMO with respect to ITER implies an increased potential exposure of workers and members of the public, and changes in the design may lead to potential new accidents to be considered. As it is unlikely that lower leakage rates than those adopted for ITER can be targeted for DEMO equipment, DEMO tritium releases into the environment will be bigger than ITER as tritium inventory will be more significant. In accordance with the optimization principle, DEMO designers should reduce as much as possible the quantities of tritium in DEMO and optimize release paths[13] since early design stage.

DEMO wastes may be more tritiated than ITER ones. This could lead designers to add new and specific detritiation system, which must be considered when undergoing accident analysis. Also, DEMO wastes may be different than ITER ones (ex: TBM). Therefore, waste management constraints must be considered since early design stage, taking into account the Regulatory framework of the country hosting the reactor.

4.5 Main issues with worker exposure and exposures of members of the public

4.5.1 Risk of exposure to ionizing radiation in ITER

Between 2019-2022 the IRSN conducted an in-depth assessment of the radiation protection measures adopted for ITER by ITER Organization, as a part of the Assembly Hold Point technical instruction.

In his file, ITER Organization claims that the time of radiation exposure of a worker at ITER corresponds to the cumulative duration of plasma pulses, as for medical center which uses

[13] In particular, releases associated with the cooling system for tritium breeding blankets and with the transfer of internal components to the hot cells and their processing in these cells

ionizing radiation (ex: radiography, tomography scan…). Also, radiation protection is a safety function only when it involves the protection of the public.

For the IRSN, even if plasma discharges are intermittent on ITER, it cannot be considered as a radiation generating facility for radiational exposure risk because of the aim of the installation, the number of daily pulses and the long duration of plasma pulses (7 minutes). Exposure risk lasts after the pulses, and this is due to the decay time of the activation products created by the neutron flux[14], combined with the progressive cumulation of activation products with a non-negligible half-life time (like ^{60}Co). Finally, workers, as well as the public, must be considered for ITER's ionizing radiation protection design criteria and shielding.

4.5.2 Risk of exposure to ionizing radiation in DEMO

For the IRSN, DEMO can even less be considered as a radiation generating facility for radiation exposure risk.

As for ITER, workers exposure risk in DEMO will mainly be associated with human intervention during maintenance. Just as for ITER, heavy maintenance (cask transfers) will rely mainly on robotics.

Since risk of exposure to ionising radiation during plasma pulse is proportionated to the power (energy and duration) of plasma pulse, at the equilibrium, DEMO will have, compared to ITER:

- more (short and long-lasting life) high-energy γ-emitter activation products;
- higher doses rates for in-vessel components (blanket modules and diverter cassettes).

Increased quantity of high-energy γ-emitter activation products combined with longer operating times with plasma may lead to the use of massive biological shielding to limit workers exposure risk during light maintenance.

Higher doses rates for in-vessel components implies that design provisions must be made to limit worker exposure during cask transfers between the VV and the hot cells, as well as during their maintenance and storage in these cells.

Also, the activation of TBM under neutron bombardment must be taken into account for DEMO. For example, the activation of the LiPb eutectic liquid produces ^{203}Hg and ^{210}Po, radioactive isotopes of mercury and polonium whose doses factors per ingestion are 100 and 100000 times higher than for tritiated water.

It is here reminded that ITER dose targets are 2.5 mSv/y average; 10 mSv/y maximum. To have comparable dose targets in DEMO, designers must undergo, since the preliminary design stage, a process optimization for the doses received by operators (use of lower neutron activation material, massive biological shielding, cask transfer and maintenance design optimization…).

[14] ^{16}N, ^{17}N and ^{19}O produced by neutrons reaction with the oxygen in water.

5 References

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Radiological inventories and source terms

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Abstract

An assessment of a fuel cycle concept intended for a compact, 500 MW, high magnetic-field tokamak has shown that a tritium plant with a 500-gram tritium inventory is feasible if one excludes tritium stored as a backup to maintain a high plant duty cycle. A tritium release during an unexpected air ingress event the poses the highest risk and drives the need for an attended emission mitigation subsystem. Experiments are required to better define the fraction of tritium that can be released from plasma facing tungsten components. Enclosing all gaseous systems within gloveboxes using helium cover gases and online real time tritium recovery systems without conversion to tritiated water have a significant impact on reducing both accidental and chronic tritium releases to the environment. These systems offer the advantage of facile tritium recovery for re-use. Accidental tritium releases from the fuel cycle are projected to be less than 0.5 grams and consequently are unlikely to require evacuation plans at the fence line. Chronic releases due to maintenance and repair operations will most likely be the source of emissions unless infrastructure is installed within the tritium plant to accommodate appropriate decontamination technology.

1 Background

High temperature superconducting magnets have fundamentally altered the design of tokamak based fusion power reactor designs (Sorbom, 2015). Fusion power and plasma volume are strongly dependent on the toroidal magnetic field (Creely, 2020). A twofold increase in the magnetic field is expected to increase fusion power sixteen-fold which in turn implies the plasma volume can be reduced by a similar ratio. Such dramatic reductions will have ripple effects on the tritium inventories and the scale of the tritium handling systems.

2 Fuel Cycle

A conceptual fuel cycle to deliver, recover, purify, and breed tritium is provided in the block diagram illustrated in Figure 1. This architecture echoes that discussed for the DEMO reactor (Day, 2022) in that three loops with different processing times are envisioned: an inner loop with processing times on the order of 25 minutes, an outer loop fed by a slipstream drawn at two locations from the inner loop with processing times on the order

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of hours, and a tritium breeding loop whose characteristics have yet to be defined but whose processing time is expected to be in the several hours range. A notable difference in this configuration is the inclusion of a Trace Tritium Recovery system and Water Treatment System to mitigate emissions from the plant. The processing time for this portion of the loop is expected to be in the range of several days.

The key features of the loop presented in Figure 1 are as follows. Fuel is injected into the torus either by a pellet injector (Baylor, 2017) or by gas puffing. Injection efficiencies for the two techniques are in debate. Most of the propellant used to launch the DT ice into the torus is reclaimed although some of the propellant will be entrained with the DT ice. Effluent from the torus, comprising helium ash, unburnt fuel, protium (H) contamination, and inert plasma enhancing gases, are recovered from the divertor outlet via metal foil pumps (Peters, 2017) or cryosorption pumps (Willms, 1995). The efficacy of the two approaches is a topic of current research with a focus on balancing processing time, working inventory, and system complexity. The function of the block is to separate hydrogenic species from all non-hydrogenic species such as chemically bound tritium and inert gases. Short processing times are required to maintain inventories as low as possible. For example, a reactor burning 0.89 mg/s (comparable to 500 MW of fusion power), with a processing time of 25 minutes in the inner loop, an injection efficiency of 40 %, a burn-up fraction of 2 %, and a tritium breeding ratio of 1.2 will require a circulating inventory on the order of 138 grams of tritium. Increasing the injection efficiency to 50 % and the burn fraction to 3 % reduces the circulating inventory to 73 g.

Hydrogenic species leaving the 'Super-permeation/Cryogenic Molecular Sieve (MS) Bed' block are returned to the 'Gas manifold'. A slipstream is extracted from this flow to reduce protium (H) build-up in the DT fuel and to help manage the deuterium/tritium ratio. Non-hydrogen species are directed to the Torus Exhaust Purification (TEP) system where inert plasma enhancing gas are recovered for re-use and chemically bound tritiated species are decomposed as the first step in tritium recovery. Hydrogen species are directed to an isotope separator for protium stripping and fuel rebalancing. Chemical species containing tritium not processed in TEP are directed to the Trace Tritium Recovery system.

The outer loop comprises five subsystems: Torus Exhaust Processing, Protium Stripping, Trace Tritium Recovery, Water Treatment, and the Gas Manifold. The Trace Tritium Recovery System accepts gas from TEP as well as an array of other clients. Its function is to reduce chronic emissions from all processing systems. Gas entering the Trace Tritium Recovery System is converted to tritiated water and elemental species such as nitrogen and trace quantities of carbon dioxide. Tritiated water recovered from the Trace Tritium Recovery Systems as condensate is directed to the Water Treatment System while non-tritiated species are exhausted to the environment.

Presently the Water Treatment System is based on the Combined Electrolysis Catalytic Exchange (CECE) process (Boniface, 2014) where the activity of the tritiated water is increased several hundred-fold over wet-proof catalyst (Butler, 2006). The degree of enrichment has a direct impact on the size of the Isotope Separator in this loop. Increasing the water activity in the electrolyzer into the tens of terabecquerels per liter, or higher, reduces the size and throughput requirements of the Isotope Separator. On the other hand, high activity water poses potential dose issues to workers and can accelerate corrosion effects. Tritium purified in the Isotope Separator is directed to the Tritium Storage system.

The breeder blanket loop comprises heat extraction for power generation (not shown), tritium recovery from the blanket medium, and coolant purification to remove any tritium that has permeated through the hot metal walls. The figure assumes water is the coolant and consequently a water distillation process would be required to manage the tritium activity in the water. Tritium extracted from the breeding medium is directed to the Isotope Separator system for purification and storage. This loop is the least defined system of all the systems outlined above. The technology readiness level of breeding loops is in the 2 – 3 range. Many studies have focused on this topic (Forsberg, 2019) but very few experiments have been performed. The major impediment to date has been the lack of a sufficient strong 14 MeV neutron source coupled to a test breeding blanket test loop.

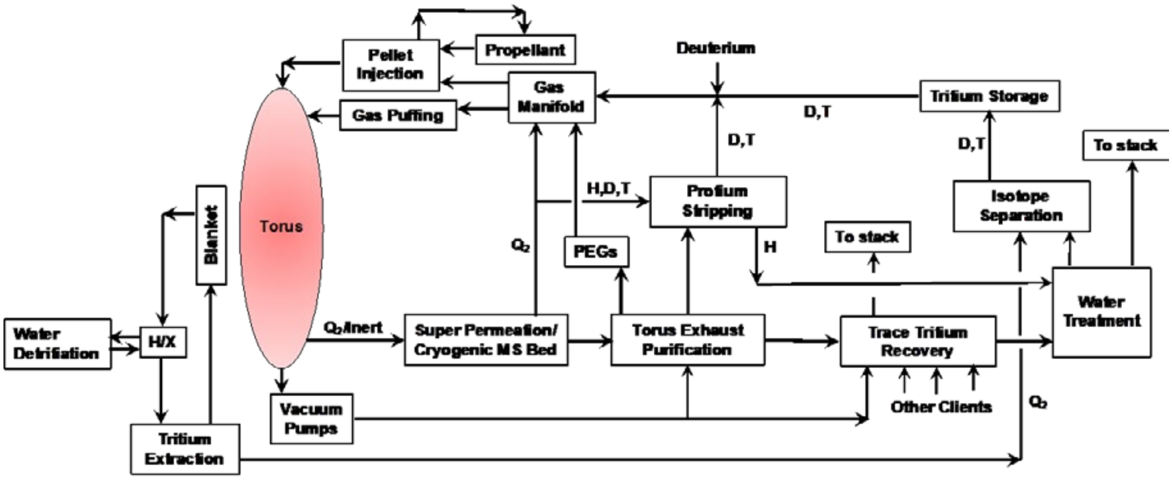

Figure 1. *Conceptual Fuel Cycle*

2.1 Inventories

The inventory distributed throughout the fuel cycle of a 500 MW high-field tokamak has been estimated by comparing the key parameters of a high-field tokamak (Sorbom, 2015) with the EU-DEMO (Day, 2022) listed in Table 1.

Table 1: *Comparison of key parameters of a high-field tokamak and EU DEMO*

	High-field tokamak 500 MW	**EU DEMO 2 GW**
Tritium consumption (g/day)	74	320
Wall area (m^2)	450	2625
Power density (MW/m^3)	3.7	0.7
Torus volume (m^3)	180	2500
Gas Throughput ($Pa\bullet m^3/s$)	95	380

2.2 Inner Loop

For a circulating inventory of 73 grams of tritium in a high-field tokamak, it is reasonable to assume two or possibly three pellet injectors with total working inventory on the order of 100 grams with 80 grams in the extruders and 20 grams in the internal loop of the injectors (Baylor, 2017; Gouge, 1991). Estimating the tritium inventory in tungsten plasma facing walls is a challenge due to the limited data. Preliminary estimates have been provided by Arredondo (2021). Scaling these estimates by the ratio of the power densities or the ratio of the wall areas suggests the 500 MW reactor would contain on the order of 280 to 300 grams of tritium. Assuming the torus prefill prior to a discharge is similar for both tokamak sizes, on the order of 6 Pa (Day, 2022), the initial tritium inventory in the 500 MW reactor case will be about 1.4 grams. During the discharge, however, the inventory will increase to about 3.5 grams assuming a plasma volume of 93 m^3 and a pedestal density of 1.5×10^{22} particles/m^3.

The expected working inventory of the effluent treatment systems depends on the technology selected. In the super-permeation approach, (Hanke, 2020; Giegerich, 2019) scaling to the gas throughput seems appropriate. For similar performance (Day, 2022) the number of 6.6 Pa•m^3/s metal foil pumps would be reduced from fifty-units down to fourteen for a total working inventory of 1 g. On the other hand, relying on the higher technology readiness level approach of cryogenic molecular sieve beds (Willms, 1995) followed by palladium-silver permeators (Hoerstensmeyer, 2020), the working inventory would be in the range of 24 grams assuming a 10-minute regeneration cycle for the cryogenic molecular sieve beds.

2.3 Outer Loop

The inventory of the outer loop of fuel cycle is distributed through the Torus Exhaust Purification system, the Protium Stripper, Trace Tritium Recovery, Water Treatment, and the Gas Manifold feeding both the Pellet Injectors and the Gas Puffing system. Untreated tungsten at 300°C outgasses at a rate of $5.54 * 10^{-5}$ Pa•m^3/s•m^2 after 10 hours (Battes, 2015). This translates to a steady state impurity concentration on the order of 3 % in the effluent that needs to pass through the Torus Exhaust Purification (TEP) system and the Isotope Separator. Assuming 1 % of the effluent is directed through TEP and the processing time is on the order of 1 hour, then the working inventory will be: 0.95 sL/s• 3600 s/h•0.02•3 g/mol/22.4 sL/mol = 9 grams. The remaining 2 % are assumed to be directed to the isotope separator for protium removal and isotopic re-adjustment. Based on the Thermal Cycle Adsorption Process (Heung, 2011) and a 25 % inventory draw per cycle, the working inventory of the Protium Stripper will be: 0.95 sL/s•3600 s/min•0.02/22.4 sL/mol•4•3 g/mol/2 = 18.3 grams of tritium where the factor of 4 accounts for drawing only 25 % of the isotope separator inventory and the factor of two assumes only half the gas is tritium with the balance being deuterium. Including the input from TEP, the total inventory of the isotope separator will be 27 grams of tritium. The Gas Manifold supports both the Gas Puffing and Pellet Injector systems and is assumed to operate at 2 atmospheres with a working inventory of 22 grams.

Trace Tritium Recovery (TTR) system is based on a classical burn and dry approach. All tritiated species leaving the various systems are converted to tritiated water over hot palladium catalyst and separated from non-tritiated species by absorption on molecular sieve driers operating at room temperature. These driers are expected to be regenerated monthly, generating on the order of 50 kg of water containing approximately 400 GBq/kg of tritium. The working inventory is projected to be less than 1/20 of a gram of tritium. The condensate from the TTR driers is directed to the Water Treatment subsystems where the activity of the water is increased at least tenfold into the 4 TBq/kg range using the CECE process. (Boniface, 2014). A working inventory of 1.5 grams of tritium in the electrolyzer is projected assuming the alkaline cells are charged with of 300 liters of KOH.

2.4 Breeder Blanket Loop

As discussed earlier, the breeder blanket loop has the greatest degree of uncertainty with different options under consideration by different investigators. The fivefold increase in neutron wall loading (= 3.7/.7) for high-magnetic field compact tokamaks requires a reassessment of heat management and materials selection considered in the current blanket designs. While numerous design studies have been carried out (Forsberg, 2020; Boccaccini, 2022; Sawan, 2005; Spagnuolo, 2021; Cismondi, 2020; Hernandez, 2020, to name a few studies), the technology readiness level of these designs is low. For the purposes of this report, a working inventory of less than one gram is assumed to be resident in the breeder system in a high-field tokamak.

For a tritium breeding ratio (TBR) of approximately 1.2, the quantity of tritium generated within the breeding medium needs to be on the order of 1.2 • 74 = 89 grams per day. Assuming a TCAP based isotope separator, the working inventory will be 16 grams for this system.

Excess tritium is collected in the Tritium Storage System. If retaining a reservoir of three operational days to account for unforeseen downtime in the outer loop is envisioned, this system will have a stored inventory of 3 days • (89-16) g/day = 219 g of tritium.

Table 2 summaries the inventories in each systems comprising the fuel cycle for a 500 MW high-field tokamak.

Table 2: *Tritium inventories in the various systems of a High-magnetic field, compact tokamak*

	Inventory (grams)
Pellet Injectors	100
Plasma Facing Wall	300
Torus content during a discharge	3.5
Effluent Separation based on Metal Foil pumps	1
Effluent Separation based on Cryogenic Molecular Sieve Beds	24
Gas Puffing System	1.4
Tokamak Exhaust Purification	9
Protium Stripper	27
Gas Manifold	22
Trace Tritium Recovery System	< 1
Water Treatment System	1.5
Blanket and Extraction System	~ 1
Blanket Isotope Separator	16
Storage (3 days)	219

3 Source Terms for Emissions

On the order of 725 grams of tritium are distributed through the fuel cycle and the reactor torus in a high-field tokamak. Setting aside the reserve to increase the reactor on-line time, the working inventory is on the order of 506 grams with most of the tritium, 80 %, residing in the Pellet Injectors and the plasma facing tungsten components. Of the systems listed in Table 2, only the inventories residing in the torus, the torus volume and the Trace Tritium Recovery system lack secondary containment. All the other systems are expected to reside within gloveboxes with attendant, online clean-up systems. Tritium releases due to primary process equipment containment failures, chronic emissions because of permeation and accidental releases during in box maintenance can be captured without conversion to tritiated water and recovered for re-use if helium is the glovebox cover gas. These glovebox clean-up systems can recover between 99.5 % and 99.9 % of the elemental tritium from the helium cover gas depending on the size of the initial release (Shmayda, 1992).

The inventory within the Trace Tritium Recovery resides as water on driers or as a condensate in the tank. An accidental spill of tritiated water due to plumbing failure will result in tritium release by evaporation. Such spills are local and manageable with the ability to recover most of the spilled liquid.

The torus volume operates at very low pressures compared to ambient conditions. Any rupture in the torus vacuum envelope will result in an inrush of air and a subsequent slow diffusion of tritium from the torus into the torus hall. Oxidation of fine tungsten is expected to generate some heat but should not generate volatile gases to pressurize the torus. One of the functions of the Trace Tritium Recovery (TTR) System is to draw air through the torus breach to draw tritium within the torus into the TTR where the gas is converted to water and sequestered on driers. The throughput of TTR should be designed to have air flow speeds above 100 m/s analogous to chemical fume hood face velocities designed to preclude tritium diffusion from the torus into the torus hall for all credible breach cross-sectional areas.

Although the tritium inventory of the plasma facing components is high, there is great uncertainty regarding the quantity that can be released during any credible air ingress accident scenario. Experiments are needed to assess what fraction remains bound with the tungsten under credible upset conditions. The release fraction is unknown currently.

Table 3 suggests that engineered systems have a strong impact on reducing emissions to the environment from systems enclosed in gloveboxes under credible accident scenarios on the assumption of single mode failures.

Experience in tritium handling facilities, however, has shown that maintenance operations tend to drive the majority of the emissions unless infrastructure is installed within the tritium plant to accommodate insitu and external decontamination protocols. This equipment does not form part of the fuel cycle but is required when components need replacement, repair, or upgrading. The design of the tritium handling infrastructure is a vital element of the overall objective of reducing emissions from a fusion power plant and tends to be overlooked.

Table 3: *Estimated tritium release inventories with and without emission mitigation equipment from a high-field tokamak (Shmayda, 1992)*

	Vulnerable Inventory without emission mitigating equipment (grams)	**Vulnerable Inventory including emission mitigating equipment (grams)**
Pellet Injectors	50†	0.05
Plasma Facing Wall	100††	0.5
Torus content during a discharge	3.5	0.3
Effluent Separation based on Metal Foil pumps	1	~ 0
Effluent Recovery based on Cryogenic Molecular Sieve	24†	0.02
Gas Puffing System	1.4	0.1
Tokamak Exhaust Purification	9†	0.05
Protium Stripper	27†	0.3
Gas Manifold	22†	0.2
Trace Tritium Recovery System	< 1	~ 0
Water Treatment	1.5	~ 0
Blanket and Extraction System	~ 1	?
Blanket Isotope Separator	16†	0.02
Storage (3 days)	219†	0.05

† system housed in a glovebox filled with helium and fitted with a tritium recovery system

†† assume 30 % is vulnerable to release (Andrew, 1999)

4 Conclusions

The tritium inventory of a 500 MW fusion power plant can be reduced to about 500 grams for a compact, high-field tokamak excluding any reserve tritium that is stored as back up fuel. Most of this tritium is not vulnerable to accidental release. Engineered emission mitigating infrastructure can reduce the inventory vulnerable to release by an additional factor ranging from 500 to 1000 depending on the system being protected. For accidental releases below 0.5 grams it is highly likely that evacuation plans are not required at the fence boundary, that is to say, doses to the general public due to accidental releases will not exceed 10 mSv (1 rem) at the fence line.

Unabated tritium release from tungsten used in plasma facing components in the torus during an air ingress accident represents the highest uncertainty and potentially the largest quantity of tritium vulnerable to release. The design of a suitably scaled Tritium Recovery System needs to address the release pathway. Experiments are required to assess the tritium releasable fraction from tungsten with properties expected to arise in the hostile environment of the torus volume.

All gaseous tritium systems should reside in gloveboxes using helium cover gases and independent tritium gas recovery systems without conversion to tritiated water. This proven technology (Shmayda, 1992) has an extensive operational history. As a side benefit, captured tritium can be recovered and re-used.

The system inventories and the release fractions vulnerable to release are based on the selected technological approaches discussed in the body of the paper. Novel, lower technology readiness level approaches need to be evaluated against those discussed above with an eye to further reducing plant inventories.

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Occupational radiation exposure at the experimental fusion facilities

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Abstract

The fusion facilities in operation nowadays are experimental devices aimed to investigate the different technological aspects to be solved to produce net energy from nuclear fusion processes. Despite the experimental nature of these plants, workers involved in the various activities are exposed to ionizing radiation that must be considered for assessing the occupational dose. Since more than 40 years, nuclear fusion plants have been designed and installed, and the related occupational radiation exposure has been assessed both for normal operation and for emergency situations. The data collected in this period are useful to evaluate the radiological impact of this kind of facility. It has been shown that individual worker exposure can be limited by applying the usual ALARA process, but the reduction of the collective dose is a real challenge for the fusion facilities, if compared with the lower values reached by fission plants. Nevertheless, the efforts in this aspect are evident, as it is shown in the current short review, and the experiments and studies toward the future demonstration fusion plant are going on with ambitious goals.

1 Occupational Radiation Exposure

The occupational radiation exposure (ORE) refers to the exposure of workers to the ionizing radiation (Natalizio, 2002). In this context, both the individual and the collective doses could be considered, as the former represent an indicator of the individual risk and the latter can be assumed as a parameter that represents the radioprotection performance of an installation.

The radiation exposure of workers is ultimately linked with maintenance experience, as maintenance activities for the experimental fusion facilities are the largest contributor to worker dose. Therefore, the definition of the maintenance activities and the identification and characterisation of the related radiation sources are the starting points for assessing the ORE.

In its most general term, maintenance refers to all the works performed on the fusion facility that are required to maintain it in a state of readiness, to meet the performance demands imposed by the physics program. This includes repairing and upgrading machine

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components as well as changing the machine configuration, such as installation of different divertors.

Considerations and partial results reported in this document refer to the magnetic confinement fusion (MCF) devices, based on the Tokamak (toroidalnya kamera ee magnetnaya katushka – torus-shaped magnetic chamber) configuration. Tokamak fusion technology is considered the most promising design, and research is continuing on various tokamaks around the world. Nevertheless, ORE assessment is similar for other fusion devices like stellarators, reversed field pinch (RFP), inertial confinement systems and magneto-inertial fusion (MIF).

2 Reference facilities

In the last decades, many countries have taken part in nuclear fusion research to some extent. The principal facilities were designed by European Union, USA, Russia and Japan, but intense programmes with valuable results are also under way in China, Brazil, Canada, and Korea. ORE has been considered in many related designs, the main one being ITER (International Thermonuclear Experimental Reactor) facility now under construction in France (ITER, 2010). The Parties to the ITER agreement, or the ITER members, are now the European Atomic Energy Community (Euratom), China, India, Japan, Korea, USA and Russia. ITER's conceptual design work began in 1988, and the final design was approved in 2001. The designing activities about this machine provided an important number of publications related to the ORE that are still useful for the assessment of the radiological exposure of the workers at fusion plants. Valuable experimental data come from the fusion facilities that have been, and are, in operation, like TFTR (Tokamak Fusion Test Reactor, USA), JET (Joint European Torus, UK), JT60 (Japan Torus-60, Japan), EAST (Experimental Advanced Superconducting Tokamak, China). Many minor experiments are under way or in an advanced design stage in different countries worldwide, among these Italy has a consolidated tradition in this field with FTU (Frascati Tokamak Upgrade), RFX (Reversed Field eXperiment), and the DTT (Divertor Test Facility) project. The executive design of the DTT has been completed and it is being built in Frascati, near Rome (Martone, 2019). It will explore the physics and technology of the plasma thermal power exhaust which could be used in the European DEMO, the post-ITER demonstration fusion power plant. DTT will operate without tritium as fuel and will have a maximum neutron yield higher than 10^{17} n/s.

JET, located near Culham, UK, is the largest and most powerful tokamak in operation in the world, and is currently the only machine capable of operating with the deuterium-tritium fuel mix of future commercial reactors (Rebut, 1992). In operation since 1983, as a joint venture, JET is collectively used by more than 40 European laboratories. JET was designed to study plasma behaviour in conditions and dimensions approaching those required in a fusion reactor. Today, its primary task is to support construction and future operation of ITER, acting as a test bed for ITER technologies and plasma operating scenarios. The sharing of collective dose at JET by workers' group is showed in Table 1 (Natalizio, 2003), pointing out that the responsible for the higher dose contribution are the maintenance activities. Additionally, experience working at JET indicate that tritium exposure is kept as low as desired by using adequate ventilation systems, detritiation equipment and protective clothes.

Table 1: *Share of collective dose at JET by workers' group.*

Group	Average
Administrative support	0.3 %
Engineering support	17.9 %
Facility support	4.9 %
Maintenance	**66.9 %**
Operations	2.5 %
Safety	1.3 %
Scientific support	6.3 %
Total	100.0 %

3 Radiological sources

The radiation sources that could contribute to the exposure of workers at the nuclear fusion plants are (Maubert, 2007):

- the primary neutron field resulting from the fusion reaction $^{2}H + {}^{3}H \rightarrow {}^{4}He + n$ occurring in the vacuum vessel,
- the gamma radiation emitted by activated products, including plasma facing components, vacuum vessel structures, loose contamination from activated dust generated in the vacuum vessel, activated corrosion products generated in the cooling loops and activation of the inner wall of cooling water pipes,
- tritium used as fuel for the fusion reaction. The inventory of tritium is distributed equally in the long-term storage, the tritium building, the tokamak and the hot cells and radwaste facility,
- wastes, containing tritium and gamma emitters.

The neutron field is usually well shielded by the so-called bio-shield, located around the Tokamak. Nevertheless, the neutrons diffusion throughout many service ports and supporting devices provides a potential exposure to the workers.

Neutron activation of plasma facing components, vacuum vessel structures, dust generated in the vacuum vessel, coolant, and corrosion products, are probably the most important sources of exposure. Among these, dust and coolant activation are a limited source of exposure, as the studies related to the ITER fusion plant demonstrate that even a conservative inventory of these contributions does not present a risk of exposure for workers and population. When the coolant material is water, its activation produces two main radioisotopes: ^{16}N and ^{17}N (see figure 1). The first one decays producing high energy gamma rays (6-7 MeV) but has a short half-life. The second one produces a delayed neutron that activates the components of the Tokamak inner walls and could increase the dose rate. Usually, this contribution can be neglected owing to the distance from the Tokamak core to the premises external to the bio-shield.

Tritium can be part of the fuel mixture and in principle can contaminate the atmosphere of tokamak and service buildings. Evaluations related to the design of the ITER plant showed that its contribution to the occupational exposure could be of the order of 20-30 % of the total. On the other end, the lesson learned during the experiments at the JET facility showed that adequate radiation protection measures (typically, fully pressurized plastic suits and ventilation systems), worked well at reducing the tritium dose to a small percentage of the total.

Radioactive wastes are produced during the operation, mainly due to the removal and replacement of in-vessel components such as first wall tiles, divertor cassettes, heating systems, and diagnostic probes. All these components, after the first years of activity, are highly activated and could be contaminated by dust and tritium. Usually, the dose rates from these materials are so high that personnel is prevented from staying near them, and remote handling tools are needed for the maintenance operations. The assessment related to the DTT facility (see Figure 2), shows that after 18 months of operation is already impossible to handle manually on many in-vessel components, even after some months of cooling down.

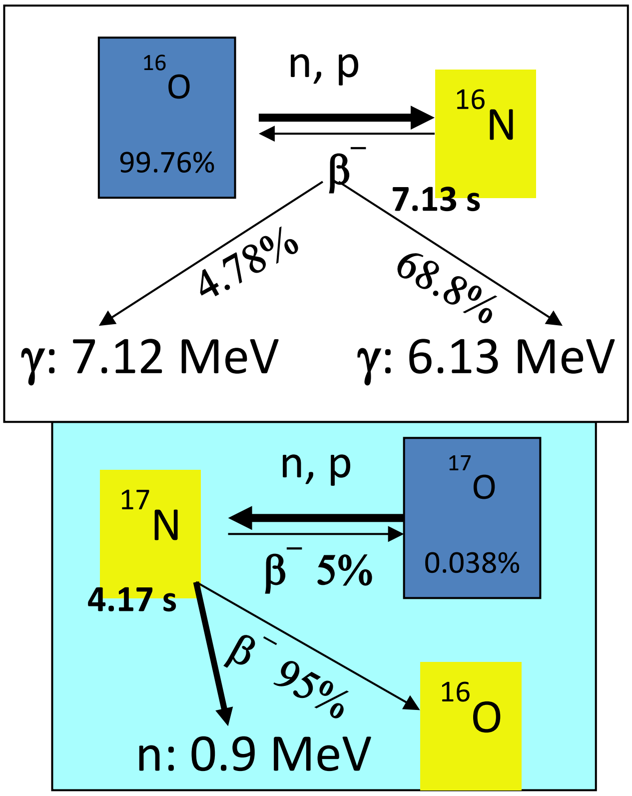

Figure 1. *Water activation products.*

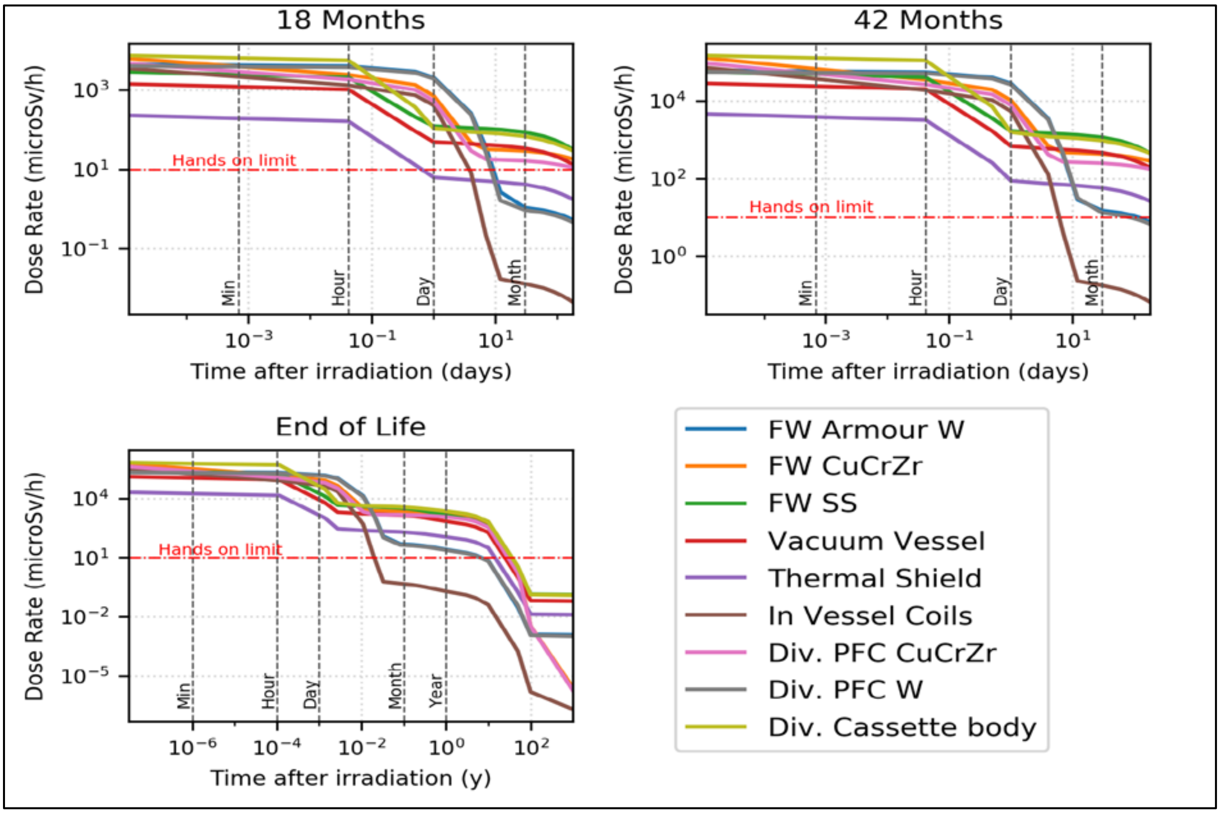

Figure 2. *DTT in-vessel calculated contact dose rate for different components after 18 months, 42 months, and end of life.*

4 Individual and collective doses

The standards and guidelines for radiological safety adopted at the nuclear fusion facilities are based upon international standards and recommendations from IAEA and ICRP. An ALARA (As Low, As Reasonably Achievable) assessment is usually established for operations and works yielding the higher doses. This ALARA process provides an opportunity for changes in the design and operational approach. It is used in the design process as an instrument to optimize shielding and maintenance procedures. This process led to reductions in the anticipated ORE for high-risk systems. In the first step of the ALARA process, the studies are mainly focused on 4 items: shielding, time lag between shutdown and intervention, design and use of remote handling, control of airborne tritium. Table 2 shows the typical guidelines for doses from occupational exposure stated for some fusion facilities.

Table 2: *Typical guidelines for doses from occupational exposure at fusion facilities.*

Quantity	Value
Annual individual limit for exposed workers	20 mSv/y
Annual individual dose constraint for exposed workers	10 mSv/y
Individual dose for any single shift	0.5 mSv/shift
Collective annual worker dose	500 mSv/y
Maximum individual worker dose after an incident	10 mSv

The radiological exposure to the workers was assessed in the design and during the operation of many nuclear fusion systems, with major results for the following working activities:

- Normal surveillance in the tokamak building
- Routine and special maintenance of the cooling system components
- Hands-on assistance for RH maintenance of the in-vessel components

The studies about worker safety at ITER showed that during the normal surveillance the operator would receive a total annual effective dose of 2 mSv, based on an estimated average effective dose rate of 1 μSv/h and an annual working time of 2000 hours. For the same facility the maintenance of the water-cooling system components requires different kind of operations, and the dose rates range from about 1 μSv/h to some tens of μSv/h. For some special operations the worker could be required to go inside the Channel Head of the Heat Exchanger (HX) where the estimated dose rate is as high as some hundreds of μSv/h. Around the components the dose rates are estimated to be quite high as can be seen in Table 3.

Table 3: *ITER estimated dose rates around cooling system components, 5 days after shutdown at end of life.*

System-Component	Dose rates at 30 cm μSv/h	Dose rates at 1 m μSv/h
Heat exchanger	7.4	4.8
Pressurizer	2.3	0.8
Pumps	2.5 – 8.1	1.1 – 3.1
Heater	3.8	1.4
Valves, welds and pipes	2.3	1.1

The most exposed operator could receive an annual effective dose of about 10 mSv and the major quantity of the workers involved in this activity should stay below an annual effective dose of 6 mSv. Similarly, the hands-on assistance to the RH for in vessel components includes many working tasks and the same considerations already issued for the maintenance of the cooling system apply to this working activity, responsible of high dose rates (see the result of a preliminary assessment in Figure 3). Also in this case, the most exposed operator could receive an annual effective dose of about 10 mSv, but the average worker should stay below an annual effective dose of 6 mSv.

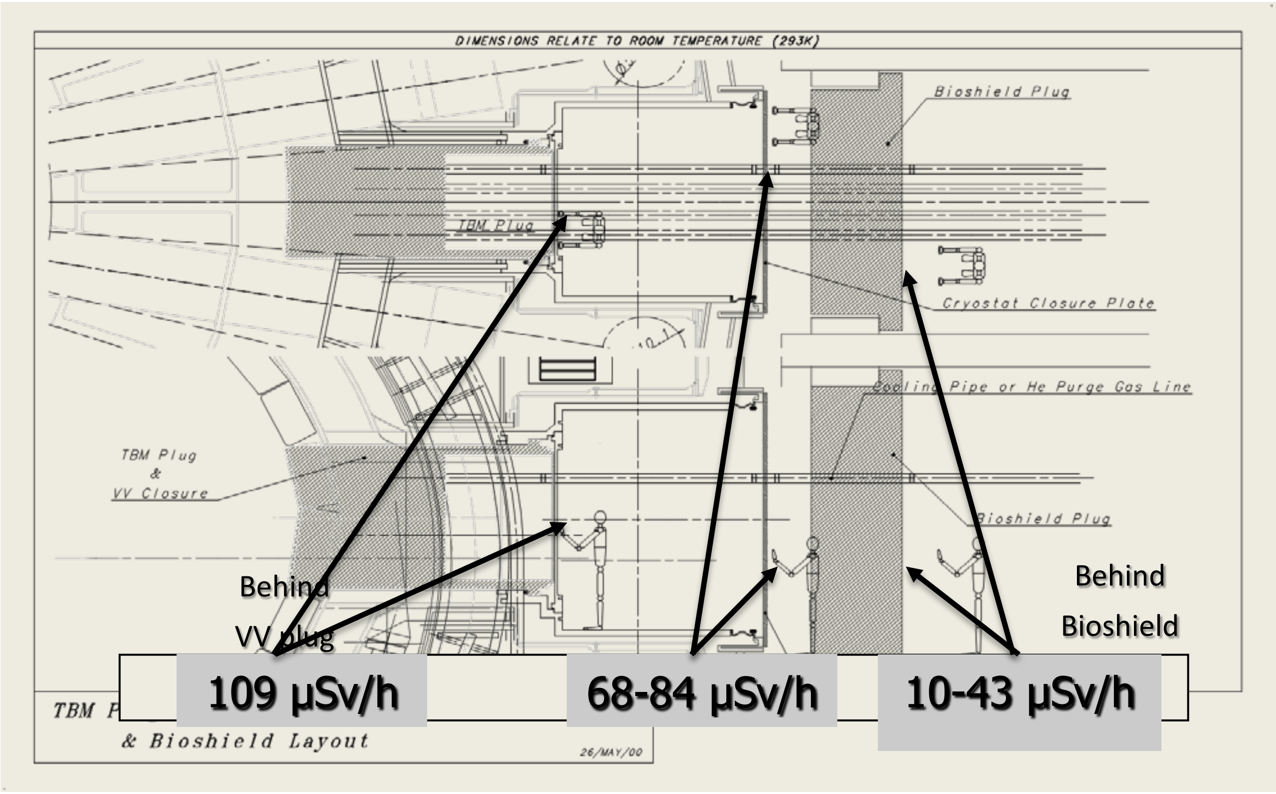

Figure 3. *ITER dose rates for the RH assistance (Sandri, 2005).*

The radiological consequences of an accident, for the personnel likely to be present at the time of the accident itself, have been assessed in some cases.
For ITER the basis accidents considered the following conservative hypotheses:

- detritiation system operation within 5 minutes of detection signals requiring startup,
- personnel may be present in the building at the time of the accident and evacuate the location within 10 minutes after detection of the accident,
- possibility of returning to a safe state without the presence of personnel.

In Table 4 the related results in term of effective dose to the worker are reported.

The meaning of the ORE is usually the collective dose (Sandri, 2002), assessed multiplying the dose rate of the working task by the time required, the number of workers involved (person power), and the task frequency (see Figure 4).

The collective dose is the parameter used to characterize the safety performance of a nuclear plant and it is frequently used to compare the fusion plants to the fission ones. The ORE of the nuclear fission plant is continuously monitored by ISOE, the Information System on Occupational Exposure (NEA, 2014).

Table 4: *Individual effective doses after design basis accidents in ITER.*

Accident scenarios (ITER design basis accidents)	**Max effective dose to a worker [mSv]**
Simultaneous rupture of a helium cooling line and of the cryogenic distillation columns	0.54
Simultaneous rupture of a fuelling line and of the second confinement barrier	0.43
Double ended guillotine break in the local air cooler (LAC) duct	0.90
Leak of high activity tritium in the Tritium Building	0.20
Accidental opening of the largest opening in a glove box	0.034

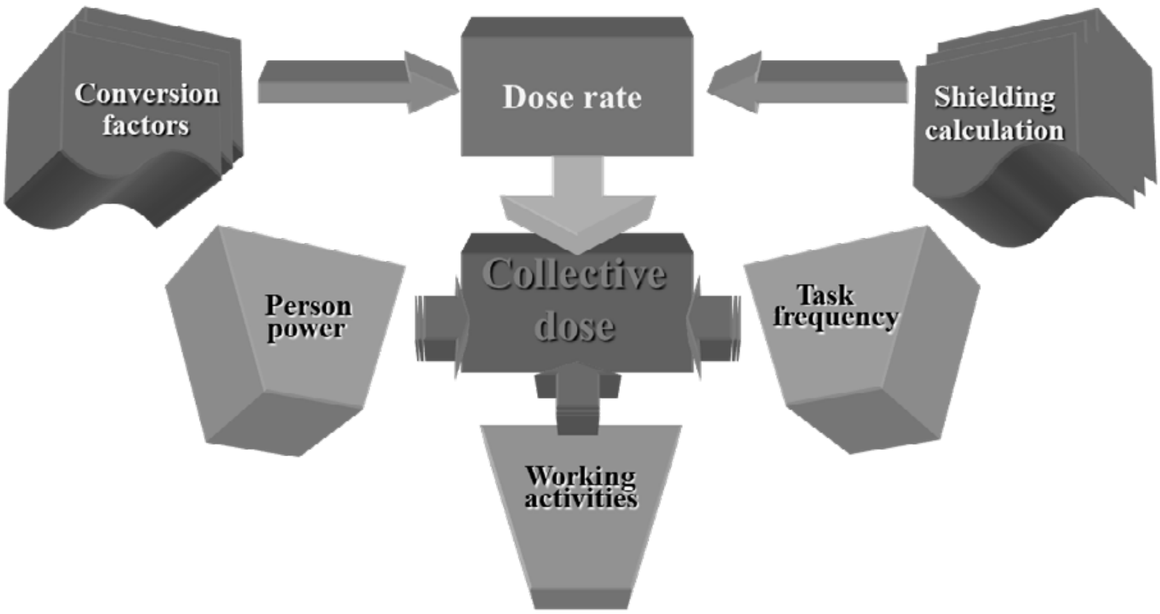

Figure 4. *Assessing the collective dose.*

5 DEMO: the future nuclear fusion reactor

The design of a demonstration nuclear fusion reactor (DEMO) is under study in different countries.

DEMO will be the pathfinder to a First-of-a-Kind Fusion Power Plant.

The goals of the European (EU) DEMO are the following:

- Ignition
- Conversion of fusion heat into electricity (~500 MWe)
- Achieve tritium self-sufficiency
- Reasonable availability
- Several full power years
- Minimize activation waste, no long-term storage

For the EU DEMO a maximum value of total collective dose of 700 p-mSv/y was stated (Caruso, 2022). This value is about the same as for the fission nuclear plants with the best performance (NEA, 2014) (figure 5).

Preliminary evaluations indicate that DEMO will have a low reliability due to the high number of anticipated yearly failure events, requiring frequent stops of the reactor.

Such frequent reactor shutdown could be overcome with corrective maintenance, that could be done with a relative short time of outage, usually between 3 days and 2 months, but for the events with higher frequency in the range from 3 days to 2 weeks. These fast but frequent maintenance activities, are expected to be important source of radiological exposure to the workers. Figure 6 (Caruso, 2022) shows that the dose rates around the Tokamak, inside the biological shield, after the shutdown of the EU DEMO will be very high and it would be very hard to fulfil a target of 10 μSv/h after 1-day cooling inside rooms with active loop components.

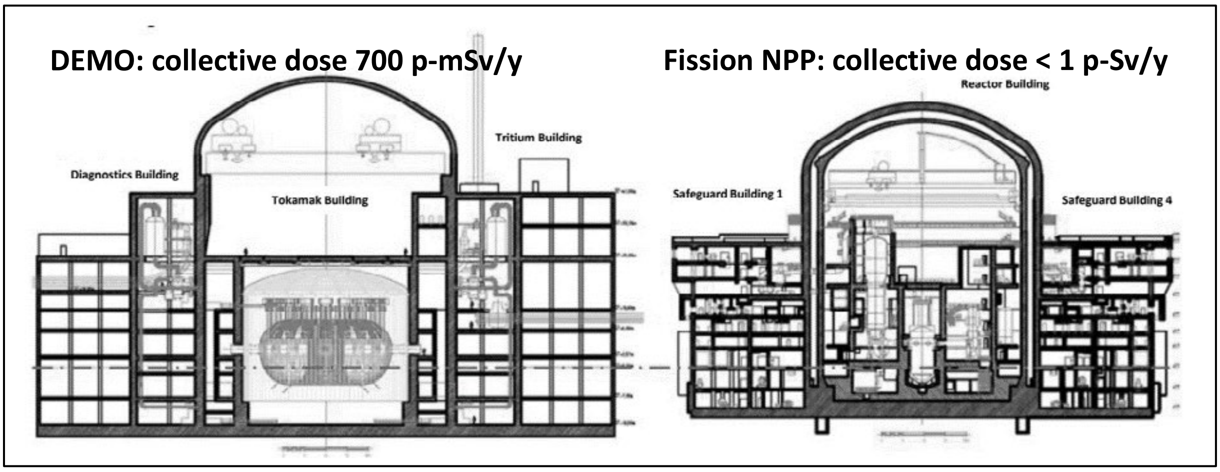

Figure 5. *Schematic comparison between the European DEMO and a fission reactor* (Federici, 2018).

Few studies related to the anticipated accidents in DEMO show a situation like that of ITER, therefore, the individual dose objective set for DEMO's workers following an accident is the same as that of ITER: ≤ 10 mSv.

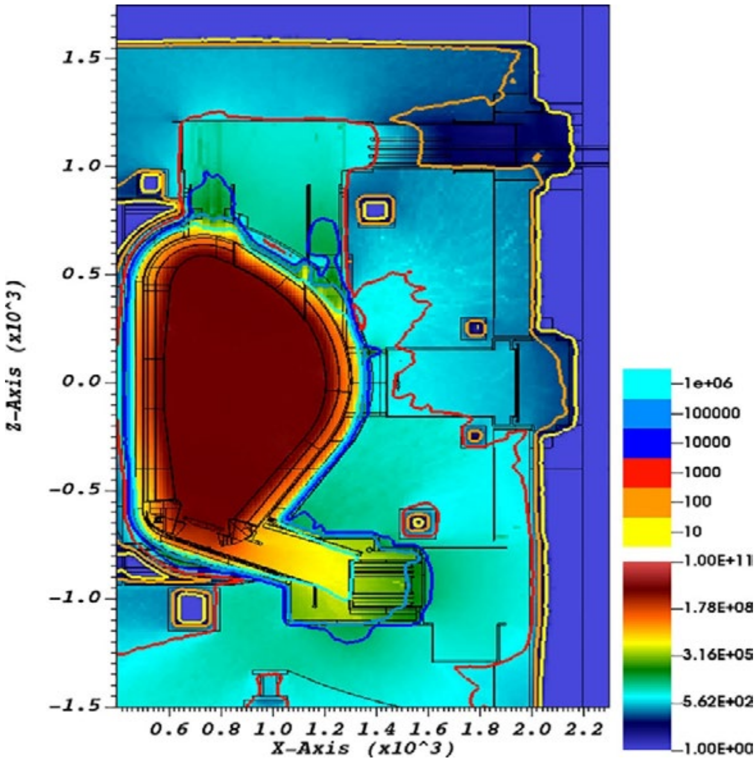

Figure 6. *Provisional biological Shutdown Dose Rates [µSv/h] in EU DEMO after end of life and 12 days cooling (upper color scale for isolines, lower color scale for dose mapping)* (Caruso, 2022).

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Environmental releases and public exposure in normal and accident situations

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Abstract

Fusion reactors are based on self-sustaining deuterium-tritium fusion reactions (D-T), in a plasma within a vacuum vessel with specific temperature and particle density conditions, for creating enough energy to allow electrical power production. These reactions and the process to achieve them will generate hazardous inventories for which safety provisions will be implemented for limiting the exposure of workers, members of the public and the environment for both routine and accident conditions.

The potential exposure of members of the public is driven by the radioactive inventories in these facilities, their chemical and physical properties, and the contamination transfer mechanisms between the inventories locations and the release points in routine and accident situations. These mechanisms will be guided by the efficiency of the confinement function associated with these nuclides in those situations.

Tritium is the expected driver of the public exposure for most of the situations; but there are postulated accident scenarios for which other nuclides have also to be considered.

Based on the lessons learnt from existing Tokamaks or under construction, the document indicates the expected releases from fusion reactors in routine situations and the potential releases in postulated accident scenarios, and their effects on members of the public.

1 Radionuclides involved in fusion reactors

Foreseen technologies of fusion reactors involve the use of deuterium (D) and tritium (T) fuels, tritium being radioactive with a half-life of around 12.3 years. Fusion reactors aim to create a self-sustaining deuterium-tritium fusion reactions (D-T), in a plasma within a vacuum vessel with specific temperature and particle density conditions, for creating enough energy to allow electrical power production. These reactions and the process to achieve them will generate hazardous inventories of radioactive materials to be presented hereafter.

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1.1 Initial deuterium-tritium fusion reaction

The initial deuterium-tritium fusion reaction, primary goal of fusion reactors, will create 17.6 MeV energy, composed by the energy coming from a helium nucleus (3.5 MeV) and around 14.1 MeV neutrons, as exposed in figure 1.

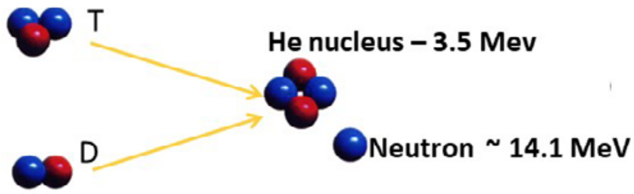

Figure 1. *D-T plasma reaction*

The primary neutron radiation is produced in the vacuum vessel located in the Tokamak Building during plasma operation, and in particular during plasma reactions. Fusion reactors consider D-T reaction for long duration sequences. Fr the European DEMO reactor, it is expected to reach more than 2 hours, with short dwell periods in between (quasi-continuous process).

1.2 Materials and by-products induced by the fusion reaction

1.2.1 Tritium

Tritium is used as a fuel in large quantities, enough to produce the fusion plasma. In theory, Tritium can be found either as its elementary form (HT, DT, T_2) for which tritium is the least radiotoxic element on earth, or under an oxidized chemical form (HTO, DTO, T_2O), or under an organic form with carbon (e.g. CH_3T) or nitrogen (e.g. NH_2T). It may also be found as adsorbed on aerosol, becoming a tritiated aerosol. For the last three forms, Tritium is more radiotoxic than its elementary form, but still less radiotoxic than many other radionuclides.

The ICRP 134 provides the dose coefficients for the different tritium chemical forms as shown in figure 2. To be noted that this figure is based on ICRP134 on occupational activities, since the values are being changed for the environmental impact, but not published yet.

	Dose coefficient (Sv/Bq)
HTO or gas/vapour unspecified	**2 E-11**
HT	**2 E-15**
CH4-xTx	**5.9E-14**
Tritiated aerosols (AMAD 5 microns - Unspecified)	**2.4E-11**
OBT (ingestion)	5 E-11
OBT (inhalation as tritiated aerosol AMAD 5 microns)	3.5 E-11
Max coefficient: inhaled tritiated aerosol AMAD 5 microns	2.6E-10 (type S carbon/hafnium tritide)

Figure 2. *Tritium radiotoxicity according to ICRP 134 (occupational activities) [4]*

1.2.2 Neutrons

Access to the Tokamak Building is strictly forbidden during D-T plasma operating periods such as the neutrons are not a health issue for the workers. The shielding materials of the Tokamak building, composed by biological shielding walls, will prevent neutrons from escaping this building. The neutron dose rates are expected to be low outside the bioshield; direct neutron radiation is therefore not a major source of occupational exposure during the plasma operation phases in the Tokamak Building. Therefore, neutrons should not pose a direct health issue on workers.

The neutrons will however have the ability to create radioactive materials (this is called neutron activation) as seen in the next chapter.

1.2.3 Radionuclides created from neutron activation

Neutron activation of nuclides in the materials by the neutron flux from the plasma is a major source of exposure, since the activated materials can emit radiation. Some of the materials used in fusion reactors will be different from the ones used in the nuclear fission industry, in particular the plasma facing components, while some others would be identical to the ones that might be created in the nuclear fission industry.

<u>New types of fusion materials</u>

The "plasma facing components" will first be activated by neutrons. There should not be fission reactions, except the potential impurity that may be present in these fusion materials.

For fusion reactors (demonstrators and reactors), the materials that would be used directly to face the plasma are mostly composed of Tungsten, known to be a very hard and dense materials.

Therefore, the radionuclides that would be created are mostly derived from Tungsten (^{185}W, ^{187}W, ^{185m} W…), and the adjacent radionuclides in the Mendeleev table (Tantalum with ^{182}Ta-, Rhenium with ^{186}Re, etc.). After activation, the nuclides could become activated dust, as fusion plasma may erode surfaces.

A fusion reactor specificity will be the use of self-sustained reactions with tritium via the use of breeding component products (e.g. Lithium) that will create tritium following neutron activation. Among the material candidates for creating tritium are Lithium-Lead liquid mixtures or ceramics pebbles containing a mixture of lithium, silicon and titanium. Therefore, the nuclides would be derived from Lithium activation (mainly tritium), Lead activation (^{203}Pb, ^{209}Pb), Silicon activation (^{31}Si), Titanium activation (^{45}Ti, ^{51}Ti) and the ones adjacent in Mendeleev table (Bismuth with ^{210}Bi, Aluminium with ^{29}Al, Magnesium with ^{27}Mg, Scandium with ^{48}Sc, ^{47}Sc).

Alpha emitters might exist at trace levels (^{210}Po) if lead is activated.

In any case, the control of impurity in materials is important for limiting the production of these nuclides.

<u>Other "traditional" radionuclides</u>

Behind the Tungsten layers, there would be other structural materials such as stainless steel, and for the European models, a specific material called EUROFER-97 specifically selected in order to reduce its activation against neutrons. For both metals, the "classical" nuclides that are also present in fission reactors from steel activation will be found: iron (^{55}Fe), Manganese (^{54}Mn, ^{56}Mn), Cobalt (^{58}Co, ^{60}Co), etc. These nuclides could be created inside cooling loops via corrosion, they are also called activated corrosion products (like for fission reactors).

There will be also conventional water activation nuclides (^{16}N, ^{17}N…), water chemical impurity or additives nuclides (e.g. ^{14}C), air activation (notably with Argon nuclides leading to noble gases). Some other species may also naturally exist, such as radon.

Therefore, in fusion reactors, the following categories of radioactive materials can be met:

- Tritium
- Noble gases (from air activation or from the inert gases injected into the plasma)
- Carbon-14 (e.g. from cooling loops additives),
- Very short-lived nuclides, e.g. from water activation in cooling loops (^{16}N, ^{17}N) or from helium activation (^{6}He)
- Beta-gamma emitters
    - Activated dust from plasma-facing components or bulk activation, mainly tungsten,
    - Metal bulk or particles from stainless steel or Eurofer or tritium breeding loops
    - Activated corrosion products from cooling loops

Figure 3 provides an overview of the nuclides radiotoxicity (taken from ICRP database for members of the public) that would be met in fusion reactors (in blue in the figure), compared to the ones that could be seen in fission facilities (in yellow in the figure). The radiotoxicity is given in logarithm scale.

It can be shown that the adequate selection of fusion materials will help in reducing the radiotoxicity of the nuclides potentially present in the fusion activities.

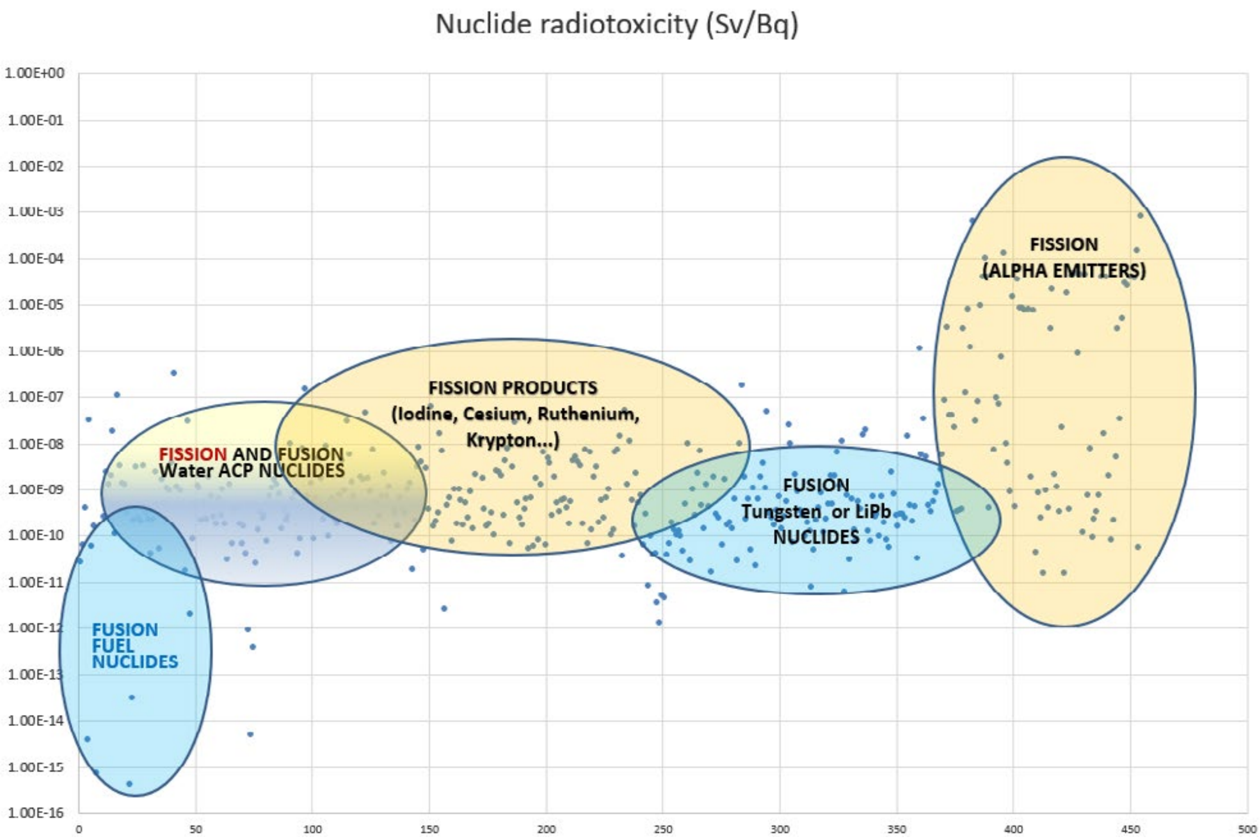

Figure 3. *Nuclide radiotoxicity (Sv/Bq) ranked on their Z (low Z at the left, high Z at the right) – from ICRP database on dose coefficient for members of the public*

Nota: The abscises in this figure are based on Z number (low Z at the left, high Z at the right), but the abscises numbering is meaningless since nuclides have different radiotoxicity upon their chemical form: a same nuclide may have several radiotoxicity put one after the other in this graph. That explains why the abscises numbers are above 450.

2 Releases and public exposure from fusion reactors

2.1 Confinement of radioactive materials

Fusion reactors will be equipped with several confinement systems able to protect both the workers and the members of the public and the environment. Several stringent requirements will be implemented in order to reach ambitious safety objectives. These stringent safety requirements are based on defense in depth principles: prevention by minimizing the likelihood of abnormal and accident events, early detection of any potential abnormal event, and mitigation of the consequences of any abnormal situations. The mitigation of these potential situations is performed by designing robust confinement systems for any type of radioactive materials in normal operations and during postulated incident and accident scenarios.

The confinement principles rely on the implementation of several confinement systems, each of them comprising confinement "static" barriers as wells as ventilation and detritiation systems ensuring a dynamic confinement. Detritiation systems are ventilation systems ensuring the dynamic confinement of all the rooms in which tritium is likely to be released during postulated accident scenarios.

The confinement design requirements will rely on the state of the art in ISO standards on confinement of nuclear facilities (ISO 17873 [5] and ISO 16646 [6]).

For each principal inventory, two confinement systems are provided such that sequential barriers are provided for each inventory. Appropriate treatment of penetrations through confinement barriers ensures that the confinement objective is met.

These standards describe in particular all the design and operation issues associated with the confinement systems, the static confinement barriers and dynamic ventilation systems:

- Negative pressure range,
- Air change rates,
- Filtration of radioactive materials,
- Design against external hazards (earthquake, loss of electrical power...),
- Design against internal hazards (fire, flooding, other pressurization sources...).

These confinement systems will help in reducing the releases for both routine and accident situations.

The confinement of tritium is one of the most important safety functions: detritiation systems (DS) provide detritiation and depression functions within the different rooms. Certain parts of these detritiation systems form an integral part of the first confinement system since they provide the detritiation of the air in the glove boxes or the confinement of the water to be further managed in water detritiation systems, whilst other parts are part of the second confinement system since they handle the environment in zones where personnel access is possible.

DS technologies are based first on the oxidation of any gaseous form into tritiated vapour and second on the isotopic exchange between tritiated water vapour and liquid water.
To convert hydrogen-containing gaseous species into water vapour prior to their injection to scrubber column conventional catalytic convertors operating over a range of 20 °C to 470 °C are generally used:

- Low temperature catalytic reactor (recombiners operating at around 170 °C) able to oxidise gaseous tritium (HT) into tritiated vapour (HTO),
- High temperature catalytic reactor (recombiners operating generally at more than 300 °C) able to crack tritiated hydrocarbon forms into vapour species and releasing CO_2.

$$HT_G + \frac{1}{2}O_2 \rightarrow HTO_V$$

at around 170 °C with catalytic materials

$$CH3T_G + 2O_2 \rightarrow CO2_G + H2O_V + HTO_V$$

at more than around 300 °C with catalytic materials (tritiated methane example)

The isotopic exchange technology employs scrubber columns and catalytic reactors for conversion of hydrogen containing gaseous species to water vapour. The technology is widely used in industry, particularly for gas purification in the chemical industry. The following technologies are used in order to support this detritiation/ water collection function:

- Scrubber columns,
- Molecular sieves driers
- Bubbling technologies

High Efficiency Particulate Air (HEPA) filters are used for any radioactive aerosols (activated dust, activated corrosion products) with at least 99.9 % efficiency.

2.2 Routine releases and public exposure from fusion reactors

Thanks to the good efficiency of confinement systems, routine releases from fusion reactors will be:

- Tritium,
- Noble gases (argon nuclides)
- Carbon-14,
- Beta-gamma emitters.

As any nuclear facility, a fusion reactor would have to meet discharges authorisations for gas and liquid routine releases established by a national regulator. In any case, in Europe, these discharge authorisations are driven by a dose limit for the public of maximum 1 mSv/year or by a lower dose constraint decided/imposed by the regulatory authority.

Even though not specifically considered as a fusion reactor, the example of ITER research facility can be provided. ITER has published its foreseen routine releases [2]. It has shown that the consequences on members of the public in routine conditions were in the order of magnitude of around 1 μSv per year for the most exposed reference group, for normal maintenance years, and are quasi totally due to tritium, even by considering conservatively that tritium is under HTO chemical form (as shown in figure 4).

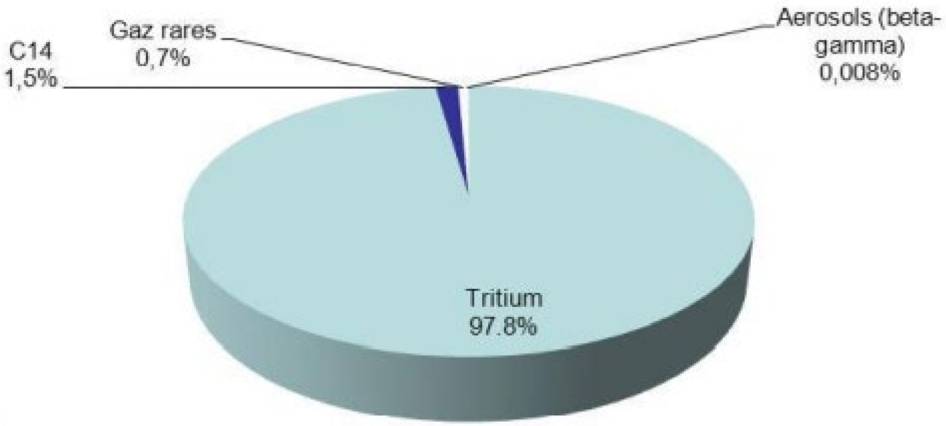

Figure 4. *Dose contributors to ITER routine gaseous and liquid releases [2]*

Tritium appears to be the almost single driver of the low doses from routine releases to the most exposed members of the public, showing the necessity to reduce tritium routine releases ALARA via an adequate use of atmosphere and water detritiation systems. This covers both gaseous and liquid releases, gaseous releases contributing to around 95 % of the dose. The incorporated dose is well below (by a factor of around 1000) the maximum dose limit for the public mentioned here above.

In addition, it is important to remind that liquid releases contributors are also driven by tritium. The World Health Organization [7] has specified drinking guidance level for tritium at 10 000 Bq/L. This guidance value (not a limit) is a rounded value of the tritium concentration in water that would, for the daily consumption of 2 L (730 L per year), would lead to an incorporated indicative dose of 0.1 mSv.

European commission has also issued an EURATOM drinking water directive [1] establishing a parametric value for tritium in water intended for human consumption at 100 Bq/L (100 times less than the WHO guidance value), above which investigations and analyses are required with regards to the presence of other artificial nuclides. This parametric value may have no utility for fusion reactors in that sense since mostly tritium would lead to an impact as shown in figure 4.

2.3 Accident releases and public exposure from fusion reactors

Similarly to other nuclear facilities, all potential accidental situations are studied too for fusion reactors.

Many accident situations are similar to the ones studied for these other nuclear fission reactors:

- External hazards (severe climatic conditions, earthquakes, external flooding, tsunamis when applicable, commercial airplane crash, etc.)
- Internal hazards (fire, explosion, flooding, load drops, etc.)
- Postulated initiating events (loss of off-site power, loss of coolant accidents, primary to secondary leaks, scenarios during maintenance, etc.),
- Design extension conditions (DEC) associated with multiple failures.

But some accidental events are new compared to fission facilities, such as:

- Plasma events (disruptions, loss of magnetic confinement, etc.)
- Loss of vacuum inside the vacuum vessel,
- Breeding blankets events (pipe ruptures, exothermic reactions, etc.)
- Cryogenic gases pipe rupture
- Detritiation systems failure
- Etc.

The radioactive releases from these accident scenarios depend on the way the radioactive materials can be involved and mobilised during the scenario, and on the efficiency of confinement systems.

Based on the accident analyses made for ITER research facility, it appears that tritium is often the driver of accident radiological consequence doses, but that, depending on the different accident scenarios, some other nuclides can also be drivers of the radiological consequences (activated dust or activated corrosion products), as shown in figure 5.

This example of some design basis accidents of this ITER facility shows the necessity to consider safety provisions in the confinement of all type of radioactive materials in fusion reactors for coping with accident situations.

Figure 6 shows the radiological consequences comparison of routine releases events, as well as several accident scenarios. The incorporated doses for the most exposed members of the public are all below 0.5 mSv including for the most severe design extension conditions (in comparison of the French regulations for general members of public requiring to lead to doses less than 1 mSv/y for routine situations).

These examples of accident scenarios show how a careful selection of confinement systems can reduce the radiological consequences from fusion facilities.

	Contribution to the dose for reference groups		
DBA	tritium	dust	ACPs
Large DV ex-vessel coolant pipe break at baking	6%	0%	94%
Loss of vacuum through one VV/cryostat penetration line	100%	0%	0%
Failure of fueling line	100%	0%	0%
Stuck divertor cassette and failure of cask	100%	0%	0%
Large DV ex-vessel coolant pipe break	91%	6%	3%
Coolant pipe break inside Port Cell (normal operation)	33%	0%	67%
Failure of transport hydride bed	100%	0%	0%
Leak of tritiated water from WDS	100%	0%	0%
Large VV coolant pipe break	60%	0%	40%
Loss of confinement in hot cell	44%	56%	0%

	Contribution to the dose for reference groups		
BDBA (DEC)	tritium	dust	ACPs
Fire in the waste processing area plus propagation to buffer storage room in the hot cell	28%	72%	0%
Confinement Failures in the Tritium Plant	100%	0%	0%
Hydrogen and dust explosion in the vacuum vessel	99%	1%	0%
Fire in the T-plant	100%	0%	0%
Damage to VV and cryostat resulting in large holes of 1 m2	1%	99%	0%
Hydrogen Deflagration and Detonation in the Tritium Plant	100%	0%	0%
Fire in the hot cell buffer storage room	66%	34%	0%
Multiple failure of FW cooling loops inside VV+ failure of both windows in RF heating line	16%	84%	0%
FW Ex-Vessel Loss of Coolant with Failure of FPTS	7%	0%	93%
Loss of vacuum through 1 VV penetration line + 2 hours blackout and in-vessel coolant leak	93%	7%	0%

Figure 5. *Examples of dose contributors in ITER accident scenarios [3]*

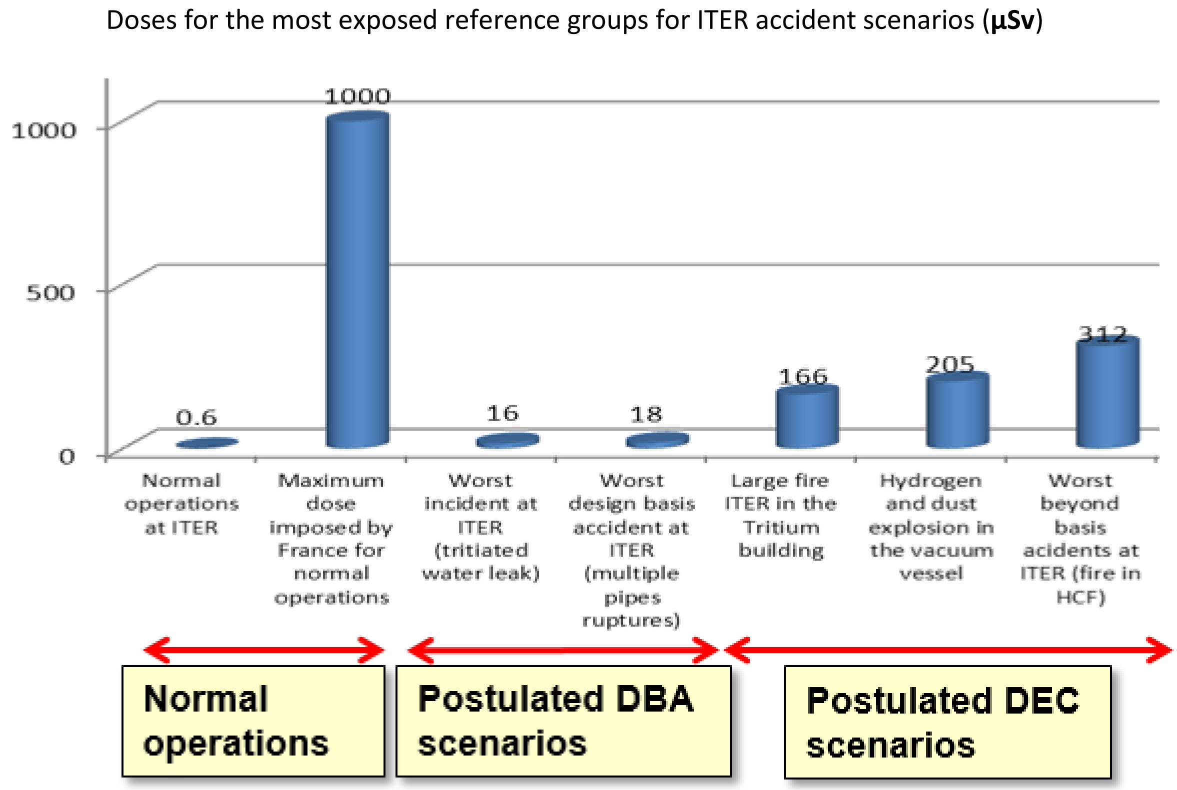

Figure 6. *Examples of estimated doses from ITER postulated accident scenarios [3]*

3 Conclusion

Fusion reactors are based on self-sustaining deuterium-tritium fusion reactions (D-T) leading to neutron activation of materials, creating activated dust and activated corrosion products. The fusion fuels materials have a very low radiotoxicity. Thanks to the adequate selection of materials, the other radionuclides created from neutron activation have a low radiotoxicity too.

The choice of state of the art confinement standards for radioactive materials is made for limiting and reducing the consequences on workers and on members of the public. Tritium is the expected driver of the public exposure for routine situations, and for several accident situations; but there are postulated accident scenarios for which other nuclides (activated aerosols) have also to be considered. This requires using adequate confinement systems against all types of radionuclides.

Given an adequate use of confinement systems and the lower nuclides radiotoxicity, the public exposure impact for fusion reactors would remain low for both routine and accident situations.

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3. Progress on safety design principles against tritium at ITER research facility - Tritium conference – Nice - Oct 2013

4. ICRP 134 - Occupational Intakes of Radionuclides: Part 2

5. ISO 17873:2004 Nuclear facilities — Criteria for the design and operation of ventilation systems for nuclear installations other than nuclear reactors

6. ISO DIS 16646 Nuclear installations — Criteria for the design and operation of confinement and ventilation systems of tritium fusion facilities and fusion fuel handling facilities

7. World Health Organization Guidelines for drinking-water quality:2022 Fourth edition incorporating the first and second addenda

Summary

Prepared by Dr. Laurence Lebaron Jacobs

on behalf of the

Working Party "Research Implications on Health and Safety Standards" (WP RIHSS) of the Group of Experts referred to in Article 31 of the Euratom Treaty[19]

1. Introduction

This chapter provides the rationale of EU Scientific Seminars, summarises the individual presentations, and the roundtable discussion on policy implications and research needs of this year's Scientific Seminar on *Radiological protection considerations for fusion reactors*. It takes into account the discussions that took place during the seminar, although it is not intended to report in an exhaustive manner all the opinions that were expressed. These proceedings have been submitted for comments to the lecturers and round-table participants, as far as their contributions were concerned.

2. The Article 31 Group of Experts and the rationale of the scientific seminars

The Article 31 Group of Experts is a group of independent scientific experts referred to in Article 31 of the Euratom Treaty, which assists the European Commission in the preparation of the Euratom Basic Safety Standards for the protection of the health of workers and members of the public against the dangers arising from ionising radiation. This Group of Experts has to give priority to protection of health, to safety and to development of the best available operational radiation protection. To this end, the Group of Experts is committed to proactively scanning new or emerging issues in science and technology, and ongoing developments in the area of radiation protection and informing the European Commission on potential policy implications.

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[19] The Scientific Seminar was chaired by R. Wakeford and L. Lebaron-Jacobs was acting as rapporteur. In addition, the following members of the Working Party on Research Implications on Health and Safety Standards of the Article 31 Group of Experts contributed to the preparation of this overview P. Olko, F. Bochicchio, I. Prlić, A. Dumitrescu. They were assisted by F. Tzika and S. Mundigl from the European Commission.

In this context, a Scientific Seminar is devoted every year to emerging issues in Radiation Protection – generally addressing new research findings with potential policy and/or regulatory implications. Following suggestions from the Working Party Research Implications on Health and Safety Standards (WP RIHSS), the Article 31 Group of Experts selects the topic of the seminar. After selection of the topic and approval of the programme by the Article 31 Group of Experts, the WP RIHSS deals with the preparation and the follow up of the seminar. Leading scientists are invited to present the status of scientific knowledge in the selected topic. Additional experts, identified by members of the Article 31 Group of Experts from their own country, take part in the seminars and act as peer reviewers. The Commission usually convenes these seminars in conjunction with a meeting of the Article 31 Group of Experts to allow the Group to discuss potential implications of the presented scientific results. Based on the outcome of the Scientific Seminar, the Article 31 Group of Experts may recommend research, regulatory or legislative initiatives. The Experts' conclusions are valuable input to the setting up of the European Commission's radiation protection programme, and to the process of reviewing and potentially revising European radiation protection legislation.

3. Key highlights of the presentations at the Scientific Seminar on Radiological protection considerations for fusion reactors

Richard Wakeford – Introduction

Radiological protection for fusion reactors should consider five issues:

- Tritium

  Tritium is a pure beta-particle-emitter with the electron emitted during decay having a low energy. This low-energy electron has a relatively high linear energy transfer (LET), i.e., a relatively high ionization density for a beta-particle, which suggests a relative biological effectiveness (RBE) of the electron emitted by tritium is >1 when compared to higher energy beta-particles. An RBE value of 2-3 has been suggested by reviews of the evidence using high-energy gamma radiation as a reference radiation. Since tritium is a radioisotope of hydrogen, the effects of the presence of tritium in DNA has been discussed. Organically bound tritium (OBT) and its retention in the body is also a subject of some discussion. Given that commercial fusion reactors will have an inventory of kilogram quantities of tritium, it seems inevitable that the impact of any release of tritium will come in for close scrutiny – this was recognized by Willard Libby half a century ago.

- Relative Biological Effectiveness (RBE) of 14 MeV neutrons and scattering of neutrons to lower energies

  The RBE of neutrons is a subject that has been much discussed, and these discussions continue – it is one of the matters presently being considered by ICRP Task Group 118. Currently, the radiation weighting factor (w_R) to apply to the absorbed dose of neutrons, as adopted by the ICRP 2007 Recommendations, varies with neutron energy, with the maximum of 20 for intermediate neutron energies of a few MeV. The 14.1 MeV neutrons emitted during fusion have a radiation weighting factor of ~5, but these neutrons downscatter to energies with higher w_R values before the w_R then decreases to 2.5 for lower energy neutrons. However, the appropriate radiation weighting factors for neutrons for the purposes of radiological protection continues to be assessed.

- Gamma radiation from fusion and neutron activated products

  Nuclear fusion reactors generate photons with a range of energies, from the high energies emitted during the fusion process itself, through intermediate energies generated by downscattering of these primary gamma-rays, to the photons emitted during the decay of radionuclides produced by neutron capture. It is this last category of sources of gamma-rays that is likely to produce the main problems for radiological protection because this source remains when the reactor is shut down and being maintained or decommissioned.

- Shielding against a range of neutrons and photon energies

  Nuclear fission reactors and other sources of neutrons and gamma-rays have provided a wealth of experience of shielding these radiations against unacceptable levels of radiation in working areas (or the environment). However, certain problems have to be borne in mind, such as streaming of high energy neutrons through shielding and "skyshine". Neutron activation presents more of a difficulty because of operations handling these gamma radiation sources when the reactor is shut down, and portable shielding methods will be required. Again, experience of operating nuclear fission reactors will be invaluable under these circumstances.

- Radioactive waste disposal

  Inevitably, fusion reactor maintenance and decommissioning activities will require the handling and disposal of radioactive materials. Tritium contaminated materials are an obvious source of radioactive waste, and the 12.3-year half-life of tritium probably implies interim storage before final disposal of the most contaminated items. Neutron activation products will also have to be dealt with, and these will have a range of activities and half-lives. Some of these radionuclides resulting from neutron capture will have been recognized before reactor commissioning, but surprises are likely, such as neutron activation of contaminant nuclides in structural materials. These aspects of fusion reactors need to be carefully considered.

Christian Grisolia - An introduction to basics of fusion technologies

The principle of thermonuclear fusion combines two light nuclei to form a single heavier one releasing massive amounts of energy. This reaction needs very high temperatures, a stable and confined plasma by means of a strong magnetic field in order to produce energy. This can be achieved in the configuration of a tokamak, a vacuum chamber containing different low atomic mass ions heated and confined. In ITER, the reaction involving deuterium (D) and tritium (T) will produce an alpha particle and a 14 MeV neutron. Some tokamaks operate in the world as JET (Culham, England), which uses tritium and is the only one able to perform D/T fusion reactions. ITER (Cadarache, France) is under construction and should show the scientific and technical feasibility of fusion as a source of energy and test the breeding blanket for tritium production. DEMO whose characteristics are still under discussion should be an electricity-producing reactor coupled to the grid.

An important aspect to be noted here is that uncontrolled fusion cannot take place in a tokamak, mainly due to the very small amount of tritium in the plasma. However, tritium can contaminate materials and the environment via permeation due to its large re-circulation in the tokamak, in the tritium plant and auxiliaries. In a fusion reactor, long-lived waste will not be produced. However, in ITER special materials will have to be managed for which R&D is necessary. In addition, studies on management of tritium releases and wastes with high tritium content should be assessed.

Elisa Mancia - Basic radiation protection issues in future fusion reactors

In the Sun, deuterium combines with tritium to produce a helium nucleus and a neutron. The mass of the atomic nucleus or nuclei produced by the fusion process is less than the mass of the atomic nuclei involved. Therefore, the difference is the release of a massive amount of energy.

To reproduce and maintain nuclear fusion for a net power gain, high temperatures, intense pressure, and a technology for plasma confinement are needed. Different fusion technologies include magnetic confinement fusion and inertial confinement fusion. The most widely studied magnetic confinement fusion device is the Tokamak.

Different radioactive materials can be found in fusion facilities. Tritium is present as a gas (T_2), easily diffusing across materials and seals, or as tritiated water (HTO) which is very corrosive. Thermal desorption is used to remove tritium from internal components of the first wall of the vacuum vessel in order to avoid losing too much fuel and inducing serious consequences in the event of an accident involving tritium. Therefore, detritiation systems are key elements regarding the confinement of tritium. There are also activated materials or products resulting from interactions between neutrons from fusion reactions in the plasma and the nearby environment: activated structural material, which constitute the major part of the waste, activated dust generated by the erosion of the first wall of plasma-facing components, water activation products (tritium, ^{14}C, ^{16}N, ^{17}N and ^{19}O), activated corrosion products in water, and activated gases (e.g. ^{14}C and ^{41}Ar).

ITER (Cadarache, France) is an international nuclear fusion research and engineering project and the world's largest experimental tokamak nuclear fusion reactor. The objective is to demonstrate the scientific and technological feasibility of fusion power and allow for the subsequent construction of commercial fusion reactors. ITER requires a construction license decree under the French nuclear facilities regulation, inter alia because of its foreseen tritium inventory of up to 4 kg.

The main ITER safety issues:

- Tritium confinement
- Radiation protection of public and workers (high radiation exposure risk mainly due to 14 MeV neutrons, tritium and neutron activated materials).

In accordance with the IRSN technical advice on radiation protection on ITER (2019-2022), exposure to ionising radiation will last after the pulses due to the presence of activation products created by the neutron flux combined with a progressive accumulation cumulating of activation products with a long half-life (e.g. ^{60}Co).

Although the conceptual design of future demonstration reactors is the same as that of ITER, the goal is to achieve tritium self-sufficiency and longer operating times, which could have a significant impact in terms of design, safety and radiation protection of workers and public.

DEMO, the European DEMOnstration power plant and ITER's successor, should produce electricity. As DEMO will have longer operating time, the exposure of the public and/or workers to ionising radiation will be increased. In addition, due to the presence of 14 MeV neutrons flux and longer operating time, faster deterioration of the initial properties of structural materials surrounding the plasma should occur. Therefore, safety recommendations on the choice of materials to withstand the intense neutron flux and to reduce maintenance operation time have to be assessed. The use of massive biological shielding to limit worker exposure risk during light maintenance will be required.

In future fusion reactors where tritium breeding blankets would be used to ensure tritium self-sufficiency, due consideration must be given to the choice of TBM type, in relation to its impact on tritium releases into the environment (releases associated with the cooling system for tritium breeding blankets and with the transfer and processing of internal components to the hot cells), as well as to radionuclides produced from neutron activation of TBM materials (e.g. ^{203}Hg and ^{210}Po produced via activation of LiPb eutectic liquid), which may have significant radiological impact.

Preliminary studies have shown that most of the potential accidents to be considered in DEMO are identical to those considered for ITER. Nevertheless, due to the increased complexity of DEMO, the larger radioactive inventory, the higher magnetic energy involved during operation, as well as potential additional release path of tritium (via the TBM cooling system), some other types of accidents should be also considered.

It is worth noting that the fusion reactor needs to comply with relevant requirements of the host country’s regulatory system, e.g. those on radioactive waste management.

Walter Shmayda – Radiological inventories and source terms

The highest tritium inventories are estimated to be in the torus wall and in the isotope separation systems. The torus inventory estimate has the highest uncertainty. The current EU DEMO fuel cycle relies on three loops to decrease the plant tritium inventories. Reducing the power from 2000 to 500 MW will decrease the plant inventory at least 5-fold for a compact tokamak. Some technologies could potentially reduce potential tritium releases.

Gas handling systems should reside in gloveboxes using helium cover gases and independent recovery systems of tritium gas to mitigate accidental releases without conversion to tritiated water. System inventories in the presentation were based on selected technological approaches and can be affected by changes in the processing concept.

Some future research needs to decrease radiological plant inventory susceptible to release include:

- Improving performance of tungsten-like walls exposed to tritium;
- Improving hydrogenic isotope separation systems.

Sandro Sandri – Occupational radiation exposure at the experimental fusion facilities

JET's primary task is to support construction and future operation of ITER, acting as a test bed for ITER technologies and plasma operating scenarios. In JET, workers receive the most significant dose during maintenance activity (66.9 %). In addition, using adequate detritiation systems and protective clothes allow to keep tritium exposure as low as below recommended dose constraints/values.

The primary neutron field resulting from the fusion reaction occurring in the vacuum vessel, gamma radiation emitted by neutron activated products, activation of the cooling water, tritium used as fuel for the fusion reaction, and waste containing tritium and gamma emitters are the radiation sources to be considered regarding workers exposure.

Based on studies for ITER, dust and coolant activation do not represent a risk of exposure for workers and population because an administrative limit of 1000 kg of dust was defined for ITER at the end of life. According to this assumption, the materials and characteristics of the process in the plant, the percentage composition of the dust is supposed to be: 50 % Beryllium, 30 % Tungsten, 20 % Tritium.

Neutrons in the tokamak can generate some radionuclides by activation of air (e.g. ^{3}H, ^{14}O, ^{41}Ar, ^{40}Cl), but they are not an issue for occupational radiological exposure. Water activation produces two radioisotopes: ^{16}N (decays producing high energy gamma rays but has a very short half-life) and ^{17}N (produces a delayed neutron that would activate the component inner walls), but their contribution can be neglected owing to the distance from the tokamak core to the premises external to the bioshield. Tritium is part of the fuel mixture and can contaminate the atmosphere of tokamak and tritium buildings. Evaluations have shown that tritium contribution to the occupational exposure in ITER was about 20-30 % of the total. Since 1997, tritium dose was around 1 % of the average worker dose by means of radiation protection measures used in JET.

In ITER, it was estimated that the annual individual dose received by an operator during normal surveillance would be 2 mSv. During maintenance, the most exposed operator could receive an annual effective dose of about 10 mSv however most workers involved in maintenance should stay below an annual effective dose of 6 mSv. The same annual individual dose of 10 mSv was estimated for RH hands-on assistance. Estimated individual doses to workers likely to be present at the time of an accident, under several design basis accident scenarios, were of less than 1 mSv.

DEMO will use low activation materials (e.g. Eurofer). However, DEMO's low reliability should need frequent maintenance activities; therefore, exposure of workers would be significant. For the moment, the target of 10 μSv/h after 1-day cooling cannot be fulfilled inside rooms with active loop components. The analysis of estimated individual doses of workers under accident scenarios for DEMO is still ongoing. The preliminary results show a situation similar to that of ITER. Therefore, the objective set for the limit of individual dose that could be received by DEMO workers during an accident is the same as that of ITER namely ≤ 10 mSv.

Pierre Cortes- Environmental releases and public exposure in normal and accident situations

In fusion reactors, tritium is used as a fuel in large quantities. Dose coefficients for the different tritium chemical forms found in the framework of occupational activities were defined in the ICRP 134 publication. Those regarding the environmental impact have not yet been published.

During D-T plasma operating periods, workers are not allowed to enter the tokamak building. As biological shielding constitutes the tokamak building walls, neutron dose rates are expected to be low outside the bioshield. Therefore, neutrons are not considered to be a health issue for public and the environment. Nuclides in materials used in fusion reactors are activated by neutrons and can further emit radiation presenting a major source of exposure.

Some radionuclides produced by neutron activation of plasma facing components will be different from those occurring in nuclear fission reactors. These rradionuclides will be mostly derived from Tungsten (e.g. ^{185}W, ^{187}W,^{185m}W), however there will be other radionuclides also, such as ^{182}Ta and ^{186}Re. As fusion plasma erodes surfaces inside the reactor, radionuclides will be embedded in dust. Other radionuclides can be produced from activation of Lithium-Lead liquid mixtures or ceramics pebbles containing a mixture of lithium, silicon and titanium, when these materials are used to create tritium, including ^{203}Pb, ^{209}Pb, ^{31}Si, ^{45}Ti, ^{51}Ti, but also alpha emitters such as ^{210}Po at trace levels. In order to control the production of these radionuclides, the control of impurities in materials is essential.

Stainless steel can also be activated resulting in the ‘classical’ radionuclides occurring also in fission reactors due steel activation. In fusion, EUROFER-97 has specifically been selected due to its low neutron activation properties. Furthermore, activated corrosion products, such as ^{55}Fe or ^{60}Co also found in fission reactors, could be created inside cooling loops. Conventional radionuclides from activation of water (^{16}N, ^{17}N), water chemical impurity or additive nuclides (^{14}C), and air (noble gases e.g. ^{41}Ar), will also be created.

Reducing the radiotoxicity of nuclides present in the operating fusion reactors will be possible through an adequate selection of materials.

In fusion reactors, ‘new’ accidental events have to be considered compared to fission facilities, such as plasma events (disruptions, loss of magnetic confinement), loss of vacuum inside the vacuum vessel, detritiation systems failure, breeding blankets events, cryogenic gases pipe rupture. Accident analyses carried out for ITER showed that, while tritium appears to have the main contribution to doses to reference groups in most accident scenarios considered, other radionuclides present in activated dust or in activated corrosion products were found to have the highest contribution depending on the accident scenarios.

In both routine and accident situations for fusion reactors, incorporated doses for the most exposed members of the public have been estimated to be below 0.5 mSv.

4. Summary of the roundtable contributions

Christian Grisolia - Issues with tritium

As fusion reactor mobilises a large quantity of tritium, which is extremely reactive and mobile because it can permeate through all materials. Tritium poses major issues in respect to fusion reactor design, operation, and dismantling. Therefore, there is a need for maintaining a tritium inventory at any time in all the parts of the fuel cycle including the tokamak, the breeding system, the tritium plant and the waste, but also for appropriate predicting tools. The issues of controlling tritium permeation and releases (confinement) during operation, maintenance and waste management needs also to be further considered.

Improving the knowledge on tritium management in fission and fusion facilities was addressed within the European projects TRANSAT (closed) and TITANS (ongoing). The projects cover a range of topics, including development of active and passive permeation barriers against tritium permeation, measurement of tritium inventory in soft waste, development of a safe container for tritium waste storage, modelling of tritium inventory and migration using state-of-the-art tools to predict tritium releases, as well as studying the behaviour in the environment and the toxicity of tritiated steel and cement particles produced during tritium plant dismantling.

The outcomes of TRANSAT showed that an active barrier concept was assessed to be suitable for mitigation of tritium permeation. In addition, using data collected from biokinetic studies on inhaled tritiated steel dust in rodents, the project resulted in the definition of a Committed Effective Dose coefficient, of 5.6×10^{-12} Sv/Bq, which could be applied to workers during dismantling operations in case of accidental exposure.

Lee Packer – Dosimetry

Currently there are different fusion strategies around the world, with EU having the first comprehensive roadmap towards fusion power plants (in an 'evidence driven' approach), and USA and UK developing strategies based on parallel development processes (in a 'schedule driven' approach). UK government's fusion strategy targets operation of a prototype fusion power plant (STEP) to 2040.

A wide range of evidence and operational experience from experimental activities at JET and MAST is available and includes data collection from planned and unplanned maintenance activities, as well as aspects of tritium (and Be) monitoring, operational personal dosimetry as well as area monitoring.

Understanding the neutron spectrum in fusion workplace environments is needed to perform accurate dosimetry. Neutron fluence to dose conversion factors used to calculate ambient dose equivalent or effective dose are given by ICRP 74 and ICRP 116. ICRP 74 H*(10) units are mostly conservative for energies < 14 MeV, apart from ICRP 116 AP at energies of ~few MeV.

Some developments so far in JET and MUST facilities include development of neutronics models of facilities to predict neutron spectra, and development of a passive neutron spectrometry (PNS) diagnostics system which aims to measure neutron spectra and fluxes which are essential for dose assessments. Furthermore, a prototype virtual JET environment with dose information has been developed for dose minimisation during maintenance procedures. In regards to neutron yield measurement, a primary measure of the fusion power, a EUROfusion project was conducted at JET with resulted in calibrating JET neutron diagnostics using a DT neutron generator with accuracy <10 %, a level of accuracy which is also target for ITER.

- Experience for current fusion experiments (e.g. JET and MAST) and projections for next step fusion reactors (ITER, DEMO, STEP), including considerations during plasma operations and residual fields during shutdown, raised research needs in terms of dosimetry: dosimetry and radiation metrology capabilities in tritium, neutron environments, and residual radiation fields),
- use of virtual (digital) nuclear environments to minimise doses during maintenance operations based on merging facility geometric information with high fidelity radiation maps,
- measurements of low doses received by workers during plasma operations including validation of neutronic models' predictions and of diagnostics helping to measure dose levels,
- shielding measurements considering the 6 MeV photons emission of ^{16}N produced from activated water,
- calibration and nuclear diagnostic design for accurate neutron yield measurements, regarded as safety measurements based on experience gained from JET and studies for ITER
- need for more nuclear benchmark experiments for streaming and shielding aspects,
- neutron activation of coolants and activated corrosion products,
- plasma interactions and activated dust due to erosion of the first wall.

Lewis Simmons- Radioactive waste issues

JET will end operations in December 2023 and a large increase in waste arising and in waste radionuclide inventory is predicted. Tritium is dominant radionuclide for the site, but tritiation of in-vessel materials is difficult to estimate. Other issues in relation to tritium to consider include: uncertainty in off-gas rates prior to breaching systems, difficulty to detect migration routes, intensive characterisation including manual material handling, often changing of gloves and other PPE routinely due to permeation behaviour. There will be also activated components that can cause issues. Neutronics and activation modelling provide useful information on expected gamma doses but carries uncertainties related to impurities activation and breaks down in areas where the models become less well defined, while offers limited capability for dose rate from β particles (which results in handling concerns). Beryllium contamination from equipment and areas will be also an issue. There are likely to be large volumes of tritiated soft waste produced at fusion facilities and tritiated soft waste is problematic to characterise. In addition, fusion uses some novel materials, which are not similar to fission radioactive materials.

Research needs have been identified in terms of radioactive waste generated by fusion reactors:

- material activation and tritiation studies,
- recycling and reconditioning of fusion materials,
- tritium detection techniques, characterisation methodologies, and protection techniques.

In terms of policy, the notion of "fusion wastes" has to be re-thought because they are driven by short-lived radionuclides, the hazard and management of this material is much more time-dependent than fission materials.

Richard Wakeford – Round table

As noted in the Introduction to the Seminar, the risks posed by exposure to tritium have been, and continue to be, much discussed – the UNSCEAR 2016 Report (Annex C) is a good recent source of information. Data derived from epidemiological studies of groups of humans exposed to tritium are highly desirable in this respect, particularly those exposed in the workplace because monitoring of exposures is likely to have occurred. A number of approaches to studying occupational exposure to tritium have been made, but the majority of dose estimates used in these studies were not tritium-specific and this is a serious limitation of these studies.

However, recently, a detailed study of the mortality of workers at three French military nuclear sites has been conducted using tritium-specific doses[20]; the 1746 workers included in this study were unexposed to external radiation sources or other radionuclides, which is a strength of this study. Only 15 % of the tritium workers had died when the study was conducted, and the numbers of deaths from particular cancer types was small, although the overall number of 124 cancer deaths was much less than expected from French national mortality rates, an example of the "healthy worker effect". Further, only 15 % of the workers included in the study had tritium doses >1 mSv. The study found indications of excess risks of cancers of the pancreas and bladder and of non-Hodgkin lymphoma, but the very small numbers of deaths of these specific types of cancer means that these findings must be viewed with considerable caution. Hopefully, follow-up of these workers will continue.

This French study underlines the need to conduct epidemiological studies of tritium-exposed workers using tritium-specific doses. Multicentre studies, involving workers from a number of countries, are highly desirable to obtain as much statistical power as possible. The risks posed by exposure to tritium continue to be debated, with suggestions that the risks may have been seriously underestimated. The nuclear fusion fuel cycle will use large quantities of tritium, and it is inevitable that the risks from tritium used in fusion will be the subject of public debate. Properly conducted epidemiological studies of tritium-exposed workers will be one substantial way that this issue can be addressed, and efforts should be made to encourage such studies.

[20] Martin S., Ségala C., Epidemiological study of mortality among workers exposed to tritium in France, (2021) Radiat. Res., 195 (3), pp. 284 - 292

Conclusions of the discussion[21]

Based on the presentations and the discussions during the Scientific Seminar, the experts of the Working Party RIHSS identified the following important *radiation protection issues in fusion reactors:*

- Tritium is the main, although not the only, radionuclide to be considered for radiation protection in normal operation, but other radionuclides may play an important role particularly in certain accidental situations.
- During normal operations, workers may be exposed to tritium, including tritium released from tungsten in the torus, to tritiated residual dust, to gamma radiation from accumulated activation products with non-negligible half-lives (e.g. ^{60}Co), and to neutrons streaming through the shielding.
- In future fusion reactors where tritium breeding blankets would be used to ensure tritium self-sufficiency, due consideration must be given to the impact of the type of Tritium Breeding Modules on tritium releases into the environment, as well as to the radionuclides produced from neutron activation of tritium breeding blanket materials (e.g. ^{203}Hg and ^{210}Po from activation of LiPb eutectic liquid) and of impurities therein, which maybe of significant radiological impact.
- During maintenance and decommissioning of fusion reactors, handling, characterisation, and disposal of radioactive materials containing tritium and neutron activation products will require special considerations of: Tritium handling challenges, presence of unexpected radionuclides from neutron activation of impurities in structural materials.
- Longer operating times and higher power in future fusion reactions (e.g. DEMO) would result in higher exposure to ionising radiation of workers and public compared to ITER. The choice of materials surrounding the plasma is important as it would affect the activation products inventory as well as the frequency of maintenance activities which would in turn impact occupational exposure.
- Due to the higher complexity of DEMO compared to ITER, potential accident scenarios, beyond the one considered for ITER, will have to be identified and assessed.
- Understanding the biokinetics of tritiated particles and their radiobiological impact in any potential release, is essential especially in future commercial fusion reactors where an inventory of kilogram quantities of tritium will be encountered.
- Understanding the neutron spectrum in fusion workplace environments is needed to perform accurate dosimetry.

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[21] These conclusions were prepared by: L. Lebaron-Jacobs, P. Olko, F. Bochicchio, A. Dumitrescu, I. Prlić. They were assisted by F. Tzika and S. Mundigl from the European Commission.

- With a view to the management of future radioactive waste generated, experience from JET shows that it is difficult to estimate the tritiation of in-vessel materials. Activated components can also cause issues. Activation modelling for dose predictions can help, when models can be well defined, but carry uncertainties from material impurities while offer limited capability for dose rate from β particles (handling concerns). Beryllium contamination from equipment and areas is also an issue. There are likely to be large volumes of tritiated soft waste, after facilities decommissioning, that is problematic to characterise. In addition, fusion uses some novel materials.

Based on the roundtable discussion, the experts of the Working Party RIHSS further identified the following research needs:

- Further scientific research and discussion should address biokinetics of organically bound tritium (OBT) and tritiated particles in the human body. Furthermore, biological studies should address the toxicity of tritiated particles, which can be accidentally released, on humans and environment with a view to assess the dose-response relationship and derive respective dose coefficients.

- The energy dependence of the Relative Biological Effectiveness of neutrons should be further studied.

- With a view to reducing tritium inventory and potential accidental release, hydrogenic isotope separation systems need to be improved.

- Identified research needs in terms of dosimetry include dosimetry and radiation metrology capabilities, use of virtual nuclear environments for minimising doses during maintenance, validation of methods and measurements of worker doses during plasma operations, in particular personal dosimetry capabilities in relation to the 6 MeV photon emissions of ^{16}N produced from activated water.

- Calibration and nuclear diagnostic design for accurate neutron yield measurements, nuclear benchmark experiments for streaming and shielding aspects, neutron activation of coolants and activated corrosion products, plasma interactions and activated dust due to erosion of the first wall should be further investigated.

- Identified research needs in terms of radioactive waste generated by fusion reactors include material activation and tritiation studies, recycling and reconditioning of fusion materials, as well as tritium detection techniques, characterisation methodologies, and protection techniques.

- Need of an international epidemiological study on worker exposure to tritium.

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