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# Materials-related issues in the safety and licensing of nuclear fusion facilities

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# Materials-related issues in the safety and licensing of nuclear fusion facilities

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#### Abstract

Fusion power holds the promise of electricity production with a high degree of safety and low environmental impact. Favourable characteristics of fusion as an energy source provide the potential for this very good safety and environmental performance. But to fully realize the potential, attention must be paid in the design of a demonstration fusion power plant (DEMO) or a commercial power plant to minimize the radiological hazards. These hazards arise principally from the inventory of tritium and from materials that become activated by neutrons from the plasma. The confinement of these radioactive substances, and prevention of radiation exposure, are the primary goals of the safety approach for fusion, in order to minimize the potential for harm to personnel, the public, and the environment. The safety functions that are implemented in the design to achieve these goals are dependent on the performance of a range of materials. Degradation of the properties of materials can lead to challenges to key safety functions such as confinement. In this paper the principal types of material that have some role in safety are recalled. These either represent a potential source of hazard or contribute to the amelioration of hazards; in each case the related issues are reviewed. The resolution of these issues lead, in some instances, to requirements on materials specifications or to limits on their performance.

Keywords: safety, environment, licensing, hazards

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

#### 1. Introduction

#### 1.1. Principal hazards in a fusion facility

Fusion power promises to provide a method of electrical energy generation with very favourable safety and environmental characteristics. Studies of the safety of conceptual fusion power plants [1, 2] as well as those of the ITER facility [3] have shown that the consequences of postulated accidents are limited, the environmental impact of normal operation is low, and there need be no accumulating radioactive waste burden with only a limited quantity of long-lived waste. Nevertheless, to be sure to realize this potential for outstanding safety and environmental performance, the safety design must take care to minimize the inventories of hazardous materials and to make provisions to prevent their release. These provisions must address all foreseen operational conditions, including normal operation, planned and unplanned maintenance activities, and all postulated accident scenarios. The choices and deployment of materials in the design are key aspects of the strategy to achieve the safety and environmental goals.

The radioactive materials that represent the principal hazardous inventories fall into two categories: tritium and the products of neutron activation. Tritium, as a fuel for the fusion process, is both consumed and generated, and is therefore inevitably present in various parts of the plant, as discussed in section 3. The fuel cycle includes the injection systems that supply tritium (and deuterium) to the plasma, either as a gas or in frozen pellets. The tokamak exhaust and vacuum systems remove gasses from the plasma chamber, containing a large quantity of tritium fuel, as only typically 2% of injected deuterium and tritium is 'burnt' in fusion reactions in the plasma. A processing plant extracts these hydrogen isotopes from the exhaust and may also separate tritium and deuterium, sending them to storage units ready for re-use in the fuel injectors. Tritium is also generated within the breeding blankets inside the vacuum vessel (VV), from where it is removed and transported by an extraction system to the processing and storage plant. All of these systems of the fuel cycle naturally contain an inventory of tritium. Their components become contaminated with tritium, both on their surfaces and, through permeation, in the bulk of their materials. The rooms, enclosures and glove boxes containing this equipment may also become contaminated, and their atmospheres must be served by ventilation systems that can remove tritium from the extracted air, these detritiation systems themselves representing another inventory of tritium. All components inside the VV are also likely to become contaminated with tritium, in their bulk as well as on the surfaces by co-deposition with other materials (see section 4.3); the accumulating in-vessel inventory may become high enough to warrant special techniques to be employed for its removal during plant maintenance shutdown periods. Some tritium becomes oxidized to become tritiated water, HTO, and contact with other fluids, such as lubricants, may produce tritiated hydrocarbons or other compounds. There are thus very many parts of a fusion power plant that may contain a tritium inventory, even at a low level, that could in principle be partly released to the environment via normal venting systems or in an accident.

The second category of radioactive material is that resulting from neutron activation. 14 MeV neutrons from the D-T fusion reaction of the plasma, moderated by interactions in surrounding materials, provide a broad energy spectrum of neutrons penetrating far into surrounding structures and equipment as far out as the biological shield. Plasma-facing components experience the highest neutron fluence, leading to a high level of activation of their structural materials (discussed in section 2) and functional materials (section 5). Erosion of the plasma-facing surfaces generates dust that accumulates in the vessel; this is also highly active and additionally may absorb a significant quantity of tritium (see section 4). Activated to a lower level, but still significant, are the VV itself and all components within the cryostat, including magnet structures. Fluids, particularly coolants of in-vessel components, can also become activated together with the products of corrosion that may be carried by the coolant flow (see section 2.5).

In addition to these inventories presenting a radiological hazard, there may be materials that present non-radiological risks. These include conventional risks that are present in any large industrial plant, and also specific hazards such as the toxicological potential of materials such as beryllium. These must be protected from release to the environment in a similar manner to the radioactive materials.

#### 1.2. Safety approach

The top-level safety objectives for a nuclear fusion plant follow established practice at existing nuclear facilities and are based on international guidelines [4]. They may be stated as:

- To protect workers, the public and the environment from harm;
- To ensure in normal operation that exposure to hazards within the facility and due to release of hazardous material from the facility is controlled, kept below prescribed limits and minimized to be as low as reasonably achievable (ALARA);
- To ensure that the likelihood of accidents is minimized and that their consequences are bounded;
- To ensure that the consequences of more frequent incidents, if any, are minor;
- To apply a safety approach that limits the hazards from accidents such that in any event there is no need for public evacuation on technical grounds;
- To minimize radioactive waste hazards and volumes and ensure that they are ALARA.

In order to meet these objectives, safety functions are defined and safety principles adopted. One of these principles, mentioned in the second bullet point, is to maintain radiological doses resulting from normal operation and maintenance ALARA. Application of the ALARA principle involves successive steps throughout the plant lifetime, starting at the design stage, to reduce radiation exposure through improvements in design and operational procedures. Such steps are repeatedly made until further reduction cannot be made by improvements that are considered reasonable, taking into account economic and social factors.

Another strategy is that of Defence in Depth, in which multiple levels of protection are employed to prevent an accidental situation from resulting in harmful consequences. At the first level, conservative design and high quality fabrication helps to avoid failures and minimize vulnerable hazardous inventories, at subsequent levels provisions are made for detecting abnormalities, controlling the plant when in an off-normal state, preventing propagation of an accident and mitigation of the consequences. Application of this strategy leads to multiple levels of protection, for example by the provision of several independent barriers to confine radioactive inventories.

A further principle is that of passive safety, which requires that wherever possible safety functions are achieved without reliance on active systems such as those dependent on an electrical or hydraulic supply. For the confinement function, this leads to the use of strong physical barriers that are already existent in the design, such as the VV. Nevertheless, these passive physical barriers are complemented by active ventilation systems that maintain a pressure cascade between isolated volumes, so that any leakage is always in the direction towards the more contaminated space.

In order to assess the effectiveness of the application of the safety approach, criteria are set against which performance can be judged. These include limits such as dose limits for occupational radiation exposure (ORE). The 'no-evacuation criterion' implied by the fifth bullet point above enables a quantitative evaluation of the design to be performed, by analysing the worst possible consequences of postulated accident scenarios. This analysis can provide a predicted maximum radiological dose to the most exposed individual, and this can be compared with recommended guidelines for the level of averted dose at which evacuation would be triggered on technical grounds.

#### 1.3. Safety functions

The safety functions that must be provided in the design of a nuclear fusion plant are identified in the course of safety analyses. They are those functions needed to ensure that any accident initiating event does not lead to a sequence of events with unacceptable consequences. They are involved in the prevention of off-normal operation, in the mitigation of the consequences of abnormalities, and in restoring the plant to a safe state. The systems, structures and components that implement these safety functions may be dedicated safety equipment, but equally the functions may be performed by existing parts of the design, for example as already mentioned, the use of the VV for the confinement function. Whatever provisions are made to implement the safety functions, they must be done with a very high level of reliability, often requiring full redundancy in the case of active systems.

The most important safety function is that of confinement. Every inventory of radioactive material needs protection to prevent its mobilization into a volume where it could give rise to an ORE and to avoid a release to the environment and potential public radiation dose. To provide adequate leaktightness and reliability of confinement systems, it is normal to require at least two independent confinement systems for each significant radioactive inventory. Each system can comprise one or more physical barriers and possibly active filtering and detritiation systems. For example, for the in-vessel radioactive inventory of tritium and activated dust the first confinement system may be the VV and its extensions, with the second confinement provided by walls and slabs of the building coupled with ventilation systems and filtered/detritiated venting. As noted above, the Defence in Depth principle implies the use of multiple barriers.

#### 1.4. Challenges to confinement

The barriers and systems that provide the confinement function are required to do so in all situations including abnormal and accident conditions. They must therefore be designed taking into account all foreseen loads, including those categorized as unlikely or highly unlikely. The main challenges to physical barriers come from elevated pressure differentials, high or low temperatures, electromagnetic loads, mechanical loads and impacts, and the effects of corrosion. These may arise as a result of a wide range of accident situations including loss of coolant events, fire and explosion, mechanical failures, and in the case of electromagnetic loads, abnormal plasma events.

Safety systems are put in place to mitigate the risks of accidental events that may challenge confinement barriers, for example fire suppression systems, but nevertheless confinement systems must be designed to withstand all conceivable challenges including external events such as earthquake or aircraft crash. This implies that materials must be selected with properties that will remain in the required ranges in all conditions predicted by analyses of postulated accident scenarios. Since many parts of the equipment in a fusion plant play some role in the confinement function, a range of materials are subject to these requirements. For example, the first confinement barrier for the in-vessel inventory of retained tritium and activated dust is the VV itself, together with all parts of the vacuum boundary, including seals, bellows, feed-throughs and non-metallic windows in diagnostic systems and radio frequency (RF) heating system waveguides.

#### 2. Structural and cooling system materials

The materials community has responded to design safety needs with the development of new materials that can provide safety functions and perhaps be recycled after a time period for radioactive decay. These alloys are meeting safety and environmental needs as well as the structural requirements of deuterium–tritium fusion design. In this section we examine the safety functions, hazards, waste disposal, and personnel safety concerns of structural and cooling system materials.

#### 2.1. Structural and cooling systems safety functions

The safety function of certain structures in fusion plant is to provide reliable confinement of radioactive and toxic materials. These structures are called confinement barriers, such as the VV, its ports and other penetrations, some internal components, and the pressure retaining components of its cooling system. These structures must remain robust, leaktight, have low induced radioactivity and minimize tritium permeation in all conditions, including operational conditions of ionizing radiation, corrosion, cyclic stresses, and high temperature and pressure, during both normal operation and accident conditions, including plasma and seismic events. The VV cooling system must also reliably remove decay heat from the tokamak. The role that materials play in these safety functions is to ensure that these structures and systems reliably perform throughout the lifetime of the facility.

#### 2.2. The hazard potential of structural materials

2.2.1. Activation and its consequences. Since the 1970s, the fusion materials programs around the globe have been



**Figure 1.** Specific activity of FW made of SiCf/SiC, ferritic–martensitic steel, V, or W alloys (activity of T from nuclear reactions included; T deposited in FW from plasma is not considered).

developing four main classes of low-activation, short-lived structural materials that require only near-surface burial and relatively short regulatory oversight of ~100 years: ferritic/ martensitic steel, silicon carbide/silicon carbide (SiCf/SiC) composites, tungsten alloys, and vanadium alloys. Each structure favours a specific liquid breeder (Li, lead-lithium alloy PbLi, Flibe-a Fluorine, Lithium and Beryllium molten salt LiF-BeF2, or Flinabe-a Fluorine, Lithium, Sodium and Beryllium molten salt LiF-NaF-BeF<sub>2</sub>) and coolant (water or He gas) based on compatibility considerations. Solid breeders are compatible with any structure. Upon exposure to neutrons, structural materials become much more radioactive than breeders or coolants. Figure 1 shows the specific activity of the four structural materials subject to the same radiation environment at the first wall (FW) of a typical tokamak [5]. A realistic level of impurities was included in the compositions of all materials. Within one day, the SiCf/SiC activity drops significantly while the activities of all materials decrease by 6-7 orders of magnitude in 100 years after plant shutdowna salient safety feature of low-activation materials. As other sections of this paper illustrate, the activation of fusion components could impact the machine safety, environment, and health/safety of workers and general public, unless attention is carefully paid to the design to minimizing the radiological hazards.

2.2.2. Accident source term. In accident analyses, reference accident scenarios are identified on the basis of the list of radioactive inventories (or source terms) present in the plant and on the basis of confinement barriers dedicated to isolate the inventories from the environment. For each source term the failure of one, or more than one, related confinement barrier will be considered in any safety assessment. One of the practices used in applying safety culture to the design of plants such as a demonstration fusion power plant (DEMO) is the minimisation of radioactive and hazardous material inventories and energy sources that can act as accident initiators.

In a fusion power plant, there are several potential source terms for radioactive releases to the environment and dose intake for public and workers:

- the vacuum chamber, which mainly contains activated dust and tritium;
- the cooling loops, which could contain both mobilisable activated materials and tritium;
- the coolant purification and detritiation systems which could contain both activated materials and tritium;
- the fuel cycle systems, which contain primarily tritium, but activated material may also be present;
- the air detritiation systems, which mainly contain tritium, and
- the hot cells and refurbishing building, containing tritium, activated dust, and activated material.

The activated products in the cooling loops are a combination of activated coolant, coolant impurities, structural chemical and/or coolant erosion corrosion and ejected activation products from structures called sputtering [8]. In water cooled blanket models, the corrosion products are predominant and the other types of products are considered negligible; while for helium cooled designs, the corrosion products will be very limited and sputtered products are expected to be the primary source of activated material, but the quantity is still under investigation (note table 1). The same should be true for a blanket concept based on both PbLi and helium coolants (dual coolant), where helium is used to cool reduced-activation ferritic/martensitic (RAFM) steel structures, and a SiCf/SiC flow channel serves as a thermal insulator to further reduce the surface temperature of the steel in the PbLi channels to levels where generation of corrosion products are acceptable.

Neutron activation of lead-lithium generates products, including isotopes of bismuth, polonium, thallium and

**Table 1.** Source terms, derived from safety and environmental assessment of fusion power (SEAFP) study and assumed in the PPCS studies for the analysis of accident sequences of a future fusion power plant.

Source terms	Model A	Model B	Model C
Tritium in VV	1 kg	1 kg	1 kg
Dust	10 kg	10 kg	10 kg
	(7.6 kg of stainless steel, SS-dust + 2.4 kg W-dust)	(7.6 kg of SS-dust + 2.4 kg W-dust)	(8.55 kg of oxide dispersion strengthened steel, ODS-dust + 1.45 kg of W-dust)
Tritium in coolant	15 g (per loop)	1 g (per loop)	0.003 g (per loop)
Activated corrosion products (ACPs) total inventory	50 kg (per loop)	_	_
ACPs mobilization fraction	1% of 50kg (per loop)		_
Mobilization fraction, tritium and dust	1	1	1
Sputtering products		~0 g	~0 g <sup>a</sup>

<sup>a</sup> The neutronic activation calculations performed in [8] estimated a production of 1.7 kg in 30 years of operating life for a total activity of 6.6E14 Bq.

mercury, that can be deposited in the system piping and components, even though cold traps in are intended to remove them. The radioactivity of <sup>210</sup>Bi, <sup>210</sup>Po, <sup>202</sup>Tl, <sup>204</sup>Tl, and <sup>203</sup>Hg could be of special concern. Highly toxic <sup>210</sup>Po is formed again by a two-step reaction <sup>208</sup>Pb $(n,\gamma)^{209}B(n,\gamma)$ . Other contributions come from the initial Bi impurities which are between 5 and 20 appm in commercial PbLi.

When the lead-lithium is melted and some aerosols created, <sup>210</sup>Po is the product with the largest safety concern for maintenance and accident conditions. Even at temperatures when the lead-lithium is solidified, experiments show that <sup>210</sup>Po is released from PbLi, although at a much lower rate than in the liquid state [6]. However, during normal operating conditions there should be no concern because <sup>210</sup>Po is an alpha emitter and the containing structure is sufficient to safely shield and protect workers. Tritium permeating from the plasma chamber and the breeding material into the coolant will be dispersed in the whole primary coolant inventory. Tritium could be in the vapour form (HTO/DTO) or gaseous tritium (HT/DT/  $T_2$ ) depending on the moisture contained in the coolant. In accident analyses, for conservatism, it is assumed that all the tritium in the coolant is in the vapour form even for a DEMO cooled by He gas and/or PbLi.

As an example, the source terms assumed for the accident analysis of a future power plant have been summarized in table 1 for three European power plant conceptual study (PPCS) plant Models [7], e.g. Model A (PbLi as breeder and water as coolant), Model B (Be pebble beds as breeder and Helium gas as coolant) and Model C (PbLi with dual functions of coolant and breeder, plus He gas as coolant).

2.2.3. ORE from shutdown gamma dose. The experimental nature of fusion devices and the lack of a large routine maintenance experience database lead to uncertainties in occupational dose assessment. The contribution to the worker dose in fusion plants will be due to both internal and external exposure, but the internal exposure can be expected to be less than 10% of the collective dose in a typical existing operating nuclear fission plant. Looking at data from the ITER ORE assessment (figure 2) the percentage of the worker dose arising from planned maintenance of the different systems reveals the importance of the water cooling system and diagnostic

systems as the main contributors [8]. Additional contributions will arise from internal doses, doses during routine operation and in corrective maintenance, each expected to contribute about 8% to the total collective dose.

Focusing attention on the materials used in the cooling system, it should be pointed out that the gamma dose rate is generated by the deposition of the activated corrosion products (ACPs) on the pipe, valve, pump and heat exchanger walls and by the direct activation of the pipe walls induced by the delayed neutron emitted by the decay of  $^{17}$ N, an activation product of  $^{16}$ O [9]. In a water-cooled DEMO or power plant, such as the water-cooled lithium lead (WCLL) concept, the same problem arises, while the contribution of the cooling system to the worker dose is less significant in a helium cooled plant such as the helium-cooled pebble bed (HCPB), helium-cooled lithium lead (HCLL) or dual-coolant lithium lead (DCLL).

Structural material is also the origin of a high gamma dose rate in the vicinity of plasma heating and the current drive systems (electron and ion cyclotron heating and neutral beam injection), as well as components related to remote handling maintenance, hot cell and radwaste processes. This may arise both from activation of the structural materials in the maintenance zones of these systems and from dust generation and contamination due to machining. The use of lower activation materials may significantly reduce the collective dose in a fusion plant for the above systems, if extensively applied.

2.2.4. Radioactive waste. Fusion power plants will be likely to generate a sizable amount of radioactive material. To put matters into perspective, we compared in figure 3 the power core volumes of ITER [10], the advanced reactor innovations and evaluation study (ARIES) tokamaks (ARIES aggressive and conservative tokamak designs, ARIES-ACT-1&2) [11, 12] and the European PPCS Model C [7] to the economic simplified boiling water reactor (ESBWR)—a Gen-III<sup>+</sup> advanced fission reactor [14]. Figure 4 illustrates the volumes of individual components comprising the fusion core of ARIES-ACT-1 with R = 6.25 m [11] and ARIES-ACT-2 with R = 9.75 m [12]—recent power plant designs in the ARIES series [14]. The FW/blanket/divertor components will be replaced several times during the plant life, but the total



Figure 2. ORE contribution per ITER system to the total collective dose [8].



Figure 3. Comparison of radioactive waste from power core of fusion and fission designs (actual volumes of components; not compacted; no replacement; no plasma chamber).

volume of the replaced items, after compacting, is expected to be similar to the volumes shown here. These in-vessel components are likely to be classified as Intermediate-Level Waste, whereas permanent components outside the vessel including the bulky magnets and their supporting structure will be Low-Level Waste. Surrounding the fusion power core is the bioshield—a 2 m thick, steel-reinforced concrete building that essentially protects the public and workers against radiation. Being away from the plasma, the bioshield contains very low radioactivity, but its large volume (not shown in figure 4) dominates the waste stream.

#### 2.3. Minimizing the hazard potential

2.3.1. Low activation materials, optimization of composition. Materials with low radioactivity from neutron activation can



Figure 4. Volume of ARIES-ACT-1&2 individual components (actual volumes; not compacted; no replacement).

be a benefit to fusion device design in several ways, including enhanced safety in accidents by reduced radioactivity, reduced volume of waste, reduced radiation exposure during facility maintenance, and reduced toxicity of facility effluents [15]. Fusion researchers have explored use of exotic materials that have low neutron activation: V alloys, RAFM, SiC, and W alloys. These materials could be used in tokamak blanket modules and other in-vessel components. Testing in Test Blanket Modules will help to determine which of these exotic materials show the greatest promise for use in future devices. Not all fusion materials must be so exotic if they are farther from the tokamak, where operating environments have reduced neutron exposure and reduced neutron energy. Engineering alloys for use in fusion, such as austenitic or ferriticmartensitic stainless steels, have been studied for optimization by reducing the composition of impurities, which reduces the overall radioactivity of the material at the end of its service life. An example is reducing the silver impurity in stainless steel [16] and Nb and Mo impurities in RAFM [17]. With good planning, low activation materials and optimized engineering alloys will reduce the waste burden during decommissioning, perhaps even leading to free release of portions of the decommissioning debris for use in commercial industry.

2.3.2. Radwaste strategies, clearance and recycling. The potential problem of sizable fusion radwaste volumes was recognized even in early fusion design studies [18]. Disposing of such a large amount of radwaste in geologic repositories is not a viable nor an environmentally attractive option for two main reasons: existing commercial repositories will tend to be reserved to their maximum capacity by fission waste needs, even before building the first fusion power plant (for example in the US by 2050–60), and in many territories the political

difficulty of opening new repositories has lately proven to be extremely difficult with strong negative public perception.

Since burying the hefty bioshield, in particular, in geological repositories is impractical, the United States Nuclear Regulatory Commission (US NRC) [19], the European Commission [20] and the International Atomic Energy Agency (IAEA) [21, 22] proposed the clearance concept where components with traces of radioactivity (such as the cryostat and bioshield) could temporarily be stored for the activity to decay, then released to the commercial market for reuse as shielding blocks or containers, concrete walls or rubble base for roads, deep concrete foundations, dams for flood control, etc.

During the decade of the 1990s and continuing to the present, strategies have been pursued to reduce the quantity of waste requiring disposal by the maximum use of recycling (reuse of activated materials within the nuclear industry) and clearance (release of slightly activated materials to the commercial market after meeting strict criteria for the nuclide inventory) [23–36]. Several fusion power plant design studies have employed an integral waste management strategy, showing that the recycling and clearance approaches are technically feasible, and identifying several critical issues that need to be addressed with a dedicated R&D program. This effort has been strongly international in nature [37–42]. If successful and integrated properly at an early design stage, the recycling/clearance approach will allow the minimization of the quantity of radioactive material assigned to repositories.

#### 2.4. Impact of damage, loss of integrity

Radiation hazards include radioactive materials that can produce radiation exposure through inhalation, ingestion or skin contact, when such materials are released from their normal



Figure 5. Corrosion phenomena in coolant piping.

location in the plant. In addition, there are sources of radiation that can produce a radiation exposure at a distance, referred as direct radiation fields.

A substantial inventory of tritium and activation products generated in the plasma chamber and the plasma facing components is confined inside the VV. A part is mobilised from the VV volume into systems interfacing with it, including the vacuum pumping system, the cooling systems, the tritium extraction systems (TESs), the liquid metal breeder (e.g. PbLi) system and the purification systems. The vacuum exhaust, the coolants, the purging and the liquid metal breeder are the mediums transporting tritium and activated products beyond the first vacuum boundary defined by the VV and its port plugs. If there is a loss of integrity of the VV or any part of the vacuum boundary, release of radioactive products towards the plant area and the environment will result. Thus the performance of the materials of the components that comprise this boundary, or support it, is highly important. Any degradation due to irradiation effects such as embrittlement or swelling will have potential safety consequences if the confinement barrier is affected.

Besides these two types of direct releases of radioactive products, if an ex-vessel rupture of the systems interfacing with the VV is followed by a consequential in-vessel rupture, then a bypass between plasma chamber and the external plant area can occur. As an example, an ex-vessel loss of coolant accident (LOCA) can be followed by in-vessel break either because of high thermo-mechanical stress in the coolant passages if the plasma is not promptly shut down or by plasma disruption caused by the fast plasma shutdown system. In these cases, air and coolant or other gas enter the VV and pressurize it. Tritium and activated materials contained inside the plasma chamber are mobilised both by the ingress of air and coolant or other gas in the VV through temperature/pressuredriven volatilisation. Then the in-vessel radioactive materials can be transported through system channels and ducting to the location of the external break where differential pressure between the vessel and the external room cause these particles to spread into the room as aerosols.

In case of a radioactive release inside the plant the first humans at risk are workers, but clearly if the enthalpy of driving fluids (as coolants) is high and/or electric or chemical energies are present to aggravate the accident consequences, then the last confinement barriers can be impaired and an environmental release can also occur. In this case, atmospheric dispersion and dose to the public will follow. Dose to the maximally exposed individual (MEI) at the site boundary has to be evaluated to determine if these bounding events will exceed the safety limits and the plant safety objectives.

#### 2.5. Cooling systems, corrosion and ACPs

In a water cooling system such as a pressurized water reactor (PWR) primary system, an ITER primary heat transport system (PHTS) cooling loop or a water-cooled breeder blanket, the formation and transport processes of ACPs is a complex process. It involves many different mechanisms that react among each other. The first mechanism is the uniform and generalized corrosion of metallic alloys. For stainless steel materials, this leads to the generation of a dual oxide layer: an inner compact layer (chromite) and an outer porous layer (ferrite). The inner layer is a passive oxide layer, which limits—but does not preclude—ion exchange between metallic alloys and primary coolant: ions are released in the primary coolant. The quantities of released materials are small (of about several mg dm<sup>-2</sup>/month) and do not alter component integrity [43].

The primary coolant transports ions generated by the corrosion-release phenomenon or by oxide dissolution. When the coolant becomes supersaturated in corrosion products, ions can precipitate on the walls or in the bulk of the fluid to form particles. Flow-induced erosion processes also generate particles. Transported by the primary coolant, particles are deposited inside the coolant flow circuits or they can agglomerate. Dissolution and precipitation depend on the corrosion product equilibrium concentrations, which depend on coolant chemical treatment (pH,  $H_2$  concentration (or Redox potential),

temperature). A schematic of the phenomena involved for iron based piping is depicted in figure 5 above.

Two types of ACP formation coexist. On the one hand, the activation of corrosion products occurs when they are deposited on surfaces under neutron flux. On the other hand, the corrosion of structural materials under neutron flux is accompanied by a release of radioactive corrosion products.

The use of austenitic steels, such as type 316 or type 304, is an effective means to reduce the problem of material corrosion in cooling water systems. Type 316 steel is a chromium-nickel steel that contains 2-3% molybdenum. The Mo content increases corrosion resistance, improves resistance to pitting in chloride ion solutions and increases strength at high temperatures. On the other hand, the type 304 has good corrosion resistance as well. Type 316L and 304L stainless steels are extra-low carbon versions of the 316 and 304 steels, respectively. Their lower C content minimizes deleterious carbide precipitation as a result of welding. Consequently, 316L or 304L are used when welding is required in order to ensure maximum corrosion resistance. The corrosion rates for 304L and 316L are typically of the order of few tenths of  $\mu m \text{ yr}^{-1}$ (1  $\mu$ m yr<sup>-1</sup> corresponds to 6.5–6.7 mg dm<sup>-2</sup>/month for stainless steel and ~7.4 mg dm<sup>-2</sup>/month for Cu alloy).

Copper alloys are also used in cooling water systems for their good resistance to corrosion coupled with combinations of other desirable properties, such as superior electrical and thermal conductivity, ease of fabricating and joining, and their wide range of attainable mechanical properties. Experiments carried out with coolant water chemistry similar to those envisaged for ITER (neutral conditions) with stagnant and flowing coolant showed larger corrosion rates for Cu alloys with respect to austenitic steels.

The corrosion of copper has some implications on the cooling water systems (like the ITER PHTS). At first there is the corrosion of the copper alloy itself as mentioned above. There are two main secondary issues that can occur independently of the corrosion mechanism on the copper material itself [44]:

- Galvanic corrosion between the copper alloy and the surrounding stainless steel structure, which is a local corrosion effect at the joints of the materials.
- Influence of the copper present in the water on the stainless steel corrosion, which can happen anywhere in the system.

Furthermore, copper alloys are known to be sensitive to erosion–corrosion that may occur at large coolant velocities  $(>10 \text{ m s}^{-1})$ .

The Cu-alloy release rates are of the order of 10  $\mu$ m yr<sup>-1</sup> for ITER PHTS cooling system baking (200–240 °C) and burn periods. The release rates are extremely low during the other operating periods (0.01–0.1  $\mu$ m yr<sup>-1</sup>).

For austenitic steels it is extremely important to reduce the level of material activation by a proper composition of the alloying elements (cobalt, nickel and niobium) and impurities.

For the Cu-alloys it is more difficult to control the activation levels by alloying element composition as the dominating element is Cu and 'longer lived' activation products of Cu are cobalt isotopes, neglecting <sup>64</sup>Cu, which at shut down is the dominant, because it has a half-life of 12.7 h. It is well known from the feedback of experience from operating fission reactors that <sup>57</sup>Co, <sup>58</sup>Co and <sup>60</sup>Co are the principal isotopes responsible for increased ACP deposits.

An important issue for the cooling water system is to avoid the occurrence of radiolysis induced by neutron causing dissociation of water molecules. When exposed to radiation, water undergoes a breakdown sequence into  $H_2O_2$ , hydrogen radicals and assorted oxygen compounds, such as ozone. The possibility that enhanced concentration of hydroxyl present in irradiated water in the inner coolant loops must be taken into account to prevent coolant loss resulting from corrosion.

The effect of material corrosion and generation of ACP for the other two main coolant systems envisaged for DEMO (helium coolant system and dual coolant system) are discussed in section 2.2.2.

#### 3. Tritium

In this section, the safety characteristics and at-risk inventories of tritium in a fusion facility, excluding those contained within the VV, are discussed. Safety approaches presented include minimizing tritium inventories, reducing permeation of tritium from these systems, and decontamination of these systems for maintenance or for waste disposal.

#### 3.1. Characteristics of tritium

The general characteristics of tritium are described in various sources, e.g. [45, 46]. Key characteristics germane to this paper are summarized here.

Tritium is radioactive and decays as follows:

$$T = \beta + {}^{3}He.$$

The  $\beta$  particle has a range of energies with an average of 5.7 keV and a maximum of 18.6 keV. Tritium produces no penetrating radiation. The range of the  $\beta$  particle in air is about 5 mm. The radioactive half-life is 12.3 years.

Tritium chemistry is very similar to that of normal hydrogen (i.e. protium). So, in its pure form at room temperature, tritium exists as  $T_2$  gas. And, like hydrogen, it can react with oxygen to form water ( $T_2O$ ), C to form methane ( $CT_4$ ) and N to form ammonia ( $NT_3$ ). Tritium in all of these species may be substituted with hydrogen and/or deuterium. Substitution or exchange reactions are important for understanding and controlling tritium. For instance, tritium gas might not be captured to an appreciable extent by a vessel's surface, but tritium can exchange with water or oil on the vessel's surface resulting in appreciable contamination. In like manner, this phenomenon can be exploited to decontaminate surfaces.

Also, like hydrogen, tritium is relatively mobile. As a gas it disperses to fill its container, be that a process vessel, a glovebox, a room or beyond. As a small atom, it can move through normally impervious solids such as gloves and metals. And, as a reactive element, it enters and leaves water, oils and particles which can be mobile carriers of tritium.

At times there are safety implications of the fact that each tritium decays to helium. For instance, tritiated water may be adsorbed on a solid resulting in no contribution from water to pressure in the vessel. But, as tritium decays to He, matter moves from the adsorbed phase to the gas phase and pressure increases. In certain cases pressures can be appreciable and require controls.

The beta decay from tritium has sufficient energy to break chemical bonds. Thus, tritiated water undergoes radiolysis resulting in generation of hydrogen gas. Under certain conditions sufficient hydrogen can accumulate to pose an explosion hazard that must be mitigated. Also, the tritium beta can catalyse reactions which would otherwise not take place. For instance, at room temperature the tritium beta will promote reforming and exchange chemistry which normally requires elevated temperatures to proceed.

The beta decay energy is not sufficient to pass through the skin's dead layer of cells, so tritium is only hazardous when it actually enters the body. This can be by skin absorption, ingestion, breathing tritiated gas or breathing solid tritide particulates. In all but the latter case, tritium becomes part of the body water which has a biological half-life of about 10 d. The chemical form of tritium that a person is exposed to strongly affects the radioactive dose. For instance, exposure to tritium as water is much more hazardous than HT gas since water is more easily absorbed into the body. Dose considerations for the remaining exposure pathway, solid tritide particulates, are different since these remain in the body much longer than those associated with body water. A preliminary assessment of the dose conversion factor for some metal tritides has been carried out in the frame of a literature survey on biological effects of tritiated dust [47, 48].

Owing to tritium's radioactivity, it is easily detected to very low concentrations. This is readily done for gases (e.g. by ion chamber), liquids (by scintillation counting) and surfaces (by swipes/scintillation counting). However, tritium is more difficult to detect in bulk solids. In many cases the bulk content can be inferred from surface measurements and knowledge of operating conditions. But, if this is insufficient, more aggressive methods such as thermal desorption spectroscopy can be used.

Together, these characteristics lead to the key strategies for managing tritium's radioactive hazards. Key controls for some other forms of radiation such as shielding and to some extent high efficiency particulate air filtering are not key for tritium. Rather, the objective is to minimize or eliminate direct personal contact with tritium as follows:

- Limit the tritium inventory
- Contain the tritium with one or more barriers considering the following phenomena:
  - Permeation
  - Outgassing
  - Bulk retention

These points are discussed in the following sections.

## 3.2. Tritium inventories in a fusion facility and associated hazards

A magnetic fusion facility has a rather small amount of tritium inside the fusion device. At ITER, a  $0.5 \text{ GW}_{\text{thermal}}$  facility, this quantity will be about 0.2 g tritium. But to maintain this value

requires considerable inventory outside of the tokamak. Since only a small fraction of the tritium burns on each pass through the device [49], a significant recirculation rate with associated processing equipment is required. The plant inventory can be sizable. For instance, in the current, most advanced designs such as ITER this is because:

- Cryopumps used to maintain tokamak vacuum work in a pump/regenerate sequence resulting in tritium accumulation
- Cryogenic distillation used to separate hydrogen isotopes requires a significant inventory to function as needed
- Other fuel cycle processing steps require accumulation in buffer volumes to maintain performance
- Inside the VV, tritium accumulates on surfaces and within components, and tritium recovery can only be performed infrequently with limited effectiveness using techniques currently available
- When components are sent to waste, they hold tritium inventory which takes considerable time to recovery
- Sufficient tritium reserve must be maintained to make up for decay, tritium burned and losses

The required plant tritium inventory will depend on factors such as the fusion power, the plasma burn fraction and approaches to the factors listed above [50]. ITER will perform its mission within a site inventory limit of 4000 g tritium. For a fusion power plant some inventories could be larger or smaller, depending on the fusion concept used and the design of the tritium processing systems. For example, some design concepts for power plants and DEMO do not employ cryopumps, and may recirculate a significant part of the D–T mixture in the plasma exhaust without separating the isotopes, so a smaller tritium processing system may suffice.

The primary nuclear hazards in a tritium facility are exposure of workers and/or the public to tritium.

Tritium facilities are categorized for nuclear hazards largely by the site tritium inventory. In France, any facility with a tritium inventory exceeding 2g requires regulatory authorization, which is how ITER comes to be classified as an Installation Nucléaire de Base (Basic Nuclear Facility). Similarly, US Department of Energy (DOE) facilities with 1.6–30g of tritium are categorized as Hazard Category 3 nuclear facilities meaning that the facility shows the potential for only significant localized consequences, while facilities above 30 g tritium are Hazard Category 2 meaning the facility shows the potential for significant on-site consequences. Tritium inventory does not result in a Hazard Category of 1, meaning that the facility shows the potential for significant off-site consequences. Fusion facility tritium inventories place them well into Hazard Category 2, meaning that the measures necessary to control these inventories will be significant. In order to maximize facility safety, it will be key that the principles of ALARA and defence in depth are applied in the design and operation of tritium processing systems. Such principles are implemented via minimization and segregation of inventories throughout the plant, and through a robust confinement strategy based on the use of multiple tritium barriers.

One key to controlling these hazards is inventory minimization, segregation and confinement. In a tritium facility the largest set of hazards is present when it is in full operation with tritium-containing gas circulating in a loop. In this mode of operation, a full set of controls is in place to ensure the hazards are properly handled. But, should an accident such as a pipe break occur, it would in theory be possible for significant tritium to be released. Thus, a set of isolation valves are in place which can be activated to further limit the amount of tritium that can be released. The facility can quickly transition to this isolated mode, as needed. But, in this mode there is still gaseous tritium at risk, so as needed, all tritium can be placed on hydride storage beds which store tritium with negligible tritium overpressure at room temperature. In this mode the tritium inventories are confined and the facility is passively safe from most credible accidents. To date, depleted uranium has proven to be the most effective tritium storage medium because of its absorption and regeneration efficiency [51]. However, uranium is pyrophoric and any ingress-of-air accident could possibly cause the bed to heat and release the tritium stored on the bed. Other options have been developed that are not as pyrophoric as uranium with some success, including the following materials: Zirconium Iron, Nickel, Titanium, and Zirconium Cobalt, but these materials are still not as effective as uranium. The effort to find a more effective storage media than uranium should be continued for this safety reason.

One example of a promising new technology which could result in lower tritium inventories is a continuously regenerable cryopump [52]. Normal cryopumps follow a sequence of pump, warm-up, gas removal, cool-down. In certain cases, four pumps are required to supply continuous pumping. This scheme results in a significant accumulation of tritium inventory. The new technology reduces the accumulation by using a regenerative head that continuously moves across the cryocondensation surface and removes the hydrogen isotopes while the pump is pumping. This could greatly reduce the tritium inventories associated with systems based on normal cryopumps. Technologies such as this merit future investigation.

#### 3.3. Permeation, permeation barriers

At high temperatures (such as those present in a fusion power plant), tritium is extremely mobile, and can readily permeate through many solid materials. Permeation can be an important loss term (the tritium must ultimately be recovered and processed) but also presents a safety concern, since it may permeate through barriers and into areas where it is neither desired nor intended. Analysis of permeation is thus an important part of the facility design, as are systems designed to mitigate it.

Permeation through metals is governed by three processes: (1) dissociation of tritium (or HT) atoms at a surface, (2) diffusion of the atoms through the solid material, and (3) recombination of surface T atoms to form gaseous, diatomic molecules again. If the dissociation and recombination are almost in equilibrium with each other on each side of a material, the concentrations at each side are given by Sieverts' Law, and the resulting concentration gradient across the layer gives the permeating flux, which is proportional to the square root of the tritium pressure. This is the 'diffusion-limited' regime. The constant of proportionality is the 'permeability', the product of diffusivity and solubility (Sieverts' constant). Both parameters are typically well described by Arrhenius functions and increase with temperature (though the opposite trend is observed for the solubility in the hydride formers).

The amount or tritium present in coolant streams may be large from a radiological perspective, but it represents only a very small partial pressure. At such low pressures, the aforementioned equilibrium at surfaces is not necessarily reached. If dissociation or recombination is the limiting process, the concentration gradient across the solid will be nearly flat, and the permeating flux is linearly dependent on pressure. This is the 'surface-limited' regime, which has been observed in a number of experiments (see e.g [53].). The transition from surface-limited to diffusion-limited permeation does not occur at a specific pressure, but is identified by a dimensionless parameter (W') that depends on the pressure and the material properties (permeability, dissociation rate constant, and wall thickness) [54]. In the diffusion-limited regime the hydrogen can suppress tritium permeation and it is expected to reduce tritium concentrations in the solid (but not the permeating flux) in the surface-limited regime [55, 56].

A permeation analysis, then, requires some knowledge of the diffusivity, solubility, dissociation rate constant, and recombination rate constant for each constituent material as a function of temperature. There is a substantial database of diffusivity and solubility parameters, especially for common materials such as steels and nickel alloys; see [57] for a comprehensive list and references. These are thermodynamic properties of a material, and since measurements can be made at high pressures, they are at least in principle straightforward to measure accurately. Dissociation and recombination rate constants are a different story. Aside from the inherent difficulty in making measurements at vanishingly small partial pressures, these parameters can vary widely based on the condition of a surface, and as a consequence repeatable results may be difficult to obtain. There is little if any data available on these parameters for most materials (we note that they have been measured for a few materials including tungsten [58], MANET, a martensitic Cr-Mo steel alloy [59], Inconel 625 [60], 316L stainless steel [60], OPTIFER-IVb, a martensitic chrome steel [61] and Incoloy 800 [62]), and what has been measured can differ by orders of magnitude from other measurements or theory [63]. Since surface-limited permeation is always less than diffusion-limited permeation (for a given material), analyses based on the latter will at least be conservative. But if surface-limiting conditions are actually present, tritium permeation will be mitigated to some degree.

Since many candidate materials (e.g. RAFM steels) do not have a low enough permeability at power-plant relevant temperatures to limit tritium permeation to the extent necessary, some other strategy must be employed. One such strategy is to apply low-permeability coatings (such as the oxides  $Al_2O_3$  [64],  $Cr_2O_3$  [65], and  $Er_2O_3$  [66]) to them so as to form permeation barriers. The effectiveness of such barriers is usually described in terms of a permeation reduction factor (PRF), the factor by which the flux is reduced in otherwise identical conditions upon application of the barrier. Though several barrier materials have performed extremely well in laboratory tests (PRFs of 1000–10000 have been achieved [64–66]), their performance in a fusion environment (and particularly in a radiation environment) and over long periods of time has not been adequately demonstrated [57].

Reviews of permeation barrier development and performance under irradiation are given in [57, 67]. The PRFs of 1000-10000 observed in some laboratory experiments could not be reproduced in-pile; some near 100 were observed, but others were as low as only ~3. Three models were proposed that might describe the mechanism by which permeation barriers work, that may shed some light on how radiation affects them. The first is a composite diffusion model, in which tritium is assumed to permeate though both the barrier and substrate; permeation through the barrier is rate-limiting. The second is an area defect model, in which the barrier is essentially impermeable, and any permeation occurring is through a small fraction of the total area exposed by defects. Finally, transport may be surface-limited, e.g. by the recombination rate. Some evidence is offered in support of both the area defect model as well as surface-limiting; on the one hand, activation energies have been observed not to change upon barrier application (indicating the substrate is still the limiting factor), but in other cases, a transition to linear dependence of the permeation rate on partial pressure has been observed, indicative of surface-limited permeation. All three mechanisms might be influenced by radiation. Radiation creates defects that might lead to formation of blisters, embrittlement, and cracking; this would increase permeation under the area defect model. Recombination rates might increase via photodesorption [68], and radiation-enhanced diffusion has also been reported in the literature [69]. Any of these might contribute to the poor performance under irradiation.

More recently,  $Er_2O_3$  has received considerable interest as a permeation barrier, because of its mechanical stability under thermal loads, compatibility with Li, and stability of its insulating properties under irradiation [70], and it has since demonstrated very high PRFs (~10<sup>5</sup> [66]). It has also shown to degrade significantly when exposed to PbLi [71] however, and we are aware of no studies on its efficacy under irradiation. Given that permeation barriers may be necessary to reduce tritium permeation to acceptable levels in a power plant-scale fusion reactor [57], demonstrating this fact remains an important challenge and should be a focus of research efforts.

#### 3.4. Outgassing of tritium

A challenge to tritium containment is that it outgasses from surfaces. Generally, pure gaseous tritium has little affinity for typical containment vessel surfaces. However, tritium does have affinity for surface contaminates such as oils. Tritium may accumulate in such surface contaminants. Furthermore, tritium may be mixed with impurities, such as water, which have affinities for surfaces. Besides containment vessel walls, tritium comes in contact with surfaces of active materials within vessels such as adsorbents and catalysts. These have high surface areas, which greatly increase the amount of tritium that may be retained. The processes of surface contamination are reversible, so even after tritium gas is removed from a vessel, an amount of tritium will remain in the gas phase as it outgasses from surfaces.

These effects must be taken into account when designing and operating tritium equipment. Consideration must be given to tritium that will be held on surfaces and its effect on maintenance, operations and disposal.

Maintenance operations on tritium processing equipment pose safety concerns since the tritium confinement barrier is breached, for instance, when replacing a pump. Prior to performing such operations, tritium on surfaces should be recovered to the extent possible. This involves at least evacuating tritium from the gas phase. For systems where significant tritium is expected, additional tritium can be recovered by repeated backfilling and evacuation. Inert gases may be used for backfilling, but in some cases gases which exchange with surfaces materials such as hydrogen or humid gases are needed. In addition, especially for high surface area components, heating is effective. Each evacuation can be monitored to gauge the progress of tritium removal.

These considerations also apply to opening of enclosures (e.g. gloveboxes) to room air. Generally, enclosures operate for long periods of time with dry, inert gases which contain small amounts of tritium. This tritium exchanges into hydrogen-containing substances such as organics or permanently bound water on enclosure surfaces. When this environment is exposed to humid gas such as room air, the reverse occurs as tritium exchanges out of surface materials. Thus, especially for enclosures which have been used with significant tritium over extended periods of time, they should be exposed to room air (or humid gas) prior to breaching the enclosure to reduce the surface contamination and gauge what will happen when the breach occurs.

Room surfaces such a painted walls can also retain some tritium, but generally room conditions are such that this has little consequence.

It is also of note that under certain conditions, even at room temperature, standard process component surfaces (e.g. stainless steel) can cause significant impurities such as methane  $(CH_3T, etc)$  to be formed in tritium gases [72].

#### 3.5. Detritiation of solid materials

Tritium can enter the bulk of most materials in significant quantities either by a physical solution process or a chemical hydride process. In some cases large amounts of tritium can be stored. For example uranium tritide stores tritium in less volume than if the tritium were  $T_2$  liquid. In other cases such as stainless steel, the amount that can be loaded into the material is much less, but it is still at levels which require significant safety controls. Practically speaking, any process component which has been used to contain significant quantities of tritium cannot be decontaminated to a level below the threshold at which it could be removed from regulatory control (sometimes referred to as 'clearance' or 'free release'). But, components which have only had incidental contact with tritium in a tritium facility's room air can generally be freely released.

There are two approaches to decontaminating solid materials. First, in the extreme case of hydrides which readily accept tritium, the bulk tritium can be removed by heating under vacuum. But, a certain amount of tritium remains in the hydride as a 'heel'. While this cannot effectively be removed by vacuum heating, it can be removed by exchanging the tritium through repeated loading with protium and unloading of the resulting HT. Each cycle of loading and unloading reduces the amount of tritium in the bulk by a simple dilution model.

The load/unload process is effective because tritium rapidly diffuses throughout the material. But this is not true for materials such as stainless steel at ambient temperatures. For this class of material, diffusion and solubility are relatively low, but not zero. The material falls in a regime in which tritium slowly diffuses into the material over an extended period. When tritium is removed from the gas phase, tritium will begin diffusing back out of the material. This also occurs over an extended period. Efforts can be made to remove tritium from the surface (e.g. solvents and abrasives), and this will only be briefly effective. But, relatively soon, tritium will diffuse back to the surface and outgassing will resume. For these materials, the strategy is often to first perform an active cleaning, and then allow the material to outgas in a controlled manner (perhaps into a glovebox atmosphere) until outgassing slows to acceptable rates. Thereafter, the material might outgas into a hood for a period of time or be packaged in a container for waste. In some cases, more aggressive decontamination procedures are used with, for instance, heating to drive bulk tritium to the surface.

#### 4. Plasma-facing materials (PMFs)

Inventories of radioactive tritium (T) and dust need to be controlled and minimized. T will be trapped and retained on surfaces (for C and Be) and in the bulk (for W) of PFMs. Erosion of PFMs generates dust that becomes activated by neutrons and retains T in the VV. The strategy at ITER for in-vessel inventory control of T and dust is to remove T and dust to the maximum extent possible on each occasion that the VV is vented to air for maintenance. Before such opening, the divertor will be baked at 350 °C in vacuum to outgas T from the divertor PFMs and from dust on the divertor. This is expected to remove a sufficient amount (>50%) of the T.

Plasma wall interaction (PWI) plays an extremely important role in selection and safety assessment of candidate PFMs in fusion devices since it determines the dust generation and T retention in PFMs. Several review articles have discussed the physics and safety issues (e.g. dust generation and T retention) in candidate PFMs in the past [73–75] and the most critical materials-related issues have been identified as: (1) Lifetime of PFMs, (2) Dust generation from eroded PFMs, and (3) T inventory in PFMs. Carbon/graphite (C), beryllium (Be) and tungsten (W) have been extensively investigated as candidate PFMs. A decision was made by the ITER Organization (IO) to exclude C from divertor PFMs and to start with a Be FW and a full W divertor from the beginning of H/He phase. Although carbon has excellent thermo-mechanical properties for high heat loads during off-normal events, a safety concern that precludes its use is the inventory of T that can be retained in C/T co-deposited layers [74]. Beryllium is not considered to be a candidate PFM for DEMO and future fusion power plants due to its toxicity, high physical sputtering, brittleness under neutron irradiation, low melting temperature [76], and in the case of a plant with water cooling, the potential for Be/steam reactions following an in-vessel coolant leak (see section 4.2.2). W and/or W alloy are considered as candidate PFMs in DEMO and fusion power plants [77].

Transient wall loading by edge localized modes (ELMs) and disruptions must be strongly limited (<3 disruption every 200 discharges) in ITER to obtain the required lifetime its PFMs. The 1000kg safety limit of mobilisable dust in VV appears not to be problem, even for conservative estimates. However, dust at this quantity will cause plasma operation problems. The 1 kg safety limit for mobilisable T in VV should not pose a safety issue until 10000 discharges in ITER [73, 74]. Gas balance and post-mortem analyses from JET ITER-like-wall (ILW) shows a factor of more than 10 reduction of the longterm fuel retention in JET-ILW with respect to JET-C [78, 79]. Recent prediction to ITER with these JET ILW results indicates that mobilisable T should not pose a safety issue until 3000 D-T discharges in ITER [80]. T retention in ITER's VV will be dominated by Be co-deposition and bulk retention in W; but T retention in neutron-irradiated W poses a greater challenge for DEMO and fusion power plants because of the large uncertainty associated with T trapping in this neutron damaged material. The subsections that follow describe the current understanding of dust generation mechanism, safety risks associated with dust, T retention in low Z co-deposit PFMs, and summarizes T retention in FW & Divertor and impact of radiation damage, respectively.

#### 4.1. Dust generation

Interaction of energetic ions with PFMs produces dust in fusion devices, which has implications for safety but also affects plasma performance and operation. For example, accumulation of dust can impede plasma startup, and impurity injection from flaking deposits/co-deposit can disrupt the plasma instantly. Recent review articles have summarized the current understanding of mechanisms for dust production and the impacts of dust on safety and operation [73-75, 81-83]. Blistering and fracturing of deposited layers, generation of reactive species in edge plasmas, arcing, explosive ejection and brittle destruction of surface imperfections, and nucleation of vaporized materials are known dust generation mechanisms in fusion devices. Physical and chemical sputtering via high flux of energetic plasma particles erode PFMs, and most of the eroded material is re-deposited at or close by its origin. For a mixed material environment such as in ITER, eroded/sputtered materials also form mixed material co-deposited layers, which are subject to break-up by mechanical and thermal stresses. Thin films (e.g. boronization) are also applied for the wall conditioning. Dust produced from arcing during the plasma startup or rapidly

varying plasma currents is more likely to be spherical [84]. Steady state erosion does not pose any serious problem for the lifetime of plasma-facing divertor components. Off-normal plasma events such as ELMs, vertical displacement events (VDEs), runaway electrons, and disruptions that deposit large quantities of energy onto materials can melt PFMs and generate dust, and are the primary dust generation mechanisms of concern. In ITER, several GW m<sup>-2</sup> of particle energy is expected to be deposited in hundreds of microseconds on PFMs during ELMs, and hundreds of GW m<sup>-2</sup> in tens of milliseconds during disruptions. This results in vaporization and melting, generating dust by condensation of the vaporized material, pressure-driven ejection of melt layer material, and explosive brittle destruction by heating of gas pockets near the material's surface. Despite extensive research, more attention is still needed to understand the formation and transport mechanism(s) of dust and the assessment of dust on hot surfaces.

#### 4.2. Dust issues

The presence of dust creates two different types of safety issues for fusion plant. The first is that the dust is potentially easy to mobilise and transport. Because it will be radioactive (resulting from both neutron activation of the constituent material and retention of tritium within the solid particles), it is thus a potential addition to the source term during an accident. In addition to the radioactive hazards, it may also be necessary to consider chemical toxicity, in the case of beryllium, to which even a very small exposure can cause lung disease.

The second issue is that dust provides a rather large specific surface area for chemical reactions. This can lead to an undesirable energy release up to and including a dust explosion, which could challenge structures providing a safety function, for example the VV that is the first confinement barrier. In systems which use water as a coolant, there also exists the possibility of hydrogen generation, which is an explosion risk in its own right, but which may also compound the severity of a dust explosion. Such events are obviously capable of degrading confinement boundaries, so it is important to understand and mitigate these hazards. In the following subsections some fusion-relevant materials and their impact on these issues are discussed.

4.2.1. Accidental mobilization and release. A mobilized quantity of tokamak dust constitutes an aerosol, the transport of which is governed by the well-known aerosol mechanics [85]. Understanding the behavior of fusion aerosols thus requires some knowledge of their basic properties, including especially the distribution of particle sizes, and also their shapes and densities.

Efforts to measure these properties of dust sampled from operating tokamaks has been ongoing over the last 15–20 years. Much of the earlier work (see e.g. [86]) was performed on samples collected by vacuuming through filters, which were subsequently analyzed by optical microscopy. More recent collections have used *in situ* collection techniques [87, 88] (e.g. silicon wafers positioned during operation [84, 89–91]) and have been analyzed by scanning electron microscope

(SEM), and in some cases these have found a preponderance of ever-smaller particles. In general though, it can be said that fusion dust particle sizes have a broad distribution (over several orders of magnitude) and a median size on the order of a micron. Recent work as the Axially Symmetric Divertor Experiment—Upgrade (ASDEX-U) has demonstrated by way of SEM and energy-dispersive x-ray spectroscopy (EDX) [92] shows that different dust particle materials have different size distributions and morphologies; tungsten particles, for example, tended to be spherical and had lognormal distributions with a smaller spread than other materials. This is supposed to be a result of different generation mechanisms affecting different materials, e.g. arcing on tungsten walls. It should also be noted that material collected from non-nuclear tokamaks may primarily consist of 'debris' created during hands-on maintenance, the quantity and character of which may not be representative of future, remotely-maintained devices.

Inasmuch as they both involve aerosols, there are some similarities in the analyses of fusion and fission accidents. This was a primary motivation in the adaption of MELCOR for fusion safety analysis [93]; MELCOR possessed a suite of aerosol models from the MAEROS code. While the transport phenomena may be similar, the source of the aerosol is rather different. Unlike fission product aerosols, fusion aerosols are created when already-deposited particles are resuspended. It is precisely this resuspension that is a concern, for example, in a loss of vacuum accident, in which there is air ingress into the VV. Resuspension is a competition between adhesive (van der Waals) and friction (fluid drag and lift) forces on a particle. Its essential feature is that, even for the same particle properties and flow conditions, only a fraction might be resuspended. In the development of complex models [94, 95], this has been ascribed to a distribution of surface roughness features, though these are ultimately correlated empirically to particle size [96]. Such models have been implemented in MELCOR-fusion [97], though it is emphasized that the empirical constants have not been determined from fusionrelevant particles, flows, and surfaces, and so their predictive power is somewhat uncertain. There is additional uncertainty regarding what fraction of in-vessel dust might be mobilizable, given that much of it may be relatively tightly bound in co-deposited layers.

4.2.2. Hydrogen generation in reactions with steam. In any fusion power plant design in which water is used as a coolant, there exists the possibility that steam will oxidize materials, and generate hydrogen in the process. In solid metallic structural materials, oxidation can be self-limiting; an oxidized layer may form, but in many cases this is self-protecting against further oxidation since reactants are separated by this boundary [98]. It is possible that metallic dusts are not protected from oxidation in this way, due to dust particle's high porosity and specific surface areas- on the order of several m<sup>2</sup> g<sup>-1</sup> [86]. Since dust inventories may be on the order of 100s of kg, this is a potentially large source of hydrogen, perhaps sufficient to exceed the lower explosion limit (LEL) (4%) of hydrogen in volumes comparable to a VV or a tokamak

building. Tungsten and beryllium both react exothermically with steam [99] to produce hydrogen. The beryllium-steam reaction has a higher energy release and reaction rate [99] and so is a greater concern, thus much study has been devoted to it in particular, which is discussed in section 5. Some of these studies have been devoted to dust in particular, see [100–102].

4.2.3. Dust explosion risk. A potentially severe example of the chemical reactivity hazards of dust is a dust explosion. A wide variety of materials including metals, plastics, and organics, when finely divided and suspended in sufficient concentrations, can explode. Such events have occasionally plagued industrial, manufacturing, and agricultural facilities over the last several centuries [103] and can result in significant property damage and loss of life. In the case of an in-vessel explosion in a fusion plant, the risk is a blast pressure that challenges the integrity of the first confinement barrier. Of course, such an explosion hazard only exists if there has been a significant leak of air into the vessel, or during maintenance periods when the vessel is deliberately vented with an air atmosphere.

In order to assess the relative hazards and consequences of such events, standards [104, 105] and standard devices [106] have been developed to conduct bench-scale dust explosions. Standard metrics determined in such devices include the explosion indices,  $P_{\text{max}}$  and  $K_{\text{max}}$ , which are the maximum pressure and maximum rate of pressure rise attainable in this standard device for a given dust; the limiting oxygen concentration (LOC); the minimum ignition energy (MIE); and the LEL, the threshold dust concentration that is required for an explosion. These parameters are specific to the configuration in which they are tested, and to the characteristics of the dust tested. Dusts of identical composition, for example, can be expected to have different characteristics if their size or surface area distributions differ. Dusts with diameters over ~63  $\mu$ m are generally not explosive [103].

A series of experiments at the Karlsruhe Institute of Technology (KIT) has investigated the properties of tungsten and carbon dust explosions [107, 108]. Both of these do explode over a range of concentrations given sufficient ignition energy, but the values of  $K_{\text{max}}$  (which is taken as an indication of the explosion severity), 68 and 71 bar-m  $s^{-1}$  for graphite and tungsten respectively, indicate they are within the least severe class of explosions, classified as only 'weakly explosive'. Still, since substantial ignition sources (high currents and high temperatures) will always be present in a fusion environment, such an event cannot be ruled out. A scenario of particular concern is a 'hybrid' explosion, in which hydrogen (generated by chemical reactions as described above or outgassed from cryopumps following an air ingress) first ignites, which provides the ignition source for a subsequent dust explosion. The pressure resulting from such an event is higher than is possible for a dust or hydrogen explosion alone. Hydrogen/ dust explosions have also been studied at KIT [109, 110].

Finally, it should be noted that beryllium, which is more chemically reactive than carbon or tungsten, can be expected to be the primary dust explosion hazard in devices in which it is found. Experiments on beryllium dust explosions are being performed at INL.

#### 4.3. Tritium co-deposition

The accumulation of T in PFMs is a significant materialrelated safety concern. T retained in in-vessel components or in dust will become a source term for possible environmental release during off-normal accident and routine replacement or refurbishment. T will be outgassed and tritiated dust will be removed during routine replacement or refurbishment in the hot cell facility, creating a possible safety risk for operational safety and environmental release.

For ITER, to minimize the potential risk of T release, the FW and Divertor will be baked at 240 °C, and 350 °C, respectively to remove >50% of the retained T. For low Z PFMs (C and Be), the build-up of the T inventory is determined by co-deposition with eroded material. For W, where erosion and T co-deposition are expected to be small, bulk trapping becomes the dominant T retention mechanism. T inventory in Be co-deposit is considered as predominant mechanism of in-vessel T inventory in ITER [73–75]. Recent post-mortem analysis from JET ILW shows that highest retention values were measured from regions with highest deposition [79].

Accurately estimating erosion of PFMs is key for the safety assessment of T retention in co-deposit. High energy (keV range) physical sputtering of Be and W are well described by theory and can be well reproduced by Monte Carlo modeling, while the low energy (below 1 keV) especially for light ions is not well understood especially its threshold effects. There exists differences/discrepancies in the erosion rate of Be among laboratory experiments and tokamak experiments. Critical and insufficiently known parameters are the threshold behavior, the contribution of heavier hydrocarbons and radicals, the influence of chemical reactivity due to simultaneously incident energetic impurity ions and the origin of the flux dependence.

Material migration from regions of high plasma interaction to regions of low plasma interaction is by re-erosion of deposited layers. Co-deposition in these layers can incorporate T in deposited layers located in remote areas, not seen by the plasma for further re-erosion. The role of oxygen in Be-rich co-deposits does not appear to be as large as that of the temperature of the layer and the energy of the incident particles. These low values combined with the very low erosion yields of W for both fuel particles and impurity ions leads to the conclusion that co-deposition with W or WC will not be a critical process for ITER. Progress has been recently made in Be co-deposition by laboratory experiments [83, 111, 112].

Baking at 350 °C baking should be sufficient to remove T from the FW and Be-rich deposited layers formed at low temperature areas, but it may not be effective in removing T from C-rich co-deposited layers and/or PFMs exposed above 350 °C [111]. Thick built-up Be co-deposited layers require a long bake-out (~1 d) to effectively remove T from Be co-deposit [112]. Transient thermal loading is not sufficient to remove T from thick Be co-deposit layer [113]. Accurately predicting T retention in fusion devices is still subject to large uncertainties, as local deposition conditions are difficult to assess: power and particle flux on the complex 3D geometry of PFCs, including gaps, composition of the incident flux in terms of fuel particles

and impurities, and local surface temperatures that depend on poorly characterized thermal properties of the deposited layers.

#### 4.4. Tritium FW and divertor, impact of radiation damage

Use of C as the divertor PFM would reach the ITER limit for T inventory in the VV within a few hundred to a few thousand discharges [83]. The decision by the ITER Organization to eliminate C and start with a Be FW and a full W divertor increases the number of discharges to reach the limit for T to a number that is closer to the expected lifetime of the divertor PFMs in ITER [73, 74].

A high flux of energetic ions at the surface of a PFM creates damage that becomes trapping sites for T. This damage is limited to a region that is a few microns of surface thickness. In addition, 14 MeV neutrons on W will produce damage and T trapping sites throughout the bulk of the W. For ITER, the irradiation damage at the end of the ITER lifetime has been estimated to be 0.6 displacements per atom (dpa) in the divertor and 1 dpa at the FW [114], but the microstructure and its relation to T trapping is still largely unknown, especially the synergistic effect of both ion and neutron damage. Progress has recently been made in this area using high-energy ion beams as a surrogate for producing ion surface damage [115–118] and fission neutrons as a surrogate for producing bulk 14 MeV neutron damage [119–123]. For ITER W, a saturation concentration of neutron-produced traps of 0.1-1% in W and enhancement of T retention were observed by both surrogate high-energy ion and fission neutron studies. Based on these studies and given the pulsed plasma operation of ITER, T retention in W should not pose a significant safety concern.

However, for DEMO and fusion power plants, T inventories for neutron damaged tungsten/tungsten alloys will be an inventory concern because FW and divertor temperatures, plasma burn duration and neutron fluence will increase in comparison to ITER. T will migrate deeper into the W and encounter higher levels of neutron damage sites. In the absence of high-flux 14 MeV neutron sources, such as the proposed international fusion materials irradiation facility (IFMIF), higher temperature exposure results to date come from fission neutron irradiations, which show deep migration and trapping of deuterium in neutron-irradiated W and raise the safety concern of possible large T inventories in neutron-irradiated PFMs for DEMO and fusion power plants. Clearly, this neutron irradiation effect requires more experimental validation especially with a more realistic fusion neutron spectrum and more detailed code simulation before a final conclusion can be drawn.

A further potential issue for plasma-facing tungsten surfaces, particularly at the divertor, is the interaction with helium exhaust from the plasma to form a nanostructure porous layer known as tungsten fuzz [124]. The formation of this layer has been observed but the mechanism is not yet fully understood. Apart from enhancing erosion of the surface, this porous layer could be a site for the trapping of additional tritium inventory.

#### 5. Functional materials

Functional materials are materials in a fusion facility that perform a specific plant process function, for example provide tritium breeding, neutron multiplying, blanket cooling, insulating, diagnostic access to the plasma (windows), etc. In practice, they represent any material that does not serve a confinement or structural purpose. Because this material class includes a very broad spectrum of materials found throughout a fusion facility, some of which are addressed in other sections of this paper, in this section we will briefly discuss only those functional materials that have a significant potential safety impact. In particular, tritium breeding, neutron multiplying, shielding, and diagnostics port viewing. The discussion regarding these materials is based on material attributes that promote inherent safety, and allow higher public and worker safety without the need for engineered safety features (ESFs). The material attributes of interest that will be addressed are: (1) induced short and long-term radioactivity, (2) radio-toxicity, (3) chemical reactivity, (4) chemical toxicity, and (5) tritium inventory (reduced permeation/ease of extraction) in the order of safety importance for that material.

#### 5.1. Tritium breeding materials

At the present, three categories of breeding materials are being studied by the international fusion community: (1) liquid metals, (2) Li ceramics, and (3) molten salts.

5.1.1. Lithium. Liquid lithium was recently proposed for the laser inertial fusion engine (LIFE) lithium self-cooled blanket concept [125]. It has also been proposed in past design studies [126, 127]. Lithium is a very chemically reactive metal. Like all alkaline metals, it reacts in air to burn, reaching measured flame temperatures of 1260 °C [128]. Lithium is the only alkaline metal that reacts with both oxygen and nitrogen after a spill, producing lithium oxide and lithium nitride at the pool surface and lithium aerosols, primarily lithium oxide, at air concentrations of  $15 \text{ g m}^{-3}$ , which is well above the US DOE public Protective Action Criteria 2 (PAC-2) public exposure limit of 1 mg m<sup>-3</sup> for lithium oxide exposure without protective measures [129]. Lithium reacts with water or steam to produce lithium hydroxide at low temperatures and lithium oxide and hydrogen at high temperatures [130, 131]. High temperature spills on concrete can produce vaporization of the water in the concrete, and then reaction with the water vapor to produce hydrogen [128].

However, more positive attributes of lithium are: (1) chemical reaction rates are slower than all other alkaline metals including sodium [132], (2) it does not require a neutron multiplier to achieve tritium breeding ratios (TBRs) greater than unity, (3) except for bred tritium or structural corrosion products in lithium, it does not become radioactive under neutron irradiation, (4) permeation losses of the bred tritium are extremely low because of lithium's high hydrogen specie solubility [133], (5) tailoring of the TBR can

be achieved during operation, and (6) can be easily recycled to reduce waste burial volumes. A plant based on the use of lithium breeder may need to rely heavily on ESFs to maintain safety during accident and normal operating conditions, and limit the inherent safety potential offered by a fusion device. To advance the deployment of this breeder, in the event that other breeding candidates prove not to be viable, the proposed tritium extraction method for lithium needs to be developed beyond the proof-of-principle experiments conducted by [134] to date.

5.1.2. Lead-lithium. A PbLi breeding blanket was proposed for a number of European power plant concepts PPCS-A, PPCS-C and PPCS-D and is currently being considered for a European DEMO [135] and US ARIES power plants ARIES advanced tokamak (ARIES-AT) [2], ARIES spherical torus (ARIES-ST) (see [136, 137]), ARIES compact stellarator (ARIES-CS) [138], and ARIES aggressive and conservative tokamak (ARIES-ACT) [11, 12]. In China, the FDS series of power plant concepts were studied, also based on PbLi breeding blankets [139, 140]. All of these concepts foresee the use of eutectic PbLi breeder material (melting point of 235 °C) and RAFM steel. The simplest concept is a self-cooled configuration (self-cooled lead lithium, SCLL). In order to avoid magnetohydrodynamic (MHD) effects, other options have been proposed that use Helium (HCLL) or water (WCLL) as a coolant for the whole blanket, or in combination with PbLi as a dual-cooled blanket (DCLL). In the DCLL concept, the structural material compatibility with PbLi (~500 °C) is overcome by SiCf/SiC flow channel inserts that maintain the RAFM steel/PbLi temperature at 500 °C, while producing PbLi temperatures of up to 1100 °C within the channel formed by the SiCf/SiC insert [141].

Safety concerns of using PbLi are related to (1) lead chemical toxicity; (2) MHD effects of flowing liquid metal, (3) lead activation products; (4) corrosion control; (5) possible exothermic reactions with steam/water and hydrogen production at elevated temperatures during accidents, and (6) tritium permeation control.

Lead-lithium alloy is considered a toxic material because of the health hazard related to its inhalation and ingestion, mainly in the form of fumes or particulates (dust). This is of safety concern during maintenance and accidental conditions as stated in section 2.2.2 above.

The use of insulating walls greatly reduces the MHD effects on the PbLi flow [142]. The use of SiCf/SiC composite can be viewed as a high-risk high payoff endeavour [143]. The latter is linked to the superior safety characteristics of SiC arising from its low induced radioactivity and after heat, and to the possibility of high performance through high temperature operation. The high risk is associated with the uncertainty about SiCf/SiC behaviour and performance at high temperature and under irradiation. Based on US and France studies SiC is likely to be compatible with PbLi alloy at 800 °C [144] significantly higher than the compatibility of PbLi with RAFM steel (~500 °C).

Corrosive behaviour experiments of RAFM steel EUROFER-97 [145] revealed noticeable corrosion is observed

after 2500 h when the surface layers are not more resistant and do not prevent dissolutions of steel components.

Activation product radionuclides of bismuth, polonium, thallium and mercury represent a potential hazard (see section 2.2.2 above). Bismuth is either an original impurity of lead or a daughter produced from lead during neutron irradiation. Experiments have shown [6] that the release of polonium from PbLi in the liquid state is determined by the vapour pressure of an intermetallic Po–Pb compound which is orders of magnitude lower than that one of polonium. Nevertheless, it is desirable to develop online Bismuth removal to a level of 1 ppm [146] However, so far such technology, without also removing PbLi, has not been demonstrated by an online system. Since the <sup>203</sup>Hg is produced by lead activation, the mercury content does not depend on the impurities added in PbLi alloy.

The maximum temperature which can be reached by chemical reactions is much lower for PbLi alloy than for lithium because the lithium content per volume is roughly 1/6 compared to pure lithium [146]. Another important difference is the liberation of hydrogen in the alloy reaction with water, while in case of lithium most of the hydrogen is bound to the liquid metal as LiH until the temperature level exceeds 1000 °C. Hydrogen liberation is of special concern in the case of an air environment because an explosive mixture of the two gases has to be avoided in any case. Although the self-sustaining reaction of PbLi pool with steam cannot occur because of the formation of lithium oxide layer at the surface, if a PbLi spill results in many small pools, the possible reaction with steam could be more extensive. An experimental study of PbLi/water interaction with DEMO WCLL blanket micro-leaks [147] concluded that due to formation of a protective layer of solid reaction products the reaction even at intermediate water leak rates is quite benign and less violent than the water-sodium reaction for similar conditions. Other experiments [148] have shown that melted alloy pool (up to 700 °C) does not burn in contact with air. Only oxidation of a fraction of lithium occurs.

The eutectic lead–lithium alloy is an attractive breeder material due to its high tritium breeding capability, its relatively large thermal conductivity, and its immunity to irradiation damage [143]. It can lead to tritium self-sufficiency without employing additional neutron multipliers and allows for tritium extraction outside the VV. Moreover, it offers unlimited lifetime of the breeder material due to the possibility to replenish on-line the <sup>6</sup>Li burn-up which implies that the liquid breeder can be even re-used in new power stations. PbLi has also the advantage of being almost inert in air and of having only a relatively mild and controlled reaction with water.

5.1.3. Solid ceramics. A lithium ceramics breeder blanket has been proposed for the European power plant concept PPCS-B (He cooled), and is being considered for a European DEMO (HCPB) [135], Japanese DEMO SlimCS (water cooled) design [149], Chinese, Indian and Korean DEMO concepts. These concepts require a multiplier and enriched Li to achieve acceptable tritium breeding rates [150]. Tritium is bred in either lithium oxide (LiO<sub>2</sub>) or ternary ceramics (lithium aluminate-LiAlO<sub>2</sub>, lithium orthosilicate OSi-Li<sub>4</sub>SiO<sub>4</sub>, lithium metatitanate MTi-Li2TiO3, or lithium metazirconate-Li<sub>2</sub>ZrO<sub>3</sub>) bed, surrounded by or adjacent to a stationary neutron multiplier bed consisting of Be or Ti-beryllides [141]. Ti-beryllides are less chemically reactive than Be (see section 5.2). A low (atmospheric) pressure helium sweep gas with 1% hydrogen is used to purge tritium from the ceramic breeder; and either high pressure (8 MPa) helium or water is employed to cool the FW and blanket internals. Five out of six test blanket modules considered to be tested in ITER are based on the use of ceramics in the form of pebble beds, with pebble diameters of about 1 mm or smaller. The main candidates for tritium breeding ceramics are MTi and OSi [151]. Li<sub>2</sub>O is more reactive with water than other solid breeders. MTi pebble has been recognized as a promising candidate material because of its chemical stability, good tritium release, and low-activation characteristics [152]. However, the mass of Li<sub>2</sub>TiO<sub>3</sub> decreases with time because of Li evaporation in a H<sub>2</sub> atmosphere and Li burnup. To prevent this mass decrease at high temperatures,  $Li_2TiO_3$  with excess  $Li(Li_{2+x}TiO_{3+y})$  has been developed as an advanced tritium breeder [152].

The merits of the various candidate breeding materials are discussed in [141].

Solid breeder blankets offer good compatibility between the breeder, coolant, and structural materials, and thus lessen the problems related to safety, corrosion, and MHD effects. The concern here is the copious amounts of H and He isotopes being produced in these substances [141]. A large tritium inventory in the breeder causes safety concerns and requires the development of tritium permeation barriers. Their major drawbacks are the limits on power densities due to low thermal conductivities of breeder materials and on blanket lifetime caused by radiation damage and burn-up of breeder materials.

The EUROFER97 steel corrosion in contact with lithium ceramics is not an issue because of the chromium rich protective layer that develops as demonstrated by [153]. Thus lower corrosion rates in ceramic breeders were observed than in contact of PbLi.

The key safety issues associated with lithium ceramics are (a) Tritium production, release and trapping characteristics; (b) Thermo-mechanical interactions between the pebbles and the structure including neutron irradiation effects; (c) Limits on allowable power density due to the relatively low thermal conductivity; (d) Limits on blanket lifetime due to irradiation damages [143, 150]; (e) neutron shielding performance in particular with He as coolant; (f) Tritium permeation to the coolant. The latter is an issue of major concern if water is used as coolant because of the difficulty in processing the tritium out of the water [143].

Although the OSi and MTi are stable at room temperatures the  $Li_2TiO_3$  exhibits very little hygroscopic characteristics, while  $Li_4SiO_4$  easily adsorbs water molecules and exhibits rather high hygroscopic characteristics. For this reason before the use of OSi pebbles they must be stored in an inert atmosphere.

Activation products are generated in the Li-ceramic inside the breeder blanket and released into the Helium purge gas of the TES. The potentially hazardous activation products are radionuclides of tritium, <sup>60</sup>Co, <sup>14</sup>C, <sup>10</sup>Be, and aluminium, which are mainly formed by activation of impurities contained in the Li-ceramics. To avoid, or to limit as much as possible, the spreading of these activation products in the TES, dedicated filters need to be considered in the design. As the activation of the ceramic breeder material is lower than that of structural materials, it will not be the dominant contributor to the gamma dose rate which could lead to the limitation of

From a safety perspective, controlling tritium permeation is essential to minimize the accumulation of tritium in certain areas of a fusion power system including the breeding blanket. However, the basic mechanisms of tritium adsorption and absorption at surfaces, diffusion kinetics in irradiated metals and ceramics, and the interaction with microstructural features such voids, helium bubbles, and defect clusters are not fully understood. For example, whereas the tritium retention in unirradiated materials generally decreases with increasing temperature (and the same is also observed in materials irradiated at low temperatures), neutron irradiation at intermediate temperatures where cavities are produced can lead to a pronounced increase in hydrogen isotope retention [155].

maintenance [154].

The concentration of products of radiolysis can reach a few percent during the exposure of OSi blankets to ionizing radiation [156]. Radiolysis of lithium-containing ceramics may lead to changes of micro-particles' surface properties, and as a result, to deterioration of the tritium thermo-extraction parameters, mechanical and thermo-physical properties of ceramics. Moreover, accumulation of radiation defects and products of radiolysis may cause significant changes in the mechanism of radiolysis at high doses and, accordingly, influence tritium release and retention. In addition, a small concentration of impurities strongly affects the OSi radiolysis [156].

5.1.4. Molten salts. FLiBe molten salt is the primary breeding material candidate for the force free helical reactor (FFHR) self-cooled blanket concept because of FLiBe's low MHD pressure drop and low solubility for tritium [157]. Even though this breeding material contains beryllium, its neutron multiplication is not sufficient to achieve TBRs greater than unity without additional beryllium present in the blanket, when the structural material of the blanket is RAFM steel. Proposed blanket concepts that use vanadium alloy (V-4Cr-4Ti) as a structural material, instead of RAFM, do not require additional beryllium to achieve required TBRs.

The key safety issues associated with FLiBe are corrosion control, tritium permeation control, and chemical toxicity. Neutron irradiation of FLiBe simultaneously produces tritium, a desired product, and free fluorine. To prevent the free tritium and free fluorine from combining to produce TF, a corrosive acid, redox agents, such as lithium and/or beryllium, must be continuously added to the FLiBe during operation. Small scale experiments have demonstrated that this can be successfully accomplished [158]. Tritium permeation can be minimized by efficient tritium extraction methods, thus limiting FLiBe tritium concentrations, similar to approaches being proposed for PbLi [159]. When V–4Cr–4Ti is used as a structural material, additional tritium control measures, not yet demonstrated experimentally, must be employed to limit tritium inventory buildup in this vanadium alloy [160]. The US DOE PAC-2 public emergency exposure limit for BeF<sub>2</sub> in air is 0.125 mg m<sup>-3</sup> [129]. During a primary confinement boundary bypass accident, [161] found that the ~30 kg of BeF<sub>2</sub> can be mobilized from the FLiBe spill during the first hour, producing an PAC-2 air concentration in the plume passing the most exposed individual at a 1 km site boundary when assuming Pasquill–Gifford atmospheric stability class F and a 1 m s<sup>-1</sup> wind velocity.

The positive safety attributes of FLiBe are: (1) low induced radioactivity and decay heating that quickly decays to negligible levels within a day after shutdown, (2) low waste disposal rating, qualifying for shallow land burial and (3) can be easily recycle for use in other fusion plant [161]. To advance the development of this breeding option, tritium extraction and redox needs to be demonstrated on a larger scale, such as part of a test blanket module ancillary system.

#### 5.2. Neutron multiplying materials

Beryllium and lead are the primary neutron multiplier material candidates for the fusion power plant. Parasitic capture of fusion neutrons in the blanket structural material might be significant. In order to reach an adequate overall tritium breeding rate it is necessary to add a neutron multiplier material in the blanket. Lead present in the PbLi eutectic and Beryllium pebbles present in the solid blankets play the role of such neutron multipliers. Lead and Beryllium are actually an effective neutron multiplier with high (n, 2n) reaction cross-section at high neutron energy thus enhancing significantly tritium breeding.

Beryllium metal, which is a conventional material for the neutron multiplier, has a drawback of chemical instability at high temperatures. This has led to development of fabrication technology of more stable beryllium intermetallic compounds (beryllides  $Be_{12}Ti$ ) [141, 152, 162].

The key safety issues associated with beryllium are (a) chemical toxicity and carcinogenicity; (b) chemical compatibility between Be and water/air (hydrogen production); (c) generation of highly toxic, radioactivity hazardous and long-lived radionuclides under the neutron irradiation; (d) Tritium inventory and tritium permeation control: and (e) Limits on blanket lifetime due to irradiation damages in beryllium pebbles.

Beryllium is not a health hazard to workers or the public as long as it is contained within the VV. Only if (i) beryllium dust/particles, which may be transported from the breeding blanket into the TES, were released through system leakage, or (ii) during the transport outside the VV of components containing Be dust/particles, or (iii) during an accident, there could be the possibility of worker/public exposure to beryllium. Beryllium is considered a highly toxic material and a carcinogen. When beryllium is inhaled in the form of fine particles (<10  $\mu$ m) it can cause occupational diseases (e.g. pulmonary berylliosis).

There is chemical energy and a hydrogen production hazard coming from the Beryllium pebbles reactions with water/steam

and air [163, 164]. These chemical reactions are of concern under specific accidental conditions when, for instance, the FW is breached and Beryllium pebbles may be exposed to the environment present in the VV. Thus, in the case of a loss of vacuum accident, the Beryllium pebbles may react with air. Moreover, in the case of a breach in the VV secondary cooling water circuit and intrusion of water/steam from the VV into the breeding blanket, the Beryllium pebbles will react with steam, producing heat and hydrogen. These cases involve the exothermal reactions between Beryllium and air [163], or steam [164] at elevated temperatures. The reaction kinetics is approximately parabolic up to 700 °C where a protective layer is formed as a result of the oxidation and the reaction rate decreases with time. At 750 °C and 800 °C a combined reaction kinetics is observed (i.e. parabolic and linear), and above 800-850 °C the reaction rate accelerates as a function of time (linear reaction kinetics) and the formed surface oxide layer does not exhibit a protective nature any longer [163, 164]. The largest chemical energy potential results from the vast amount of beryllium multiplier needed. The exothermic reaction per tonne of Be with water or oxygen generates 40 GJ or 67.4 GJ, respectively [165].

Activation products are generated in the Beryllium multiplier inside the breeder blanket and released into the Helium purge gas of the TES. These products are mainly activated impurities contained in the Beryllium pebbles. Among them are the radionuclides of tritium, strontium, caesium, cobalt, and iron, all of which represent a potential hazard. Moreover, a raw material (an ore) used for production of Beryllium pebbles can contain certain level of Uranium as an impurity and, as a consequence, limited amount of transuranium radionuclides (like Pu, Np) can be produced in the Beryllium pebbles [166, 167]. Unfortunately, present-day materials often do not meet FPP requirements, in particular with respect to impurities [42]. To avoid, or to limit as much as possible, the spreading of these activation products in the TES, dedicated filters need to be considered in the design. Highly toxic <sup>210</sup>Po, Pu, <sup>115</sup>Cd, as well radioactivity hazardous Tritium, <sup>137</sup>Cs, <sup>60</sup>Co, <sup>129</sup>I, <sup>90</sup>Sr, and long-lived <sup>243</sup>Cm, Np, <sup>99</sup>Tc, and <sup>209</sup>Bi as the precursor of to the <sup>210</sup>Po production, are of main concern. Assessment of the overall PPCS-B breeder blanket, at the end of 5 full power years operation, reported the overall <sup>239</sup>Pu production is the range 3.7–9.9kg [167] considering the realistic S-65 Be grade from Brush Wellman Co.

One possible solution to reduce hydrogen production by water–Be reaction [143] would be the use of Be<sub>12</sub>Ti, which has better compatibility with water. The chemical stability of Be<sub>12</sub>Ti in a stream of argon gas containing 1% water vapour was demonstrated in [168]. The generation of hydrogen started at a temperature near 800 °C, and terminated 10 h later. No breakaway reaction was observed. The amount of oxidized Be<sub>12</sub>Ti is far smaller than in similar conditions with beryllium. This was also demonstrated in [168] where three titanium beryllides, exposed to dry air at 1000 °C for 24 h, showed almost no mass gain, while beryllium in the same conditions has large mass gain.

The safety issues of lead are similar to those of the Pb in liquid PbLi alloy, and are considered in section 5.1.

#### 5.3. Diagnostics and RF heating windows material

Some plasma diagnostic and heating systems require access through the VV by way of non-metallic ceramic materials known as windows. These windows, and the assembly that attaches them to the VV, must also perform the same safety function as this primary confinement boundary, which is to confine the radiotoxic, chemically toxic, and chemically reactive materials confined within the VV, in particular tritium and dusts produced by plasma erosion. These windows must not only provide confinement under normal vacuum operating conditions, but also during off-normal VV pressurization events. For ITER, the redundancy of this confinement function is provided either by a window and isolation valves that attach to outside of the VV just behind the windows, or through two independent windows in the same feed though [1]. A second safety function associated these windows is the same as that of the systems of which they are components, which is plasma control.

For ITER, the confinement function must be maintained during pressurization events that can produce up to a 2 bar pressure differential across the window. The design of a window assembly that can withstand this pressure differential, given the fact that windows are often made of brittle materials, requires specific design guidelines and rules. Reference [169] addresses these guidelines and provides recommended standards for testing and evaluating ceramic window products.

Also explained in [169], the window systems for fusion devices fall broadly into two categories:

- Windows which are primarily for plasma diagnostic purposes, some being for use with laser probes (such as Thompson scattering and light detection and ranging (LIDAR) diagnostics) with high laser power throughput,
- Windows for transmission of high-power RF or microwaves to drive the plasma, operating typically, in the bands 30–55 MHz and 140–170 GHz.

Materials for windows are discussed in [170]. For windows in electron cyclotron (EC) heating and current drive (H&CD) systems, the primary candidate in window material is chemical vapor deposition (CVD) diamond material [171, 172]. CVD diamond is favored in the GHz range because of its low dielectric loss at ambient temperature, very high thermal conductivity, reasonable mechanical strength and good radiation resistance. For these windows, an Ag-Cu braze compound is being proposed to join the CVD diamond to actively cooled copper cuffs. Experience demonstrated that the elevated temperature of the brazing process anneals decreases in CVD diamond thermal conductivity for CVD diamond material irradiated to fluence  $\sim 10^{17}$  n cm<sup>-2</sup>, also indicating the possibility of in situ annealing of these windows. As of the present, it appears that a complete window assembly has not been studied yet under neutron irradiation, as compared to fused silica window example in the previous paragraph.

It is clear that a great deal of progress has been made by the materials and design communities regarding these windows. However, these windows assemblies are weak points in the primary confinement boundary. This becomes an even larger concern for DEMO concepts that use high temperature and pressure coolants, such as water or helium. The question for these communities is if more consideration should be given to developing other joining material for these windows that would also allow high temperature window operation, thus providing the potential for *in situ* annealing of these windows, reducing their impact on plasma diagnostic systems needed for plasma control, thereby reducing the need for window replacement. Reference [173] has successfully demonstrated that a silica based compound named SAY (SiO<sub>2</sub>-Al<sub>2</sub>O<sub>3</sub>-Y<sub>2</sub>O<sub>3</sub>) can be used to form pressure-less joints for SiCf/SiC materials, which perform well under neutron irradiated to fluences of  $10^{21}$  n cm<sup>-2</sup> at temperatures up to 820 °C. While it is not known if this material can be used to join silica windows to suitable VV collar materials and produce window assemblies that tolerate high temperatures and pressures, research in this area should be continued for window applications for DEMO.

#### 5.4. Shielding and reflector materials

Neutron and gamma-ray shielding is required to reduce radiation fluxes originating from the plasma as well as other sources, principally materials that have become activated by neutrons. For example, neutron activation of oxygen in water coolant yields <sup>16</sup>N which is a powerful gamma emitter and may lead to a need to shield parts of the cooling circuit to minimize personnel exposure. Apart from reducing dose rates in ORE, the shielding function is necessary to reduce materials damage by neutrons (particularly at the VV), to protect electronic components including those in safety equipment, and to minimize nuclear heating in cryogenically-cooled superconducting coils.

Conventional materials for neutron shielding usually include a mixture of low-Z materials (typically H and/or C) to rapidly moderate the neutrons, and a material to capture the neutrons once at low energy. A mixture of iron and water remains one of the most effective neutron shields, so that water-cooled blankets with steel structure, together with the breeding material, can be designed to have good shielding efficiency. However, He or PbLi coolants yield relatively poor shielding effectiveness, so that more novel shielding materials are of interest.

By employing TiH<sub>2</sub> pebbles mounted in a SiCf/SiC holder as neutron shield, a low nuclear heating rate (<0,1 mW cm<sup>-3</sup>) can be maintained in the superconducting winding [143]. Removable shielding blocks made of tungsten carbide (WC) that can radiate their volumetric heat to the cooler surroundings without active cooling had been proposed for the ARIES-CS design [143]. Other neutron shielding options may utilize boron carbide, silicon carbide, etc. Lead is also a very effective gamma radiation shield, for which the general requirement is a high-Z material.

The amount of beryllium needed for neutron multiplication might be reduced by replacing some of the shield behind the blanket with a graphite reflector. While the TBR remains almost unchanged, there is an added advantage of a relatively low <sup>6</sup>Li enrichment of 40% [174]. The graphite reflector is also used in a pebble-bed form in order to accommodate any possible geometrical changes during a neutron irradiation [175]. SiC coating on the graphite pebble is considered to prohibit the reaction of graphite with steam or air [176]. The thick graphite reflector has another advantage in that it can play a role of a heat sink in the case of a LOCA.

#### 6. Requirements and licensing

Future fusion power plant operations will offer a fundamentally different safety paradigm compared to fission reactors, and it is reasonable to expect that fusion will require a tailored licensing approach very different from fission. At present, no country has official fusion-specific regulatory framework for power plant construction and operation, although the US DOE has safety guidelines for US experimental fusion facilities [178].

The US DOE Fusion Safety Standards [177, 178] provide general safety requirements and guidance for fusion facilities, and were issued in the 1990s in an effort to develop an official regulation on safety for fusion devices. This effort was expanded to include not just US DOE and national laboratory safety professionals, but also members of the fusion community so that all facets of fusion design and operation were represented. Although the US DOE Fusion Safety Standards were intended for experimental fusion facilities, the requirements were structured to be congruent with safety regulations published by the US NRC, so that the US DOE document would remain applicable until the point that fusion energy research transitioned to commercial applications under US NRC licensing and regulation. Nevertheless, the US DOE Standards fail to provide guidance for fusion materials qualification requirements, which will be a key consideration for licensing a future fusion power plant.

Although not licensed as a nuclear facility, the JET tokamak in the United Kingdom has operated using tritium fuel and has produced a complete safety case to authorise this, of the type that would be needed for a nuclear regulator [179]. In preparation for a future deuterium–tritium campaign, a full safety case has been developed to the standard generally required for nuclear licensing [180].

In France, ITER is licensed under the same legislation and regulations as all other nuclear facilities, including fission reactors, laboratories and fuel storage facilities [181]. The ITER facility is a one-of-a-kind ITER experimental facility and not intended to demonstrate energy production in a cost-efficient way. Although important lessons can be learnt about fusion licensing from the experience of ITER, it is important to recognize the differences between ITER and DEMO or future fusion power plants [182]. The experience of licensing ITER construction has enabled potential safety issues for a fusion power plant or DEMO to be identified [183]. These include the removal of decay heat following a loss of cooling, ORE, potential for additional accident scenarios, and the environmental releases of tritium in normal operation and maintenance.

It is clear that the licensing process for a commercial fusion facility will need careful detailed planning and that final requirements are unclear at this early stage. However, initial consultations with a range of experts have provided a working assumption for planning purposes. This has revealed a range of approaches that vary from country to country. Whereas in some, fusion could in principle follow a risk-informed, performance-based approach to technical requirements, rather than the deterministic approach used for the light water reactor (LWR) fission reactor fleet, in others an approach parallel to that applied to fission plant would be expected. However, it is very likely that new legislation would have to be enacted to licence nuclear fusion plant, and an important principle is that regulations should be targeted and proportionate. This should benefit fusion licensing if the potential of its favourable safety and environmental characteristics can be fully realized [184]. In lack of operational experience, it is expected that a strong materials qualification program will be required to develop the performance basis of safety components. In the meantime, it is critical that fusion community continues to develop rules for fusion in a prudent manner with strong technical arguments that will in the future help the regulating authorities make the most informed decision possible.

### 6.1. Requirements for materials specifications and performance arising from the above

Assuming that future commercial fusion power plants are licensed under regulations as currently foreseen, they will have to use materials that are sufficiently 'qualified', this meaning that the material response is well known and uncertainties in the materials properties reduced under a range of expected conditions within a fusion device. Due to the lack of material properties data after irradiation with 14 MeV neutrons, this requirement will have to be met by choosing materials that have been tested in existing irradiation facilities up to similar radiation dose levels and in similar temperature ranges as the ones expected in the fusion chamber. Other important effects, such as the impact of helium and hydrogen retention in the materials will also have to be addressed by additional experiments and R&D. In principle, the qualification requirements will be more stringent for those components that are providing a safety function, such as credited confinement barriers to limit the spread of radioactivity, or components that control radiation exposure. It is reasonable to expect that the material qualification level can be relaxed for those non-safety-related components.

If it can be rigourously demonstrated that a fusion facility presents a lower risk when compared to fission, it is possible that nuclear grade design codes are not essential in the design of components. Non-nuclear grade codes may be acceptable. Beyond safety and licensing concerns, testing and qualification activities are required for investment protection, thus effort is needed by the relevant technology communities to develop coherent qualification strategies for the key components of the next generation fusion facility, assumed to be a fusion plant prototype or DEMO [135]. These strategies must recognize the investment protection needs, reliability requirements, and safety aspects of the components. It is easy to gravitate to the existing fission rules since they exist, are workable and have been adopted by governmental regulatory bodies, but caution must be exercised and a complete understanding of what is required when a code or standard is adopted is needed. In many cases, the fission codes and standards may look acceptable but the operational aspects of using such a code in terms of in-service inspection may be quite difficult to implement in the fusion system. The fusion community needs to develop reasonable safety requirements, demonstrate that rules to be used are prudent, identify exceptions, and justify them. Simple rules may not work for the complicated load conditions in some components like the blankets and magnets. In some cases more sophisticated design by analysis techniques may be needed.

#### 6.2. Materials qualification needs for licensing

Experience gained from fission energy technology is highly relevant for fusion materials development, but there are important differences. Considerable progress has been made in elucidating the basic mechanisms of materials degradation in an irradiation environment by utilizing a variety of irradiation sources (e.g. fission reactors, ion beams, etc) coupled with a robust theory and modelling effort, but there remains a need for an fusion relevant neutron source that can enable accelerated testing of promising materials and subcomponents. The shortcomings of the use of fission and ion beam surrogate irradiations are discussed in [185]. Ultimately, significant progress towards development of materials and structures for the fusion environment requires an intense fusion neutron source that is capable of providing an energy dependent neutron flux effectively equivalent to the FW of a fusion power plant.

In order to close the existing gap, Europe and Japan are jointly engaged in an engineering validation and design activity for such a neutron source, the IFMIF [186]. IFMIF would be the ideal neutron source for fusion material testing. However, in order to have materials qualified in time for a DEMO construction start early after ITER operation, the materials development program may need to be accelerated. Considering that at the present time it is uncertain what materials would ultimately be selected for DEMO (i.e. it is considered likely that several of the current leading candidates for particular components or concepts will require modifications in constituents, heat treatments or fabrication, to be determined by future research), it is premature to perform a materials qualification test program that would be needed for a traditional style of regulatory approval. However, some effort on establishing a framework for identifying the unique aspects of fusion energy systems, compared to fission reactors, and that builds upon experiences obtained in the regulatory approval of ITER, is useful to initiate during the near term. Separate effects and integral testing in a fusion materials irradiation facility, fission reactors, particle accelerators could provide a portfolio of high damage (>10 dpa) performance testing data for advanced fusion materials and the blanket and divertor components; combined with ITER results these data should be used to make the licensing case to qualify DEMO components. To pursue options for the timely development of irradiation facilities for the testing of candidate DEMO materials, an 'Early Neutron Source'

project has been launched within the European fusion programme [135, 185].

It is expected that before transitioning to fusion commercial operations, qualification of components for a commercial power plant in the relevant DEMO environment will be ultimately required to validate the design and to demonstrate safety roles of key components.

#### 6.3. Codes and standards

Fusion designers are starting to include fusion materials into the design codes for nuclear systems [135, 185]. The 2012 Rules for Design and Construction for Mechanical Components of Nuclear Installations (RCC-MRx) gives design rules for sodium fast fission reactors, fission research reactors, and fusion reactors (see www.afcen.org). This code includes design rules for components operating in the creep regime and in irradiated conditions [187]. Poitevin [187] describes the work done on material properties, which includes irradiation testing [187], of EUROFER so this material can be incorporated into the RCC-MRx design code. Like France, most countries have design codes and standards that are used to assure nuclear facility quality and robustness.

Another recent development is the American Society of Mechanical Engineers (ASME) Division 5 publication of an international design code for high temperature fission reactors; the latest version of this design code was published in 2013 [189]. This design code has not yet been incorporated by reference into any regulations on construction of nuclear facilities. The new code addresses several alloys (carbon steel, austenitic stainless steel, incoloy 800 H) and graphite for fission reactor use at temperatures in the ~400 °C-650 °C range and higher temperatures for some of the materials. Presently, many fusion blanket designs operate in the range of 650–750 °C, so these design rules are pertinent for fusion heat transport system design. The Division 5 rules account for creep and high temperature relaxation effects as well as ductile rupture, gross distortion, and fatigue. Like other ASME codes, it does not address radiation damage. The Division 5 rules are a conservative step forward in high temperature design, and these rules may serve as support to the RCC-MRx in fusion high temperature design. The materials addressed in the ASME Division 5 rules are suitable for fission reactors and could be used in fusion for heat transfer applications; however, the materials discussed in the code are not fusion materials. Despite the initial success with fusion materials in the RCC-MRx, more work is needed to develop fusion material properties for design code adoption, and also develop design rules for higher temperatures, such as up to the 800 °C range.

For the past several years, the ASME Division 4 committee has been working on design code development for magnetic and inertial fusion energy devices [190, 191]. This development is proceeding slowly; a roadmap plan has been approved and parties are contributing design guidance they have developed for fusion designs. As of this writing, the focus of the Division 4 effort is on design rules for superconducting magnets and VVs. Consensus data on material properties for fusion alloys should also be a part of the Division 4 code development effort.

#### 7. Conclusions

The potential for excellent safety and environmental performance of fusion power is one of the principal motivations for its development as an energy source. To fully realize this potential the design of DEMO or a fusion power plant has to provide the necessary safety functions and minimize hazardous inventories. The role played by materials in the design is crucial, both because materials failures can lead to accidental releases and radiation exposure, and because design provisions to maintain a safe state depend upon the performance of materials.

Structural and cooling system materials, such as steels and high temperature alloys, can become neutron activated and add to the radioactive inventory of a fusion facility. If these activated materials are in easily mobilizable form (such as ACPs or erosion dust) or are volatilized during accidents they would add to the radiological and toxicological hazards posed to workers and the public. Low activation materials, such as SiC or RAFM, reduce potential accident doses as well as reduce the radioactivity in waste disposal. Further materials development is needed to create LAM alloys that are compatible with selected coolants in a fusion power plant. Fusion may generate large volumes of low level waste and a smaller quantity of intermediate level waste. The amounts of fusion radioactive waste requiring near-surface burial may be reduced by recycling slightly activated components within the nuclear community. More research is needed on methods of separating elements and impurities from metals to facilitate this recycling.

The safety characteristics and at risk inventories of tritium in a fusion facility have been discussed in section 3. The primary nuclear hazards in a tritium facility are exposure of workers and/or the public to tritium. The key to controlling these hazards is inventory minimization, segregation and confinement of tritium. Recent technology developments in the areas of continuously regenerating cryopumps, permeation barriers, and detritiation of solid materials, are key for minimization of inventories and optimizing the safety and environmental characteristics of future fusion facilities.

Transient wall loading due to plasma behaviours (e.g. ELMs and disruptions) must be limited to obtain the required lifetime of PFMs. Accumulated erosion dust in vessel may cause plasma operation problems and the quantity must be limited to avoid potential safety issues. The in-vessel tritium inventory must also be limited and controlled by implementing techniques for the measurement and removal of both dust and tritium. Tritium retention in neutron-irradiated W poses a challenge for DEMO and fusion power plants because of the uncertainty associated with T trapping in this neutron damaged material.

The leading candidates for tritium breeding materials appear to be PbLi and ceramics such as Li2TiO3. A crucial system needed for the safe operation of a PbLi breeder system is an online bismuth extraction system that does not extract other key constituents from this liquid metal eutectic. This system is needed to minimize the production of <sup>210</sup>Po. For ceramic breeder options (e.g. Li<sub>2</sub>TiO<sub>3</sub>), the basic mechanisms of tritium adsorption and absorption at surfaces, diffusion kinetics, etc are poorly understood, especially under DEMO irradiation conditions. Materials research in this area is needed. Promising options exists for the materials of both diagnostic and RF heating windows. However, because these windows are part of the vacuum boundary they help to provide the safety function of radioactive material confinement. The joining of these window materials to the collars that attach them to the VV is the weak point regarding this safety function. Additional materials research is needed for these window joints.

Although the details of the regulatory framework that will be applied to future fusion power plants has not yet been developed, it is expected that fusion devices will have to use materials that are sufficiently 'qualified', this meaning that the material response is well known and uncertainties in the materials properties reduced under a range of expected conditions. Given the lower risk of a fusion facility when compared to fission, it is not clear that nuclear grade design codes must be used in design of components. Qualification strategies must recognize the investment protection needs, reliability requirements, and safety aspects of the components.

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