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Nuclear safety issues for fusion power plants

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ABSTRACT

Developing a robust safety case is a key step in the development of a fusion power reactor for electricity generation. Plans for fusion power reactors are already underway and before nuclear facilities are licensed, they must demonstrate they satisfy several safety objectives involving keeping workers and the public safe and limiting any environmental impact. In this paper the key safety issues relating to fusion power are explored, including the current approach to fusion safety and methods of accident identification. The paper draws on major studies on the safety of fusion power plant concepts and the current work being undertaken for the ITER project. As well as discussing the key safety issues and potential accident scenarios, the paper identifies gaps in current knowledge together with areas for future work, including the establishment of internationally recognised safety standards for fusion power stations.

1. Introduction

Fusion power plant concepts have been under development since the 1950s; however, until recent events in the US [1] there has been no national or international regulatory framework for fusion power plants. In spite of this, various safety concepts have been developed, concurrent with plant design, that have allowed multiple approaches to be considered to determine the most promising route.

The aim of this paper is to provide an analysis of some of the key approaches that are likely to be part of a safety case for a fusion reactor for electricity generation. Key safety gaps are highlighted together with analysis of the applicability of the current safety objectives defined (see Section 3). Where the current objectives are questionable, recommendations are given on how best to resolve the issues. A literature review was performed to establish the current progress on fusion safety and the critical areas that need to be focussed on.

Various European studies were reviewed, including the Safety and Environmental Assessment of Fusion Power (SEAFP) [2]– [4] and the Safety and Environmental Assessment of Fusion Power – Long Term Programme (SEAL) [3,5]. The Power Plant Conceptual Study (PPCS) was used as a basis for fusion reactor designs, as well as identifying key safety factors and blanket types [6,7]. The safety analysis performed in the licensing of ITER (RPrS) was also used as a major reference point, as this currently provides the most comprehensive view of a fusion safety concept in the world [8].

2. Risk analysis

Defence in depth is the basic nuclear safety principle used in fission reactor design [9, 10]. This approach utilises multiple levels of defence (e.g. confinement barriers/protection systems), so that if one system fails, another will be in place to ensure the safety consequences are limited. This concept of defence in depth can be applied to fusion reactor design to deliver high levels of nuclear safety, nuclear security, and the protection of the environment. In line with good safety practice, the number and extent of barriers required will depend upon both the frequency of the initiating event and its consequences. Whilst the role of these barriers is to prevent the release of radioactive material, there are accident scenarios where the integrity of these barriers will be challenged, hence the need for multiple independent barriers.

This safety approach requires knowledge of the probabilities of the initiating events, the probabilities of failure of the various barriers, and the consequences of failure. For example, events that have a significantly likely probability of occurrence should have minor or no radiological consequences, whilst events that have the potential to result in significant radiological consequences to the public should have a very low probability of occurrence [11]. This is illustrated in Fig. 1.

The figure depicts the relationship between acceptable risk and not acceptable risk in relation to expected dose to the public and probability of occurrence (risk is defined as probability of occurrence x consequence of failure). An accident can be plotted on the graph (see red circle) with its expected dose and probability and it can be determined if the risk is acceptable, depending on which zone it lies in.

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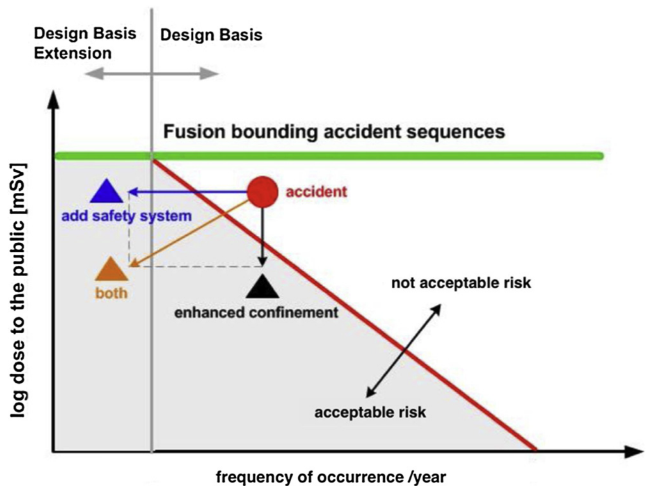

Fig. 1. Risk approach to safety. Figure taken from [11]. Original from [12] (Edited).

The red line on the graph marks the boundary between these zones. As shown, as the probability of occurrence increases, the dose to the public must significantly decrease in order to remain in the acceptable risk zone. Note, both the dose and probability are on a logarithmic scale.

When constructing a fusion safety case, a deterministic approach, complemented when necessary by a probabilistic approach, can be used to identify additional accident sequences to be further considered. If an accident scenario lies outside the acceptable risk zone, there are a number of steps that the designer can take. If the radiological release cannot be reduced, then either an additional containment structure must be provided or another mitigation tactic to reduce the consequence must be employed. Alternatively, additional protection and safety systems can be provided to reduce the probability of the release occurring. Judging what is acceptable is not easy and is often based on public acceptance of a risk of harm when compared to the benefit they gain from the activity that is producing the risk. For fission reactors operating in the UK, the law requires risks to be reduced to 'as low as reasonably practicable' (ALARP) and hence there is no simple demarcation between what is acceptable and what is not. Note the ALARP principle is a UK concept that is broadly accepted, which is similar to the principle 'as low as reasonably achievable' (ALARA) used in relation to ionising radiation exposure by other bodies nationally and internationally [13]. The HSE document "The Tolerability of Risk from Nuclear Power Stations" (TOR) [14] addresses public perceptions and gives guidance on how these perceptions can be translated into risk. Although originally produced for fission reactors, TOR is equally applicable to fusion power reactors and any safety case for a fusion power station would need to demonstrate that the risk was either broadly acceptable (10^{-6} chance of death/year of operation to a member of the public) or within the ALARP region (less than 10^{-4} but greater than 10^{-6} chance of death/year). See Fig. 2 (below).

3. Nuclear safety objectives

The top-level safety objectives for the DEMO facility are based on international guidelines and similar to those adopted by any nuclear facility [15]. These are:

- 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 minimised to be as low as reasonably achievable;
- to ensure that the likelihood of accidents is minimised 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 minimise radioactive waste hazards and volumes and ensure that they are as low as reasonably achievable.

These are worthy high-level goals and are consistent with those adopted for the PPCS; hence, similar objectives should be appropriate for a fusion reactor for electricity generation. In the case of the second principle, however, the overarching priority should be the limitation of risk as set out in the TOR document. The final principle provides a challenge to designers to choose appropriate materials that can minimise neutron induced activation.

4. Safety related inventories

To meet the above principles the design of fusion power plants must take account of the hazard potential that result from a number of features, some unique to fusion power. The inherent features of a fusion power plant that give rise to these hazard potentials are the energy and radioactive materials inventories.

4.1. Energy inventories

Energy inventories play a crucial role in the safety analysis of a fusion reactor. Stored energies have the potential to break confinement barriers and mobilise radioactive elements, releasing them into the environment.

4.1.1. In-vessel fuel energy

The SEAFP and SEAL studies [2,3] identified the various energy sources present in a commercial fusion power plant concept. The studies produced conservative estimates of the significant energy sources and showed that the in-vessel fuel inventory was not a primary safety concern. This finding was based upon the fact that in the event of a plant malfunction or accident, the fusion process would be terminated by shutting off the fuel supply to the plasma. The estimated maximum energy that could be released from the residual fuel in the fusion chamber was estimated to be some 6.5 GJ (equivalent to the energy released when a barrel of oil combusts). This is not sufficient to challenge the integrity of the vacuum vessel. Note this value does not take into account the additional energy from potential combustion of adsorbed hydrogen on PFC surfaces. Further work is needed to accurately determine these additional source terms and evaluate their energy release. The studies also found that the plasma thermal energy is not a primary safety concern. Its stored energy was estimated to be only 1–2 GJ.

4.1.2. Magnetic energy

The magnetic energy inventory in a fusion reactor is expected to be relatively large, with toroidal and poloidal coils having energies up to 180 GJ and 50 GJ, respectively. Failure of the magnet systems could result in the discharge of this energy into the wall of the first confinement barrier (the vacuum vessel) or structural components of the containment system. It is worth noting that multiple penetrations will be present in the vacuum vessel wall (for auxiliary systems such as heating and diagnostics) and these are the weakest sections of the wall. If the energy from the magnets is discharged into a small area of the vacuum vessel wall (or its penetrations) it can result in melting of the steel and the initiation of a loss of vacuum accident (LOVA) [2]. Coil quench occurs when part of the superconducting coil suddenly enters the resistive state, as a result of excursions over limits of temperature, magnetic field, and current density [16]. In this situation, the magnetic energy from the coil must be removed as soon as possible, in order to

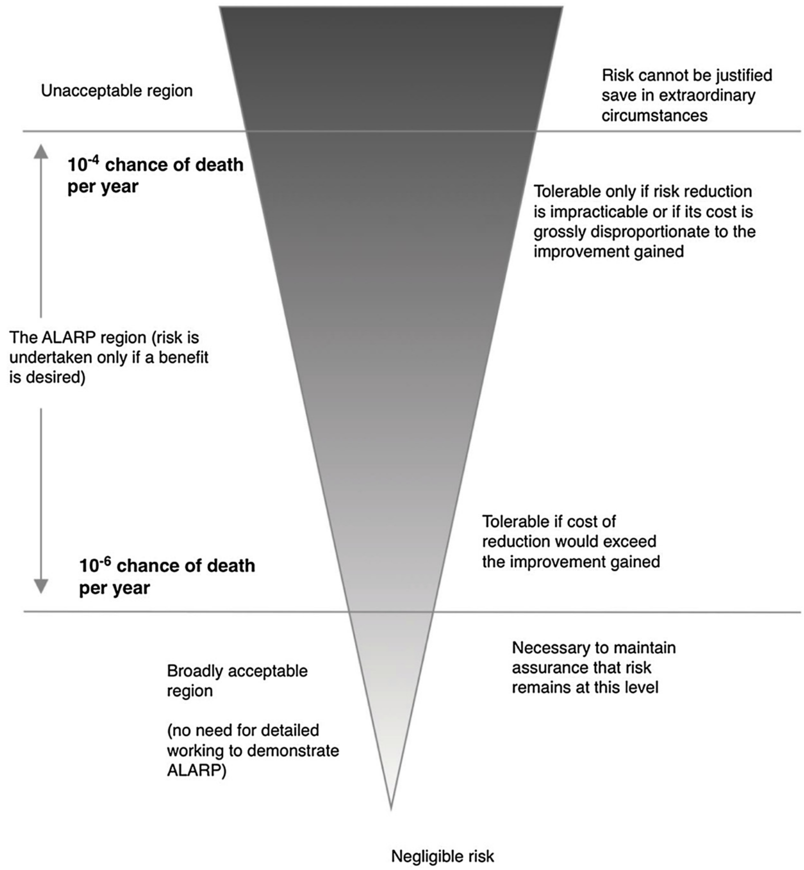

Fig. 2. Levels of risk and ALARP. Original from [14] (Edited).

prevent arcing and damage to adjacent structures. This accident scenario is recognised in the ITER safety case and the ITER design has included an accident mitigation system.

This potential hazard is common to any magnetic confinement fusion plant and hence all fusion reactor designs will need to have a similar mitigation system to that proposed for ITER. The ITER system includes real-time monitoring, plasma control, and stabilisation of magneto-hydrodynamic (MHD) modes [17]. As an example, in the case of coil quench, ITER's superconducting magnets are fitted with a fast discharge system for quench protection. This system dumps the energy safely through the use of energy dump resistors. During a quench, the flow of current is interrupted and dumped into Fast Discharge Units (FDUs). These are energy dump resistors that discharge the magnets and dissipate the stored magnetic energy as heat [16]. In ITER, the toroidal field FDUs are classified as safety important components (SICs) and perform the safety function of protecting the vacuum vessel [16].

As the magnetic energy inventories in a demonstration fusion reactor (DEMO), or any other fusion power plant, are expected to be larger than in ITER, the coil quench protection system will be an essential safety design feature, and the substantiation of its performance a major component of the plant's design and operational safety case.

4.1.3. Plasma facing component stored heat

The heat generated from the radioactive decay of activated plasma facing components (PFCs) must be taken into account because of its potential to magnify consequences of accidents. The major structural material expected to be used in fusion plants is the reduced activation martensitic steel Eurofer [18], due to its expected performance under fusion conditions. In order to reduce the erosion rates of the first wall, the current approach is to have tungsten tiles form a protective layer (or armour) on the PFCs. Tungsten (W) is also expected to be the main structural material used in the divertor, an area of the plant that will be exposed to extremely high heat fluxes (up to 20 MW/m^2 [19]) and intense radiation damage. The incoming neutrons not only cause cascades of damage in the PFCs, but also result in activation and transmutation of the structural materials.

Activated tungsten decays via β -decay to form small amounts of rhenium (Re) and osmium (Os) (expected concentrations in tungsten armour after 5 years in a fusion reactor are 3.8% and 1.4%, respectively) [20,21]. Tungsten can also transmutate to form trace amounts of tantalum (Ta) (expected concentration after 5 years 0.8%) [21]. The decay heat density of tungsten is expected to be modest, with a value for the first 12 h after shutdown between 0.2 and 0.3 kW/kg [20]. The structural material of the blankets (typically Eurofer) is expected to be a more significant source of decay heat compared with the W armour, as is, possibly, the breeding materials themselves. The Eurofer first wall is expected to have a decay heat of around 0.1 kW/kg [20], albeit with a much higher inventory than the W armour. The impact of this decay heat will depend upon the accident scenarios that are identified in the design safety case. Further work is needed to ensure that decay heat effects can be accurately modelled in accident scenarios that have the potential to thermally threaten the integrity of the vacuum vessel.

4.2. Radioactive materials inventories

Fusion is a nuclear process that uses deuterium and tritium as fuel and results in the production of high-energy (14.1 MeV) neutrons that can activate non-radioactive materials. Tritium is a major radioactive source term in a fusion power plant and can be found in the vacuum vessel, coolant, breeding blankets, and tritium plant. Understanding and quantifying the potential radioactive source terms from neutron-activated materials is another crucial safety analysis requirement. The amount of radioactive material present determines the hazard potential of an accident, not only to workers but also to the public if radioactive material is released into the atmosphere. The other major source terms identified are activated dust (W or Be) and activated corrosion products (ACPs). Dust refers to the products formed due to the erosion of plasma facing components, whilst ACPs are defined as the products of corrosion within the water cooling loops. Depending on the breeding blanket type used in a fusion power plant, there may be additional source terms that are not mentioned here. As things stand, there are four design options with different levels of design and technology readiness being considered for DEMO using helium, water, or lead-lithium (PbLi) as a potential coolant [22]. Until a final decision is made on the breeding blanket type, it remains difficult to identify the radioactive source terms present in a fusion plant blanket architecture.

4.2.1. Tritium

Whilst there is only a few grams of tritium fuel in the plasma at any one time, the tritium consumption in the vacuum vessel (VV) amounts to ~125 kg per year (in a standard $1GW_e$ fusion reactor) and can lead to a build-up of tritium in the VV and fuel and coolant system over time. The use of reduced activation martensitic steel in the vacuum vessel should result in a relatively low level of tritium absorption, due to its high diffusion coefficients under fusion conditions. However, the majority of tritium build up will be due to absorption in the W/Be armour and co-deposited tritium in dust [23]. In the PPCS and SEAFP studies [2,6], the maximum tritium inventory that is able to be mobilised in the event of an accident is assumed to be 1 kg, which results in a releasable inventory of 3.57E + 17 Bq.

4.2.2. Activated dust and corrosion products

Quantifying the inventories of radioactive dust and mobile corrosion products in a fusion reactor remains problematic. Due to the lack of information over the wide range of phenomena taking place during dust and corrosion product production, mainly the plasma-material interactions and the physical and chemical processes involved, the inventory at any one time is based on approximate assumptions and does not take into account the engineering parameters of different plant designs [24]. Nevertheless, an attempt has been made in [25] to identify the potential source terms that can be produced in a fusion power plant along with their activity.

The maximum expected inventory of dust in DEMO has been estimated at 1000 kg [23,26]. Whilst it remains unclear which isotopes will make up this 1000 kg at any time, a conservative assumption that the entire dust inventory is composed of W-185 (this isotope has the highest activity and decays on the timescale of days rather than minutes), would suggest that the inventory of dust available for mobilisation would be in the region of 3.7E + 16 Bq. Whilst this inventory is still lower than the releasable inventory of tritium (see 4.2.1), it is still a significant amount and the potential production of activated dust and corrosion products must be taken into consideration at the design stage as this could influence component material selection. Further work will need to be carried out to accurately estimate the composition of the activated dust and activated corrosion products to determine an accurate source term that can be used in fault analysis. This can then be used to fully evaluate the consequences of radioactive dust dispersion (as a result of fusion accidents) on the public and the environment.

4.2.3. Chemi-toxic dust

Identifying and quantifying the chemi-toxic materials in a fusion reactor that can be released in the event of an accident is also problematic, mainly due to the lack of final designs and final choice of materials used. In ITER, the beryllium dust from plasma facing components can form chemically toxic beryllium oxide. In a fusion power plant this is expected to be less of an issue as tungsten is the currently preferred candidate for plasma facing components rather than beryllium. Tungsten can, however, form tungsten trioxide (WO_3) under certain conditions. The effects of a release of WO_3 need to be better understood, particularly in the case of a loss of coolant accident (see 5.2.1). Until final designs are completed the use of beryllium cannot be ruled out and, given that a large quantity of beryllium may be used in the breeder blankets in some of the design concepts, an understanding of the effects of the release of beryllium dust is necessary. Further work is required to gain a better understanding of the potential chemi-toxic source terms and their expected compositions in a fusion reactor. Further work is also required to better understand the consequences for the public and the environment of the release of these materials in an accident scenario.

5. Key nuclear safety issues

Nuclear safety can be regarded as all those activities that are necessary to protect workers and the public from a release of radioactivity from a nuclear installation under both normal and accident conditions. Nuclear safety therefore requires a detailed knowledge not only of the radioactive inventory within the facility at any point in time, but also of how the facility will handle routine releases, as well as how it will behave under accident conditions. A list of postulated events that could cause a nuclear safety concern in a fusion reactor has been produced utilising the methods of Failure Mode and Effects Analysis (FMEA) and Master Logic Diagrams (MLD) in [27]. It is claimed that the use of both of these methods ensures a comprehensive list of postulated events.

A selection of design issues and accident scenarios is presented here with a focus on nuclear safety concerns, along with comments on the status of current understanding and the actions needed to resolve them. Whilst this selection is not exhaustive, the accident scenarios detailed below were chosen because they are deemed to be either a primary safety concern or, because there is currently insufficient information available to judge their significance, they have the potential to become a safety concern.

5.1. Thermal inertia

During normal operation the walls of the vacuum vessel and the breeder blankets store energy. Heat is removed to maintain a steady state temperature profile via the coolant. There is therefore always the potential for a power coolant mismatch should there be an unplanned loss of cooling. If such an event were to occur, there are two possible outcomes. The first is that the loss of cooling protection system fails to terminate the fusion process. The second is that the loss of cooling protection system successfully terminates the fusion process. The consequences of the former are discussed later. In the case of the latter, the loss of cooling capability, even when the fusion process is terminated, will result in a transient change in the temperature of the wall of the vacuum vessel and the breeder blankets. The safety analysis will need to demonstrate that the temperature transients do not challenge the integrity of the vacuum vessel (or its penetrations), its supporting structure, the breeder blankets, or other key safety related components. The extent and impact of the thermal inertia stored in these key components will need to be taken into consideration in the detailed design of the plant.

5.2. Decay heat removal

In a fusion plant, decay heat is not associated with the fuel, as is the case in a fission reactor, it is associated with the tritium breeding blankets, tritium that has migrated into the structural components, and the activated materials in the plasma facing components (PFCs). The impact of decay heat removal in the breeder blankets will be discussed later. In relation to the PFCs and other structural components that have become impregnated with tritium, the impact of decay heating arising from the activation of the Eurofer steel, the breeding material, the tungsten, or from tritium will depend upon the activation levels (sustained power levels) and the accident scenario. Decay heat in this context is the heat that is produced in the activated or impregnated materials after shutdown of the fusion process. As shown above in Section 4.1.3, the decay heat density from activated tungsten is around 300 W/kg. The decay heat density from tritium is similar at 325 W/kg [28]. However, due to the low levels of tritium present, the overall decay heat from tritium is expected to be low. Whilst the decay heat density from Eurofer steel appears low at 100 W/kg, the large inventories expected means that the overall decay heat from Eurofer will be significant. The impact of decay heating on the course of accident scenarios, especially in relation to the release of radioactive materials, needs further investigation. The following sections on loss of coolant accidents look at the impact of decay heating.

5.2.1. Loss of coolant to breeder blanket and divertor

The breeder blankets contain structural materials (Eurofer steel) as well as breeder materials (e.g. Li_4SiO_4) that contribute to the overall decay heat. From [27], the primary safety concern with this decay heat is a potential loss of coolant accident (LOCA). Coolant in this sense refers to the fluid (water, helium, liquid metal etc.) that is used to cool the breeder blankets or divertor during normal operation. The PPCS study [6] investigated a bounding LOCA accident, resulting in a total loss of cooling from all loops in the plant, with added assumptions of no active cooling, no active safety system operation, and no intervention for a prolonged period. Temperature transients in the blanket structures were then obtained for a period of 100 days after the accident, with contributions due to thermal inertia in the structure, decay heat, and tungsten activation and subsequent decay heat. Fig. 3 shows the poloidal temperature profile in PPCS Model A 10 days after the hypothetical accident.

The PPCS analysis shows that the tungsten first wall is expected to reach a maximum temperature of ~1200 °C 10 days after the postulated LOCA. This value is significantly lower than the melting point of tungsten, inferring the component should not fail at this temperature. The Eurofer steel has a melting point of 1325–1530 °C and should also not melt at this temperature. However, it is not sufficient to look solely at melting temperature as an indicator of safety: there are other factors that have to be taken into account when substantiating the adequacy of the design in the safety case. One such factor is the formation of tungsten trioxide (WO_3). Analysis by the Materials Assessment Group (as part of the EU Fusion Roadmap process) found that in the event of an air ingress into the vacuum vessel (probable due to failure at penetration at this temperature), significant quantities of highly volatile WO_3 could form at a rate of 10–100 kg/h for a surface area of 1000 m^2.

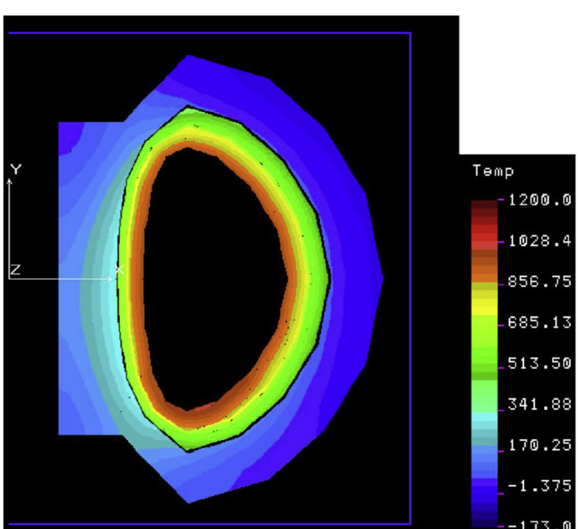

Fig. 3. PPCS Model A poloidal temperature profile 10 days after a total LOCA occurs [6]. Note, the temperature scale is in degrees Celsius and Y denotes the vertical direction.

It is clear therefore, that if a LOCA challenged the integrity of the vacuum vessel and caused deterioration of confinement barriers, a fraction of this radioactive WO_3 could escape and disperse into the environment [29–31]. Further work on this is needed to evaluate the likelihood of this event, the amount of WO_3 that could be released, its radioactivity source term, and the associated impact on people and the environment resulting from exposure to WO_3. It would be expected that the safety case would be based upon worst-case weather conditions in order to calculate the expected doses to the most-exposed individual at the site boundary. If the level of risk to the public was too great, the design would need to include the provision of an emergency cooling system to remove the decay heat and limit the PFC temperatures to reduce the production of WO_3. Further work is needed on the range of potential LOCA accident scenarios to examine not only WO_3 releases, but also the potential for hydrogen explosions due to air ingress into the vessel (see 5.4).

5.2.2. Loss of coolant to vacuum vessel

An in-vacuum vessel loss of coolant accident (in-VV LOCA) has been identified as one of the key safety concerns for a fusion reactor. As the accident sequence in a fusion reactor for electricity generation is expected to be similar to that used in the ITER safety analysis (due to the similarities in expected initial reactor designs and the final design for ITER), the analysis performed for ITER has been used here as a basis to investigate the impact of a loss of coolant to the vacuum vessel and provide an estimate for the radiological consequences.

The key steps and safety responses to an in-VV LOCA scenario are detailed in the ITER RPrS [8]. Initially, a coolant pipe rupture causes the LOCA. This results in coolant ingress into the VV, which in turn causes a plasma disruption: terminating the plasma with a rapid release of thermal energy and potentially resulting in electromagnetic loading on the VV and its supporting structural components. These loads would need to be substantiated to give confidence that the integrity of the primary confinement barrier (the VV) is not significantly challenged. In order to be explicit, in ITER the primary confinement barrier is defined as the VV and any extensions (i.e. any system that enters the VV or has a barrier that may fail such as first wall/blanket cooling loops).

The hot water entering the VV undergoes rapid evaporation, producing steam which pressurises the vessel. To reduce the potential to over-pressurise the VV, drain and suppression tanks, connected to the VV via rupture discs, are used to enable the steam in the VV to be drained, and the steam to be condensed. This is known as the vacuum vessel pressure suppression system (VVPSS). However, these actions result in a significant inventory of radioactive material (maximum estimates are almost 1 kg of tritium and hundreds of kilograms of dust) being transferred to the drain and suppression tanks.

It is expected that the mobilised radioactive inventory of tritium, activated corrosion products, and dust will be initially trapped in both the drain and suppression tanks. On the basis of assumptions used in the analysis, it is suggested that the pressure increase in either the drain or suppression tanks will be such that pressures will be maintained below atmospheric pressure (the pressure in the VVPSS is maintained at the level of about 4 kPa to effectively depressurise the VV). The implications resulting from the removal of contaminated liquors from the tanks will need to be assessed, especially in relation to the need for shielding and radioactive waste treatment, which could influence the design of any commercial power plant.

From the analysis in the ITER RPrS, the mobilised radioactive inventory is not released from the drain or VVPSS tanks in the adjacent rooms since pressure remains below room pressure. Given that there are no workers present in the VVPSS tank room or the drain tank room during plasma operation, and that the return to safe state does not require the presence of workers in these rooms, there are no significant radiological consequences for workers [8]. Following the event workers will be exposed to ionising radiation as part of the clean-up and plant recovery activities. However, during these activities worker exposure will be controlled by normal radiation protection procedures, which in the case of ITER will limit worker doses to less than 10 mSv/year [8].

Given that there is no failure of primary confinement barrier (the radioactive inventory cannot escape the VV or travel further than the cooling pipework which, as defined earlier, is part of the primary confinement), and hence there are no leaks into adjacent rooms and no uncontrolled leaks into the environment, the only potential environmental releases are controlled releases via the suppression tank detritiation system (ST-VS). The calculated radiation doses for most-exposed persons arising from the radioactive release associated with this accident are 9.7E-05 mSv at 200 m and 6.6E-05 mSv at 2.5 km [8]. Such exposures are very low when compared with the 1 mSv limit for members of the public and are orders of magnitude below the evacuation limit (50 mSv).

Whilst the 1 mSv dose limit for the public is generally associated with routine releases during normal operation, comparing this value with the dose predicted for members of the public arising from accident scenarios is useful as it puts the postulated consequences into perspective. As the predicted inventories of the DEMO fusion reactor and ITER are expected to be similar, it seems reasonable to assume that even with the added complexity of an advanced reactor (longer running times, presence of breeding blankets etc.), the radiological dose from an in-VV LOCA would not reach levels where an evacuation may be necessary.

There is another accident scenario in which the VVPSS fails to activate (e.g. a mechanical fault occurs in which the rupture discs fail to burst at the specified pressure). The consequences of this could lead to a higher release. This accident needs to be investigated to demonstrate that either 1) the VV will not reach overpressure in the absence of this safety system, or 2) the radiological release due to overpressure has no significant impact on workers or the surrounding public. Until one of these points is met, it remains unclear if an in-VV LOCA is a primary safety issue, or whether the response satisfies the safety objectives outlined.

5.2.3. Loss of cooling during transfer of blanket sectors

The role of the breeding blanket in a fusion reactor is to absorb high-energy neutrons produced in the plasma, extracting heat as well as producing tritium to be used as fuel. Due to this intense neutron bombardment and the activation of materials in the first wall, the blanket sectors will need to be removed and replaced at various points throughout the lifetime of the reactor. The current conceptual design for DEMO suggests that specially designed ports in the roof of the vacuum vessel will be used for the remote removal of the breeder sectors (or half-sectors). Given the levels of radioactivity of these reactor components (dose rates of DEMO half-sectors during maintenance have been estimated at around 3 kGy/h [32]), and given that the lethal dose is 5 Gy, changing of the sectors will have to be performed using robotic handling (RH), to ensure worker exposure to ionising radiation is kept as low as reasonably practicable.

Due to their large size, it is expected that the decay heat of each blanket sector will be significant (around 4.55 MW per sector just after shutdown [33]) and therefore will require active cooling during their transfer from the vacuum vessel to the hot cells. Ref [33] investigated the decay heat of reactor components following shutdown on the former Japanese SlimCS DEMO reactor project. This analysis showed that to ensure the decay heat of the blanket had reduced to acceptable levels ($<$ 0.5 MW per sector), it was necessary to wait at least one month after shutdown before transfer of any sectors is carried out. The availability of a power plant is a hugely significant factor and whilst this paper focusses on safety, it seems likely that new solutions will need to be found for tritium breeding in order to reduce this outage time significantly, if fusion is to be economically competitive.

Assuming the current conceptual design for DEMO requires breeder blanket sectors to be removed, it would appear that some form of active heat removal will be necessary during the blanket transfer process. Analysis in [31,33,34] suggests that without this the temperature of the blanket could reach ~1000 °C after around 40 days – a figure considered too high for a component in a zone outside the primary confinement barrier. Designing a safety critical active cooling system for the breeder sectors to enable removal transport to the tritium treatment plant will be challenging. The complexity of such a system will inevitably give rise to safety challenges associated with loss of cooling as shown above.

The safety analysis will require an evaluation of the potential causes of failure of the transport-specific cooling system along with their consequences. However, as this will be a new concept, there will be little if any component failure rate data; as such, a reliable probability of failure analysis will be difficult to obtain. When studying the potential consequences of this loss of transport-specific cooling, a number of factors will need to be considered, namely: recovery times, transient temperatures within the breeder sector, tritium release pathways, location of blanket sectors within the building, and the building containment and ventilation capability.

The decay heat removal system is clearly safety critical. The substantiation of the design of the system will be a major part of the safety case for a fusion reactor and it is clear that the analysis of potential consequences of a LOCA, either during normal operation or during breeder blanket transfer, needs further work.

5.2.4. Loss of cooling in a dual coolant lead Lithium (DCLL) blanket

The makeup of the breeding blankets can have a significant effect on the safety case. For example, activation of the DCLL blanket produces ^{203}Hg and ^{210}Po, whose respective dose factors per ingestion are 100 and 100,000 times higher than for tritiated water [31]. The primary concerns with these radioisotopes are potential spills and releases during maintenance operations. For the test DCLL blanket module (TBM) being developed for DEMO, the end-of-life production of ^{203}Hg and ^{210}Po equates to activities of 1332 GBq and 66.6 GBq, respectively [35]. If an accident were to occur that resulted in the deterioration of confinement barriers and the release of the entire ^{210}Po inventory to the environment, assuming average weather conditions (P-G stability conditions D with a wind speed of 4 m/s), the dose at the site boundary would be 0.08 mSv [35]. Similarly, if the entire ^{203}Hg were to be released, the dose at the boundary would be 0.002 mSv. Whilst these doses are low, it is worth bearing in mind these estimates are for a single blanket module. Investigation of the potential consequences of an unplanned release of inventories of ^{203}Hg and ^{210}Po in a fusion power plant needs further work, in order to inform the design of the blanket cooling systems to avoid a LOCA resulting in an unacceptable release of radioactivity.

During operation, the only release pathways for the lead-lithium (PbLi) coolant in the DCLL blanket are through potential leaks in the pumping systems. This is a similar concern for the release of tritium during operation and will need to be addressed in the pre-construction safety case. Potential spills of the PbLi during maintenance activities will also need to be investigated, in order to protect workers. Ref [36] reports work on the modelling of the blanket in a conceptual 1000 MWe fusion plant design to identify safety issues and develop mitigation strategies. The most promising approach is the introduction of online bismuth removal to 1 ppm. As ^{209}Bi acts as a precursor to ^{210}Po, reducing the ^{209}Bi can limit the ^{210}Po inventory. Due to the volatility of these radioactive isotopes, there is a potential for an off-site release. A detailed safety analysis of the accident scenarios is required in order to determine the appropriate containment system for a fusion reactor.

5.3. Loss of vacuum vessel integrity

5.3.1. Failure of penetration

Failure of penetrations in the vacuum vessel (VV) can result in a loss of vacuum and ingress of air into the vessel itself. These are typically called loss of vacuum accidents (LOVA). As part of the RPrS at ITER [8], an assessment of a LOVA found that if a single penetration line is assumed to fail, the resulting air ingress will trigger a disruption, resulting in an immediate termination of the fusion power.

From the analysis, the tritium and dust masses in the vessel that are likely to escape outside the bioshield are very small (0.32 mg and 6 mg, respectively). Given that there are no workers present in these areas during operation, and the return to safe state does not require workers to be in these areas, there are no significant radiological consequences for personnel.

The calculated radiation doses for most-exposed persons arising from the radioactive release associated with this accident are 0.012 mSv at 200 m (short term) and 0.013 mSv at 2.5 km (long term) [8]. Such exposures are very low when compared with the 1 mSv limit for members of the public and are orders of magnitude below the evacuation limit (50 mSv). As the predicted inventories of the DEMO fusion reactor and ITER are expected to be similar, it seems reasonable to assume the radiological dose from a single failure of penetration would not reach levels where an evacuation may be necessary.

5.4. Hydrogen and dust explosion

Within a fusion reactor where there is tritium and deuterium there is the potential for an energetic hydrogen interaction should a failure of the VV result in significant air ingress. Failure of the VV causing significant ingress of air resulting in a combined hydrogen and dust explosion was considered to be a beyond design basis event for the ITER design.

Nevertheless, this event was considered in the ITER safety analysis [8]. The accident sequence chosen considered multiple failures in one of the penetration lines connecting the VV to a port cell, resulting in rapid air ingress into the VV. Hydrogen from the cryopumps was assumed to mix with the air. As the ignition energy required for a hydrogen explosion in air is so low (0.02 mJ), an explosion can spark on any hot surface.

Within the VV of a fusion reactor there is the potential for a large quantity of dust to accumulate. This dust is composed of Be and W that is eroded when the plasma hits the VV walls; in ITER and in EU-DEMO the maximum limit for dust in the VV is 1000 kg. In the ITER analysis, it is assumed that the hydrogen explosion provided enough energy to initiate a more severe dust explosion (expected energy of around 14 GJ). The combination of these explosions resulted in multiple failure of confinement systems (windows or valves) between the VV and several Port Cells, providing a release pathway for any radionuclides into the atmosphere.

In the ITER analysis, as this scenario was classed as a beyond design basis accident, no worker doses were calculated; however, the calculated radiation doses for most exposed persons are 0.33 mSv at 200 m and 0.20 mSv at 2.5 km. Radiation doses at this level would again not result in the need to evacuate people in the surrounding areas. Given that the probability of the initiating event is extremely low, even at these dose levels the risks are likely to be in the broadly acceptable region. However, for a commercial fusion power plant, the probability of the initiating event would need to be evaluated to demonstrate this is the case, and that any additional safety measures needed to reduce the risk further would need to satisfy the ALARP criteria.

Nevertheless, such explosions have the potential to compromise the integrity of the VV and the containment/confinement vessel and result in multiple release pathways for radioactive materials. Given the larger size and added complexity of DEMO, or other commercial fusion power plants, the consequences of a potential hydrogen/dust explosion could be more severe than that shown in the ITER safety analysis. Whilst avoiding ignition sources is not a practical solution (the ignition energy required for a hydrogen explosion is extremely low), mitigation systems that aim to limit the consequences of an explosion are currently being explored. Examples of mitigation tactics for future fusion reactors include igniters within the VV (which ignite a small amount of hydrogen/air mixture as soon as the lower flammability limit is reached resulting in a less severe combustion), or rapid injections of inert gas to reduce the rate of pressure increase [26]. Another option for designers is to reduce the potential for dust accumulation through material selection for components within the VV and dust extraction systems.

5.5. Loss of plasma control

Plasma instabilities and disruptions can lead to physical phenomena such as thermal shocks, electron beams, eddy currents, etc. that can, if uncontrolled, threaten the integrity of the VV (e.g. due to electromagnetic loads in VV components and on the vessel itself) [31]. Such instabilities also have the potential to accelerate production of dust from erosion of the first wall and damage the VV cooling system causing coolant ingress (as discussed above).

The ITER safety analysis looked at a scenario that began with an "over-fuelling" of the plasma, resulting in a loss of plasma control and an increase in fusion power. A simultaneous failure of the Fusion Power Termination System (FPTS) and failure of all three first wall cooling loops into the VV were postulated as aggravating factors. This scenario was chosen as a bounding case for events related to loss of plasma control to demonstrate the safety margins of the reactor design.

In the event, it is assumed the FPTS fails to stop the plasma on the indication of an increase in fusion power. If this occurs, the FPTS has a backup system in which it stops the plasma burn after receiving a signal that the outlet (VV) water coolant temperature exceeds 170 °C. As it takes roughly 40 s to reach this temperature, it is assumed that both the coolant spilled inside the VV and the in-vessel components are at significantly higher operational temperatures than normal. Future work should investigate the consequences of a loss of plasma control in which the FPTS fails completely.

The ITER RPrS [8] finds that both the temporary increase in fusion power and the increase in temperature and pressure have no significant effect on the VV. Whilst the failure of the cooling loops demonstrate failures of safety critical components, assuming the VVPSS operates correctly, the VV will not reach overpressure and there is no significant release of radiological material.

Similar to the point made in 5.2.2 (above), further work on accidents/events in which the VVPSS fails is necessary to fully evaluate the potential threat from these types of event. The safety analysis for a fusion power plant must be able to demonstrate that in the event of a loss of plasma control (resulting in an in-VV LOCA), failure of the VVPSS system will not result in overpressure of the VV, or result in a radiological release that has a significant impact on workers or the surrounding public.

5.6. External hazards

In addition to designing fusion power plants to cope with a range of plant modifications and accident initiating events, it is necessary to consider the challenges posed by external hazards [37]. External hazards can generally be split into two categories: natural events such as earthquakes, extreme temperature, high winds, flooding, precipitation, and forest fires etc.; and man-made events such as aircraft crashes, external explosions, loss of off-site power etc. [38]. In the SEAFP studies, only preliminary consideration was given to the role of external events such as those described above.

5.6.1. Bounding event

In the SEAFP studies [2], an unspecified ultra-energetic event was postulated, resulting in the complete destruction of confinement barriers. In this scenario, the radiological consequences of the release of the full inventory of tritium would almost certainly require evacuation of the public in the surrounding area, if worst-case assumptions are retained. In order to prevent this, it is clear that any fusion power plant would need to be designed to limit this uncontrolled release of radioactivity in line with the Tolerability of Risk concept [14] and in conformance with Fig. 1 (above). Consideration of this worst-case scenario is useful to put the potential consequences into perspective and enable appropriate protection and confinement systems to be built into the power plant design [11].

In [2], it was concluded that only certain ex-plant events have a potential for breaching the primary radioactivity confinement barrier. It was suggested that aircraft impact and earthquakes be covered by the design basis [11]. However, for any fusion power plant design, the range of external hazards to be considered will depend upon the country and location that the plant is sited in.

5.6.2. Seismic events

The design requirements to withstand seismic events depend upon a number of factors including the consequences of an uncontrolled release of radioactive materials and the seismicity of the area in which the plant is located. In the case of ITER, the French regulators required that buildings that contain radioactive inventories have earthquake protection [39]. This was to ensure that, in the event of an earthquake, safety important components are not impaired and retain their function. The analysis performed at ITER found that an earthquake itself would not initiate an accident that has not already been covered by the safety case; however, internal and external hazards can act as aggravating factors in an existing situation, for example loss of electric power following an earthquake [8].

The approach adopted for ITER is understandable, but it should not be regarded as a precedent for future fusion power stations. Seismic protection can be costly and can increase design complexity. To justify special design measures to withstand seismic events, it is essential to understand the potential consequences of failure. As such seismic design requirements for fusion power stations should be risk based, designers of future fusion power stations will need to evaluate containment integrity based on the radiological release consequences. It is entirely possible that enhanced seismic design requirements may not be justified on safety grounds alone but rather for asset protection reasons.

5.6.3. Aircraft impact

Prior to the attacks on the Twin Towers in New York in 2001, the traditional approach to aircraft crash assessment was to consider the likelihood of an aircraft falling out of the sky and impacting on an installation. These probabilities were generally very low and hence in most, but not all, cases no special design measures were required. However, things have changed, and fission power stations now need to demonstrate resilience against a direct aircraft impact. Assessments of aircraft impacts on reactor buildings were performed in the safety analysis at ITER [8]. A range of aircraft families were analysed and the probability of a hazard relating to general aircraft impacting on the Tokamak Building was calculated at 1.2×10^{-6} per annum. As this value was above the 10^{-7} per annum limit for a radiologically controlled building (as stated in the Fundamental Safety Rule (RFS) [40]), the hazard must be taken into account in the design of the facility. The analysis showed that the design and layout of the buildings ensures that any impact from a general aircraft would not impair safety important components (SICs) or result in a release of radioactive material. This is generally due to the concrete in the roofs and walls of reactor buildings being sufficiently thick to withstand an aircraft crash or the impact of structures, liable to fall on them, without causing major cracks or perforations [8].

However, aircraft impact protection is costly and can increase design complexity and hence aircraft protection for any future fusion power station must be justified. The potential radiological release consequences of an aircraft crash must determine the extent to which the power plant is designed to protect against aircraft impact.

5.7. Internal hazards

5.7.1. Fire hazards – reactor (tokamak) building

Fire within a power station is a recognised internal hazard and, as such, all nuclear installations are designed to limit the initiation and consequences of fire. The preliminary safety analysis (RPrS) of the ITER design [8] addressed the fire risk and showed that it is possible to design a fusion facility so that a fire in the tokamak building (i.e. the building housing the fusion reaction) would not result in a loss of vacuum vessel integrity, and that the loss of safety functions from damage to safety important components (SICs) was very unlikely [8]. This analysis has shown that with the application of the appropriate fire standards, the risks associated with internal fire hazards in fusion power stations can be managed. The radiological consequences of a fire breaking out in the tritium plant are discussed in 5.7.2 (below).

5.7.2. Fire hazards – tritium plant

The impact of a fire in the tritium plant was modelled as part of the RPrS at ITER [8]. The analysis assumed the failure of a glove box confinement which resulted in a release of tritium. It was assumed that the entire tritium inventory in the glove box (70 g) was instantaneously released into the room as the fire began. The temperature increase led to a pressure increase; however, it was assumed that the detritiation systems will be able to cope with the room pressurisation during the fire and maintain it under depression. The maximum quantity of tritium as HTO released into the environment was calculated to be 7.3 g.

The calculated radiation doses for most-exposed persons arising from the radioactive release associated with this accident are 1.07 mSv at 200 m and 0.17 mSv at 2.5 km. Whilst these exposures are on the same order as the 1 mSv limit for members of the public, they are significantly below the evacuation limit (50 mSv). Given that this is classed as a beyond design basis accident, i.e. a hypothetical event sequence postulated by adding a series of independent aggravating failures, the likelihood of the overall sequence transpiring is extremely low [25]. The objective must always be to make any fault sequence extremely low if the consequences result in the risk not being ALARP/ALARA.

For ITER there is a comprehensive fire detection and suppression system together with a robust defence in depth approach to fire protection [8]. It is clear that there is a potential for radiological release from a fire in a fusion reactor tritium handling plant and hence the fire safety design will require robust substantiation.

5.7.3. Electromagnetic discharge

As discussed in 4.1, the magnetic energy inventory in a fusion reactor is expected to be large, with toroidal and poloidal coils having energies up to 180 GJ and 50 GJ, respectively. Failure of the magnet systems can result in discharge of this energy in arcs leading to significant damage to the first confinement barrier (the VV wall). Energy from the magnet is discharged in a small area and can result in a hole forming in the wall, initiating a LOVA [2]. As detailed in 4.1.2, ITER's magnet system incorporates separate monitoring, fault detection, and protection systems that act to minimise the likelihood of magnetic energies damaging the first confinement barrier.

A bounding accident related to this hazard was included in the ITER RPrS [8]. In the scenario, two 1 m^2 holes appear simultaneously, one in the wall of the VV and one in the wall of the cryostat, providing potential release paths to the environment. The hole in the VV wall causes coolant ingress into the VV, causing a pressure rise and effects similar to those discussed in the in-VV LOCA in 5.2.2. As this scenario was classed as a beyond design basis accident, no worker doses were calculated; however, the calculated radiation doses for most-exposed members of the public are 3.0 mSv as 200 m and 0.13 mSv at 2.5 km. Radiation doses at this level would again not result in the need to evacuate people in the surrounding areas. However, the safety case for any future fusion power station will need to address this accident scenario to ensure that the design is robust to reduce the consequences of this type of accident such that the risks to workers and the public are ALARP. Fusion power stations such as DEMO will have larger magnetic energies compared to that in ITER. Additional work in this area is currently being carried out [11].

5.8. Component failure rates

Evaluating risk requires knowledge of the probabilities of the initiating event and the subsequent performance of the protection systems. Currently there are large gaps in component failure rate data for evaluating accident probabilities. Failure rates are generally based on empirical data where available. A fusion-specific database has been developed as part of an international collaboration, based on data from typical equipment used in other areas of engineering (such as pipes, valves, ducts etc.) [41]. Many fusion-specific systems (as they are new), however, have no empirical data and hence cannot be assigned an accurate component failure rate. In these circumstances judgement has to be used to assign failure rates. In the SEAFP studies [2], failure rates were used to form bands of probabilities defining events as:

- incidents – 1 to 10^{-2} per annum;
- design basis accidents – 10^{-2} to 10^{-6} per annum; and
- beyond design basis accidents - $< 10^{-7}$ per annum.

The current international fusion safety community, as illustrated in the ITER project, uses a similar technique but without indicating numerical values for occurrence rates [11]. Instead, ITER defines an incident as an unplanned event that can nevertheless be expected to occur at least once in the lifetime of the reactor. An accident is defined as an event that is not expected to occur; however, precautions are taken in the design to mitigate the consequences if it does. A beyond design basis accident is defined as an accident with multiple aggravating factors that is not expected to occur and has such a low probability that it is generally not taken into account during design [11].

Looking towards DEMO and future fusion power stations, it is imperative that the consequences of accidents where safety systems are impaired or fail to act are established. Using the in-VV LOCA as an example (see 5.2.2), if the VVPSS rupture discs fail to burst, the consequences of an unmitigated pressure rise need to be established. The design pressure limit of the VV will be verified (e.g. in ITER it is expected to be 200 kPa), but there will need to be an analysis of what the peak pressure would be in an in-VV LOCA with failed VVPSS rupture discs and if this peak pressure is sufficient to cause failure of the VV. Whilst rupture discs tend to have a low rate of failure, in order to determine if this rate is acceptable, one would need to know the probability of the initiating event coupled with the probability of the failure of the bursting disc, as well as the consequences of the likely release. The assumptions around the size of the water ingress in the case of an in-VV LOCA will also need to be substantiated, along with the ability of the VVPSS to cope with a range of water ingress events. This will ensure the VVPSS has been designed to cope with the design basis event and there was no cliff edge present beyond the design basis. Given this it is not unreasonable to suggest that more work is needed to identify the range of challenges from the design basis water ingress assumptions to the VVPSS and the ability of the proposed design of the VVPSS to cope with the design basis challenge.

As discussed above, the reliability of the plasma control system is vital to the safe operation of a fusion power plant. In a fusion power station, the control and protection system is likely to be more complex than that in ITER and hence the potential for malfunctions of the plasma control system could potentially increase. Initiating events could result in a rapid increase in fuelling rate or a rapid increase in auxiliary heating [42]. New systems will probably have to be developed that can monitor and control the plasma to limit the likelihood and consequences of these types of events. As these will be new fusion-specific systems, again there will be little empirical failure rate data, which will make reliability assumptions difficult to verify in the early stage of fusion power station development.

Without accurate system and component failure rates, the reliability of fusion reactor control and protection systems will be difficult to verify. To compensate the lack of component failure rates, the operations at ITER will have to be scrutinised in order to provide further input to be used for safety and reliability assessments at future fusion facilities. This work is vitally important to the demonstration of the safety of fusion power and can be used to help develop, build, and maintain a comprehensive failure rate database for evaluating accident probabilities.

6. Conclusion

A review of the key nuclear safety issues associated with fusion power plants has been performed in this paper. From the evidence gathered, the indications are that on current knowledge the use of fusion energy for power production does not present significant off-site radiological risks for the public. A number of fusion reactor safety issues have been reviewed together with their impact on public safety. It has been shown that despite the significant amount of in-vessel fuel (deuterium/tritium) energy inventory, the burn fraction of around 2% expected in a fusion power plant ensures that the maximum fuel energy able to be released under accident conditions will not challenge the integrity of confinement barriers. However, disruptions that could lead to a release of magnetic energy need to be better understood, in order to gain a better understanding of the potential risk they pose.

The large gaps in component failure rate data is a significant issue for the robust safety analysis and engineering substantiation that will be needed for fusion power plants. This is especially true for the new fusion-specific systems that are being developed. Without robust failure rate data, the probabilities of potential accidents will be based on engineering judgement rather than hard data. This will impact on the robustness of the necessary safety cases. Without a detailed knowledge of how likely an accident is, the risk approach to safety becomes less robust and subject to uncertainty. The current development of a fusion-specific database is aiming to combat this potential weakness. The work at ITER aims to fill in many of the gaps but more work needs to be focussed in this area. The production of a robust system and component failure rate database should be a main priority in the coming years to enable the early delivery of fusion power stations.

Breeder blankets and their tritium inventories pose a challenge to the design and safety analysis of fusion power stations. The removal of decay heat from these blankets, particularly when changing blankets, needs to be studied further to establish the risk associated with this activity. The potential accident scenarios associated with the transfer of blanket sectors need further investigation, especially in relation to the consequences of a loss of coolant accident during the transfer process. This is necessary in order to determine the number of engineered barriers that are required to ensure the safety of these operations.

Whilst the heat in the plasma chamber under normal operation should never reach levels that could melt the first wall, the production of WO_3 needs to be investigated further to better understand the consequences of accidents that could result in the release of this material.

Fusion reactors for electricity generation will undoubtedly require sophisticated engineering solutions to the safety issues highlighted here. However, in order to develop a robust safety case for a fusion power plant it is clear that significant further work is needed in areas including component failure rates, decay heat removal, vacuum vessel integrity, accident scenarios in which control or protection systems fail, and the impact of external hazards.

The above review has shown that whilst the hazard potential of a fusion power station is significantly less than that of a fission power station, there is the potential for the release of radioactive materials in accident conditions. Fusion power stations will also produce radioactive waste, some of which will require long-term management. There is an internationally recognised nuclear safety standards framework for nuclear fission reactors; however, the application of this framework to fusion power would, on the basis of the safety issues discussed in this paper, be disproportionate to the hazard potential.

The direct application of the fission safety standards to fusion would not only be disproportionate but would also result in unnecessary cost and complexity. It is therefore recommended that the nuclear fusion community gives serious consideration to the development of a fusion-focussed safety standards framework (similar to that which has been developed for fission power) to enable fusion power station designers to produce proportionate safety lead designs.

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