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Experiences for Consideration in Fusion Power Plant Design Safety and Safety Assessment

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EXPERIENCES FOR CONSIDERATION IN FUSION POWER PLANT DESIGN SAFETY AND SAFETY ASSESSMENT

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FOREWORD

This publication collects international experience and safety practices on experimental fusion facilities and possible design safety and safety assessment approaches to be used in fusion demonstration and power plant projects. As such, the publication can support Member States' endeavours to develop safety frameworks for future fusion power plants, as well as facilitate the early international harmonization of these frameworks.

This publication was prepared on the basis of the answers to questionnaires provided by experts from various Member States and international organizations and was complemented by further information provided by experts in the field of safety of fusion facilities. The intention of this publication was to collect Member States' current practices and experiences regarding the safety of experimental fusion facilities, with an attempt to focus on the aspects that are most relevant to future fusion power plants. The publication also identifies some of the knowledge gaps related to safety approaches for fusion.

The IAEA wishes to express its appreciation to the participants and participating organizations for their contributions, in particular B. Colling (United Kingdom). The IAEA officers responsible for this publication were G. Choi, D. de Caires Watson, J. Luis Hernandez, Z. Stone and P. Calle Vives of the Division of Nuclear Installation Safety and S. Gonzalez de Vincente and M. Barbarino of the Division of Physical and Chemical Sciences.

CONTENTS

1 INTRODUCTION

1.1 BACKGROUND

Over the past four decades, the fusion community has accumulated substantial knowledge and experience of fusion science and technology through the construction and operation of experimental fusion facilities [1, 2]. This includes the design, construction, operation and decommissioning of the Tokamak Fusion Test Reactor (TFTR) in the United States of America (USA), the Joint European Torus (JET) in the United Kingdom (UK), the National Ignition Facility (NIF) in the USA, superconducting facilities like the Experimental Advanced Superconducting Tokamak (EAST) in China and the Korea Superconducting Tokamak Advanced Research (KSTAR) facility in the Republic of Korea, as well as the design and construction of the international fusion facility, ITER, in southern France and JT-60SA in Japan. More recently, steps have been taken toward the design of several fusion demonstration power plants (known as ‘DEMO’ plants) in China, Europe, Japan, the Russian Federation, the UK and the USA. As well as public sector activities, substantial private sector efforts have emerged in some Member States. These private sector activities include the planning and design of various experimental and prototype facilities for a diverse range of fusion design concepts. Private sector companies have also gained experience constructing and operating experimental fusion facilities.

It is expected that fusion technologies will evolve in the coming years from experimental facilities towards prototype and electricity-generating facilities, and there is increased global interest and investment in the field [3]. This is driving a need for Member States that will host such fusion facilities to establish appropriate regulatory frameworks and expectations for the safety assessment of their design, construction, operation and decommissioning. Although the IAEA has established general requirements that are applicable to all facilities and activities using radioactive sources, aspects of existing design safety standards and guidance for radiation sources such as particle accelerators or nuclear installations such as nuclear power plants (NPPs) may not be appropriate for some fusion applications. The IAEA is therefore considering whether fusion-specific design safety and safety assessment Safety Standards may be needed. This TECDOC gathering experiences and perspectives from the Member States is to be used to inform this future work.

In parallel to the production of this TECDOC, a second TECDOC is in publication, titled International Experience in the Regulation of Fusion Facilities, that will collate current practices and proposed approaches of Member States for the regulation of fusion facilities.

1.2 OBJECTIVE

The objective of this publication is to provide insight into how safety may be addressed for prototype, demonstration or commercial fusion power plants, referred to collectively in this publication as fusion power plants (FPPs – see below for definition). This publication intends to achieve this by capturing safety approaches from a range of currently operating and planned experimental fusion facilities, as well as those approaches under development for proposed FPPs. This information has been captured in such a way (see Section 1.3) that it can help to inform on how safety is addressed for FPPs, but in many places is relevant to other types of fusion facilities. Additionally, and where possible, this publication aims to identify commonalities and differences in approaches within the captured information.

The information within this publication also aims to improve the general understanding of the safety considerations for FPPs and, as such, can inform on any future decision of the IAEA and its Member States on whether there is a need for fusion-specific design safety and safety assessment safety standards. No recommendations are made for the implementation of design safety and safety assessment for FPPs. This publication is intended for use by public and private sector organizations developing FPPs, regulatory bodies, technical support organizations (TSOs), research organizations, prospective operating organizations and other stakeholders involved in addressing the safety of fusion technologies.

Fusion facility: for the purposes of this publication, a fusion facility is any facility making use of fusion reactions.

Fusion power plant (FPP): for the purposes of this publication, a fusion power plant, or FPP, is any prototype, demonstration or commercial fusion facility for energy production. The hazard potential of FPPs may be larger than that of experimental facilities due to the operational conditions required to achieve, or demonstrate, commercial energy production.

The term 'FPP' applies to the entire power plant, i.e. the fusion device, supporting systems such as heat removal, tritium handling (where relevant) and magnets, as well as radioactive waste handling and active maintenance areas.

Fusion device: for the purposes of this publication, a fusion device includes the fusion reaction-containing vacuum vessel and the key supporting systems needed to create and sustain fusion conditions.

The information provided in this publication in relation to identified good practices represent expert opinion but does not represent a consensus of all Member States.

1.3 SCOPE

This publication is intended to provide information on safety considerations for future FPPs. As at the time of writing there were no FPPs in construction or in operation, and those being proposed are generally at an early stage of design. The limited experience related to FPPs is supplemented with experiences from current and near-term experimental facilities, as well the judgement of fusion safety experts. The information was gathered in the following two main ways:

(a) Responses to a questionnaire sent to experimental fusion facility operators, FPP developers, regulatory bodies and TSOs. The organizations that responded are acknowledged in the List of Contributors at the end of this publication. The questionnaire is provided in full in ANNEX IV and is in two main parts:
   - Part 1: gathers public sector and private sector experts' views on the general safety design approach for FPPs in view of their safety characteristics.
   - Part 2: collects information on design safety and safety analysis relevant to FPPs from projects planned and under development (public and private). The questionnaire also gathers knowledge from existing and shutdown experimental fusion devices by asking respondents to consider which aspects of their experience might be relevant to FPPs.

(b) Expert judgement from this TECDOC's participating experts via attendance at IAEA meetings and/or reviewing and drafting of the text.

For ease of language, the survey responses and expert judgements are hereinafter collectively referred to as ‘the contributing experts’. The full list of questionnaire respondents and contributing individuals are listed at the end of this publication.

The information gathered relates to a range of fusion concepts and technologies. However, this publication does not fully represent the wide range of fusion concepts proposed or under development worldwide (for example, only limited experience was gathered from the contributing experts relating to inertial and magneto-inertial concepts). This publication is generally structured assuming FPPs will be for electricity production, however many aspects of this publication will be relevant for other applications such as process heat or radioisotope production. The information presented in this publication consists of examples only and is not intended to represent best practices.

This publication focuses on radiological safety; other types of safety, such as industrial (sometimes known as ‘conventional’) health and safety, are only addressed at a high-level. Aspects relating to the regulatory approach for fusion is out of scope and will be covered by another publication for fusion regulation. Matters relating to decommissioning and management of radioactive waste from fusion are covered by other IAEA activities. Security and safeguard aspects are out of scope of this publication, although the importance of having an integrated approach to safety, security and safeguards (known as the ‘3S’ concept) is recognized. Fusion–fission hybrid systems (i.e. any fusion device that also makes use of fission reactions and uses ‘nuclear material’, as defined in the IAEA Safety and Security Glossary [4]) are also not within the scope of this publication, although aspects of this publication may provide useful background for discussions about design safety and safety demonstration for such facilities.

1.4 STRUCTURE

This publication consists of 4 main sections—referred to as the main body—complemented by a set of annexes.

Section 1 describes the background, objective, scope and structure of the publication.

Section 2 provides a general description of fusion technologies and key safety considerations. The principles of fusion are described together with a general overview of the different approaches to achieving useful fusion power. The different fusion fuels and technology approaches are briefly presented, along with some possible systems and functions of an FPP. To set the rest of the publication in context, the key safety considerations for fusion are described in a general sense, along with the main radiological hazards to the public and workers. A general and simplified description of the safety demonstration process is described in Section 2.5, providing useful background to the collated information in Section 3.

Section 3 collates the information received from the contributing experts about their view or experience in the approach to safety for FPPs, drawing on identified commonalities. This section presents the information gathered from the contributing experts on their view or experience in approaches to safety demonstration, including use of safety functions, postulated initiating events, application of defence in depth (DiD), assessment of hazards and other safety-related topics. Some participants provided additional detail about their approach that has been recorded within the annexes of this publication.

Section 4 provides a summary of the findings, common approaches and issues identified and comments on possible future work needed for the development of fusion-specific safety standards.

There are four annexes: Annex I provides additional detail relating to specific responses and information received from Member States through questionnaires. Being the largest current international collaboration on fusion, Annex II is dedicated to information on the safety approach at the ITER project. Annex III provides an overview of plasma control at the ITER and JET facilities. Annex IV is the questionnaire that was sent out to relevant stakeholders.

2 GENERAL DESCRIPTION OF FUSION TECHNOLOGIES AND SAFETY CONSIDERATIONS

2.1 PRINCIPLES OF FUSION ENERGY

Fusion reactions involve two lighter nuclei colliding and fusing together to form a heavier nucleus, releasing energy in the process. For fusion to occur, a fusion device needs to create conditions that impart the light nuclei with enough energy to overcome the repulsive forces (Coulomb force) acting between them and approach each other at sufficiently close range to be attracted (Strong force) and fuse. To be a useful energy source, the fusion reactions need to meet several criteria, including being exothermic and using low atomic number nuclei (so that repulsive forces can be overcome). The probability of the fusion reaction depends upon the thermal agitation of the nuclei and their cross-sections. The fusion reactions that are most commonly referred to in relation to FPPs are as follows:

$$\mathrm{D + T \rightarrow {}^{4}He\ (3.5\ MeV) + n\ (14.1\ MeV)}$$

$$\mathrm{D + D \rightarrow T\ (1.01\ MeV) + p\ (3.02\ MeV)\ (50\%)}$$

$$\mathrm{\rightarrow {}^{3}He\ (0.82\ MeV) + n\ (2.45\ MeV)\ (50\%)}$$

$$\mathrm{D + {}^{4}He \rightarrow {}^{4}He\ (3.6\ MeV) + p\ (14.7\ MeV)}$$

$$\mathrm{p + {}^{11}B \rightarrow 3\ {}^{4}He + 8.7\ MeV}$$

Achieving the required conditions for fusion reactions is often described through the approach to ‘plasma confinement’, i.e. the process of bringing the ions close enough together to fuse.

The two main plasma confinement processes for fusion are magnetic or inertial, although some concepts are variations of these, or even a combination, such as magneto-inertial approaches.

- Magnetic confinement – uses strong magnetic fields to confine a high temperature plasma (e.g. within a tokamak or stellarator) which is then heated by external means.
- Inertial confinement – rapidly compresses and heats up a fuel capsule to a high density and temperature (e.g. using laser beams).
- Magneto-inertial confinement – a combination of magnetic and inertial approaches (e.g. a plasma accelerator that merges plasmoids at high speed (inertial compression) and then further compresses the resultant single plasmoid using magnetic forces to achieve fusion conditions).

The main elements to consider for the plasma are (a) its temperature, (b) its density, and (c) the energy confinement time. The triple product is what is known as the Lawson criterion. This criterion emphasizes the efficiency of the medium with the ratio of the energy produced by the fusion reactions to the power lost by the medium. The energy confinement time is the rate at which the plasma loses its energy to the surroundings. It is the energy density divided by the power loss density. A review of the fusion R&D progress as measured against the Lawson criterion can be found in Ref. [5].

2.2 OVERVIEW OF DIFFERENT CONCEPTS

Concepts for FPPs based on a diverse range of technologies have been proposed by national laboratories and private companies around the world, with most of them oriented to the production of energy. Some key characteristics of the different approaches are the fuels, plasma confinement technology and operational mode (i.e. continuous or pulsed).

The most commonly referred to fusion fuels and reactions are given in Section 2.1. While much of the experimental experience to date relates to fusion of hydrogen and deuterium (D), the fusion fuel proposed for many FPPs is deuterium-tritium (D-T), with D-^{3}He and proton-boron (p-^{11}B) fuels offering some potential advantages but also some additional challenges. Developers of FPPs using tritium as a fuel will need to consider breeding a self-sufficient supply (as this isotope of hydrogen is not naturally occurring in sufficient quantities). Any FPP relying on ^{3}He as a fuel would face similar supply challenges given the extremely low natural abundance of this isotope.

The most common plasma confinement technology design explored in experimental devices has been the magnetic confinement tokamak, such as TFTR, JET, and ITER. However, there are a wide range of confinement concepts being considered. For simplification, Fig. 1 places the different concepts into three approximate groupings: those using magnetic plasma confinement, those using inertial plasma confinement and those using magneto-inertial plasma confinement, sometimes referred to as ‘alternate confinement’.

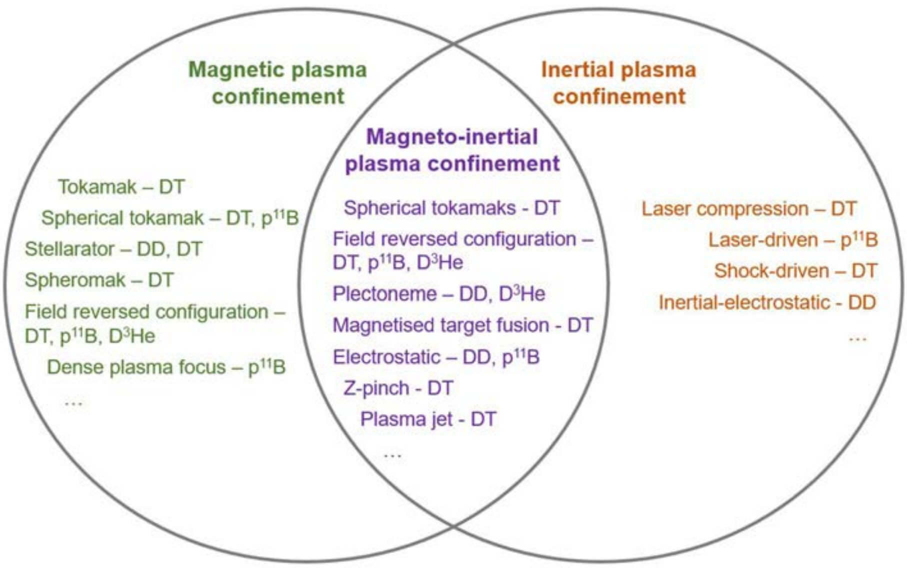

FIG. 1. Simplified summary of the various fusion approaches being considered across the globe for FPPs (based on Fusion Industry Association survey [6]).

The operational mode of a future FPP will likely be very different to that of an experimental fusion facility. A commercially viable FPP might need to operate either continuously or for a long duration or high frequency pulsed operation.

The fusion fuels (reaction type), plasma confinement technology and operational mode are just three key characteristics that make up an FPP concept; however, there are many other design options that can have an impact on safety. These include materials selection, coolant choice, the size and power of the fusion device and its siting. How these options are chosen and assessed is part of the safety design approach. Although there will be specific hazards for different FPP concepts, the key radiological hazards relating to the public and environment will be due to the fusion fuel/reaction selection (i.e. if tritium is used or produced) and related material activation from neutron irradiation (see Section 2.4).

2.3 POTENTIAL FUSION POWER PLANT SYSTEMS AND FUNCTIONS

In general, an FPP of any technology type will consist of some key systems of similar high level function and characteristics, although the specific naming and design details may change between technologies. Each FPP concept will need some form of vacuum vessel or chamber to facilitate the fusion reactions, shielding to protect equipment and personnel from radiation, a fuel cycle system to introduce fuel into the fusion device and process unfused material, an energy extraction and conversion system, and cooling to extract the heat produced.

The main systems and structures that are typical of an FPP concept (regardless of the technology but noting terminology differences) are described in the remainder of this subsection. In addition to these systems, FPP concepts that will use or produce tritium and activated material will have some structures and systems in place to confine this material, thereby minimizing mobilization and limiting the size and likelihood of any potential release. Inventory confinement is discussed further in Section 2.4. There may be other designated shielding structures such as those providing radiation shielding for equipment and personnel, or thermal shielding structures to protect equipment such as superconducting magnets from high temperatures.

Plasma-facing wall

The first material surrounding/facing the fusion reaction is often referred to as the first wall. Depending on the FPP concept, this first wall will need to withstand high heat and neutron fluxes, and may be operating at temperatures of up to 1,000 °C or more. For ITER, the first wall panels will experience temperatures from 110°C up to 880°C (further detail provided in Annex I). As an example, developers of some magnetically confined tokamak FPP concepts are considering using tungsten as a first wall material due to good plasma impurity control, low physical and chemical erosion and low tritium permeation [7]. Developers of some FPP concepts are considering the potential for using liquid metal first ‘walls’, such as flowing lithium–lead or lithium–tin [8]. Pulsed magnetic fusion concepts would require a dielectric for the first wall, and at least one approach is considering silica for its plasma-facing wall [9].

Vacuum vessel/chamber

A vacuum vessel or chamber, of some form, is used in all fusion devices to allow the fusion process to occur. This vacuum vessel also confines the fuel.

Fuel systems

The fuel cycle will include the introduction of fuel (e.g. D-D, D-T or D-^{3}He) into the vacuum vessel/chamber, the extraction, processing, recirculation and storage of both unfused fuel and helium products. In some FPP concepts, the fuel systems may include a breeding process, for example some D-T FPP developers propose breeding tritium using lithium (see 'fusion blanket or shielding' below).

Plasma confinement technology

The systems providing the plasma confinement differ for each technology. Examples of plasma confinement technologies include superconducting magnets, lasers and high-velocity pistons.

Energy extraction systems

The energy released in the fusion reaction, primarily high energy neutrons for the D-T reaction, is absorbed in the plasma-facing components and fusion blanket and removed via a primary cooling system. For most designs, the heat removed by the primary cooling system is transferred to a secondary circuit where it is used to raise steam and produce electrical power via a relatively conventional steam turbine cycle. Alternate secondary cooling systems, such as superheated CO_2 cycles, could also be used.

Those FPP concepts that propose using the D-^{3}He or p-^{11}B reactions would principally generate charged particles rather than neutrons (although some neutrons would be generated by side reactions). Developers of these FPP concepts are thus exploring whether they could eliminate the need for steam generating (or other secondary side) plant using the properties of energetic charged particles to induce electrical currents directly within stator coils surrounding the vacuum vessel.

Fusion blanket or shielding

For FPP concepts that need to breed tritium fuel, the material surrounding the fusion reaction is often referred to as the 'fusion blanket', 'breeding blanket' or simply 'blanket'. In other cases, there may be a shield made of borated concrete or hydrogenous material that surrounds the fusion device. The amount and type of shielding required depends on the intensities and types of radiation generated. Additionally, the blanket may also serve as the main energy extraction system through heating of coolant.

Therefore, the blanket could have any (or all) of the following key roles, depending on the concept: (a) to breed a self-sufficient supply of tritium fuel, (b) extract fusion energy for power production (or other application), and (c) to provide shielding.

Tritium breeding can be achieved through the interaction of neutrons (from the fusion reaction) with a lithium-based material within the blanket[1].

[1] To achieve a self-sufficient supply of tritium, more than one tritium atom needs to be produced per neutron to make up for tritium losses through the fuel cycle. To increase the tritium production in the blankets, a number of neutron multiplying materials such as beryllium or lead are proposed; these materials will increase the total number of neutrons available and hence rates of tritium production.

For some of the FPP developers proposing to use direct energy conversion (i.e. non-D-T FPP concepts), it may be that the main shielding is not integrated into the vessel structure as a blanket but is more akin to the radiation shielding provided for some tokamak concepts.

Active maintenance/radioactive waste facility

For FPP concepts that produce neutrons (including low neutronicity[2] FPPs), some components will become activated. In concepts using D-T reactions, the materials will have been subjected to high energy neutrons and are also likely to be contaminated with tritium. The longer the exposure time, the greater the level of activation. Maintenance or processing as part of a waste management strategy of any activated or contaminated components is likely to occur in a dedicated area with suitable shielding, heating, ventilation and air conditioning (HVAC) systems providing tritium decontamination, and any other controls needed to ensure radiological safety.

2.4 SAFETY CHARACTERISTICS AND CONSIDERATIONS

2.4.1 General considerations for safety

The following are examples of some general safety characteristics of fusion devices, including discussion of some inherent safety features of FPPs.

In an FPP there would be no potential for runaway chain reactions, which can be a concern for certain NPP designs. An inherent safety feature of a fusion device is that departures from normal conditions tend to cause the fusion reaction to cease or substantially degrade (this is one reason why achieving fusion conditions has, up to now, proved such a scientific and engineering challenge). This means conditions such as excessive plasma temperature gradients, the presence of impurities, deviations in fusion confinement (magnetic field) or the vacuum, plasma instabilities (from a loss of plasma control or originating from certain operational conditions), off-normal operation of the laser systems, or a change in fuel mixture leads to fusion reaction termination.

Another characteristic of a fusion device is the small amount of fuel present within the vacuum vessel at any given time (a few milligrams to tens of grams)[3]. To maintain the fusion reaction within an FPP, the fuel mix and density would be precisely controlled and constantly replenished, e.g. via gaseous injection or using target capsules. In contrast, for most operating NPPs, the nuclear fuel remains within the reactor core for several years allowing the buildup of radioactive fission products within the fuel.

FPP developers may need to consider the generation of ‘decay heat’. Unlike an NPP where most of the decay heat is generated by fission products in the fuel, for an FPP most of the decay heat arises from activated materials, particularly in plasma-facing materials. The rate of decay

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[2] Neutronicity is the fraction of the fusion energy released through neutrons and is therefore indicative of the levels of radiation damage to materials and neutron activation that can be expected.

[3] Noting that although the fuel introduced to the chamber at any one time is low, where tritium is used as a fuel or produced, there can be tritium buildup within the structures comprising the device due to tritium diffusion and/or codeposition of tritium with eroded material. Additionally, any device with reactions that can produce neutrons will result in some level of activated material. Therefore, the total radioactive inventory in an FPP is more than just the fuel. This is discussed further is Section 2.4.5.

heat production within an FPP would depend on the materials selected and the neutron fluence (how long the materials have been exposed to neutrons, and at what flux).

In accident scenarios where the decay heat removal function is impacted, the buildup of heat could cause damage to confinement barriers and other systems. Temperature rise from decay heat can also impact the mobilization of radioactive material, for example through increased volatilization. Some FPP developers plan to claim passive heat removal of residual decay heat.

FPPs using the D-T reaction will produce the most decay heat due to the high neutron fluxes of these reactions. For example, immediately following shutdown, the experimental device ITER when operating with a D-T fuel mix will generate a maximum of 11 MW of decay heat [10], which is ~2 % of the 500 MW thermal power generation. For nuclear fission the equivalent maximum decay heat from 500 MW thermal energy is ~35 MW, which is ~7 % of thermal power [11]. Typically, the maximum decay heat from nuclear fission is three times greater than that of a D-T fuelled FPP concept for a given level of thermal power. For fusion reactions with low neutronicity (e.g. D-^{3}He and p-^{11}B), the levels of activation, and therefore decay heat, will be much lower than that of D-T-fuelled reactions[4].

Another inherent safety feature of FPPs is that the reaction vessel operates under a vacuum. This eliminates some hazards related to operating the reaction vessel at higher pressures by most of today's operating NPPs (although some innovative non-water-cooled reactor designs do not operate at pressure). In the event of a loss of vacuum, air would move into the vessel/chamber and terminate the fusion reaction. Only after reaching pressure equilibrium could radioactive material (fuel, dusts) be mobilized from the vacuum vessel/chamber into the surrounding structure or building. Some FPP concepts include pressure suppression systems to mitigate overpressure in the vacuum vessel and control the mobilization of radioactive material within the vacuum vessel. Note that while the fusion reaction occurs in a vacuum, supporting systems may operate at pressure (for example, pressurized water-cooling systems).

2.4.2 Fusion-specific hazards to be considered in FPP design

Achieving controlled fusion means imparting the fuels with sufficient energy to overcome Coulomb repulsion (see Section 2.1). The means of achieving these energies, referred to in this publication as the plasma confinement technology, can differ amongst FPP concepts. As the plasma confinement technology is necessarily a source (or container) for large amounts of energy, it is also a hazard source. Example plasma confinement technologies include high energy magnets, high-powered lasers, compression systems, ion beams and projectiles.

Developers of FPP concepts with a superconducting magnetic plasma confinement approach may need to consider the magnet and its supporting systems as hazard sources. The loss of superconductivity, known as magnet quenching, can cause temperature, voltage and mechanical

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[4] The risks posed by decay heat depend substantially on the materials the device is comprised of, as this determines which radionuclides may be produced through neutron activation. Counterintuitively, a material with activation products of shorter half-life (i.e. that is 'hotter') may pose less of a decay heat challenge than a material with a longer half-life. For example, a device using aluminium magnets and silica walls—whose most common activation products have half-lives on the order of minutes and hours, dependent on neutron energies —will have a relatively low decay heat contribution. This is because the short half-lives mean their concentrations within the materials quickly reach equilibrium during operation so the inventory at time of shutdown is relatively low and rapidly reduces. Longer half-life materials will build up a larger inventory of radioactive products by the time of shutdown and therefore make a larger contribution to decay heat.

stresses to increase in the magnets, which can cause structural damage and potential vaporization of the magnet coolant. A large amount of magnetic energy is accumulated in the superconducting coils of such FPP concepts e.g. ~135 GJ in the European Demonstration power plant (EU DEMO) toroidal field coils.

Magnets used in other FPP approaches such as in the beam accelerator and focusing systems of ion beam driven devices could also pose a hazard if they fail catastrophically. However, their failure might be less likely to lead to radiological release if such an event does not damage radiological containment structures.

Any FPP operating superconducting magnets will likely be working with very large electrical currents, so electrical hazards such as arcing faults may need to be assessed.

Some developers of inertial confinement FPP concepts propose using high-powered lasers. However, there is potential for the confinement to become damaged or cause the target to become a projectile due to a misalignment of the beams or an unsymmetrical response of the targets.

Alternate plasma confinement technologies such as compression, ion-beam and projectile approaches also introduce energetic phenomena to be considered as part of the internal hazards assessment (see Section 3.6.1).

Depending on the FPP characteristics, the sudden termination of a magnetically confined fusion plasma might be a potential initiating event that could cause damage to radiological barriers (confinement) and safety systems. For example, plasma disruptions in a tokamak from losing plasma control can cause thermal loads in plasma-facing materials. To respond to this type of event, some FPP developers may include systems such as 'disruption mitigation systems' or 'fast plasma discharge' to prevent potential damage and extend component lifetimes[5]. Some of the contributing experts have found, through analysis, that the impact from any loss of control of the fusion reaction does not represent a safety challenge for the fusion device. For example, the analysis performed so far for EU DEMO suggests that control of the fusion reaction does not represent a safety challenge, with the first confinement barrier (the outer shell of the vacuum vessel) remaining intact in the event of sudden termination of the plasma.

FPP developers may need to consider the potential for exothermic chemical reactions between materials that could be brought into contact in accident scenarios. For example, if beryllium, tungsten, and lithium containing materials interact with air or steam at elevated temperatures, this may release energy and/or hydrogen, which poses an explosion hazard. Leakage of hydrogen, deuterium or tritium from the fuel cycle equipment can result in a flammable or explosive hazard.

The materials surrounding the fusion reaction will be hot during operation of an FPP so leakage from the cooling system into the vacuum vessel can lead to sudden vaporization resulting in a rise in pressure.

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[5] ITER will have a disruption mitigation system for the purpose of asset protection. Further detail on the system can be found in Annex I.2.

2.4.3 Fusion component material and coolant selection

FPP materials will need to be designed to operate in very demanding conditions. Aspects to consider when selecting materials include the following:

- Mechanical performance of structural components under fusion conditions, including accounting for effects such as embrittlement, cracking and loss of ductility.
- Sensitivity of components to ionizing radiation, e.g. superconducting magnets and fibre optics.
- Compatibility of different materials and/or coolants combinations. For example, there is the potential for chemical reactions between water and beryllium and lithium metal/alloys, if these materials can come into contact with each other during accident scenarios. The corrosion characteristics of lithium metals and salts when operating at high temperature with realistic impurities and interacting with embrittled structural materials also needs to be understood. Such compatibility issues need to be considered for all plant states.
- The potential for neutron induced damage (e.g. embrittlement) and material activation, noting the potential for the activation of impurities in the structural materials, as well as the main alloy itself.

Some of the most challenging material selections are for plasma-facing components such as the first wall, divertor and limiters in magnetically confined concepts and breeding blankets for D-T concepts. The structural materials within these components face particularly challenging environments for most of the FPP concepts. An example plasma-facing material proposed in several magnetically confined D-T FPP concepts is tungsten, or tungsten-based composites or alloys, due to several properties of these materials, including: a high threshold for sputtering (reducing impurities within the plasma), high melting point, low tritium retention and high thermal conductivity. Further developments of tungsten-based materials include the advancement of self-passivating alloys that could have a significant safety advantage over pure tungsten. Self-passivating tungsten will form a protective oxide layer on the surface when subject to air ingress in a loss of vacuum event, thereby avoiding the formation of the mobilizable tungsten trioxide [12, 13], which could contain tritium contaminates and be radioactive due to neutron irradiation. For inertial confinement FPP, an additional challenge is to understand the response to a pulsed neutron load that has a very high instantaneous flux and can drive a thermo-mechanical response that exacerbates material failure modes such as fatigue.

Further detail on materials for fusion can be found in references [14, 15, 16, 17].

Most of the FPP developers propose the use of either a water, helium, molten salt or liquid metal-based coolant. Helium cooling is not corrosive and is therefore less likely to degrade piping. However, it is a light gas and will likely need to be pressurized and a very large pumping capacity will be needed to remove sufficient heat. Water has a greater capacity for heat transfer and would therefore need lower flow rates. However, in loss of coolant accidents, depending on the size of the leak, the water may flash to steam as it is released and pressurization may be lost, reducing heat removal capacity. Furthermore, water is corrosive, requiring chemical treatment and management of activated corrosion products, and can impose difficulties in preventing tritium mobilization to the environment. Liquid metal or molten salt-based coolants also have high heat capacities, but the flow of liquid metal coolants may be affected by magnetic forces from magnets if used in the plasma confinement. Both water and liquid metal coolants can accumulate activated corrosion products and impurities under neutron irradiation, which means shielding and/or delay (hold-up) tanks may be necessary for radiation protection purposes.

2.4.4 Main radiological hazards for the public and the environment

Most FPP will be larger in scale than existing experimental facilities in terms of characteristics such as fusion power, fuel throughput, operational time and neutron flux (and therefore material activation).

Developers of any FPP making use of D-T fuel will need to consider how to safely handle, store and transport tritium**.** Furthermore, developers of any FPP concept that makes use of a fusion reaction that produces neutrons will need to consider which materials may become activated[6]; this includes structural and mechanical components but also the coolant, depending on the design.

Tritium is a radioactive isotope of hydrogen that decays to a stable helium atom through low energy beta radiation with a half-life of 12.3 years. Tritium is rare in nature, being naturally produced mostly high in the atmosphere due to the interaction of cosmic radiation with nitrogen and oxygen, but it is often created as a by-product from nuclear fission[7]. Tritium has various industrial uses such as in luminescent devices and medical diagnostics.

Due to its low energy beta decay, tritium is only harmful to human health once inside the human body. Tritium can enter the body via inhalation of gaseous tritium or tritiated water vapour, ingestion of tritiated food and drink, and absorption of tritiated water through the skin. Tritium has a high solubility and diffusivity and in an FPP may penetrate into the surrounding materials, including, for example, coolants. Tritium is highly mobile within the environment and within all biological systems (where it will exchange with hydrogen in other molecules), existing in three main chemical forms[8]:

- Tritiated water (HTO) is naturally the most abundant form of tritium and is extremely mobile, integrating quickly into the water cycle. HTO can be easily absorbed into the body (through inhalation, ingestion or absorption) where it quickly diffuses throughout the body.

- Gaseous tritium (HT) makes up only a small fraction of tritium currently released into the environment (releases from fission are dominated by HTO). When in air, it can be deposited into the soil through dry or wet deposition, but its radiotoxicity is lower than that for HTO by a factor of 10,000.

- Organically bound tritium (OBT) is produced through the incorporation of tritium into nutrients such as carbohydrates, fats or proteins, and may be retained within the body for longer than HTO.

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[6] There will be other radiological aspects to consider, such as gamma ray and X ray production but the key radiological safety considerations in respect to public and environmental aspects will be from the tritium and activated materials.

[7] Around a few kilograms of tritium is produced each year from heavy water fission reactors, such as CANDU, through neutron activation of deuterium in the heavy water. Light water reactors and other fission technologies produce tritium at lower rates.

[8] Among the contributing experts, it was expected that the dominant species of tritiated gas and tritiated water would be HT and HTO respectively, although other diatomic species are possible.

Although tritium, particularly HTO, is readily absorbed into the body, the radiotoxicity of tritium is low due to its short biological half-life (HTO is 10 days and OBT is typically 40 days) compared to its 12-year radioactive half-life, and the low energy of its beta decay.

The inhalation hazard posed by a release is highly dependent on the molecular composition of the tritium. If tritium is present as HT gas it is far less hazardous (by a factor of 10,000) than if it is released as HTO or converted to HTO following release. The factors that influence the conversion of HT to HTO and the speed of this conversion in relation to the exposure period are therefore important to understand. The HT conversion rate depends on the temperature of the release, the presence of air and humidity levels. An example of some of the assumptions made about the chemical form of tritium for some fault sequences in the JET safety case is provided in Annex I. In some accident assessments, such as those relating to fire scenarios, 100 % conversion to HTO is often conservatively assumed.

The activation of materials is closely related to the neutronicity of the fusion reaction and the neutron flux, i.e. the higher the neutronicity and the more neutrons there are (flux), the more activation of the surrounding material. Most of the activated products will be bound within the solid structures that comprise the in-vessel components, however some could be mobilized, such as dust or active corrosion products, or the activation of the coolant itself (depending on coolant choice). Hazards relating to the activation of material will be different depending on the plant state, for example whether it is normal operation or accident conditions; this is explored further in Section 2.4.7. Examples of the activation of plasma-facing materials include the activation of tungsten to produce ^{187}W, ^{185}W, ^{181}W and ^{182}Ta and the activation of stainless steel to produce ^{60}Co, ^{58}Co, ^{55}Fe and ^{54}Mn.

Many materials used in structural components are alloys or composites, and the minor components of these materials need also to be considered for their potential for activation. For example, aluminium alloys are much stronger than pure aluminium, but the small quantities of magnesium and sometimes zinc present in the alloy cause its activation with ^{24}Na and ^{65}Zn. Where lead is used for shielding it is usually alloyed with antimony to improve its strength (unless supported by other structures). This will create radioactive tin and antimony isotopes from neutron activation. Fast neutrons (>1 MeV) will produce radioactive ^{60}Co in stainless steel as it contains ~ 10 % nickel. It is possible for the minor impurities present within materials to be a major contributor to the overall level of activation. For example, the presence of trace levels (few parts per million) of Nb, Tb, Ho in many common steels is a significant source of activation for this alloy. This issue has led to the ongoing development of reduced activation materials specifically intended for fusion applications. These include reduced activated ferritic martensitic (RAFM) steels like EUROFER97 and F82H.

Side reactions are a potential additional source of tritium and neutrons. For example, in FPP concepts that plan to use the D3-He reaction, there are other reactions that can occur, e.g. D-D, that lead to the production of tritium and neutrons. In a steady state FPP concept, this tritium would build up with operating time and could also cause D-T reactions producing 14.1 MeV neutrons. In pulsed FPP concepts, tritium could be removed (e.g. as the vacuum chamber is exhausted) between pulses. If the pulse duration is sufficiently short, D-T reactions may not occur, or could be very limited in nature.

$$\mathrm{D} + {}^{4}\mathrm{He} \rightarrow {}^{4}\mathrm{He}\ (3.6\ \mathrm{MeV}) + \mathrm{p}\ (14.7\ \mathrm{MeV})$$

$$\mathrm{D} + \mathrm{D} \rightarrow \mathrm{T}\ (1.01\ \mathrm{MeV}) + \mathrm{p}\ (3.02\ \mathrm{MeV})\ (50\%)$$

$\rightarrow {}^{3}\mathrm{He}\ (0.82\ \mathrm{MeV}) + \mathrm{n}\ (2.45\ \mathrm{MeV})\ (50\%)$

The primary p-^{11}B reaction does not produce neutrons as part of the main reaction but it does have possible neutron production pathways via side reactions.

Additionally, there are potential neutron production pathways through interactions with impurities within the plasma. For example, concepts making use of beryllium could produce additional neutrons through the direct interaction of fast particles (H and ^{3}He from plasma heating systems) with beryllium impurities in the plasma [18, 19].

Neutronicity is indicative of the levels of radiation damage to materials and neutron activation that can be expected, and is described in detail in Ref. [20]. Concepts making use of the D-^{3}He reaction are often referred to as low neutronicity, whilst those of the p-^{11}B reaction are often referred to as aneutronic (assuming less than 1 % of the energy is released through neutrons) [20]. The calculation of neutronicity involves assumptions about the contributions from side reactions; Table 1 lists the approximate neutronicity of the four most important fusion reactions [20].

TABLE 1. NEUTRONICITY OF THE FOUR MOST IMPORTANT FUSION REACTIONS [20]

Fuel	Neutronicity
D-T	0.80
D-D	0.66
D-^{3}He	~0.05
p-^{11}B	~0.001

2.4.5 Typical sources of radiological inventory

The locations and sizes of radiological inventories are highly dependent on the FPP design. This section is intended to provide examples of the kinds of radiological inventory that might be present within an FPP, and where they might be located. It is mainly of relevance to neutronic and low-neutronicity FPP concepts, although for low neutronicity concepts the sizes of radiological inventories may be considerably lower than for high neutronicity concepts. The description is based on a simplified typical plant description (see Fig. 2) whose key areas include the vacuum vessel, cooling systems, fuelling building and related systems, and active maintenance/waste facilities. These inventories are not necessarily mobilizable in accident or abnormal events. However, when considering safety implications and regulatory oversight, it is the hazard potential (the unmitigated risk) arising from these inventories that is important as this determines the kinds of measures that need to be in place to control the hazard. The fraction of material vulnerable to mobilization is often referred to as the mobilizable inventory, and that which is postulated to be released, as the source term.

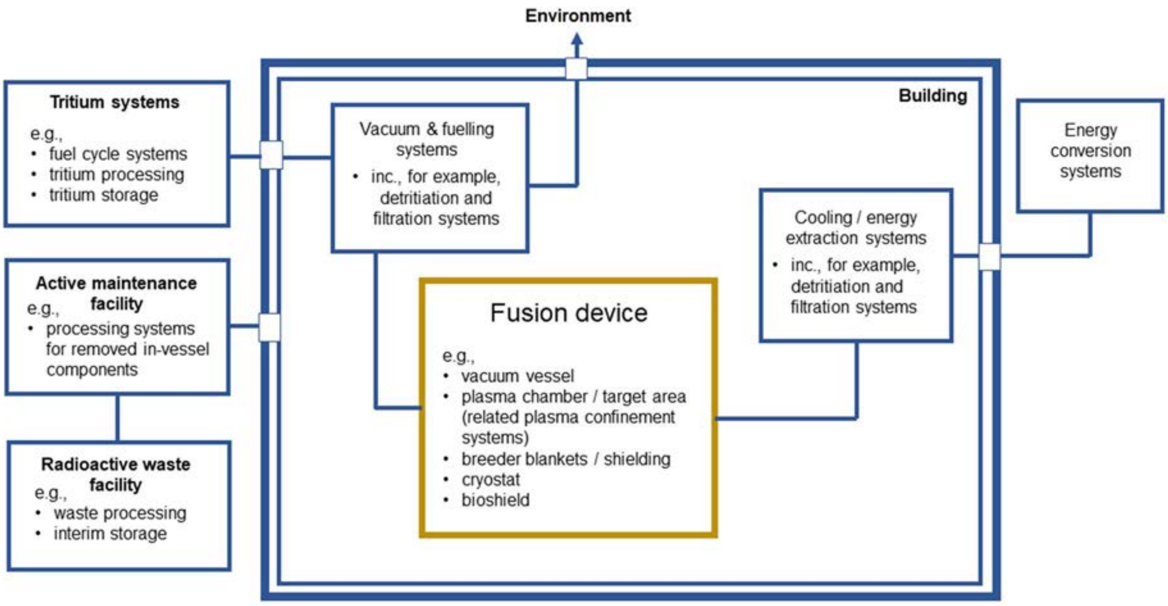

FIG. 2. Illustration of key systems in a typical FPP layout (reproduced from Ref. [21] with permission courtesy of UKAEA).

2.4.5.1 Vacuum vessel

FPP concepts that make use of tritium fuel (i.e. the D-T reaction) will only need a small amount (a few milligrams to tens of grams – see Section 2.4.1) of tritium fuel to be present in the vacuum vessel at any given time. However, not all the tritium fuel will undergo fusion (with predicted burn fractions varying amongst FPP concepts from 1–5 % for typical tokamaks, to 15–20 % for laser fusion designs). Some of the unfused tritium will be exhausted and processed in a fuel cycle system, but some will diffuse into the materials surrounding the fusion reaction. Some FPP developers propose using a liquid 'first wall' with the intention of removing the tritium from the vacuum vessel within this liquid. For inertial fusion, an additional complication is the potential for tritium to bond to the other materials in the target (e.g. carbon) to form a complex gaseous exhaust.

FPP concepts aiming for a tritium self-sufficient cycle will need to breed tritium fuel. The material surrounding the fusion reaction is often referred to as the 'blanket' or 'breeder blanket', which may, or may not, have a designated 'first wall'. The lithium material within the blanket will create tritium due to interactions with neutrons. The tritium produced within a blanket will need to be removed and processed within the fuel cycle systems. Some of the tritium will be considered 'trapped' within the materials as it cannot be completely extracted. The trapped tritium will build up over the operational lifetime in any static materials within the vacuum vessel. Additionally, the trapped tritium can potentially permeate further through the materials.

In FPP concepts where tritium is produced as part of side reactions, for example D-D and D-^{3}He (the latter through D-D reactions), then tritium extraction methods could be used to minimize D-T reactions and reduce the buildup of tritium within the fusion device (the tritium could be used for other commercial applications or maintained in a storage facility where it would decay to ^{3}He that could be used for future fuel). As with D-T FPP concepts, some tritium will diffuse into the surface of the materials.

In the vacuum vessel, activated products will also build up during the operation of the FPP due to neutron bombardment. Most of the activation products will be bound within solid structures, although some mobilizable products will be found as dust, corrosion/sputter products circulating in coolants, or as activation products within coolant material. For magnetic confinement FPPs, dusts are created when small pieces of the plasma-facing materials break away (through several mechanisms) under the extreme conditions experienced inside the vacuum vessel. For some inertial confinement systems, dusts may be generated from the target products, and due to neutron irradiation and tritium, if present, it is likely to be radioactive or contaminated. In some accident scenarios with very large temperature transients, activated products could be mobilized through volatilization resulting from oxide-driven mobilization. For FPP concepts using targets, the target materials (used to encase the fuels) will become activated and fragmented.

Both design and operational choices can affect the size of the inventories of tritium and activated products within the vacuum vessel. For example, careful material selection can reduce activation, dust production and/or tritium penetration and retention. Scheduled maintenance may include vacuum vessel cleaning to remove dust. The buildup and permeation of tritium can be limited by regular tritium cleaning from the walls. For example, using a non-fusion, deuterium plasma (e.g. a glow discharge) to remove tritium from the walls to be subsequently evacuated with the pumping system. In pulsed systems, these cleanup pulses could, in principle, take place between every energy production pulse. More frequent maintenance and cleanup reduces long term buildup and permeation however any decontamination benefit needs to be balanced against the lost revenue from stopping energy production.

2.4.5.2 Coolant systems

Radioactive inventories within coolant systems could include permeated tritium (in FPP concepts where tritium is present), active corrosion or sputter products, and the neutron activation of the coolant material.

Coolant purification and filtration systems are possible ways to minimize the circulating tritium and activated corrosion products. Permeation barriers are being considered for some FPPs as an additional measure that could be developed to reduce the rate of tritium entering the coolant.

2.4.5.3 Tritium systems

For an FPP using or producing tritium, some dedicated systems will be required to process, store and/or recirculate the tritium as fuel. For FFP concepts with tritium breeder blankets, there may be a need to have buffer storage of tritium to account for the lag time between fusion and tritium extraction and processing from the blankets. The efficiency of the tritium extraction system will influence lag time inventory needs within the buffer storage [22].

The minimization of the tritium fuel cycle inventories is pursued through development of the fuel cycle architecture and system efficiencies. For example, through developing fuel cycle designs with direct recirculation to minimize isotope separation process inventories.

2.4.5.4 *Inventories and source terms for FPP*

As previously described, the inventory of radiological material is the estimated quantity present whereas the source term is the amount of radioactive material postulated to be released, either through operational or accident scenarios. In preliminary accident analysis, the inventory is sometimes used as a pessimistic source term to provide a bounding, worst case consequence. To provide a more realistic estimate of the source term, factors such as the properties of the material are considered to establish the fraction of material that might be mobilized and then released. Depending on the type of analysis being performed, and perhaps the categorization of the accident scenario, conservative or best estimate source term quantities may be used.

Given the design maturity of today's FPP concepts, there is considerable uncertainty surrounding the potential inventory and source terms of FPPs, however some indicative values can be obtained from the published literature.

For D-T FFP concepts, the total on-site tritium inventory is likely to depend on some key factors: device power and fuel throughput, tritium retention within plasma-facing components, efficiency of tritium extraction from breeder blankets and tritium lag time[9] within the fuel cycle processing systems. For the larger DEMO devices, total site tritium inventory could be on the order of a few kilograms, although much of this would likely be within the systems of a tritium handling facility and not necessarily the main fusion plasma systems (lower power FPPs would likely have correspondingly smaller tritium inventories). Some activated products within cooling systems are likely to deposit onto cooling circuit surfaces and collect within filtration systems. Typical inventories of activated products in the coolants of a large DEMO device could be of the order of a few grams to a few kilograms. The production of dust is an area of significant uncertainty, with some estimates showing significant dust inventories – perhaps even beyond that possible for a functioning fusion device as dust resuspended into the plasma will inhibit fusion conditions. Efforts are ongoing within the fusion community to improve estimation methods for dust inventories and source terms in accident scenarios. For example, estimates of dust inventories for D-T magnetically confined tokamak concepts range from tens to thousands of kilograms, depending on the size of the FPP, material choices, tokamak conditions and material behaviour [23].

For EU DEMO and ITER, inventories will be minimized whenever possible. In volumes where a significant inventory may arise, design targets are being set. For example, there are targets for less than 1 kg of tritium and 1000 kg of dust within the vacuum vessel; these bounding values have fed into some of the initial safety analysis until inventories and source terms can be refined as the design matures [24, 25].

As noted previously, in FPPs using D-^{3}He fuel, side reactions of D-D could produce tritium. For pulsed concepts, tritium extraction methods are being considered by some FPP developers to limit the tritium buildup.

2.4.6 Confinement of radioactive material

Radiological material confinement is a key strategy to prevent mobilization of radioactive material and to limit any releases to protect workers, the public and the environment.

[9] The 'tritium lag time' is the typical time it takes for tritium produced in breeder blankets to be processed and be made available as fuel.

Developers of FPP concepts that make use of fusion reactions using or producing tritium will need to consider the high mobility of tritium and devise a suitable confinement strategy. Additionally, developers of FPP concepts that use reactions producing neutrons will need to consider material activation, particularly where this material could be mobilized in the event of an accident scenario.

The approach to confinement will vary depending on the type and design of the fusion device and the hazard potential (as per the graded approach to design safety). There may be multiple layers of confinement and these layers may be passive or active in nature. As an example, a simplified illustration of a confinement approach is shown in FIG. 3, which provides an example of a two-layer confinement. The example assumes the use of ventilation systems with exhaust from ventilation zones routed through filtration and detritiation systems to support the static barriers provided by the vacuum vessel and building.

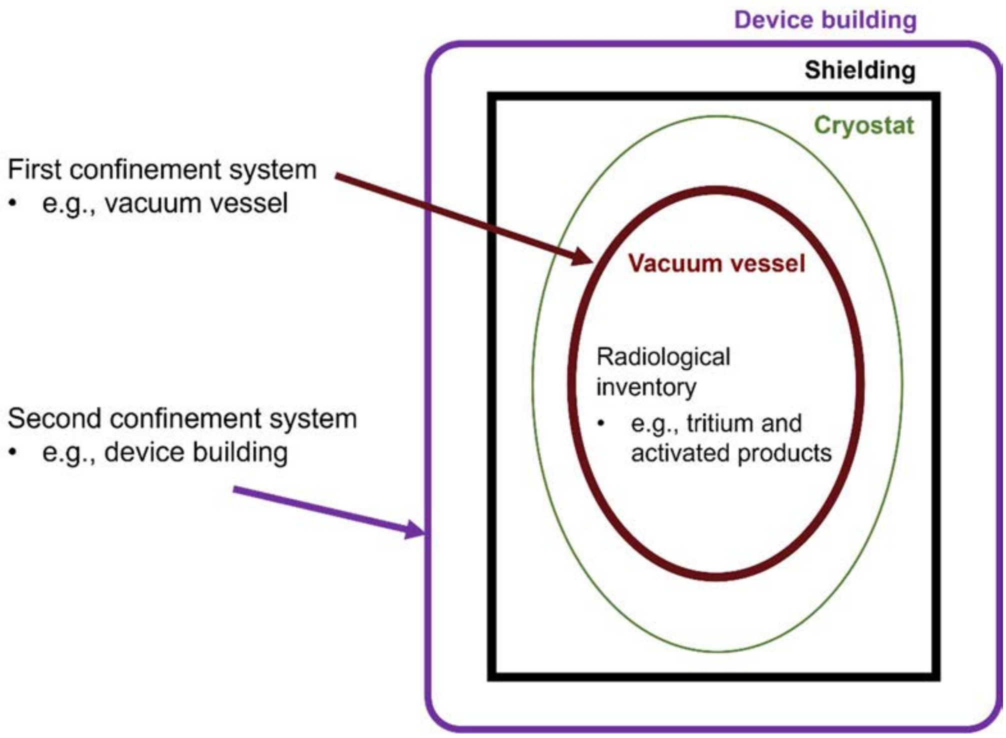

FIG. 3. Illustration of multiple layers of confinement with the vacuum vessel forming the first confinement, and the building around the fusion device as the second confinement. (reproduced from Ref. [21] with permission courtesy of UKAEA).

A strategy for confinement is important not only for the fusion device but for all supporting systems where radiological material is handled or stored, including fuel cycle facilities (see Section 2.4.5).

2.4.7 Radiological hazards during operational and accident conditions

FPPs may present radiological hazards for workers, the public and the environment. These hazards will likely differ in nature and in magnitude for operational releases (including discharges) compared to releases during accidents. A discussion of plant states (i.e. operational states and accident conditions) is provided in Section 3.3.1.

The following subsections provide an overview of some of the main radiological hazards for operational states and accident conditions based on the information received. The sizes and types of hazards varies between FPP concept type and will be influenced by design (particularly materials selection) and operational choices.

2.4.7.1 Radiological hazards for workers in operational states

Operating FPPs will generate radiation that can be a hazard to workers. Developers of FPP concepts that produce neutrons from the main fusion reaction and/or side reactions (see Section 2.4.4) will need to consider the production of neutrons, gamma rays and X rays. Hazards associated with neutron fluxes (direct radiation and activated materials) will be most significant for D-T FPP concepts but still have to be considered for lower neutronicity concepts where neutrons could be produced through side reactions. For some FPP concepts using alternative fuels such as p-^{11}B, the key radiological hazards for workers will be X rays and gamma rays. A discussion of the radiation protection considerations for radiation (e.g. shielding) is given in Section 3.7.

Activated materials and radioactive products represent another radiation and contamination hazard that needs to be considered. Developers of FPP concepts using or producing tritium will need to consider the mobility and different forms of tritium (see Section 2.4.4). It can be challenging to estimate the relative proportions of HT and HTO in any release. Given its mobility and the ease with which it is absorbed into the human body, it is often conservative to assume a high proportion of the tritium is in HTO form. Work is ongoing in the field to develop a better understanding of the HT to HTO conversion process. Many of the FPP concepts include ventilation systems with moisture control to limit the conversion of HT to HTO, along with monitoring systems to evaluate if additional protection is needed to access certain areas for maintenance or inspection activities (see Section 3.7).

For FPP concepts that will produce neutrons, some materials may become activated. Some of the key hazards to consider for workers in these concepts are as follows:

- Gamma radiation emitted by activated materials within device components, structures and shielding.
- Where present, activation of cooling water to create gamma-emitting nitrogen-16 and possibly other radionuclides.
- Radiation from the accumulation of activated dusts, most, if not all, of which remain in-vessel and therefore are likely to pose only limited risk to workers during operation. There is greater risk to workers during maintenance and inspection activities. Hazards include radiation from dusts and the potential for inhalation or for skin deposition of circulating or remobilized dusts.
- Radiation from active corrosion or sputter products circulating in coolant loops (which may be outside the main shielding area), and the accumulation sites, such as activated corrosion product deposits, and filtration systems, particularly if there is a requirement for maintenance or inspection (see discussion on remote maintenance in Section 3.8). In lithium–lead coolants, examples of radionuclides that could be produced under neutron irradiation are tritium, ^{204m}Pb, ^{203}Pb, ^{204}Tl, ^{203}Hg, ^{207}Bi and ^{209}Bi [26]. There is also potential for creating quantities of ^{210}Po due to neutron interactions with lead [27].
- Activated air, due to interactions with neutrons, for example the production of the beta and gamma emitter, ^{41}Ar, with a 1.8-hour half-life.

Radionuclides can also be produced in other FPP concepts, for example, an FPP using the p-^{11}B fuels produces ^{11}C, with a short half-life of about 20 minutes, decaying through positron emission. It may be that the majority of the ^{11}C is embedded within materials and considered immobile, however if the ^{11}C combines with hydrogen then radioactive methane could be produced. Depending on the operating temperatures, much of this may be coated onto the surrounding materials and again be considered immobile in routine operations.

2.4.7.2 Radiological hazards for the public in operational states

During the routine operation of an FPP, the hazards that pose a risk to workers will generally also have the potential to affect the public, although in different ways (see Section 2.4.7.1). Additional considerations for hazards include:

- The location of the site boundary with respect to hazard sources, including radiation from activated coolants, corrosion and sputter products circulating in coolant loops (which may be outside the main shielding area) and at accumulation sites.
- The potential for sky shine[10], including any analysis of off-site effects and optimization of the design to minimize these effects.
- Whether normal seepage and discharges of tritium, activated dusts or other radiological substance are within acceptable limits. This includes considering long term public exposure to tritium and other substances in the environment.
- In routine maintenance activities, there may be potential for raised discharge levels, for example where the vacuum vessel is opened (such as for routine component replacement) there is potential for mobilization of activated products (such as dust). Release pathways and release fractions need to be developed and analysed as appropriate for the various FPP concepts. Example radionuclides in dust could include ^{31}Si for concepts that use a silica material.

2.4.7.3 Radiological hazards for workers in accident conditions

Direct radiation from the fusion reaction process will stop when the fusion reaction ceases (in many FPP concepts, it is assumed that the reactions stop at or before the accident). This means that the hazards in accident scenarios are limited to the release of tritium (if present in significant quantities) and activated materials (dusts, corrosion products, oxides, volatilization products), as well as direct radiation from static activated materials. These are hazards also present during operations, but additional considerations during accident conditions include:

- The potential for mobilization of tritium and releases from the inventory within the confinement, particularly if there is damage to the vacuum vessel (or other primary confinement). Mobilized inventories may be in a different form during accidents compared to operations. Releases could also occur from supporting systems such as tritium process plant.
- The impact of damage to shielding on gamma radiation hazard, including sky shine.
- Releases from damaged pipework carrying activated coolants.
- Releases of activated dusts due to damage to confinement structures. This includes potential for remobilization of dusts that had accumulated inside the confinement.

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[10] Radiation sky shine effects are related to the scatter of radiation with air, with scatter both up and then downwards back towards the ground.

- Degradation in performance of other systems involved in radiation protection such as HVAC filtering.

2.4.7.4 Radiological hazards for the public in accident conditions

Given that most direct radiation from an FPP will cease when an accident occurs (see Section 2.4.7.3), the hazards for the public during an accident scenario are generally similar to those for workers, although the mechanisms and timescales for these hazards may differ (e.g. long term exposure from land or water contamination). The safety assessment is to assess how radioactive substances such as tritium and activated dusts might be dispersed off the site and identify the pathways for their uptake from the environment by the public (see Section 2.4.8).

2.4.8 Exposure pathways and dose calculations

Many factors influence the potential exposure pathways for worker and the public following a release of radioactive material. These factors differ depending on the radionuclide (including its chemical form). An approach to modelling the dispersion of radioactive material via air and water is described in detail in IAEA Safety Standards Series No. NS-G-3.2 [28], and typically accounts for factors such as release elevation, presence of obstructions in the dispersion wake, orographic elevations, and meteorological conditions (based on local meteorological data or investigations). An approach to estimating worker and public doses is provided in IAEA Safety Report Series No. 19 [29], with a compendium of dose coefficients provided by the International Commission on Radiological Protection (ICRP) [30].

2.5 DEMONSTRATING SAFETY

To ensure the safe deployment of FPPs, designers, operating organizations, managers and operating personnel will need to understand the hazards associated with their facility and to demonstrate how those hazards are being managed so that the risk to workers and the public is as low as reasonably achievable (ALARA). Two approaches to achieving safety include designing the facility to meet an appropriate set of standards, requirements and principles, referred to as design safety, and carrying out analysis that shows that the facility can be safely operated, referred to as safety assessment.

Within the nuclear power plant industry, design safety requirements have been developed and documented in IAEA Safety Standards Series No. SSR-2/1 (Rev. 1) [31], Safety of Nuclear Power Plants: Design. Some FPP developers, in the absence of any FPP specific safety standards, have adopted a similar approach through tailored design safety principles based on a number of the NPP general plant design requirements. Some FPP developers have modelled safety requirements based on industrial facilities and standards (e.g. ISO standards) or standards for non-nuclear installations such as particle accelerators. Note that discussion of the existing regulatory frameworks for fusion facilities in Member States will be presented in a separate TECDOC, International Experience in the Regulation of Fusion Facilities, which is currently in development.

Common considerations within design safety can include, for example, categorization of plant states, postulated initiating events (PIEs), internal and external hazards, identification and categorization of safety functions, identification and classification of safety structures, systems and components (SSCs), defence in depth, physical separation and independence of safety systems, common cause failures, single failure criterion, fail-safe design, operational limits and conditions, and equipment qualification. These considerations apply to various facilities handling radioactive material, and not just nuclear power plants.[11]

In accordance with Requirement 1 of IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [32], a graded approach is to be used in determining the scope and level of detail of any safety assessment, consistent with the magnitude of the possible radiation risks arising from the facility or activity[12]. The safety assessment process in GSR Part 4 (Rev. 1) [32] includes: assessment of possible radiation risk, assessment of safety functions, assessment of site characteristics, assessment of the provisions for radiation protection, assessment of engineering aspects, assessment of human factors and assessment of DiD. GSR Part 4 (Rev. 1) [32] includes both deterministic and probabilistic approaches to safety assessment.

Demonstrating that the design safety approach has been suitably implemented in achieving the safety objective (see Section 3.2) is an iterative process and varying levels of effort and detail in supporting safety assessment and analysis may be needed for different FPP concepts. For example, higher neutronicity FPP concepts are likely to involve more supporting analysis than lower neutronicity concepts to demonstrate radiation safety.

In this section, an example of a general and simplified process for demonstrating safety is described in the context of an FPP. This would include aspects such as: design of safety related structures, systems and components; specification of design quality requirements; specification of the limits and conditions of operation; establishment of appropriate continuous review stages; and consideration of aspects over the lifetime of the facility, such as safety in construction and decommissioning[13]. The safety assessment and design safety processes are iterative; initial findings from the analysis are intended to feed back into the design. Licensing and regulatory requirements vary among Member States and can include country-specific design safety requirements and a prescribed scope and content for safety assessments[14].

Some of the steps below may not be applicable, or are only partially applicable, for some FPP concepts. For example, if analysis demonstrates a low hazard potential, then detailed categorization and classification of SSCs may not be proportionate.

The generalized process can be described as follows:

1) Develop a preliminary understanding of the plant design and safety considerations, assuming a level of knowledge of key concept and technology decisions, such as fusion fuels/fusion reaction, plasma confinement technology (i.e. magnetic, inertial), and technologies such as energy extraction systems and breeder blanket materials (if used). An example of how to develop this preliminary understanding includes the following iterative process:

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[11] Necessity and the scope of taking into account such considerations as physical separation and independence of safety systems, common cause failures, single failure criterion and others need to be considered on the basis of the graded approach.

[12] A graded approach is applicable to safety assessment process as well as establishing regulatory requirements. e.g. if the facilities in question did not surpass certain conservative and public risk thresholds, then simplified approaches for regulating them as well as for safety assessment could be applied.

[13] Decommissioning is out of scope of this TECDOC.

[14] Discussion of the existing regulatory frameworks for fusion facilities in Member States is presented in the TECDOC 'International Experience in the Regulation of Fusion Facilities' which is now under development.

- Define any high level safety objectives, such as safety acceptance criteria (see IAEA Safety Standards Series No. SSG-2 (Rev. 1) for guidance on setting acceptance criteria [33]) and develop a high level safety policy document defining the approach to achieving and demonstrating safety.
- Determine what the key radiological hazards are, for example through preliminary hazard identification (HAZID) and functional failure modes effect analysis (FFMEA). These analysis methods will likely be repeated at varying stages of the design safety process, with maturing inputs and design information.
- In view of the hazards, define an appropriate strategy for the demonstration of safety, for example define a set of safety acceptance criteria against which the design can be assessed.

There may be multiple iterations of this process as understanding of the hazards and design matures.

2) Identification and categorization of safety functions, and classification safety structures, systems and components for all plant states. Through development of the plant functional understanding, such as through plant breakdown structures, functional breakdown structures, master logic diagrams (MLDs), etc, along with plant modelling and plant layout information, the internal and external hazards to the plant can be identified. Through a combination of appropriate techniques, such as HAZID, hazard and operability (HAZOP), unmitigated analysis, FFMEA, failure modes and effects analysis (FMEA), event and fault tree analysis:
    - The hazards can be identified and assessed to develop a set of PIEs.
    - From the set of PIEs, the key safety functions can be identified.
    - The SSCs required to perform the safety function can be identified.
    - The safety functions can be categorized, often based on the safety significance of the function.
    - The SSCs required to perform the safety functions can be classified through consideration of the safety function category within which the SSC is performing the function, and taking into account other factors, such as the consequence of failure to perform the safety function.

    There may be multiple iterations of this process as understanding of the hazards and design matures. The suitability of the safety classification is typically verified through safety analysis and if necessary, another iteration of the identification and classification process.

3) Safety analysis to verify that the design meets the safety acceptance criteria, follows the safety objective and safety principles with an adequate safety demonstration is performed through appropriate analysis methods. Safety analysis methods are usually categorized as either deterministic safety assessment (DSA) or probabilistic safety assessment (PSA), and often a combination of the two are used to complement each other.

    Deterministic methods could include, for example, structured fault identification techniques such as HAZOP and FMEA. DSA could also include some analysis of the initiating event frequencies and some probabilistic assessment of overall accident sequence frequencies (such as limited event tree analysis) to provide a probabilistic risk assessment informed support to complement DSA.

Probabilistic methods could include, for example, analysis of specific accident sequences through to the development of a full scope PSA assessing risks to workers and the public (analogous to a full scope, Levels 1, 2 and 3 PSA for an NPP). The level of detail needs to be proportionate to the risk, as per the graded approach to safety demonstration.

Understanding the safety functions and related SSCs necessary to achieve the safety objective is a key part of the safety demonstration. By identifying the safety functions and related SSCs, it is possible to apply a ranking of importance for those SSCs based on concepts such as consequence of failure, frequency of occurrence and risk acceptance, and therefore determine their design quality and related requirements (such as for manufacturing, construction, installation, commissioning, operation, environmental qualification and inspection and maintenance). Identification and categorization of functions and classification of SSCs is an iterative process that may involve review throughout the facility's lifetime. The approach to identification, categorization and classification needs to be consistent with the safety objective and safety principles. An approach to safety classification for mechanical components for fusion devices is proposed in IAEA-TECDOC-1851 [34] and could be used as the basis of a safety classification method more broadly (see Section 3.2.3 for further discussion).

The above approach is similar to that used in the safety analysis of NPPs. However, it is important to ensure that the effort spent on demonstrating safety is proportionate to the hazard potential of each FPP concept and taking account of operating experience from existing fusion facilities, in accordance with the graded approach discussed at the beginning of this section.

3 APPROACHES TO SAFETY FOR FPP CONCEPTS

The information in this section was gathered according to the steps outlined in Section 1.3 of this publication.

3.1 FUNDAMENTAL SAFETY OBJECTIVES, PRINCIPLES AND APPROACHES

The IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [35], establishes the fundamental safety objective and ten safety principles. Section 2 of SF-1 [35] states:

> **"The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation."**
>
> "2.1. This fundamental safety objective of protecting people - individually and collectively - and the environment has to be achieved without unduly limiting the operation of facilities or the conduct of activities that give rise to radiation risks. To ensure that facilities are operated and activities conducted so as to achieve the highest standards of safety that can reasonably be achieved, measures have to be taken:
>
> a) To control the radiation exposure of people and the release of radioactive material to the environment;
> b) To restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation;
> c) To mitigate the consequences of such events if they were to occur."

The fundamental safety objective applies to all circumstances that give rise to radiation risks. The safety principles are applicable, as relevant, throughout the entire lifetime of all facilities and activities (existing and new) utilized for peaceful purposes, and to protective actions to reduce existing radiation risks. Therefore, SF-1 [35] is applicable to FPPs.

It was noted from the contributing experts that, for FPPs, the provisions of para. 2.1(b) of SF-1 [35] are to be interpreted in a general sense given that FPPs do not have a reactor core or make use of nuclear chain reactions.

Based on the information received, as with the fundamental safety objective, the ten safety principles established by the IAEA in SF-1 [35] are also applicable to FPPs, however the measures taken when applying these principles may differ. For example, for the 'prevention of accidents', the DiD principle might be applied to design safety, although the application of the principle in different FPP concepts might differ based on the hazard potential (see Section 3.4).

The ten safety principles as established by SF-1 [35] are listed below:

> **"Principle 1: Responsibility for safety - The prime responsibility for safety must rest with the person or organisation responsible for facilities and activities that give rise to radiation risks."**
>
> **"Principle 2: Role of government - An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained."**
>
> **"Principle 3: Leadership and management for safety - Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks."**
>
> **"Principle 4: Justification of facilities and activities – Facilities and activities that give rise to radiation risks must yield overall benefit."**
>
> **"Principle 5: Optimization of protection - Protection must be optimized to provide the highest level of safety that can reasonably be achieved."**
>
> **"Principle 6: Limitation of risks to individuals - Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm."**
>
> **"Principle 7: Protection of present and future generations - People and the environment, present and future, must be protected against radiation risks."**
>
> **"Principle 8: Prevention of accidents - All practical efforts must be made to prevent and mitigate nuclear or radiation accidents."**
>
> **"Principle 9: Emergency preparedness and response - Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents."**
>
> **"Principle 10: Protective actions to reduce existing or unregulated radiation risks - Protective actions to reduce existing or unregulated radiation risks must be justified and optimized."**

3.1.1 Design requirements

The IAEA safety standards include requirements and recommendations which, if followed, are a means to achieve the fundamental safety objective and safety principles discussed in Section 3.1. These requirements and recommendations cover topics such as the application of DiD, radiation protection, safety classification, determining plant states, identifying PIEs, redundancy, diversity, and protection against internal and external hazards. Whilst there are specific requirements for the design and operation of NPPs [31, 36], research reactors [37] and fuel cycle facilities [38], there are currently no requirements or recommendations specifically for FPPs. The remainder of Section 3 provides discussion on how topics covered by NPP design requirements and safety assessment standards might be applicable to FPPs.

3.1.2 Deterministic and probabilistic analysis

Safety analysis is used to demonstrate the safety of the design and satisfaction of the safety criteria. Requirement 15 of GSR Part 4 (Rev. 1) [32] specifies the use of both deterministic and probabilistic approaches for the safety analysis. Deterministic and probabilistic approaches have been shown to complement one another and can be used together to provide input into an integrated decision making process. The extent of the deterministic and probabilistic analysis carried out for a facility or activity is to be consistent with the graded approach.

The convention of dividing PSAs for NPPs into three levels representing core damage, loss of containment and release to the environment may not be directly applicable to FPPs; some adaptation may be needed, including considering alternate structures, to ensure the level of effort is proportionate to the hazard potential (in line with the graded approach).

The IAEA Nuclear Safety and Security Glossary [4] provides definitions for the following terms:

> "Deterministic analysis **-** Analysis using, for key parameters, single numerical values (taken to have a probability of 1), leading to a single value for the result.
>
> - In the safety of nuclear installations, for example, this implies focusing on accident types, releases of radioactive material and consequences, without considering the probabilities of different event sequences.
> - Typically used with either ‘best estimate’ or ‘conservative’ values, based on expert judgement and knowledge of the phenomena being modelled.
> - Contrasting terms: probabilistic analysis or stochastic analysis.”
>
> “Probabilistic analysis **-** Probabilistic analysis is often taken to be synonymous with stochastic analysis. Strictly, however, ‘stochastic’ conveys directly the idea of randomness (or at least apparent randomness), whereas ‘probabilistic’ is directly related to probabilities, and hence only indirectly concerned with randomness.
>
> “A natural event or process might more correctly be described as ‘stochastic’ (as in stochastic effect), whereas ‘probabilistic’ would be more appropriate for describing a mathematical analysis of stochastic events or processes and their consequences (such an analysis would, strictly, only be ‘stochastic’ if the analytical method itself included an element of randomness, e.g. Monte Carlo analysis).”

“Probabilistic safety assessment - A comprehensive, structured approach to identifying failure scenarios, constituting a conceptual and mathematical tool for deriving numerical estimates of risk.

“Three levels of probabilistic safety assessment are generally recognized:

- Level 1 comprises the assessment of failures leading to determination of the frequency of core damage.
- Level 2 includes the assessment of containment response, leading, together with Level 1 results, to the determination of frequencies of failure of the containment and release to the environment of a given percentage of the reactor core’s inventory of radionuclides.
- Level 3 includes the assessment of off-site consequences, leading, together with the results of Level 2 analysis, to estimates of public risks.”

Deterministic methods could also include some analysis of initiating event frequencies and some probabilistic assessment of overall accident sequence frequencies (such as limited event tree analysis) to provide a probabilistic risk assessment-informed support to complement the DSA.

Most of the contributing experts to this TECDOC envisage FPP developers demonstrating safety using conservative deterministic approaches, although some will include elements of probabilistic analysis.

There were some limitations identified around the use of safety analysis methods for FPPs, including:

Limitations for the DSA with conservative approach:

- Conservative estimation of consequences due to use of conservative assumptions made to give confidence that the model has captured unknown risks.

- Difficulty capturing the role of active monitoring or autonomous actions to limit consequences or limit event sequences.

Limitations for the PSA:

- A lack of operational experience in fusion conditions makes estimates of probabilities difficult.

Limitations common to both DSA and PSA:

- Need for commonly accepted ‘best estimate’ weighting factors, such as the rate of conversion of HT to HTO in the case of a release.

3.1.3 Radiological acceptance criteria

Requirement 4 of IAEA Safety Standards Series No. GSR Part 4 (Rev. 1) [32] states:

“**Safety assessment of facilities and activities: the primary purposes of the safety assessment shall be to determine whether an adequate level of safety has been**

achieved for a facility or activity and whether the basic safety objectives and safety criteria established by the designer, the operating organization and the regulatory body, in compliance with the requirements for protection and safety as established in Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series No. GSR Part 3 [39], have been fulfilled."

Among the information gathered from the contributing experts, both quantitative and qualitative radiological acceptance criteria are being used or proposed to be used as a means of achieving the safety objective for FPP concepts. The use of acceptance criteria as part of deterministic safety analysis is discussed in [31]. Common quantitative criteria used are individual worker doses, maximum collective dose, maximum dose rates in different zones of the plant, maximum releases to the environment, maximum predicted individual public dose resulting from an accident and limits for exposure of workers to non-radiological hazards including toxic materials, lasers, magnetic fields and radio-frequency fields (see Section 3.8).

The approach for deriving radiological acceptance criteria may differ for each FPP concept, being influenced, for example, by regulatory requirements and relevant good practice within the State where the FPP is planned to be constructed. For some developers of FPP concepts, numerical values for dose limits are based on an interpretation of existing requirements and recommendations from the IAEA and the ICRP. Some other FPP developers aim to use radiological acceptance criteria defined within the relevant national legal framework. For example, within the EU, radiological acceptance criteria are transposed from the European Union Basic Safety Standards directive, or Basic Safety Standards (BSS) (Council Directive 2013/59/Euratom [40]) and are based on the recommendations and dose coefficients of the ICRP. Article 30 of the Euratom Treaty provides that the 'basic standards' are meant to include "maximum permissible doses compatible with adequate safety" [41]. The BSS lays down basic safety standards and dose limits for this purpose, for the protection of persons subject to occupational, medical and public exposures.

Due to the lack of fusion-specific criteria and considerations, many FPP developers use the same quantitative acceptance criteria as for NPPs. For example, numerical safety criteria based on the safety assessment principles for nuclear facilities [42] are used for JET. Some FPP developers complement the NPP criteria by setting more ambitious, or else complementary, targets, such as:

- No need for sheltering or evacuation following any credible accident scenario.
- No potential for cliff edge effects [15], even following combination events. The occurrence of two concurrent, independent failures should not lead to unacceptable consequences or cliff edge effects.
- Criteria related to limiting the time/cost to restore the environment after an accident and for final decommissioning.
- Criteria for resource consumption efficiency (e.g. freshwater consumption) and impact on environmental receptors.
- Constraints on the amount of radioactive waste after 100 years of radioactive decay (considering recycling, reuse and clearance as much as possible).

[15] Noting that 'cliff edge' effect may have different meaning between nations and FPP concepts, but in general is used to describe an abrupt transition of the plant state.

- In addition, some FPP concepts envisage establishing quantitative acceptance criteria based on the dose rate and probability of an event leading to exposure.

3.2 SAFETY FUNCTIONS AND ASSOCIATED SYSTEMS

3.2.1 Identification and definition of safety functions

Requirement 7 of GSR Part 4 (Rev.1) [32] states that: "**All safety functions associated with a facility or activity shall be specified and assessed.**"

The identification and definition of SSCs delivering these safety functions, are a key part of designing for safety in an FPP. Consideration of safety functions and SSCs is common among all facilities that use radioactive material and is not exclusive to nuclear installations.

The IAEA Nuclear Safety and Security Glossary [4] provides definitions for the following terms (notes omitted):

> "Safety function **-** A specific purpose that must be accomplished for safety for a facility or activity to prevent or to mitigate radiological consequences of normal operation, anticipated operational occurrences and accident conditions."
>
> "Structures, systems and components **-** A general term encompassing all of the elements (items) of a facility or activity that contribute to protection and safety, except human factors."

The majority of contributing experts, proposed FPP safety functions derived from the provisions of the fundamental safety functions presented in SSR-2/1 (Rev. 1) [31] but with some interpretation for FPPs. Many stated that the following safety functions are in line with the definition provided in IAEA-TECDOC-1851 [34]:

- "The confinement of radioactive material: to prevent mobilization and dispersal of radioactive material within the plant, and the avoidance of the leakage of any part of this radioactive inventory to the environment.
- "Limitation of exposure to ionizing radiation: to minimize occupational radiation exposure of personnel arising from radiation from all radiation sources including secondary activation and mobile source terms."

Some contributing experts also consider the removal of heat as a safety function depending on the design characteristics, as the temperatures reached in case of loss of the cooling function may differ. Factors to consider include the residual and decay heat from the components, cooling system design and whether a passive system can be implemented (e.g. using natural convection). Some contributing experts consider heat removal to be a supporting function to help protect the confinement systems from damage. Determining the need for heat removal systems is likely to involve thermohydraulic analysis.

The control of plasma[16] is considered a safety function for some FPP concepts. This is because, for some designs, the consequence of plasma disruption or uncontrolled events may lead to

[16] Some of the questionnaire responses, referred to the control of the fusion reaction: however, based on the information provided, this is better termed 'control of plasma', as there are additional plasma related factors that affect the production of plasma disruptions, not just the fusion reaction.

unacceptable damage to the systems and plasma-facing components of the vacuum vessel, potentially compromising the performance of the confinement. Some plasma events may also contribute to the production of mobilizable inventory such as dusts.

Some contributing experts list additional safety functions related to the control of toxic materials and other industrial hazards.

The safety functions for FPP concepts are often expanded into a range of principal and supporting safety functions (referred to hereafter as Principal safety functions and Supporting safety functions). The supporting safety functions are generally design-specific and are usually determined as part of the safety analysis. There is a relatively wide range of supporting safety functions identified among the FPP concepts considered for this publication.

Many of the supporting safety functions that have so far been identified for the FPP concepts relate to the elimination or mitigation of challenges to the principal safety function 'confinement of radioactive material', with some related to 'limitation of radiation exposure'. Examples of these supporting safety functions (based on a tokamak concept) are provided in Table 2.

TABLE 2. EXAMPLES OF PRINCIPAL AND SUPPORTING SAFETY FUNCTIONS FOR FPPS

Principal safety functions	Supporting safety functions
Confinement of radioactive material	Control of plasma energy Control of thermal energy Control of coolant energy Control of chemical energy Control of magnetic energy Control of electrical energy Control of mechanical energy
Limitation of exposure to ionizing radiation	Limitation of exposure to workers Limitation of exposure to the public Limitation of airborne and liquid releases to the environment

3.2.2 Structure, systems and components

For many of the FPP developers, it is too early in the design process to identify the SSCs necessary to fulfil the safety functions, however example tables of main SSCs, functions, subsystems and components and related design requirements are given in Annex I. Any numerical values provided in Annex I may evolve as the designs progress.

For the example of ITER, the vacuum vessel and buildings are considered as part of the physical confinement barrier, while monitoring, control and cooling systems provide the safety functions to preserve normal operation and limit the escalation of an anticipated operational occurrence to an accident scenario (for example, radiation monitors will send a signal to automatically shut down the plasma on detection of high radiation levels). Note that some cooling systems may not be considered safety-related if a loss of cooling does not contribute to a release of radioactive material.

Examples of SSCs for fusion technologies and their main functions to achieve safety are given below:

- Blanket – to provide shielding for components against heat and radiation;
- Vacuum vessel – to provide confinement and shielding for components;
- Primary cooling systems – to remove heat from the blanket (and divertor, where applicable), and provide confinement;
- Tritium cycling system – to control tritium emissions, and provide confinement;
- Diagnostic and control systems – to provide accurate measurements of plasma behaviour and, for some FPPs, to detect abnormalities and initiate mitigating systems[17];
- Vacuum vessel pressure suppression system – to limit the pressure in event of an in-vessel loss of coolant accident (LOCA), and to provide confinement;
- Limiters – to reduce neutron flux and contribute to neutron/gamma radiation shielding;
- Remote maintenance systems – to facilitate replacement of components in high dose rates;
- Active maintenance systems – for storage of radioactive components, maintaining a negative pressure, providing confinement and radiation protection;
- Radioactive waste treatment and storage – for decontamination, maintaining a negative pressure and providing confinement.

For magnetic FPP concepts, SSCs may include:

- Divertor – to provide heat extraction and contribute to neutron/gamma shielding;
- Magnet systems – to confine and stabilize the plasma;
- Thermal shields – to limit heat load to the superconducting coils;
- Cryostats – to provide a vacuum to avoid excessive thermal loads.

For inertial FPP concepts, SSCs may include:

- Rotating or moving shields – to protect the target injection system and/or allow insertion/removal of batches of targets or exchange of final optics;
- Optical windows on the vacuum chamber – to enable transmission of the lasers and provide a seal for tritium confinement.

SSCs supporting FPP are likely to have many design requirements covering material properties, performance, reliability and other aspects. Among the contributing experts, the following were identified as aspects typically covered by design requirements:

- Structural strength;
- Seismic resistance;
- Prevention of contamination spread;
- Pressure tightness;
- Leakage limits;
- Fire prevention;
- Radiation shielding;

[17] For example, some FPP concepts may include a fusion power shutdown system through gas injection, and/or a disruption mitigation system through cryogenic pellet injection

- Emission reduction.

3.2.3 Safety classification of structures, systems, components

Safety function categorization serves to identify those functions most important to achieving the safety objective and development of a balanced approach to DiD (described further in Section 3.4). SSCs that are relied on to fulfil a given safety function are assigned a classification in line with the importance of the safety function and the importance of the SSC in fulfilling that safety function. The process ensures that SSCs are designed, manufactured, installed, commissioned, operated and maintained to a level commensurate with their safety importance.

Most of the contributing experts have not yet got to a stage where they have defined the safety classification of SSCs, and based on the information received for this publication, there is no identified documented process for which the identification and classification of SSCs is performed. ITER is an example of a large fusion experimental facility for which the safety categorization of functions and the classification of SSCs has been performed. Most of the participating FPP developers considered it to be too early in the design to identify and classify SSCs. Some FPP developers either follow, or plan to follow, an approach similar to that outlined in IAEA-TECDOC-1851 [34]; a publication that was significantly influenced by both the ITER design and the IAEA Safety Standards Series No. SSG-30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants [43]. Whilst SSG-30 [43] provides guidance on the safety classification of SSCs in NPPs, and some of the principles could be relevant for FPPs, many of the safety-related functions and failure consequences are not applicable to fusion (i.e. those related to reactivity control and core melt). IAEA-TECDOC-1851 [34] provides information for the safety classification of mechanical components for fusion applications, limited in scope to magnetically confined tokamaks, although the principles for classification could be relevant for other SSCs and FPP concepts.

IAEA-TECDOC-1851 [34] specifies three classes of 'safety important component' (SIC): SIC-1 is the highest safety class, SIC-2 the intermediate safety class and SIC-3 the lowest safety class. These three classes are similar to the safety classes 1, 2 and 3 specified in SSG-30 [43] for NPPs, although their design standards may differ given the different hazard potential of NPPs compared to FPPs.

The classification schemes outlined by some of the contributing experts were broadly similar, in that components are classified in decreasing order of importance based on the consequences associated with their failure, i.e. class 1 components are those whose failure has the highest potential consequences.

As an example, the top-level criterion used to assign each SSC to a safety importance class in EU DEMO is as follows [44]:

- SIC-1: its failure could lead to an event with consequences exceeding the limits set out in the EU DEMO Plant Safety Requirement Document.
- SIC-2: it is needed to prevent, detect or mitigate an incident or accident, although not needed to reach a safe state.
- SIC-3[18]: although not needed to prevent, detect or mitigate an incident or accident, the SSC can help to further reduce the consequences of such an event.

[18] This is called 'safety related' (SR) for EU DEMO.

At ITER, in accordance with the French regulation for civil land-based nuclear installations, components for protecting public safety, health and sanitation, the protection of nature and of the environment are relevant for classification under regulatory requirements and are referred to as 'protection important components' (PICs). PICs cover both radiological and non-radiological risks for the public, as well as environmental discharges. The provisions to be implemented are commensurate to their role for the safety. As part of the PIC, SIC include all structures, equipment, systems, materials, components and software that implement a safety function necessary to keep the device within the general safety objectives and contributing to prevent, detect or mitigate incidents or accidents (SIC-1 and SIC-2). Additional information on how this relates to the licencing requirements for ITER is given in Annex II.

In at least one Member State, the national nuclear regulatory guide separately classifies preventive and mitigative SSCs with three safety classes PS-1, PS-2 and PS-3 for preventive systems and MS-1, MS-2 and MS-3 for mitigative systems.

Some of the key details provided by the concepts are listed in Annex I. This includes an example of a categorization and classification scheme being applied to UKAEA fusion new build projects.

3.2.4 Operational limits and conditions

Operational limits and conditions are identified as part of the safety demonstration and ensure that the plant is operated within its safe working envelope. Working within the safe working envelope enables the plant to respond effectively to initiating events within the design basis. Operating limits and conditions are derived from the plant safety demonstration and relate to normal operational modes, including startup, power operation, plasma-off and maintenance, and need to be defined to ensure that safety systems can perform the function when needed.

In accordance with Requirement 28 of SSR-2/1 (Rev.1) [31], NPPs are required to establish a set of operational limits and conditions for safe operation. Paragraph 5.44 of SSR-2/1 (Rev.1) [31] states:

> "operational limits and conditions…shall include:
>
> (a) Safety limits;
> (b) Limiting settings for safety systems;
> (c) Limits and conditions for normal operation;
> (d) Control system constraints and procedural constraints on process variables and other important parameters;
> (e) Requirements for surveillance, maintenance, testing and inspection of the plant to ensure that structures, systems and components function as intended in the design, to comply with the requirement for optimization by keeping radiation risks as low as reasonably achievable;
> (f) Specified operational configurations, including operational restrictions in the event of the unavailability of safety systems or safety related systems;
> (g) Action statements, including completion times for actions in response to deviations from the operational limits and conditions."

This list provides a possible starting point for developing limits and conditions for FPPs. These operational limits and conditions will be developed as part of the safety demonstration process (see Section 2.5), reflecting provisions made in the plant safety design. Where relevant, the operating limits and conditions may need to take account for any operator actions needed to ensure safety. Limits and conditions are used to establish operating rules and operating instructions.

At ITER, the central solenoid will induce most of the magnetic flux change needed to initiate the plasma, generate the plasma current, and maintain this current during the burn time. To maintain safety, the electrical current and the discharge profile is not to exceed 15MA of plasma current; this is an example of an operating limit that might apply for an FPP.

3.2.5 Inherent and passive safety

Paragraph 5.8 of SSR-2/1 (Rev. 1) [31] states:

> "The expected behaviour of the plant in any postulated initiating event shall be such that the following conditions can be achieved, in order of priority:
>
> (1) A postulated initiating event would produce no safety significant effects or would produce only a change towards safe plant conditions by means of inherent characteristics of the plant.
> (2) Following a postulated initiating event, the plant would be rendered safe by means of passive safety features or by the action of systems that are operating continuously in the state necessary to control the postulated initiating event.
> (3) Following a postulated initiating event, the plant would be rendered safe by the actuation of safety systems that need to be brought into operation in response to the postulated initiating event.
> (4) Following a postulated initiating event, the plant would be rendered safe by following specified procedures."

All fusion devices have the inherent safety feature of ceasing to operate upon a breach of the vacuum vessel (as fusion conditions are no longer sustained). This feature is discussed in Section 2.4.1.

Some FPP developers make use of the design principle of 'passive safety'. Prioritizing passivity would mean, for example, ensuring the confinement function by using a permanent, static barrier rather than by the closing of an isolation valve. Passive safety may play a role in mitigating certain accident sequences. For example, heat removal using passive rather than active cooling systems is preferred. Passive cooling relies on heat transfer by conduction and radiation, and coolant circulation by natural convection, gravity, or evaporation and condensation. A strong demonstration may be necessary to provide confidence that natural circulation of coolant (and other passive, as opposed to inherent, safety features) will be effective in all credible accident scenarios for which it is relied on in the safety analysis.

3.3 POSTULATED INITIATING EVENTS AND ACCIDENT SCENARIOS

3.3.1 Plant (facility) states and related safety criteria

The IAEA Nuclear Safety and Security Glossary [4] contains definitions of a range of plant states (for nuclear power installations) and facility states (for research reactors and nuclear fuel cycle facilities) from normal operation through to design extension conditions (DECs). The characteristics of fusion facility states more closely match those a nuclear fuel cycle facility, so the term 'facility states' is used hereafter.

All contributing experts describe multiple facility state categories (a minimum of three) in their safety assessments covering normal operation through to accident conditions. Some of the contributing experts subdivide facility states more than others, for example by frequency of event (examples of additional breakdown of the categories can be found in Annex I).

Generally, the use of facility states among the contributing experts was found to align with that in the IAEA Nuclear Safety and Security Glossary [4] for nuclear fuel cycle facilities. Different names were used for some states, but the underlying meaning may be the same as provided in Fig. 4.

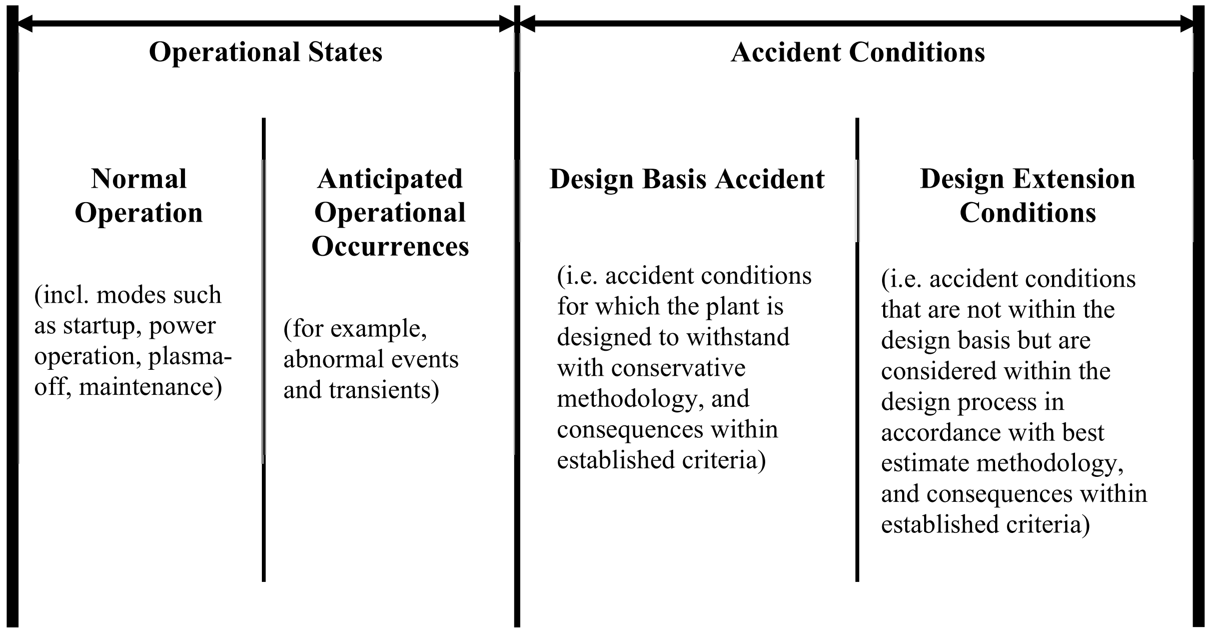

FIG. 4. Facility states considered in design for an FPP (adapted from [4]).

Among the contributing experts, the following facility state model was found to be generally applicable:

- **Normal operation:** The specific modes considered as being normal operation may differ among the FPP concepts, but in general includes operational modes such as: startup, power operation, plasma-off and maintenance.
- **Anticipated operational occurrence**: Deviations from normal operation could be abnormal events, transients, anticipated operation occurrences or off-normal conditions that do not lead to an accident, sometimes referred to as an 'incident'.
- **Accident conditions:** Most contributing experts consider the PIEs that result in any level of negative impact on physical barriers important to safety to be accident condition facility state categories. An example of such a PIE might be an ex-vessel LOCA from the primary cooling loop of the vacuum vessel. There are some differences amongst the contributing experts as to what accidents are considered within the design basis. The concept of DECs was introduced in the IAEA safety standards to encourage assessment of an NPP's ability to cope with less frequent postulated events that are considered beyond the design basis accidents. Two subcategories of DEC are defined [4]: those without significant fuel degradation, and those with core melting. In an FPP there is no core melt scenario. Instead, the assessment of DECs for FPPs might be to demonstrate

that the design can cope with common cause or multiple failure events considered to be beyond the design basis accidents. Some of the contributing experts refer to 'hypothetical bounding event', which are the lowest frequency/highest severity scenarios considered as part of the beyond design basis assessment. For example, the safety demonstrations for some FPP concepts include best estimate analysis of an event that leads to a significant impact to all credited confinement structures.

Safety criteria

Among the contributing experts there is no common approach to the definition of safety criteria and for some developers, such criteria are yet to be established. Some concepts describe the plant states though an assigned frequency of event, in accordance with IAEA-TECDOC-1851 [34]. These types of quantitative criteria are defined such that as the frequency of the event for each plant state increases, the acceptable dose will decrease.

Some of the contributing experts have defined safety objectives for accident consequences. For example, EU DEMO has the following safety objectives for design basis accidents [24]:

- No need for immediate or delayed public protection measures in nearby homes (neither sheltering nor evacuation);
- No need for restrictions on vegetable or animal consumption in nearby agricultural areas.

ITER has a similar objective that no design basis accident will necessitate any off-site response such as public sheltering or evacuation. The accident scenarios considered in the ITER preliminary safety report [25] are estimated to lead to off-site doses of a few mSv or less. ITER does allow for off-site response such as sheltering for its hypothetical ($<10^{-6}$/yr) scenarios, but any such response needs to be limited in duration and area of coverage.

Additional information on the acceptance criteria and related considerations for ITER can be found in Annex II.

Maintaining a safe state

Maintaining a safe state will depend largely on either preserving or, if damaged, re-establishing the systems of confinement. This is common across the concepts, although a precise definition of a 'safe state' is not always provided.

Some examples of systems that might be needed to achieve a safe state would be systems provided for emergency plant shutdown, emergency heat removal, control of coolant enthalpy, controlling the energy of the magnetic field and radiation monitoring.

3.3.2 Practical elimination by design of conditions that can lead to early radioactive release or a large radioactive release

The IAEA Nuclear Safety and Security Glossary [4] contains definitions for the following terms:

- "Early release of radioactive material - A release of radioactive material for which off-site protective actions are necessary but are unlikely to be fully effective in due time."

- “Large release of radioactive material – A release of radioactive material for which off-site protective actions that are limited in terms of times and areas of application are insufficient for protecting people and the environment.”

IAEA Safety Standards Series No. SSG-88, Design Extension Conditions and the Concept of Practical Elimination in the Design of Nuclear Power Plants [45] provides the following definition:

- “Practical elimination – The concept of practical elimination applies to plant event sequences that could lead to unacceptable consequences (i.e. an early radioactive release or a large radioactive release) that cannot be mitigated by reasonably practicable means. Practical elimination implies that those plant event sequences have to be demonstrated to be either physically impossible or, with a high level of confidence, extremely unlikely to arise by implementing safety provisions in the form of design and operational features.”

The concept of ‘practical elimination’ arose from the desire to preclude major impacts in the environment and the population from the occurrence of the most severe types of accidents for NPPs and is intended to complement the defence in depth concept. These most severe accidents involve large releases of energy leading to critical damage to containment (or other confinement) structures and resulting in large releases and early releases of radioactive material. Although primarily developed for NPPs, the concept of practical elimination can be applied to the design safety of FPPs. Generally, the safety demonstrations for FPPs need to include an assessment of plausible event sequences leading to critical damage of barriers performing the confinement function and assess how their failure might result in large releases and early releases of radioactive material for their proposed design. Where they are considered credible, a strategy is needed for demonstrating their practical elimination. For example, similarly to NPPs, an assessment of bounding case scenarios (i.e. rupture of major equipment containing radioactive material, e.g. reactor pressure vessel in NPPs) could be considered as an appropriate initial approach.

For D-T FPP concepts, many of the contributing experts considered a large external event, such as an earthquake, as the potential starting point for an accident that might lead to an early and/or large release of radioactivity. An internal event that could lead to a similar outcome, considered by some of the concepts, is the buildup of an explosive mixture of gas and then its detonation with an explosion of sufficient magnitude such that it significantly damages the systems of confinement barriers.

For low neutronicity FPP concepts, there were no reported events (from the information received) that could lead to an early and/or large release of radioactivity. This is due largely to the predicted low inventories of radioactive material in such FPP concepts as they use alternative fusion reactions to D-T FPPs.

Some FPP developers stated that they were not yet at the stage of identifying events that may lead to an early and/or large release of radioactivity.

3.3.3 Postulated initiating events

In accordance with Requirement 4 of GSR Part 4 (Rev. 1) [32], the safety assessment for NPPs is required to include anticipated operational occurrences, accident conditions and also address any failures that might occur and the related consequences. In accordance with Requirement 16 of SSR-2/1 (Rev. 1) [31], safety assessments for NPPs are required to apply a systematic approach to identifying a comprehensive set PIEs.

Several of the participating FPP developers are not at a level of design maturity that allows for detailed identification of PIEs. One of the FPP developers assessed a worst case hypothetical event instead of carrying out a detailed PIE study, and from this it was decided that further PIE analysis was not necessary at this time. Of the FPP developers and other contributing experts that have performed some work in this area, there are some common themes. For example, there is a common approach to the identification of PIEs through techniques such as FFMEA and, as the design progresses, use of FMEA and HAZOP studies. Once PIEs are identified they are categorized by likelihood of occurrence (see Section 3.3.1 on plant states).

Most contributing experts also recognize that there will be a very large number of PIEs and that event tree analysis, often combined with a top-down approach such as MLDs, will be used to derive a reference set of worst case accident sequences. This reference set is used as the basis for the design of SSCs.

The following list is a typical reference set for an FPP (not intended to be comprehensive):

- Loss of flow in the primary cooling loop of the vacuum vessel;
- Small or large ex-vessel LOCA from one of the primary cooling loops;
- Loss of heat sink in all primary cooling circuits;
- Complete loss of power (total blackout);
- Loss of confinement in rooms permanently contaminated;
- Hydrogen isotope leaks inside rooms leading or not to an explosion;
- Hydrogen and/or dust explosion in vacuum vessel or in other process systems;
- Fire from combustible materials in various rooms (with a notable scenario being a fire from a chemically reactive coolant/breeder such as liquid lithium metal);
- Primary to secondary leaks;
- Steam line break;
- Maintenance accident scenarios (e.g. in shutdown states) leading to confinement losses.

For magnetic FPP concepts, the PIE reference set may also include:

- Plasma events with specific reaction forces on supports or on penetrations;
- Plasma event leading to a loss of vacuum accident (LOVA) and/or an in-vessel LOCA;
- Large cryogenic line break inside rooms;
- Quench of superconducting magnets.

For inertial FPP concepts, the PIE reference set may also include:

- Rupture of the first wall/blanket system and/or vacuum chamber windows due to penetration by a ‘dud’ target, leading to a LOCA/LOVA;
- Disruption of the chamber gas handling system due to buildup of reactive or flow-restricting compounds from unwanted chemical reactions between target by-products.

An example PIE identification process is given for ITER in Annex II. This process determined a set of PIEs and then classified them as either an incident or an accident based on likelihood of occurrence. Some PIEs were combined with an ‘aggravating’ (additional) failure and/or a loss of power event. The determination of the PIEs starts from a reference list obtained by expert judgement and by considering the output of HAZID studies. The classification of the reference sequences is mainly based on the use of deterministic rules and comprehensive, bottom-up methods (i.e. starting from individual component failures or part-failures) such as FMEA. These were supplemented by top-down, plant-level analysis such as MLDs to check the completeness of the FMEA results. From this, ITER identified a list of reference PIEs and their plant state categories, as documented in the preliminary safety report [25] (a review of these accidents is in progress as part of ongoing ITER efforts).

Other examples of PIE reference lists are provided in Annex I.

3.4 APPLICATION OF DEFENCE IN DEPTH

3.4.1 Concept of defence in depth

Principle 8 of SF-1 [35] states that "**All practical efforts must be made to prevent and mitigate nuclear or radiation accidents.**" SF-1 specifies that DiD is the primary means of preventing and mitigating the consequences of accidents. SF-1 also identifies that protection for a defence in depth level can be provided by one or more barriers.

The general concept of DiD, i.e. to use independent layers of provisions and measures to prevent and mitigate accidents, is considered in some form by all FPP developers that contributed to this publication. However, some contributing experts mention that the concept of DiD might be applied in a different way as the five layers set out in IAEA SSR-2/1 (Rev. 1) [31] for NPPs. A graded approach to the application of DiD is envisaged by some of the contributing experts; it was suggested that FPPs could be categorized based on the potential for on-site and off-site consequences, and that this could be used to determine the approach to DiD.

3.4.2 Applications of defence in depth

For NPPs, the five levels of DiD are defined in IAEA SSR-2/1 (Rev. 1) [31]. An excerpt is provided below:

> **Level 1**
>
> "The purpose of the first level of defence is to prevent deviations from normal operation and the failure of items important to safety. This leads to requirements that the plant be soundly and conservatively sited, designed, constructed, maintained and operated in accordance with quality management and appropriate and proven engineering practices. […]"
>
> **Level 2**
>
> "The purpose of the second level of defence is to detect and control deviations from normal operational states in order to prevent anticipated operational occurrences at the plant from escalating to accident conditions. This is in recognition of the fact that postulated initiating events are likely to occur over the operating lifetime of a nuclear power plant, despite the care taken to prevent them. […]"
>
> **Level 3**
>
> "For the third level of defence, it is assumed that, although very unlikely, the escalation of certain anticipated operational occurrences or postulated initiating events might not be controlled at a preceding level and that an accident could develop. In the design of the

plant, such accidents are postulated to occur. This leads to the requirement that inherent and/or engineered safety features, safety systems and procedures be capable of preventing damage to the reactor core or preventing radioactive releases requiring off-site protective actions and returning the plant to a safe state."

Level 4 (without footnotes)

"The purpose of the fourth level of defence is to mitigate the consequences of accidents that result from failure of the third level of defence in depth. This is achieved by preventing the progression of such accidents and mitigating the consequences of a severe accident. The safety objective in the case of a severe accident is that only protective actions that are limited in terms of lengths of time and areas of application would be necessary and that off-site contamination would be avoided or minimized. Event sequences that would lead to an early radioactive release or a large radioactive release are required to be 'practically eliminated'."

Level 5

"The purpose of the fifth and final level of defence is to mitigate the radiological consequences of radioactive releases that could potentially result from accidents. This requires the provision of adequately equipped emergency response facilities and emergency plans and emergency procedures for on-site and off-site emergency response."

Below is an example of interpretation of how to apply DiD to FPPs based on the information received from the contributing experts. Further examples are provided in Annex I. Interpretations of the levels of DiD differ somewhat from those usually seen for NPPs:

Level 1

- Minimize potential accident initiators by reducing the inventory of radioactive and hazardous materials and hazard sources;
- Incorporating 'fail safe' designs, where possible, including passive safety features;
- Using applicable design codes and standards by either modifying existing codes or specially developing new codes;
- Providing a robust safety design that can tolerate failures of novel and first-of-a-kind components.

Level 2

- Safety system reliability through redundant and diverse systems as necessary;
- Providing access control and airlocks to protect site personnel;
- Providing safety parameter monitoring for operating personnel and minimizing operator actions in response to safety parameter monitoring;
- Undertaking periodic inspection, testing, and/or continuous monitoring of safety-related systems, structures and components.

Level 3

- Providing confinement systems to protect all radioactive and non-radioactive sources, while utilizing reliable passive barriers where possible;

- Providing a pressure suppression system to protect the vacuum vessel, which is the first confinement barrier;
- Providing free volumes to protect the other confinement barriers against overpressure events;
- Providing filters and detritiation systems to limit releases to the environment;
- Limiting the consequence of an accident by providing an off-normal rapid shutdown function for the plasma.

Level 4

- Placing the radioactive and non-radioactive sources in a safe state;
- Directing any releases that do occur towards controlled and monitored discharge points;
- Utilizing operator actions to further mitigate consequences, monitor plant conditions and validate system response.

The contributing experts did not identify any specific measures at Level 5. However, one experimental facility operating organization noted they had a high level off-site plan under national radiation protection regulations. Among the contributing experts there was general agreement that the likelihood of a Level 5 response being needed is low, however off-site prevention measures such as short term sheltering or food production bans may be needed over a limited physical area for some FPP concepts.

Some fusion developers propose an approach in which the focus is placed on Levels 2–4 (detect and control) for an early FPP, as opposed to Level 1 (prevention of failure). This was in part influenced by the inherent safety feature of fusion facilities to shut down when abnormal conditions occur (see Section 2.4) and because claims made on components, materials and failure rates for Level 1 components may be burdensome to substantiate with the currently limited reliability data for fusion environments.

Furthermore, some of the contributing experts are investigating whether the implementation of DiD could follow a graded approach by applying safety categories to different FPP designs. These categories generally would depend on the intrinsic hazard potential of the FPP.

3.4.3 Confinement barriers

It is recognized in SF-1 [35], and throughout GSR Part 4 (Rev. 1) [32], that protection through multiple confinement barriers is an important concept. Many of the contributing experts do not yet have sufficient information on the FPP designs to identify the structures and systems providing confinement functions. Most contributing experts envisage FPP concepts making use of a combination of static physical barriers and dynamic systems (e.g. detritiation system and pressure suppression system) to support the confinement function.

For example, in EU DEMO static barriers will provide the principal contribution to the confinement safety function, whilst redundant active systems are provided for increased reliability. This approach generally applies not only to the tokamak but also to other EU DEMO facilities such as the active gas handling system.

For ITER, an extensive list of safety important components contributing to the confinement safety function is presented in the preliminary safety studies (i.e. many ‘SIC’ of different class – see Section 3.2.3). For further details, see ANNEX II.

The structures and systems that comprise the confinement barriers differ amongst FPP concepts. Some FPP developers seek to have the first confinement barrier close to the fusion reaction, for example the barrier could be the vacuum vessel itself and any mechanical penetrations such as cooling pipework (this approach likely requires isolation valves important to safety for these penetrations). Others FPP developers aim to place the confinement barrier further away from the reaction, for example claiming the cryostat, the radiation shielding or the structure of the fusion device building for this confinement function. The confinement approach may make use of a combination of these barriers to provide defence in depth.

For some FPP concepts, the shielding needed to protect against gamma radiation will also provide a confinement barrier, in addition to providing mechanical strength. For one of the FPP concepts, the vacuum vessel is the primary means of preventing and limiting releases of tritium. The building and shielding structures do not need to be claimed given the very low doses calculated for the worst case accident (< 1 mSv) and only provide defence in depth for the confinement function.

Based on information provided by the contributing experts, for the EU DEMO project the confinement barriers and supporting systems might include the following.

For the primary confinement barrier:

- Vacuum vessel outer shell;
- Port closure plates;
- Port flanges;
- Primary vacuum pump flange;
- Vacuum vessel pressure suppression system;
- Ex-vessel primary piping for in-vessel component cooling;
- Electron cyclotron resonance heating windows and transmission lines between closure plate and window;
- Ion cyclotron resonance heating windows and transmission lines between closure plate and window (if this heating system is considered);
- Neutral beam injector vessel and bellows to vacuum vessel; port duct (during plasma operation and testing & commissioning) – the use of neutral beam heating in EU DEMO (if this option is retained);
- Neutral beam injection absolute valve (during schedule maintenance and in failed state) – the use of neutral beam heating in EU DEMO (if this option is retained)
- Pellet injector system/gas injector system;
- Transfer cask (during docking);

For the secondary confinement barrier:

- Tokamak complex base;
- Roof of tokamak building and tritium building;
- External walls of tokamak building and tritium building and separating wall between tokamak building and diagnostic building.

3.5 FURTHER DESIGN REQUIREMENTS FOR SAFETY

This section considers the applicability to FPPs of other well-established general design requirements that are typically applied to the design and safety assessment of NPPs. Based on the information received, most of the contributing experts considered the design requirements described in this section to be applicable, albeit applied in a proportionate manner that accounts for the nature and magnitude of the hazards at an FPP. Several of the contributing experts proposed that the application of some of the design principles might not give significant additional benefit in achieving the safety objectives given the low potential for radiological consequence. A probabilistic approach might be used in the future by some FPP developers to assess whether the separation and independence of safety systems needs to be applied, and whether the single failure criterion and analysis of common causes failures is necessary.

3.5.1 Physical separation and independence of safety systems

The application of this design requirement, as used in NPPs, is given in Requirement 21 of SSR-2/1 (Rev. 1) [31] which states:

> "**Interference between safety systems or between redundant elements of a system is prevented by means such as physical separation, electrical isolation, functional independence, and independence of communication (data transfer), as appropriate.**"

An example of how to meet this requirement would be the functional isolation of instrumentation important to safety and control from that of the normal operation, process instrumentation (i.e. separate signal channels appropriately de-coupled and shielded). This could include physical separation between the redundant channels. At ITER, two control rooms will be provided in separate locations with redundant controls to ensure safe operations can be maintained if a hazard or other scenario renders one of the control rooms inoperable.

3.5.2 Common cause failures

The application of this design requirement, as used in NPPs, is given in Requirement 24 of SSR-2/1 (Rev. 1) [31] which states:

> "**The design of equipment takes due account of the potential for common cause failures of items important to safety, to determine how the concepts of diversity, redundancy, physical separation and functional independence have to be applied to achieve the necessary reliability.**"

Among the contributing experts, the following types of common cause failure are being considered:

- Common geographic causes;
- Common equipment modes;
- Common energy supply modes;
- Common hazard causes (internal or external);
- Common operation modes.

For example, EU DEMO will make use of separation and independence, where necessary, to protect some SSCs against common cause failures, e.g. to protect against hazards affecting plant located in the same area.

In the case of ITER, measures are adopted as part of the design to prevent hazard-induced common mode failures in safety important systems. In particular, measures are taken in terms of the arrangement and dimensioning of buildings, systems and components, so as to prevent an internal hazard from causing the simultaneous loss of the redundant components of a safety system or their backup system.

In ITER, the following measures are taken to prevent this type of failure in a redundant system leading to a failure to perform a safety function:

- Spatial separation of redundant safety sets, with active components located in different areas;
- Physical protection such that an internal hazard (fire, flooding, etc.) affecting a room will result, at the most, in a partial failure of the redundant system.

3.5.3 Single failure criterion

The IAEA Nuclear Safety and Security Glossary [4] contains definitions for the following terms:

- "Single failure – A failure which results in the loss of capability of a single system or component to perform its intended safety function(s), and any consequential failure(s) which result from it."
- "Single failure criterion – A criterion (or requirement) applied to a system such that it must be capable of performing its task in the presence of any single failure."

Requirement 25 of SSR-2/1 (Rev. 1) [31] states that "**The single failure criterion shall be applied to each 'safety group' incorporated in the NPP design.**" A safety group is the assembly of equipment designated to perform all actions required for a particular initiating event to ensure that the limits specified in the design basis for anticipated operational occurrences and design basis accidents are not exceeded. This assembly of equipment could be a 'system' – in such a case single failure criterion is applied to a system, or a 'set of systems' – in such a case single failure criterion is applied not to the individual system but to such a set (i.e. only one failure is postulated in the whole of this set of systems).

Through this principle, the design is developed in a way that no single failure could result in a loss of the capability of a safety group (system or set of systems) to perform as intended, unless the time available from onset of the accident would be sufficient for operator actions.

If judged proportionate based on the graded approach, this principle could be applied to the design of FPPs.

3.5.4 Fail-safe design

Systems and components that are 'fail-safe' are designed so that their failure does not prevent the fulfilment of the supported safety function. For the example of EU DEMO, tritium-bearing pipework is provided with redundant, fail-safe isolation valves. At least one of these valves has to close in certain accident conditions (e.g. loss of confinement). Being fail-safe means that if the valves lose power they will close automatically (i.e. they fail to a safe position).

At ITER, the electro cyclotron system is one of the four auxiliary plasma heating systems for plasma startup, central heating and current drive, and magneto hydrodynamic instability control. A fail-safe isolation shutter valve is located in the secondary confinement barrier acting as a safety barrier for fire segregation and confinement. This shutter valve will close automatically in case of loss of power.

3.5.5 Equipment qualification

All contributing experts that provided information on equipment qualification consider that some form of qualification is necessary for equipment important to safety. Equipment qualification for FPPs may differ to that for NPPs, and the codes and standards used to qualify NPP equipment may not always be suitable for an FPP environment. Some of the contributing experts suggest that the qualification requirements need to be tailored to FPP-specific radiological hazards and be commensurate with an equipment's safety classification.

One of the contributing experts considered that much of the approach to equipment qualification for NPPs could be adapted to FPPs using provisions for dealing with novel components normally applied to e.g. research reactors, but that nevertheless some additional standard development may be needed.

Some of the contributing experts use terminology similar to Requirement 30 of SSR-2/1 (Rev. 1) [31] that "**items important to safety…are capable of performing their intended functions when necessary, and in the prevailing environmental conditions, throughout their design life, with due account taken of plant conditions during maintenance and testing**" when emphasizing the need for component qualification.

There are challenges to qualifying novel equipment, with many components needing to operate in complex environments. In addition, qualification through testing will be challenging for large and complex FPP SSCs, for example the vacuum vessel, or the tritium breeder blankets which are part of some FPP concept designs. Therefore, a combination of component testing and design review may well be necessary as a part of the overall qualification approach.

3.6 CONSIDERATION OF INTERNAL AND EXTERNAL HAZARDS IN THE DESIGN

Identification and evaluation of internal and external hazards is an essential part of any safety assessment. Hazards can be a source of initiating events while simultaneously degrading the plant's capability to respond to those events (by causing the failure of equipment needed to mitigate its consequences). Hazards may challenge more than one level of defence in depth. Hazards may also degrade the performance of confinement barriers.

A summary of the internal and external hazards identified within the FPP concept information received is given in the following subsections. This list is not exhaustive, nor specific to a type of FPP, and therefore some hazards will not necessarily be applicable to all concepts. The level to which the FFP concept is designed to protect against these events is planned to be determined by analysis of the consequences (both radiological and non-radiological) of each event (or a combination of events) and the frequency of the event.

For each identified hazard with a potential impact on the plant, appropriate analysis is performed to characterize the bounding effects of the hazard. Where relevant, the SSCs necessary to detect the hazard are identified, as well as any needed to control or mitigate the hazard's effects such that a safe state can be maintained or achieved (as discussed in Section 2.5). The outcome of these analyses may be additional design requirements.

Further examples of the hazards identified can be found in Annex I.

3.6.1 Identification of internal hazards

The following internal hazards were identified with some examples provided where appropriate. The list of credible internal hazards, as well as their relative importance, differs for each FPP concept. Note that many aspects of IAEA Safety Standards Series No. SSG-64, Protection against Internal Hazards in the Design of Nuclear Power Plants [46], can be adapted for FPPs. SSG-64 [46] provides specific recommendations for many of the internal hazards listed below that may aid designers of FPPs to address internal hazards that may occur at their facility.

- **Internal fire**: fires may break out wherever there is a source of fuel, oxygen, heat and an exothermic reaction. Fires might degrade the performance of one or more confinement systems. If uncontrolled, fires have the potential to spread beyond the area of origin such that many SSCs are lost at once, including potentially redundant SSCs supporting the same safety function (i.e. they are a significant source of common cause failures – see Section 3.5.2).

- **Internal explosion**: sources of explosion include flammable gases, combustible dusts (including metallic dusts) and sudden decomposition of ozone into oxygen. The main potential consequences of an internal explosion are the degradation of one or more confinement systems for radioactive or hazardous materials and the degradation of one or more components of a system performing a safety function, due to thermal and overpressure effects.

- **Internal flooding**: sources of flooding are the systems and vessels containing fluids (typically water), for example for cooling or for fire extinguishing. Flooding can cause physical damage to components and structures, as well as the failure of any electrical and electronic equipment that becomes submerged. This could lead to the failure of SSCs important to safety or the release of radioactive or hazardous materials due to the degradation of a confinement system.

- **Internal missile**: generic sources of internal missiles include pressurized (e.g. high energy pipework and pressurized vessel) and rotating (e.g. turbines, fan blades) equipment. Some inertial fusion concepts employ a high velocity (>100 m/s) low mass target (few g) into the vacuum chamber; this could act as a missile hazard with the potential to rupture a confinement barrier or coolant loop. For FPPs operating high-powered magnets, faults of these systems could impart forces onto metallic materials, potentially generating missiles.

- **Pipe breaks**: pipes operating under high pressure and high temperature are subject to potential whipping or fluid ejection if they rupture. It was noted by one of the contributing experts that cooling systems are particularly subject to potential pipe whipping which might result in damage to the pipe itself and nearby SSCs. Although the fusion reaction occurs in a vacuum, coolant systems for FPPs are often water-based and operating at high temperatures and pressures.

- **Release of hazardous substances inside the plant**: there may be some materials which are inherently chemically hazardous (such as beryllium in some breeder blanket concepts, or mercury in some cryopump designs), and those which have potential for producing a chemical energy hazard (such as hydrogen inventories). Other chemical energy hazards can result from reactions, such as beryllium, tungsten or lithium containing materials with air or steam.

- **Electromagnetic interference (EMI):** general sources of internal EMI include motor and generator brush assemblies, faults in electrical equipment such as switchgear. FPPs may operate very powerful magnetic fields, which would be a significant source of EMI. EMI has the potential to affect the performance of electrical and electronic equipment.

- **Structural integrity**: collapsing structures, falling objects and production of dusts are all examples of potential hazards that may need to be considered.

- **Mechanical hazards**: mainly associated with component handling/hoisting and transfer operations, failures in systems and equipment exposed to an internal vacuum or a pressurized atmosphere, or failures in cryogenic systems. The mechanical hazards can cause the degradation of a confinement system for radioactive or hazardous materials, and the degradation of a system performing a safety function.

- **Plasma transients:** associated with uncontrolled plasma performance and can be followed by a plasma disruption. The resulting forces and thermal loads could cause damage to the in-vessel components.

- **Quenching of superconducting magnets**: quenching occurs when some part of a superconductor switches to a normal resistive state, releasing large amounts of heat. Superconducting magnets will contain a relatively large amount of energy and the conversion of the stored energy into heat can lead to significant damage if not properly controlled. Substantial damage to the superconducting coils may occur if an unmitigated quench or electrical short circuit creates significant electric arcing, leading to localized heating within the coil [47]. These events could potentially damage confinement barriers and other nearby components important to safety.

3.6.2 Identification of external hazards

The following external hazards have been identified by the contributing experts:

- Hazards relating to the natural environment:
  - Earthquakes;
  - extreme climatic conditions, notably severe heat, severe cold, snow, wind, tornado, tsunami and lightning;
  - external flooding;
  - external fire;
  - other.

- Hazards relating to human activities:
  - aircraft crashes;
  - hazards associated with the industrial environment and communication routes, primarily external explosions;
  - malicious/terrorist acts (out of scope of this TECDOC);
  - other.

When generating a complete list of external hazards for an FPP, reference might be made to the various IAEA safety standards on design safety, external hazards and site selection for NPPs, including IAEA Safety Standards Series Nos SSR-1, SSG-79, SSG-18, SSG-21, SSG-9 and SSG-35 [48, 49, 50, 51, 52, 53].

3.7 RADIATION PROTECTION

Radiation protection design criteria are aimed at achieving the safety objective established in SF-1 [35] and apply to all circumstances involving radiation risks.

Requirements for the protection of people and the environment from ionizing radiation are established in the IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [39]. The following Safety Guides provide recommendations on meeting these requirements:

- IAEA Safety Standards Series No. GSG-7, Occupational Radiation Protection [54];
- IAEA Safety Standards Series No. GSG-8, Radiation Protection of the Public and the Environment [55];
- IAEA Safety Standards Series No. GSG-9, Regulatory Control of Radioactive Discharges to the Environment [56].

The three basic principles of radiation protection are justification, optimization of protection, and application of dose limits (and are expressed within the Safety Principles 4, 5, 6 and 10 of SF-1 [35]).

For situations involving planned exposure, justification is the process of determining whether the benefits, to individuals, society or environment, of a practice outweighs the harm. For emergency exposure situations, justification is the process of demonstrating protective actions are justified, i.e. do more good than harm, in the context of the protection strategy in the event of emergency exposure.

Principle 5 of SF-1 [35] addresses the optimization of protection, which means determining the appropriate measures needed to ensure that the magnitude of individual doses, number of individuals exposed (workers and public) and the likelihood of exposure are ALARA, after accounting for economic and social factors. Protection of the environment needs to also be considered within the process of optimization. This optimization process includes:

- Evaluating exposures to identify the need for any action;
- Identifying possible protective actions to keep the exposure ALARA;
- Selecting appropriate protective actions (taking into account economic and social factors);
- Implementing the protective actions;
- Regularly reviewing exposures to evaluate whether any changes are necessary.

Dose limits apply to the radiation exposure of workers and the public in planned exposure situations. Dose limits are established by the government or regulatory body and act as legal limits for an individual. Doses to members of the public are usually estimated through simulation predictions of radionuclide transport in the environment and verified by means of source and environmental monitoring programmes, as opposed to individual monitoring as used for occupational exposure.

The key radiological hazards relating to FPPs are described in Section 2. Many of the concepts have tritium hazards. In addition to any radiation from the fusion reaction, any FPP concept that uses a fusion reaction producing neutrons will need to consider the neutron activation of material. The remainder of this section considers the radiation protection aspects in a similar way to Section 2.4.7, looking at protecting workers and the public during operational states and during accident conditions. The information presented is a summary of the feedback gathered from the contributing experts and may not be exclusive or exhaustive.

Many of the provisions in the IAEA Safety Standards Series No. SSG-90, Radiation Protection Aspects of Design for Nuclear Power Plants [57] apply to all facilities that handle radioactive material. In this TECDOC some general points about radiation protection are provided along with some fusion relevant context and examples. Not every example will be relevant to all FPP concepts.

This section provides an indication of radiation protection aspects to consider in early stage FPP design.

3.7.1 Protecting workers during operational states

The following are aspects being considered by the contributing experts as to how to control doses to workers during normal operation and anticipated operational occurrences:

- Control of sources of radiation – e.g. minimize the production of radionuclides (such as through selection of materials, process chemistry);
- Plant layout – e.g. classification of areas and zoning and access control to minimize the time spent by workers in high dose rate areas;
- Shielding – e.g. providing sufficient shielding to optimize doses, including around penetrations;
- System and component design – e.g. avoiding activity traps, providing high reliability components to minimize maintenance, designing for ease of maintenance;
- Remote techniques – e.g. inspection, removal of equipment, in-situ repair;
- Ventilation – e.g. maintaining pressure differentials, filters;
- Handling and storage of radionuclides;
- Monitoring systems – e.g. for dose rates, contamination;
- Task and workstation studies to optimize and limit exposures;
- Education and training – to improve knowledge and understanding of radiological hazards, and how to manage them.

3.7.2 Protecting the public during operational states

Section 2.4.7.2 provides detail on the radiological hazards to the public during normal operation and anticipated operational occurrences. The main concerns are related to discharges of tritium and activated dusts and any potential for off-site radiation (e.g. sky shine). Some aspects being considered by the contributing experts as for how to address radiation protection considerations for the public (and environment) in routine operation are:

- Limiting the inventory, source terms, source reduction – to limit the hazard potential, where possible;
- Setting discharge criteria (limits and design targets) – e.g. agreeing annual discharge limits with the regulatory body;

- Shielding – protecting the public from exposure to external radiation (often already addressed for workers);
- Monitoring programmes – tracking level of radioactivity from discharges and at measuring points close to the site boundary;
- The population distribution and characteristics (e.g. location and habit) – these may influence the design or layout;
- Site restrictions and exclusion zones – considering whether to limit the kinds of activities that could occur close to the FPP;
- Waste production – considering how to minimize the generation of radioactive wastes.

3.7.3 Protecting workers during accident conditions

Appropriate assessment of potential accident scenarios needs to provide magnitudes, locations and possible exposure pathways for workers during such conditions. Aspects to consider may include:

- Assessment of potential accident scenarios and exposure pathways (prevention and mitigation);
- Plant layout – e.g. access routes, locations for response actions and locations for mustering and sheltering;
- System design – e.g. protection systems and mitigation systems;
- Shielding;
- Ventilation;
- Radiation monitoring;
- Emergency operational procedures;
- On-site emergency plans and procedures;
- Training and exercises;
- Personal protective equipment.

3.7.4 Protecting the public during accident conditions

Design targets or acceptance criteria (see Section 3.2) can be used to demonstrate compliance of design based on suitable safety analysis of potential accident scenarios and possible consequences to the public, or where additional protective features may be required. Some of the radiation protection considerations for the public in accident conditions are summarized here:

- Assessment of potential accident scenarios including the source terms, releases and exposure pathways (considering dispersion in environment);
- Design targets for doses to public;
- Off-site emergency plans and procedures;
- On-site and off-site emergency response facilities and locations;
- Emergency planning zones and distances;
- Protection strategy (covering protection systems, prevention and mitigation).

3.8 CONSIDERATION OF OTHER NON-RADIOLOGICAL HAZARDS

Although this publication focuses on radiological (nuclear) safety, the following is a high-level overview of non-radiological hazards to be considered in the design and operation of FPPs.

Some of these hazards may also be a source of PIEs but in this context are being considered for the direct harm they could cause to workers and the public.

3.8.1 Toxic materials

Some of the participating FPP developers, including spherical tokamak for energy production (STEP) and EU DEMO, plan to make use of some materials that are harmful to human health if not correctly handled. For example, due to their useful properties in fusion environments, some of the FPP concepts use beryllium, mercury and a lithium-lead eutectic. It may be that confinement systems for FPPs also have the function to confine such chemically toxic materials, as well as radioactive substances. It's possible that the risk contribution of non-radiological hazards dominates that from radiological hazards during certain operational phases, for example during non-active phases. This is particularly likely for the vacuum vessel materials.

3.8.2 Magnetic field hazards

Most of the currently proposed FPP concepts considered within this publication will generate magnetic fields. As well as being a possible initiator to PIEs (see Section 3.3), strong magnetic fields can pose a hazard to conventional health and safety. For example, any loose ferrous materials may move when subjected to magnetic fields, becoming high energy missiles that could pose a hazard to workers. FPP developers may want to consider design measures and operational controls to eliminate this hazard. Magnetic fields also pose a risk to workers with pacemakers, as they can interfere with their normal operation. FPP developers could adapt existing approaches for industrial facilities to address these hazards.

3.8.3 Other industrial hazards

Except where they could impact radiological safety (in which case, they become internal hazards – see Section 3.6.1), other industrial hazards such as fires, electrical hazards, working with cryogens or working at height or in confined spaces are typically addressed through compliance with industrial safety regulations, risk assessment and access control.

4 SUMMARY OF COMMON ISSUES AND APPROACHES TO FPP DESIGN SAFETY AND SAFETY ASSESSMENT

A large amount of experience in design safety and safety assessment has been accumulated in the design and operation of experimental fusion facilities over the past four decades. The international drive towards commercial fusion power means that fusion technologies are expected to evolve from experimental facilities towards demonstration, prototype and commercial fusion power plants. These demonstration, prototype and commercial power plants, referred to generally within this publication as FPPs, may be more demanding in terms of safety demonstration than the experimental facilities currently in operation. The objective of this publication (see Section 1.2) was to provide insight into how safety may be addressed for FPPs.

As at the time of writing there were no FPPs in construction or in operation, and those being proposed are generally at an early stage of design; the limited experience related to FPPs has been supplemented with approaches being applied on current and near-term experimental facilities, as well the judgement of fusion safety experts (see Section 1.3 for an explanation of how the information supporting this publication has been gathered). Where relevant, commonalities and differences in safety approaches among the different plasma confinement technologies and FPP concept developers have been identified.

Important topics covered by this TECDOC include: the principles of fusion energy, key safety considerations for the design and operation of FPPs and examples of safety demonstration approaches among the Member States, including application of design safety topics such as defence in depth, safety classification, internal and external hazards and radiation protection.

The following is a summary of the commonalities and differences in approaches to design safety and safety assessment from the contributing experts:

- The IAEA Safety Fundamentals, SF-1 [35] are applicable to all facilities and activities using radioactive substances, including FPPs.

- Among the contributing experts, different approaches to the derivation of radiological acceptance criteria were noted. In the absence of fusion-specific acceptance criteria, many of the contributing experts are either using or propose the use of those developed for NPPs. Some of the contributing experts envisage the pursuit of more ambitious goals (or design criteria) including other qualitative considerations, such as specific criteria on ‘no sheltering’ and/or ‘no evacuation’.

- Among the contributing experts, both probabilistic and deterministic tools were highlighted as potentially being part of the safety assessment. Most of the FPP developers envisage using mainly conservative deterministic analysis for their safety demonstration, although some will include elements of probabilistic analysis such as initiating event frequency analysis and event tree analysis. Some limitations were identified by the contributing experts around the use of analysis for FPPs, for example the impact of making overly conservative assumptions and the lack of FPP-specific component reliability data.

- The safety demonstrations for FPPs may use safety functions derived from those in IAEA SSR-2/1 (Rev. 1) [31] (which was intended for the design of NPPs) but with some interpretation or adaptation to suit the needs of FPPs. The safety functions listed in IAEA-TECDOC-1851 [34] (confinement of radioactive material and limitation of radiation exposure), which have been proposed as fusion-specific safety functions, are considered suitable by some of the contributing experts. Some of the contributing experts consider ‘removal of heat’ and ‘control of plasma’ as safety functions where there is potential for the failure to support these functions to lead to damage to confinement structures. Some concepts also include safety functions for the control of non-radiological hazards.

- For many of the FPP concepts, it is too early in the design process to identify and classify the SSCs necessary to deliver the safety functions. Among the FPP concepts considered, physical barriers are the most important category of SSCs in mitigating the impact of accident scenarios. The processes for the identification and classification of SSCs differed among the FPPs concepts. Some FPP developers follow—or plan to follow—an approach similar to that outlined in IAEA-TECDOC-1851 [34], a publication that was significantly influenced by the ITER design, and the recommendations provided in SSG-30 [43]. Note that IAEA-TECDOC-1851 [34] only covers mechanical systems.

- All contributing experts define three or more plant state categories to be used in FPP safety assessment, covering operational states as well as accident conditions. As an example, most of the contributing experts consider any PIE that has (or results in) a negative impact on the performance of physical barriers supporting the confinement function to be an accident condition.

- DECs might be used to demonstrate that the design can cope with common cause failures or multiple failure events considered to be beyond the design basis. The aspect of DECs related to fuel melt events is not applicable to FPPs.

- The safety demonstrations for some of the developers of FPP concepts who contributed to this report have not reached a level of maturity that allows for a systematic identification of PIEs. Of the contributing experts that have performed some work in this area, some common themes are identified. For example, there is a common approach to the identification of PIEs through techniques such as FFMEA and, for more mature designs, FMEA and HAZOP studies. Most of the contributing experts also recognize that there will be very large number of PIEs; consequently, event tree analysis, often combined with the use of a top-down approach such as MLDs, could be used to derive a reference set of worst case accident sequences.

- All FPP developers that contributed to this publication plan to make use of the concept of DiD as part of their safety demonstration, although not necessarily in the same way as for NPPs. Some of the FPP developers are prioritizing DiD Levels 2–4 (detect and control) over Level 1 (prevention of failure) as they believe claims made on DiD Level 1 SSCs may be hard to substantiate given currently limited data in fusion environments and considering the safety characteristics of fusion devices (see Section 2.5). On Level 5 (emergency response) no specific measures were identified by the contributing experts but there is a recognition that although the likelihood of a Level 5 response being necessary is low, off-site prevention measures such as short term sheltering or food production bans may be needed over a limited physical area. Some of the contributing experts are investigating whether the implementation of DiD could follow a graded approach by applying safety categories to different FPP designs. These categories generally would depend on the intrinsic hazard potential of the FPP.

- Based on the information received, most of the contributing experts consider that design requirements, such as physical separation and independence, common cause failures, single failure criterion and fail-safe design, are applicable to FPPs and need to be applied in a proportionate manner, considering the nature and magnitude of the hazards.

- All contributing experts that provided information on equipment qualification consider that some form of qualification is required for all equipment important to safety in FPPs, although not necessarily following the codes and standards used to qualify NPP equipment. Some of the contributing experts suggest that the qualification requirements need to be tailored to FPP specific radiological hazards and be commensurate with an equipment's safety classification to ensure that each item of equipment can meet its required quality and reliability expectations.

- For many of the contributing experts, the key principles to achieving and demonstrating radiation protection, are that risks to workers and the public are ALARA and within dose limits and targets. Many of the recommendations in SSG-90 [57] to protect workers, the public and the environment against radiological hazards, in operational states, accident conditions and decommissioning, are common across all facilities that handle radioactive material, including FPPs.

The IAEA safety standards provide a useful starting point for establishing an appropriate safety framework for FPPs. For example, SF-1 [35] establishes the fundamental safety objective and ten safety principles and is applicable to the whole lifetime of a facility. The General Safety Requirements within the IAEA safety standards will also generally apply. For example, GSR Part 4 (Rev. 1) [32], which establishes generally applicable requirements for the safety assessment of facilities and activities utilizing radioactive substances, and GSR Part 3 [39], which establishes general requirements for the protection of people and the environment from ionizing radiation, are applicable to FPPs, although no in-depth review of these Standards for their applicability to FPPs has been carried out at this time.

A general finding was that although the Fundamental Safety Principles and General Safety Requirements are common to all facilities using radioactive substances, there are fusion-specific characteristics and safety aspects that are not addressed by the IAEA safety standards. It is also clear that some of the IAEA safety standards and other publications developed for NPPs have elements that would not be applicable to FPPs. These findings will be taken forward for consideration as part of future IAEA activities in design safety and safety assessment for FPPs.

The process to develop this publication highlighted the challenges of defining a common safety approach for the diverse range of plasma confinement technologies being proposed for FPPs. It also highlighted the challenge in presenting information relevant for the safety of FPPs when there were no FPPs under construction or in operation at the time of writing, and those being proposed are at an early stage of design. For future work in this area, considering a broader set of fusion facilities that includes experimental devices as well as FPPs may lead to better insights and prove more relevant for the near-term. This could be taken into consideration when developing the scope of any future fusion-specific safety standard(s), if these are needed.

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ANNEX I. Additional detail relating to specific responses and information received

I–1. FUSION FACING FIRST WALL TEMPERATURES

In Japan's DEMOnstration power plant, JA DEMO, the maximum neutron wall load is 1.66 MW/m^2 and the heat wall load due to radiation from the plasma is 0.5 MW/m^2, when the fusion power is 1.5 GW and the major radius is 8.5 m. The coolant condition was selected to be PWR water conditions of 15.5 MPa and ΔT=35 ºC (290 ºC-325 ºC) [I–1]. The blanket first wall of the JA DEMO is fabricated by the structure material of the F82H (low-activation ferritic steel) of thickness in 10-20 mm and is coated with about 0.5-1.0 mm thick tungsten to suppress erosion by physical sputtering. Basically, the temperature of the first wall is kept at a high temperature to reduce retention as much as possible. Therefore, the first wall temperature of the JA DEMO is operated at the temperature close to 550 ºC for the F82H allowable operating temperature, as far as the thermal stress generated by the temperature difference allows. Under the current design concept, the maximum temperature is kept at about 430 ºC.

I–2. ITER DISRUPTION MANAGEMENT SYSTEM

The purpose of the ITER Disruption Mitigation System (DMS) is investment protection through a reduction of thermal and electro-mechanical loads to ensure the appropriate lifetime of all affected ITER components. This is achieved through the injection of large quantities of neon and protium into the plasma using shattered pellet injection (SPI) technology. SPI technology is based upon forming solid pellets of the impurities and/or fuel. When triggered these pellets are propelled through a series of flight tubes towards the plasma. Immediately before entering the plasma the pellets collide with a surface which cause them to shatter into a large number of smaller fragments. These fragments have a higher assimilation efficiency than an intact pellet.

The injected material dissipates the plasma energy through spectral line radiation and leads to a rapid increase of the plasma density. The radiative energy dissipation reduces localized thermal loads to plasma-facing components and allows controlling the current quench rate to minimize forces on the vacuum vessel and in-vessel components; whilst the increase in density aims to avoid the generation of runaway electrons or if they are generated to stop them from causing damage.

I–3. HT/HTO ASSUMPTIONS – JET EXAMPLE

The JET Safety Case [I–2] assumes for tokamak releases at higher temperatures (i.e. the machine is baked) a HT to HTO conversion of 10% for the calculation of exposure for public dose following a release into the torus hall. The assessments assume an initial composition of 99%HT:1%HTO, since the safety case considered that in the event of a loss of vacuum accident (LOVA), any conversion of HT to HTO within the vacuum vessel would be slow in comparison to the timescale of the event. The conversion rates presented in Refs [I–3] (which related to tritium on cold surfaces in the presence of steam) and [I–4] (which considered heated surfaces) indicate that in the initial few minutes of the accident, less than 0.2% of the elemental tritium would convert to HTO. A 99%HT:1%HTO composition was therefore assumed for the initial pressure driven release. However, in the event of a dual failure whereby the exhaust detritiation system pumping capability was inoperable, there would be a slow diffusion from the vessel back through the breach over 12 hours (following a LOVA). For this, the safety case applied a conversion rate of 3.6E-05 $TBq m^{-2} s^{-1}$ on the basis of data presented in Ref. [I–3], and calculated that in 12 hours, 7 % of the tritium in the vessel would be converted to HTO. For analysis purposes this was rounded up to 10 % and a 10 % composition of HTO was therefore assumed for operations fault sequences which result in a slow diffusion from the hot vessel.

I–4. SAFETY FUNCTIONS AND SSCS

In addition to the safety functions listed in Section 3.3, some FPP concepts have included safety functions related to hazardous/toxic materials and other industrial hazards such as electromagnetic radiation. A couple of concepts have safety functions related to hybrid FPP – although these are outside the scope of this TECDOC.

One concept divided the SSCs based on the functions implemented as follows:

- Protective safety systems – to implement the function of prevention or mitigation of the damage to vacuum vessel, blankets, as well as equipment and pipelines containing radioactive material.
- Confining safety systems – to prevent or limit the propagation fusion fuel, radioactive material and ionizing radiation beyond the boundaries and to the environment.
- Supporting safety systems – to supply power, working media to safety systems and ensure their operation.
- Controlling safety systems – to initiate actions of safety systems and control their functions.

I–4.1. UKAEA categorization and classification

An example safety function categorization scheme based on three categories:

- Category A: Any function that plays a principal role in ensuring radiological safety;
- Category B: Any function that makes a significant contribution to radiological safety;
- Category C: Any other safety function contributing to radiological safety.

Figure I–1 illustrates how categories could be designated based on consequences vs likelihood.

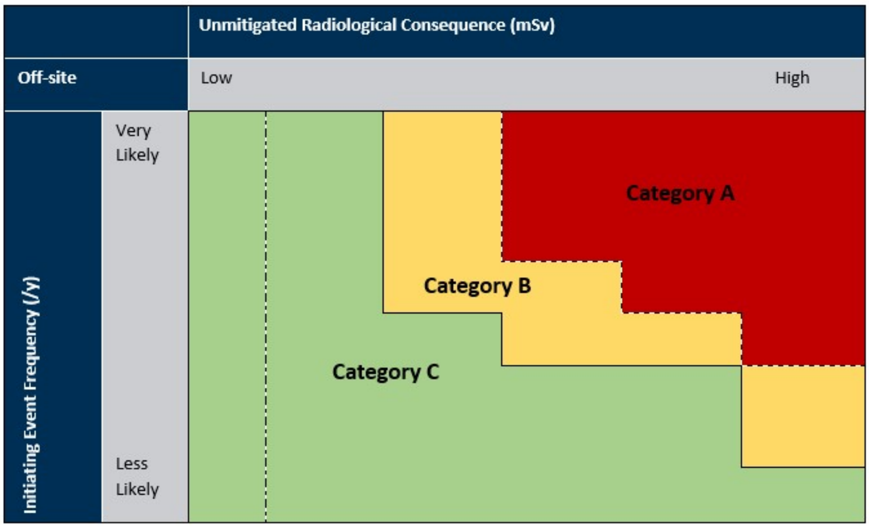

FIG. I–1. Example safety function categorization for an effective dose received by a person off-site (reproduced with permission from UKAEA).

The classification assigned to a safety measure will be based on the category of the safety function that the safety measure performs and the importance of the measure in the fulfilment of the safety function. Structures, systems and components (SSCs) that deliver safety functions will be classified using the following scheme:

- Class 1: Any SSC that forms a principal means of fulfilling a Category A safety function.
- Class 2: Any SSC that makes a significant contribution to fulfilling a Category A safety function or forms a principal means of ensuring a Category B safety function.
- Class 3: Any other SSC contributing to a categorized safety function.

Table I–1 outlines the relationship between the safety function category and SSC classification. TABLE I–2. illustrates the minimum safety measure(s) for each safety function category.

TABLE I–1. SSC CLASSIFICATION SCHEME

	Importance of the SSC in the delivery of the safety function		
Safety Function	Principal Means	Significant Means	Other Means
Category A	Class 1	Class 2	Class 3
Category B	Class 2	Class 3	-
Category C	Class 3	-	-

TABLE I–2.EXAMPLE MINIMUM NUMBER OF SAFETY MEASURES

Safety Function	**Minimum Safety Measures**	**Safety Measure Classification**
Category A	Two independent safety measures	One Class 1 measure and one Class 2 measure
Category B	One safety measure	One Class 2 measure and one Class 3 measure
Category C	One safety measure	One Class 3 measure

The classification of an SSC mandates its reliability, driving design requirements in terms of the failure frequency (for continuously operating systems), or the probability of failure on demand for demand-based systems. Table I–3 provides an indication of the minimum reliability of an SSC depending on its classification.

TABLE I–3. EXAMPLE MAXIMUM PERMISSIBLE FREQUENCY PER YEAR OR PROBABILITY OF FAILURE ON DEMAND FOR AN SSC DEPENDING ON ITS CLASSIFICATION

SSC Class	Maximum Failure Frequency Per Year	Maximum Probability of Failure on Demand
Class 1	10^{-3}	10^{-3}
Class 2	10^{-2}	10^{-2}
Class 3	10^{-1}	10^{-1}

I–4.2. European Demonstration power plant (EU DEMO)

I-4.2.1. Safety Importance Classification [I–5]

SSCs that are identified in the systematic safety analysis of the design as necessary to achieve the DEMO safety functions are assigned a SIC based upon the magnitude of the consequences of their failure.

The classification is intended to identify as SIC those SSCs whose failure can directly initiate an incident or accident, leading to significant risks of exposure or contamination, those SSCs whose operation is required to prevent or limit the consequences of an incident or accident that would lead to significant risks of exposure or contamination, and those SSCs whose operation is necessary to ensure, support or protect the functioning of other SIC SSCs.

It is important that any unintended operation or failure of active components in any system (whether SIC or non-SIC) does not prevent safety functions from being provided by SIC components when needed.

A SIC classification scheme is determined for the project based on three SIC classes and associated requirements. This is based on an adaptation of IAEA recommendations for NPPs [I–6], following more recent advice from IAEA on classification of components in fusion applications [I–7]. The main features are summarized below.

I-4.2.2. Criteria for identification of SSCs to be classified as safety important

Three criteria are applied to determine which SSCs are important for safety and will therefore be classified SIC. These are:

Criterion A: Their failure can directly initiate an incident or accident, leading to risks of exposure or contamination.

Criterion B: Their operation is necessary to limit the consequences of an incident or accident that would lead to risks of exposure or contamination.

Criterion C: Their operation is necessary to ensure the functioning of other SIC components.

I-4.2.3. Definition of Safety Important Classes

SSCs identified as important to safety according to one or more of the criteria listed above are assigned one of three safety importance classes, SIC-1, SIC-2 or SR. These are provisionally defined as follows:

An SSC is classified as SIC-1 if:

- Its failure could lead to an event with consequences exceeding the limits set out in section 5.2.1 of the DEMO Plant Safety Requirements Document;
- It is needed to prevent, detect or mitigate an accident from resulting in consequences exceeding the limits set out in section 5.2.1 of the DEMO Plant Safety Requirements Document; or
- It is needed to bring and maintain the plant into a safe state.

An SSC is classified as SIC-2 if:

- It is needed to prevent, detect or mitigate an incident or accident, although not needed to reach a safe state; or
- It is needed to ensure adequate shielding from radiation during normal operation.

An SSC is classified as SR if:

- Although not needed to prevent, detect or mitigate an incident or accident, it helps to further reduce the consequences of such an event.

All other components are described as 'non-SIC'. Future revisions of these SIC class definitions are possible.

I–5. EU DEMO EXAMPLE OF A PRELIMINARY EVALUATION OF SSCS

Numerical values are given as currently considered for the EU DEMO design but may evolve as the design progresses. The in-vessel components (breeder blanket, limiters, divertor) are not SIC. Details on the design requirements of each main SSG are provided in Table I–3.

TABLE I–3. EU DEMO Design Requirements of each main SSC

Main SSC	**Main functions**	**Main sub-systems or components**	**Design requirements**
Magnet System	• The central solenoid and the poloidal field coils initiate the plasma (break down), ramp up and drive the plasma current. • The central solenoid and poloidal field coils enable the stabilization and control the plasma shape and position. • The TF winding pack of the inner leg are designed for a maximum peak magnetic field in the superconductor of (acceptable range [9.5 – 16 T], design value: 12.2 T) • The TF system generates a toroidal magnetic field with a maximum toroidal ripple not considering ferromagnetic inserts anywhere on the plasma boundary of less than	Toroidal field coils (TFC), Poloidal field coils (PFC) Central solenoid coils (CS)	• The magnet system will be supported by the cryostat pedestal ring. • The magnet system will allow access to the vacuum chamber by providing space for port structures between the TF and PF coils. • The magnet system will withstand all loading conditions in accordance with the design criteria defined for the magnet system • The fast discharge of all TF coils will have a time constant of no less than (acceptable range [32 – 50s], design value: 35s) • After a standard pulse the CS will be recharged and ready for the next pulse within a time no longer than (acceptable range [0-1000s], design value: 600s) • The magnet system will be electrically grounded to the pedestal ring

<table>
<tr><th>Main SSC</th><th>Main functions</th><th>Main sub-systems or components</th><th>Design requirements</th></tr>
<tr><td></td><td>(acceptable range [0-0.6%], design value: 0.6%)
• The CS system generates at a nominal plasma current a total flux swing that enables a pulse length of (acceptable range [1 – 8h], design value: 2h).
• The magnet system is capable to sweep the divertor strike points on the inner and outer targets when detachment conditions are temporarily lost with a minimum frequency of (acceptable range [0.5 – 2Hz], design value: 1Hz) and a poloidal amplitude on the target of at least (acceptable range [±5-20cm], design value: ±10cm)</td><td></td><td>• The minimum temperature margin in the conductor is: (acceptable range [1.0 – 3.0K], design value: 1.5K)</td></tr>
<tr><td>Vacuum Vessel</td><td>• The vacuum vessel (VV) provides a vacuum boundary of the fusion plasma suitable for high vacuum (<10^{-5} Pa)
• The VV – together with other components –provides the first confinement barrier for radioactive inventory within the plasma chamber
• The VV contributes to the radiation shielding for the superconducting coils
• The VV contributes to the toroidal field ripple reduction, using ferromagnetic materials that are inserted in the VV.
• The VV contributes to the plasma vertical stability.
• The VV supports the in-vessel and in-port components.
• The VV contributes to the removal of decay heat of all in vessel components, even in conditions when their cooling systems are not functioning.
• The VV– together with the cryostat – provides the vacuum boundary for the superconducting magnet systems, the thermal shields and their manifolds in the cryostat.</td><td>Main vessel, port structures, VV supports.</td><td>• The VV will provide the ports for the integration of diagnostics, plasma H&CD systems, vacuum pumping, and fuelling.
• The VV will provide access ports for the in-vessel RM systems.
• The VV will be connected to a vacuum vessel pressure suppression system (VVPSS).
• The VV design or VV support design will not preclude the ability to move the whole VV during or after its assembly to achieve and maintain the prescribed tolerance relative to the magnet flux lines
• All VV ports will be connected electrically to the cryostat via metallic structures (such as the bellows)</td></tr>
<tr><td>Vacuum vessel pressure suppression system</td><td>• In the event of an in-vessel LOCA, the VVPSS limits the pressure in the VV to the peak values.
• The VVPSS provides primary confinement for the radioactive inventory inside the plasma chamber in accident conditions.</td><td></td><td>• The VVPSS will connect the plasma chamber to an expansion volume.
• The connecting duct between the plasma chamber and the expansion volume will be sealed in operational states by rupture disks.
• The VVPSS design will include bleed lines that bypass the rupture disks to avoid their opening at small in-vessel water leaks.</td></tr>
<tr><td>Breeding Blanket (BB)</td><td>• The breeding blanket provides sufficient tritium to the TER (Tritium Extraction and Removal) system to enable DEMO to operate with a close tritium fuel cycle.
• The BB system reduces the neutron flux such that the lifetime irradiation damage in the VV behind the BB is <2.75 dpa.
• In all events of category I the BB confines its coolant.
• In all events of category I and II the BB confines its process fluids (i.e. helium purge gas and at least 99.9% of its LiPb)
• In all events of category I, II, and III the BB confines at least 99% of its breeder/multiplier materials (i.e. LiPb / ceramic breeder, Be-based multiplier)
• The BB system contributes to the neutron and gamma radiation shielding</td><td>The BB is divided into segments, per sector there are three outboard and two inboard BB segments.

Manifold/backplate Breeding module first wall (FW) attachment structures</td><td>• For an operating pressure of 80 bar, the helium cooled pebble bed (HCPB) BB segment cooling loop will not cause a pressure drop inside the BB segments of more than (acceptable range [0 – 4.0 bar], design value: 2.0 bar)
• For an operating pressure of 155 bar, the water cooled lithium lead (WCLL) BB segment cooling loop will not cause a pressure drop inside the BB segments of more than (acceptable range [0 – 5 bar], design value: 2 bar)
• The HCPB BB system will limit the tritium released during ex-vessel loss of coolant accidents, i.e. the tritium inventory in the coolant will be limited to (acceptable range [0-10 g], design value: 2 g)
• The HCPB BB system will ensure that the saturated tritium concentration in the coolant during flat top causes the tritium permeation into the secondary loop – in accordance with the limits – not to be greater than (acceptable range [0-1.0 g/year], design value: to be defined (g/year)</td></tr>
</table>

Main SSC	Main functions	Main sub-systems or components	Design requirements
			• The WCLL BB system will ensure that the saturated tritium concentration in the coolant during flat top does not become higher than the level defined for the CANDU reactors (1.85E8 Bq/cm³) • The BB system will ensure that the interface to the BB primary heat transfer system (PHTS) is in the upper pipe chase • The weight of any BB segment during transport will be lower than (acceptable range [0-80 tons], design value: 60 tons) • The BB FW will withstand the design heat loads during flat-top operation and all transients as defined in the DEMO plasma facing component (PFC) heat load specification • The BB system will withstand all loading conditions as defined in the DEMO plant structural load specification. • The BB system is made of 3 outboard and 2 inboard vertical segments per VV upper port • The shape of the BB segments is to be compliant with the removal/installation kinematics • The toroidal sidewalls of the BB segments will allow for sliding contact during the extraction of BB segments • All feeding piping will be routed either through the upper or the lower VV ports. • Each BB segment will be electrically grounded to the VV

Divertor	• The divertor system intersects the plasma scape-off layer during flat-top and part of ramp-up and plasma termination
	• The divertor restricts the backflow of particles – which are neutralized at or near its targets – to the main plasma so as to be compatible with plasma operation.
	• The divertor enables a sufficient flow of neutrals to enable the performance of the torus vacuum pumps
	• The divertor enables a sufficient neutral pressure for a robust detachment
	• The divertor system reduces the neutron flux such that the lifetime irradiation damage in the VV behind the divertor is <2.75 dpa.
	• The divertor system contributes to the neutron and gamma radiation shielding
		48 cassettes
		Each cassette includes one cassette body (CB) and three plasma-facing components (PFCs), namely the inner and outer vertical targets (VTs) and the liner and reflector plates.
			• The divertor system will withstand the design heat loads during flat-top operation and all transients as defined in the DEMO PFC heat load specification, in particular:
			  • During flat top in detached condition (acceptable range [10-15 MW/m²], design value: 10 MW/m²)
			  • During sweeping: (acceptable range [0 – 80 MW/m²], design value: 70 MW/m²)
			  • During central disruption: energy deposited on the targets (tor. Angle ψ=360°) and shared in a ratio 2:1 (outer/inner target) for up to 10 ms (acceptable range [0.5-1.0 GJ], design value: 1.0 GJ)
			• The divertor system will withstand all loading conditions as defined in the DEMO plant structural load specification.
			• The divertor will be made of three cassette assemblies per VV lower port
			• The design of the divertor cassettes will aim at reducing their size and weight as much as reasonably achievable.
			• The divertor PFCs will be capable of being baked at a temperature of at least (acceptable range [220-300°C], design value: 240°C)
			• The divertor cassette body will be capable of being baked at a temperature of at least (acceptable range [150-200°C], design value: 180°C)
			• The divertor cassette will be electrically grounded to the VV both on the inboard and on the outboard side.
			• For each cassette body, the divertor system will need no more than four pipe connections to the cooling systems.

Limiters	• The limiter system, together with the divertor system, prevents the plasma SOL from coming into contact with the BB FW or the H&CD or diagnostic FW in any plasma transient classified as Category I or II event.		• The limiter system will withstand the design heat loads during flat-top operation and all transients as defined in the DEMO PFC heat load specification and in particular, the most severe foreseeable unmitigated disruption (damage of armour acceptable)The limiter system will withstand all

Main SSC	Main functions	Main sub-systems or components	Design requirements
	• The limiter system reduces the neutron flux such that the lifetime irradiation damage in the VV behind the limiter is <2.75 dpa. • The limiter system contributes to the neutron and gamma radiation shielding		loading conditions according to the design criteria defined for the limiter system. • The design of the limiters will aim at reducing their size and weight as much as reasonably achievable. • The limiter PFCs will be capable of being baked at a temperature of at least 240°C • The limiter structures other than the PFCs will be capable of being baked at a temperature of at least 180°C • The limiters will be electrically grounded to the VV
Thermal Shields	• To limit the heat load to the superconducting coils from warm internal/external sources to levels that can be tolerated by the coils and reasonably removed by the helium cryogenic system.	Cryostat thermal shield (CTS), VV thermal shield (VVTS)	• Heat to magnets (radiation and conduction): Plasma operation (flat top) ~6 kW, Baking ~6 kW • Heat to be removed from VVTS: Plasma operation (flat top) ~200 kW, Baking ~1000 kW • Heat to be removed from CSTS: Plasma operation (flat top) ~200 kW, Baking ~150 kW

Cryostat	• To provide a vacuum environment to avoid excessive thermal loads applied to the components operated at cryogenic temperature, such as superconducting magnet system, by gas conduction and convection		• Will provide support to the VV in normal and off-normal conditions
			• Will provides support to the magnet system
			• Will provides penetrations for:
			  • VV ports;
			  • Equipment inside the cryostat, i.e. magnet feeders, VV cooling pipes, instrumentation feedthroughs, pumping systems;
			  • Access for maintenance equipment and personnel into the cryostat;
			  • Access to the CS coil for direct vertical removal.
			• The cryostat top lid will allow opening and re-sealing during the non-nuclear phase

Heating and Current Drive Systems	To provide for the following: • Breakdown of plasma; • Assistance during the plasma ramp up phase; • During the flat top phase, heating of the core plasma, control of instabilities (NTM and thermal instability) • Assistance during the plasma ramp down phase.	Neutral beam (NB) system Electron cyclotron (EC) system Ion cyclotron (IC) system Ideally with only EC system, NB and/or IC if required by physics	
Diagnostic and control systems	• The diagnostic system is capable of detecting a loss of detachment causing the target heat flux to exceed the specified nominal range (max. 10MW/m²) within 0.5s, e.g. by measuring the plasma temperature at the target • The diagnostic system is capable of detecting category III slow VDEs (Vertical Displacement Events) with a current quench time >400ms and initiate a fast plasma shutdown to avoid excessive halo currents to occur. • To be further defined		• To be defined
Tritium, fuelling and vacuum systems	• To ensure a minimum dwell time in between the plasma pulses in order to enable a high availability, and to service a plasma with a range of different plasma enhancement gases for stability and power control at unprecedented amounts	Direct internal recycling loop (DIRL) Inner tritium plant loop (INTL) Outer tritium plant loop (OUTL)	• To be defined
Plant electrical systems	• To provide the electrical power to all DEMO plant loads and to deliver to the power transmission grid (PTG) the net electrical power produced	• Turbine generator TG • AC HV switchyard and power distribution HVS (green box)	• To be defined

Main SSC	Main functions	Main sub-systems or components	Design requirements
		• Coil power supply system CPS (blue box) • Steady state electrical network SSN (purple box) • H&CD Power Supply system HPS (red box)	
Cryoplant and cryo-distribution systems	• To supply the magnet systems TF & PF & CS and the thermal shields VV & Cryostat with He at 4°K and 80°K.		• To be defined
Balance of plant	• To extract the thermal power from the in-vessel components and the VV and its conversion into electrical power through the adoption of power cycles based on steam (Rankine cycles)	• Tokamak cooling systems: blanket and BB PHTS, Divertor and limiter cooling system, VV cooling system • Intermediate Heat Transfer System • Power conversion system	• To be defined

I–6. CATEGORIZATION OF POSTULATED INITIATING EVENTS AND ACCIDENT SCENARIOS

I–6.1. Plant states

Some FPP projects further break down the categories, for example, breaking down normal operation and anticipated operational occurrences[1] into substates, such as for normal operation testing, plant operation, and maintenance. Using these substates to assist in developing specific safety objectives for workers, the public and the environment.

Another example of breaking down the plant states into substates is found in FPP projects where the following are considered and can be interpreted as substates of normal or routine operation. In this example, any off-normal event is considered as an accident scenario, thereby creating two main states, normal operation and off-normal.

- Offline and standby;
- Normal operation (as understood for producing energy);
- Regularly scheduled maintenance.

In another example, principal safety functions, such as confinement, heat removal and radioactive inventory removal for accident conditions, are defined in relation to the plant state. Supporting safety category functions, such as fuel isolation and plasma shutdown, are defined for normal operation and transient conditions. An example of a type of transient event is a vessel plasma disruption. Examples for accidents are: (a) in-vessel and ex-vessel loss of coolant and

[1] In some FPP projects, the term ‘incident situations’ is used instead of anticipated operational occurrences.

loss of flow accidents; (b) magnetic accidents such as a coil short circuit; (c) cryogenic plant overpressure; and (d) load drops during maintenance.

I–6.2. Postulated initiating events

Example 1

The list of PIEs is focused on the following aspects that may cause damage and a radioactive release from the system and fusion facility:

- Loss of power;
- Loss of cooling;
- Leakage of air, water or helium;
- Failure of magnet and system control;
- Failure of decay heat removal;
- Potential risk events during maintenance and transport.

Example 2 (JET)

The list of PIEs considered in the JET safety case include:

- Loss of vacuum accidents (LOVAs);
- Loss of coolant accidents (LOCAs);
- Loss of flow accidents (LOFAs);
- Loss of plasma control (LOPC) events;
- Plasma heating and fuelling system (PHFS) events;
- Loss of electrical power (LOEP) events;
- Ex-vessel LOCA;
- Magnet events;
- Shielding events;
- Operator tasks;
- External events (extreme weather, fire, dropped loads, missiles, aircraft crash, earthquakes)

Note, in JET 'LOCAs' refer only to coolant of the neutral beam and some ancillary systems. There is no active cooling of the vacuum vessel, magnets, etc., so a LOCA in JET is not at all similar to a LOCA in an FPP where you have a high pressure, high temperature coolant.

Example 3 (EU DEMO)

In EU DEMO, in order to ensure the completeness of the safety analyses, it is essential that the identification of potential accident event sequences is comprehensive. To achieve this, PIEs are identified using systematic methods such as FMEA or HAZOP studies. At an early stage during the conceptual design, when sufficient design detail is unavailable, a FFMEA may be performed. In addition to identifying PIEs, FMEA or FFMEA may be used to assess the frequency of an event initiator, if adequate failure rate data is available.

The identified PIEs are used to develop event sequences, with the aid of event trees if appropriate. Full consideration needs to be given to consequential failures in the sequences, and a single aggravating failure needs to be included if it has an adverse effect on the accident progression. This aggravating failure may be assumed in a SIC that is not designed to satisfy the single failure criterion or for which there is still the possibility of common mode failures between trains.

Complimentary top-down techniques such as global fault trees or MLDs may additionally be used to ensure the completeness of event identification. Particular attention needs also to be paid to including all potential common cause (including common mode) failures.

The frequency category of an event sequence is conservatively assigned on the basis of the frequency of the initiating event, unless there is very strong justification for taking into account the low probability of a particular aggravation in the sequence of event steps.

From the full set of event sequences identified, a limited set of events are selected, known as the ‘Reference Events’. These are chosen to ensure full coverage of the types of accidents that have been identified for all systems of the EU DEMO plant. For each selected initiating event, the sequence with the most severe consequences is chosen as the reference.

The non-exhaustive reference PIEs likely to need design specific safety provisions are:

- Plasma events with specific reaction forces on supports or on penetrations;
- Plasma event leading to a LOVA and/or an in-vessel LOCA;
- Loss of flow in the primary cooling loop of the VV;
- Small or Large ex-vessel LOCA from one of the primary cooling loops;
- Loss of heat sink in all primary cooling circuits;
- Total blackout;
- Large cryogenic line break inside rooms;
- Loss of confinement in rooms permanently contaminated;
- Hydrogen isotopes leaks inside rooms leading or not to an explosion;
- H_2 and/or dust explosion in vacuum vessel or in other process systems;
- Fire from combustible materials in various rooms;
- Primary to secondary leaks;
- Steam line break;
- Cask stuck in rooms;
- Maintenance accident scenarios (e.g. in shutdown states) leading to confinement losses.

Example 4 (JA DEMO) [I-8]

Several accident sequences of particular concern were deduced by using the FFMEA and MLD methods, with 21 reference accident sequences identified. The reference events identified are summarized in the following Table I–4:

TABLE I–4. REFERENCE POSTULATED INITIATING EVENTS AND EVENT SEQUENCES

No.	Postulated initiating events	Event sequences
1	Abnormal increase in the fusion power	• Increase in the temperature of the divertor plate • Plasma disruption due to break of divertor cooling channels • Break of in-vessel components cooling pipes due to the disruption • Ingress of water in the VV • Pressure suppression system in operation and condensation of steam water
2	Loss of coolant flow in the blanket cooling system	• Heating of the first wall by the plasma

		• Plasma disruption due to break of a cooling channel of blanket or ingress of impurity into the plasma • Ingress of water in the VV • Hydrogen production by the channel reaction of the first wall material with steam water • Pressure suppression system in operation and condensation of steam water
3	Loss of coolant flow in the divertor cooling system	• Heating the divertor plate by the plasma • Plasma disruption due to break of divertor cooling pipes or ingress of impurity into the plasma • Break of blanket cooling channel due to the disruption • Ingress of water in the VV • Hydrogen production by the chemical reaction of the divertor material with steam water • Pressure suppression system in operation and condensation of steam water
4	Local increase in the heat load on the first wall	• Local heating of the first wall (due to the plasma vertical displace event or failure of plasma heating systems) • Plasma disruption due to break of first wall cooling channel • Ingress of water in the VV • Hydrogen production by the chemical reaction of the first wall material with steam water • Pressure suppression systems in operation and condensation of steam water
5	Transient thermal energy release due to the disruption	• Break of the cooling channel of in-vessel component due to the disruption • Ingress of water in the VV • Pressure suppression system in operation and condensation of steam water
6	Loss of coolant flow after shutdown	• Break or melting of in-vessel component cooling channel • Ingress of water in the VV • Pressure suppression system in operation and condensation of steam
7	Transient electromagnetic force due to the disruption	• Break of in-vessel components cooling channel due to the disruption • Ingress of water in the VV • Pressure suppression system in operation and condensation of steam water
8	Quench of the toroidal magnetic field coils	• Decrease in the current of in the toroidal magnetic field coil • Plasma disruption due to the loss of the toroidal magnetic field • Break of in-vessel components cooling channel due to the disruption • Ingress of water in the VV • Pressure suppression system in operation and condensation of steam water
9	Short of the toroidal magnetic field coil	• Decrease in the current of in the toroidal magnetic field coil • Quench of the coil due to increase in the current in the shorted coil • Plasma disruption due to the loss of the toroidal magnetic field • Break of in-vessel components cooling channel due to the disruption • Ingress of water in the VV • Pressure suppression system in operation and condensation of steam water
10	Break of the blanket/divertor cooling pipe	• Ingress of water in the VV • Pressure suppression system in operation and condensation of steam water
11	Increase in the coolant pressure	• Pressure relief valves open for the coolant loop
12	Increase in the pressure of the He	• Pressure relief valve open of the He refrigerating system
13	Ingress of air at the cryostat boundary	• Decompression in the building • Quench of the toroidal magnetic field coils • Decrease in the current in the toroidal magnetic field coil • Plasma disruption due to the loss of the toroidal magnetic field • Break of in-vessel component cooling channel due to the disruption • Ingress of water in the VV • Pressure suppression system in operation and condensation of steam water
14	Failure of cooling the isotope separation system	• Increase in the hydrogen isotope gas pressure in the isotope separation system

		• Pressure suppression system for the isotope separation system in operation
15	Failure of heater of the fuel storage bed	• Increase in the fuel gas pressure in the fuel storage bed • Pressure suppression system for the fuel storage system in operation
16	Break of the connecting port	• Plasma disruption due to ingress of air into the vacuum vessel • Break of in-vessel components cooling pipes due to the disruption • Ingress of water in the VV • Pressure suppression system in operation and condensation of steam water • Release of the steam in the VV to the reactor building • Diffusion and absorption of the steam, tritium and dust in the building
17	Ex-VV LOCA of the blanket cooling system	• Increase in the gas pressure in the reactor building and release of radioactive material from the stack or the blowout panel • Decrease in the coolant pressure in the cooling pipes • Heating of the first wall by plasma • Plasma disruption due to break of first wall cooling channel or ingress of impurity into the plasma • Break of in-vessel component cooling channel due to the disruption • Hydrogen production by the chemical reaction of the first wall material with steam water • Pressure suppression system in operation and condensation of steam water
18	Ex-VV LOCA of the divertor cooling system	• Increase in the gas pressure in the reactor building and release of radioactive material through the stack or the blowout panel • Decrease in the coolant pressure in the cooling pipes • Heating of the divertor plate by plasma • Plasma disruption due to break of divertor cooling pipes or ingress of impurity into the plasma • Break of in-vessel component cooling channel due to the disruption • Hydrogen production by the chemical reaction of the first wall material with steam water • Pressure suppression system in operation and condensation of steam water
19	Failure of the fuelling line	• Ingress of the tritium gas to the tritium area • Exchange of the tritium with the air in the tritium area • Tritium burning in the fuelling line • Removal of the tritium gas
20	Failure of the isotope separation system	• Ingress of the tritium gas to the tritium area • Exchange of the tritium with the air in the tritium area • Tritium burning in the fuelling line • Removal of the tritium gas
21	Break of the cooling pipe in the breeding blanket	• Ingress of the steam in the coolant to the blanket module • Break of pipes of the tritium recovery system • Break of the blanket module • Pressure suppression system in operation and condensation of steam water • Release of radioactive material through the broken pipes of the tritium recovery system • Break of the VV and release of the radioactive material (if the pressure suppression is failed due to non-condensable gas)

I–7. APPLICATION OF Defence in Depth

Concepts of defence in depth (DiD) are used in the management of sources of radiation. Some example approaches to DiD for fusion facilities are provided below, although there is no single overarching model.

Example 1

EU DEMO applies the following interpretation of the five levels of defence in depth, as articulated in detail within Ref. [I–9]:

Level 1

Use of conservative design practices, application of quality assurance and promotion of a positive safety culture:

- Minimizing radioactive and hazardous material inventories and energy sources that can act as accident initiators;
- Providing fail-safe designs where possible, including passive safety features;
- Using adapted design codes and standards applicable to the technology and the hazards involved, or specially developed specifications where no such codes and standards exist;
- Providing a robust safety design that can tolerate failures of novel and first-of-a-kind components.

Level 2

Control of abnormal operation and detection of failures that could lead to damage to confinement barriers or other safety systems:

- Maintaining normal operating conditions using instrumentation, monitoring and control systems capable of detecting abnormalities in systems and correcting them;
- Providing features to control the plasma and the plant and thus to avoid challenges to safety systems and components;
- Providing redundant and, where appropriate, diverse systems as necessary, to achieve high reliability for safety systems;
- Providing access control and airlocks to protect site personnel from hazards;
- Providing safety parameter monitoring for operating personnel, requiring few, if any, operator actions in response to safety parameter monitoring;
- Undertaking periodic inspection, testing, and/or on-line monitoring of SSCs important to safety.

Level 3

Control of accidents within the design basis, and the provision of SSCs important to safety, including:

- Providing multiple confinement systems to protect all significant radioactive and non-radioactive inventories, using reliable passive barriers where possible;

- Providing a pressure suppression system to protect the vacuum vessel, which is the first confinement barrier;
- Providing free volumes to protect the other confinement barriers against overpressure events;
- Providing filters and detritiation systems to reduce limit releases to the environment;
- Providing an off-normal fusion power termination function to rapidly shut down the plasma when this is needed to limit the consequences of an accident.

WENRA recommendations for DEC [I–10] could be included for example:

"It has been proposed to treat the multiple failure events as part of the 3rd level of DiD, but with a clear distinction between means and conditions (sub-levels 3.a and 3.b).

…….

"For level 3.b, analysis methods and boundary conditions, design and safety assessment rules may be developed according to a graded approach, also based on probabilistic in-sights. Best estimate methodology and less stringent rules than for level 3.a may be applied if appropriately justified".

Level 4

Accident management: measures are taken to preserve the integrity of confinement and actions are taken to reduce consequences following an accident including:

- Placing the radioactive and non-radioactive inventories in a safe state;
- Directing releases that might occur to controlled and monitored discharge points;
- Operator actions to further mitigate consequences, to monitor plant conditions and ensure that systems function as intended.

Level 5

Despite all the accident prevention and impact reduction measures taken in the previous levels, this fifth level allows for off-site protection measures in the event of an accident whose impact goes beyond the site boundary. These could include measures such as short term restrictions on food production and consumption over a limited area.

Example 2

TO HERE For first-of-a-kind systems where operational experience is limited (thus certain Level 1 controls are more difficult or not cost-effective), reliance on good practices and a strong safety culture, Level 2 systems to detect failures, and where appropriate Level 3 or 4 systems to have secondary confinement, may be a suitable course of action. Since fusion facilities have no risk of criticality or meltdown, and limited fuel in the vacuum vessel, monitoring for incidents and the use of secondary confinement barriers can in many cases keep overall risk adequately limited until sufficient operational experience is obtained to improve engineering design cost-effectively.

Example 3

The primary means of preventing and mitigating the consequences of accidents is the consequent implementation of the DiD concept. The main objective of defence in depth is the risk reduction of accidents with unacceptable radiological consequences by consecutive and independent levels of protection. Whereas for NPPs the defence in depth concept is well elaborated, this concept needs to be adapted to the specific needs of fusion facilities. Application of the defence in depth concept is not only required for NPPs, but also for research reactors and fuel cycle facilities. Thus, it can be concluded that the defence in depth concept is also an essential element of the safety concept for fusion facilities.

Central elements, like consecutive levels linked to certain plant states will remain and are also applicable to fusion facilities. The Table I–5 shows the proposed levels of DiD for a fusion facility and the four associated plant states. In addition, the objective of each level of defence in depth as well as the acceptable radiological consequences are shown.

TABLE I–5. PROPOSED LEVEL OF DEFENCE IN DEPTH FOR A FUSION FACILITY

<table>
<tr><th colspan="2">Level</th><th colspan="2">Plant state</th><th>Objective</th><th>Radiological objective</th></tr>
<tr><td rowspan="4">Fusion facility design</td><td>1</td><td rowspan="2">Operational states</td><td>Normal operation (planned plasma transients inclusive)</td><td>Prevention of abnormal operation and failures</td><td rowspan="2">No-off site radiological impact (i.e. no exceedance of regulatory operating limits for discharge)
Occupational exposures below regulatory limits</td></tr>
<tr><td>2</td><td>Anticipated operational occurrences</td><td>Control of abnormal conditions and failures
prevention of accident conditions</td></tr>
<tr><td>3</td><td rowspan="2">Accident conditions</td><td>Postulated single initiating events</td><td rowspan="2">Control of accidents to limit radiological releases</td><td rowspan="2">No-off site radiological impact or only minor radiological impact (i.e. no exceedance of operational intervention levels for protective actions in emergency exposure situations)</td></tr>
<tr><td>4</td><td>Postulated multiple failure events</td></tr>
<tr><td>EPR</td><td>5</td><td>Accident conditions</td><td>Accidents requiring off-site emergency response</td><td>Mitigation of off-site radiological consequences of significant releases of radioactive material</td><td>Radiological impact with protective measures limited in area and time</td></tr>
</table>

The range might be from small experimental facilities up to demonstration facilities for energy production. Consequently, the potential radiological hazard of future fusion facilities may vary.

To avoid applying overly strict safety requirements that are not commensurate with the radiological hazard, a graded approach needs to be applied.

The key element of the graded approach is the categorization of the potential radiological hazard of a certain facility. This consideration is not limited to the tritium and radioactive material inside the vacuum vessel, but also covers all radioactive material present on the site, including the tritium fuel cycles system and the waste management and storage facilities.

Three risk categories based on the potential radiological impact could be used for the application of graded approach:

(a) Off-site radiological impact exceeds operational intervention levels for protective actions in an emergency;

(b) On-site radiological impact, no off-site actions needed to protect the public;

(c) No actions to protect workers outside radiation controlled aeras.

I–8. OTHER DESIGN REQUIREMENTS

I–8.1. Application of general requirements

Example 1

EU DEMO applies all the general design criteria (listed below) to all SIC SSCs:

- Use of physical separation and independence of safety systems;
- Accounting for common cause failures;
- Applying the single failure criterion;
- Using fail-safe designs;
- Establishment of operational limits and conditions for safe operation.

It groups together physical separation and independence of safety systems with the application of common cause (including common mode) failures. The rationale for this is that common cause failures are very important mechanisms for undermining independence. EU DEMO fully applies the single failure criterion to all SIC[2] SSCs and applies this to both passive and active SIC SSCs. EU DEMO also applies the principle of fail-safe design where feasible; an example of which is the isolation valves where on loss of power the valves automatically close to the fail-safe position.

The application of operational limits and conditions for safety are a regulatory requirement for EU DEMO, an important context for which is the identification of plant states that could lead to the exceedance of every step of the DiD levels. In line with this approach, it breaks down the overall operation of the FPP into the five DiD level domains and provides a description of key criteria used to monitor for any exceedance of operational limits and conditions that indicate a challenge to remaining within the domain specified.

[2] For SIC-1 but for SIC-2, upon a case by case safety analysis depending on a graded approach

EU DEMO emphasizes the importance of applying passive safety measures where static barriers are preferred to dynamic barriers and passive cooling systems to active cooling systems.

Example 2

One of the contributing experts stated that design requirements could be extended to their four safety functional categories, which are as follows:

- Occupational radiation exposure;
- Routine releases of radioactive material in normal operation;
- Release of radioactivity in accidents;
- Management of radioactive waste

Example 3

One of the contributing experts stated that the physical separation and independence of safety systems is necessary for an FPP. The application of common cause failure analysis is a part of their assurance of the independence of safety systems. Fault tolerance of FPP safety systems is justified through applying the single failure criterion.

I–8.2. Equipment qualification

Example 1

An example of the application of equipment qualification is provided through the EU DEMO approach, where the process of equipment qualification is applied to all SSCs important to safety. This includes qualification of equipment for normal and abnormal environmental conditions (temperature, pressure, radiation etc.) under which the SSC has to continue to operate, qualifying for ageing mechanisms and, where relevant, qualification against hazards such as seismic events. The typical qualification approach for EU DEMO is based on the following:

- Tests on a representative SSCs;
- Using calculation and analysis, particularly when an SSC is too large or when the accident creates an accumulation of conditions that cannot be reproduced on a test facility;
- Use of industrial feedback, assuming the operational profile of the SSC in the industry for which feedback is being sought is close to that of the FPP;
- Continuous qualification, which is used to extend the service life for which the initial qualification was carried out.
- A combination of the above methods.

Example 2

In another example, the aim is to demonstrate that all safety important components will fulfil their safety functions in all foreseen loading conditions throughout their operational life. This may include structural integrity, leak tightness, verification of radiation shielding etc., and it may be necessary to demonstrate that performance is maintained even after radiation damage and in the presence of high magnetic fields. Qualification may also be needed to demonstrate compliance with design codes and standards, using existing codes and standards wherever possible.

Example 3

One of the contributing experts is currently actively seeking how to qualify materials for their technology and will argue the necessity of any qualification on a case-by-case basis. They also state that their approach to qualification is very likely to diverge from that of many other FPP developers due to their claims of a much lower hazard from the worst case accident and their use of diverse technology for the FPP design. They note that a major failure of their fusion machine will have a very limited safety impact and one of the more important safety considerations will be the handling and storing of the by-product tritium and therefore qualification may well be focussed on matters such as the getters for the storage of tritium.

Example 4

One of the contributing experts stated that novel FPP components need to be qualified based on NPP reactor component qualification practices. This NPP approach means that a graded approach will be used based on the safety importance classification of the SSCs. The type of FPP will also be considered, an example of a fusion–fission hybrid is given that may give rise to special qualification requirements.

I–9. INTERNAL AND EXTERNAL HAZARDS

Some examples of internal and external hazards of ITER and EU DEMO fusion power plants are provided in Table I–6.

TABLE I–6. EXAMPLES OF INTERNAL AND EXTERNAL HAZARDS OF ITER AND EU DEMO

FPP concept	Internal hazards	External hazards
ITER	Internal fire Internal explosion Thermal releases Plasma transients Internal flooding Missile effects and pipe whip Mechanical risks Chemical risks Magnetic and electromagnetic risks	External fire External flooding Extreme meteorological conditions Earthquake Hazards associated with nearby installations and communication routes Aircraft crash (small tourist plane, such as a Cessna)
EU DEMO	Internal fire Internal explosion Thermal releases Plasma transients Internal flooding Missile effects and pipe whip Mechanical risks Chemical risks Magnetic and electromagnetic risks	Earthquake Extreme meteorological conditions External flooding/water table External fire Aircraft crashes Industrial environment

A	Internal explosion Pipe whipping Internal fire Thermal release Plasma transient Internal flooding Mechanical risk Magnetic and electromagnetic risk	External fire Earthquake External flooding Aircraft crash
B	No response	Population and density of population. Natural disasters, including geologic hazard, flooding, earthquake, etc. Human-caused disasters, including exploding, aircraft crash, etc.
C	Tritium Low level radioactive waste	Earthquake Fire Conflict
D	Fire Dropped loads Pipe breaks Explosion Use of hazardous materials (Galden HT55, Beryllium, etc.) Missiles (from rotating machinery)	Extreme meteorological phenomena (Heavy snowfall, high winds, etc.) Fire Dropped loads and collisions Missiles Aircraft crash Earthquake
E	Fire Explosion Missiles Pipe breaks Collapse of structures Falling objects Electromagnetic interference (on-site cause) Flooding Use, storage and generation of hazardous materials	Earthquake Aircraft crash Extreme meteorological conditions Electromagnetic interference (off-site cause) Flooding
F	No response	Earthquake Extreme meteorological condition, intense heat, cold, rain, snow, wind and lightning External flooding External fire Hazards associated with human activity Aircraft crash Industrial environment (explosion near plant site) Accident nearby plant site

G	No response	Consider (at least) the same initiating events as for NPP Aircraft crash Act of terrorism
I	Plasma disruption Pipeline rupture Loss of fuel (tritium) supply to plasma. Change of magnetic field configuration Vacuum system failure	Airplane crash Earthquake
J	No response	Fire
K	Unplanned shut down of fusion machine	Loss of external power supplies to back up generator internal power
M	Fire Explosion Impact of flying or whipping objects Thermal, mechanical, or electrical failure of safety function equipment	Earthquake Tsunami Airplane crash

I–10.RADIATION PROTECTION AND THE CONCEPT OF OCCUPANCY

In many States (e.g. Canada, USA, UK) design dose limits are determined based on the pertinent national regulations and also justified occupancy factors, and the classification of the person exposed.

The occupancy of an area is defined as the fraction of time at work that a person is present in the area. The occupancy factor is 'a typical fraction of the time for which a location is occupied by an individual or group' [I-11].

For situations where the dose rate is predictable, and the pattern of work is stable this concept may be used to allow work in higher dose rate areas. The lower occupancy in such areas protects the workers without the need for additional design protection features.

REFERENCES TO ANNEX I

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[I–10] Western European Nuclear Regulators Association, Safety of new NPP designs – Study by Reactor Harmonization Working Group RHWG, (2022).

[I–11] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Nuclear Safety and Security Glossary, 2022 (Interim) Edition, IAEA, Vienna (2022), https://doi.org/10.61092/iaea.rrxi-t56z.

ANNEX II. Additional ITER information

ITER is considered as a so called Civil Land-based Nuclear Installation (INB) in the French regulatory framework. Therefore, the licensing process requested by the French regulatory body is close to that specified in the IAEA safety standards. The operating organization, ITER Organization has specified general safety objectives regarding public exposure, using levels and occurrence frequencies in accordance with existing regulatory limits and continuous dialogue with the authorities. Regarding the exposure of workers and the public in the event of an accident, the optimization principle (ALARA) is applied, as described in Table II.1. Nevertheless, 'hold points' are set by national legislation and regulatory requirements. These hold points give the regulatory body the power to ensure that risks to people and to the environment from nuclear installations and their activities are properly controlled by the persons or organizations responsible for the nuclear installations and their activities. At each hold point set down by the regulatory body or in the licensing process, an authorization or a licence from the regulatory body may be required. Conditions may be attached to licences granted at each step and may require that the licensee obtain further, more specific, authorizations or approvals before carrying out particular activities.

A safety demonstration was prepared following a deterministic approach including initiating events as well as taking into account a list of external and internal hazards, as required by the French nuclear safety regulation. The objectives of the safety analysis are to check and to adjust, if needed, that the ITER design has sufficient provisions to withstand accident sequences without violating the release guidelines and other safety principles. It is to be demonstrated that the probability and the consequences of any sequence are below the frequency and the dose limits of its category. In addition, it was expected to show that the response of the plant and of the systems important to safety are adequate and to define the loading and performance requirements that need to be considered in the design of these systems. The components important to safety are identified through the lists of accidental scenarios and studies within the design of the fusion machine. For these components, the safety provisions, the qualification requirements, the levels of redundancy and diversity, as well as the safety margins all have been established following a graded approach taking into account their importance for safety.

II–1. ITER SAFETY OBJECTIVE AND RESIDUAL RADIOLOGICAL RISK

For ITER, the level of residual radiological risk of ITER are indicated through its safety objectives presented below, taking into account the associated probabilities and an operating period of about 20 years:

- Incidental situation, frequency of occurrence > 10^{-2}/year
- Design basis accident, 10^{-2}/year > frequency of occurrence > 10^{-6}/year
- Beyond design basis accident, frequency of occurrence < 10^{-6}/year

TABLE II–1. GENERAL SAFETY OBJECTIVES OF ITER [II–1]

	General safety objectives of ITER	
	For personnel	**For the public and environment**
Design basis		
Normal operation	As low as reasonably achievable, and in any case less than: Maximum individual dose ≤10 mSv/year Average individual dose for workers classified for radiation exposure ≤2.5 mSv/year	Releases less than the limits authorized for the installation, Impact as low as reasonably achievable, and in any case less than: ≤0.1 mSv/year
Incident situations	As low as reasonably achievable and in any case less than: 10 mSv per incident	Release per incident less than the annual limits authorized for the installation. ≤0.1 mSv
Accident conditions	Take into account the constraints related to the management of the accident and post-accident situation	No immediate or deferred public protective actions(sheltering, evacuation) < 10 mSv No restriction of consumption of animal or vegetable products
Beyond design basis		
Hypothetical accidents	No cliff-edge effect; possible public protective actions limited in time and space	

II–2. POSTULATED INITIATING EVENTS

Safety analyses were carried out around 2007 by the ITER Organisation and its partners in order to develop the safety demonstration of the concept. For hypothetical events, so called beyond design basis events, the objectives were to show the robustness of the application of defence in depth (DiD) in ITER, that safety functions degrade gradually (i.e. no ‘cliff-edge effects’), that the need for evacuation is avoided, and that other possible protective actions such as sheltering or food restrictions are avoided or limited in time and space.

Most accident studies are limited to typical accident scenarios and use a deterministic methodology, and the total societal risk of fusion facilities is still not well recognized. In order to support the probabilistic risk assessment of fusion facilities, PIEs need to be identified.

A set of PIEs were determined and each classified as either an incident or an accident based on likelihood of occurrence. The PIE could be combined or not with aggravating failures and/or loss of power states. The determination of the PIEs starts from a reference list obtained by expert judgement and by considering the full range of hazard identification studies previously carried out. The classification of the reference sequences is mainly based on the use of deterministic rules, through comprehensive bottom-up component-level analyses using systematic methods such as FMEA. These were supplemented by a top-down plant-level analysis using MLD, as a check on the completeness of the FMEA results.

FMEA for ITER were carried out by ENEA between 2001 and 2007 on the systems listed in below Table II–2 based on the basic concept design leading to identify PIEs.

TABLE II–2. LIST OF PIE IDENTIFIED FROM FMEA FOR ITER

Systems	PIEs	Event categorization
Magnet systems		
Break in magnet cryogenic cooling line inside cryostat, inducing an arc between coils and cryostat or VV walls		Unlikely event
Vacuum Vessel		
Loss of forced cooling flow in a VV coolant circuit due to pump trip		Anticipated event
Ingress of Air in the VV - small leakage		Anticipated event
Loss of forced cooling flow in both VV cooling circuits because of pump stop due to a common cause		Unlikely event
Loss of heat sink to VV cooling loop		Unlikely event
Small rupture of VV external shell		Unlikely event
Small rupture in the internal VV shell - equivalent break size: a few cm^2		Unlikely event
Ingress of air and water in the cryostat - small leakage		Unlikely event
Large rupture of VV external shell		Extremely unlikely event
Large rupture in the internal VV shell		Extremely unlikely event
Simultaneous coolant ingress from VV circuit into plasma chamber and cryostat		Extremely unlikely event
Simultaneous coolant ingress from VV and FW circuits in VV and vault		Hypothetical bounding event
FW/blanket cooling water systems		
Loss of flow in a FW/BLK coolant circuit due to pump trip		Anticipated event
Loss of flow in a FW/BLK coolant channel		Anticipated event
Ex-vessel coolant ingress due to small rupture of FW/BLK cooling circuit pipe inside TWCS vault		Anticipated event
Small FW/BLK in-vessel coolant leakage - equivalent break size: a few cm^2		Anticipated event
Loss of flow in all primary loops due to stop of cooling pumps (with coast-down) due to common cause failure		Unlikely event
Loss of flow in a FW/BLK coolant circuit due to pump seizure		Unlikely event
Loss of all FW/BLK cooling pumps (with coast-down)		Unlikely event
Loss of heat sink to FW/BLK loop		Unlikely event
Ex-vessel coolant leakage due to rupture of tubes in a primary FW/BLK HX		Unlikely event
Rupture of one FW/BLK segment coolant loop inside VV		Unlikely event
In-cryostat coolant ingress due to rupture of FW/BLK coolant line		Extremely unlikely event
Ex-vessel coolant ingress due to large rupture of FW/BLK cooling circuit pipe inside TWCS vault		Extremely unlikely event
Coolant leakage from FW/BLK into port cell		Extremely unlikely event
Rupture of all FW/BLK cooling module coolant loops inside VV		Extremely unlikely event

Divertor cooling water system	
Loss of flow in a DIV/LIM coolant circuit due to pump trip	Anticipated event
Ex-vessel coolant leakage due to small rupture of DIV/LIM cooling circuit pipe inside TCWS vault	Anticipated event
Loss of flow in a DIV/LIM coolant circuit due to pump seizure	Unlikely event
Ex-vessel coolant leakage due to rupture of tubes in a primary DIV/LIM heat exchanger	Unlikely event
Rupture of one DIV/LIM coolant loop inside VV	Unlikely event
Small DIV/LIM in-vessel coolant leakage - equivalent break size: a few cm^2	Unlikely event
Coolant ingress into cryostat due to rupture of DIV/LIM coolant line in the cryostat	Extremely unlikely event
Ex-vessel coolant leakage due to large rupture of DIV/LIM cooling circuit pipe inside TCWS vault	Extremely unlikely event
Fuelling system	
Rupture of second confinement barrier of fuelling process line inside port cell or building room	Anticipated event
Rupture of fuelling process line with loss of Q2 isotopes into secondary containment	Anticipated event
Remote handling equipment	
Release of radioactive products (fraction of dust & tritium implanted in transported components) into port cell.	Anticipated event
Cask stop and leakage during RH transportation of divertor cassette to hot cell, inducing release of radioactive products (fraction of dust & tritium implanted in transported components) into Gallery.	Anticipated event
Cryostat	
Large leakage of cryogenic gas in a penetration towards outside CV II	Anticipated event
Large ingress of He in the cryostat	Anticipated event
Ingress of He in the cryostat - small leakage	Anticipated event
Large ingress of air in the cryostat	Anticipated event
Ingress of air in the cryostat - small leakage	Anticipated event
Ingress of gas (He and Air) in the cryostat - small leakage	Anticipated event
Large ingress of Cryogenic coolant in the cryostat	Anticipated event
Ingress of Cryogenic coolant in the cryostat - small leakage	Anticipated event
Large ingress of gas (He and Air) in the cryostat	Extremely unlikely event
Cooling Water System	
Loss of heat sink to DIV/LIM cooling loop	Unlikely event
Total loss of heat sink to all primary FW/BLK loops due to a large break in Heat Rejection System (coolant discharged in HRS room)	Unlikely event
Total loss of heat sink to all primary loops (FW/BLK, DIV/LIM and NB) due to a large break in Heat Rejection System pipes feeding main pumps from Cold Basin (coolant discharged in HRS room)	Unlikely event
Large HRS pipe rupture inside TWCS vault and consequent multiple pipe rupture in HX of divertor primary loop	Unlikely event

Large rupture of secondary loop in HRS room and consequent multiple pipe break in FW/BLK heat exchangers	Extremely unlikely event
Large rupture of secondary loop inside TCWS vault and consequent multiple pipe break in FW/BLK HXs	Extremely unlikely event
Vacuum pumping system	
Loss of cryogenic fluid from a cryopanel of the torus pump, while the pump is in regeneration	Anticipated event
Tritium plant	
Break of T process line inside Glove Box confinement of Tokamak Exhaust Processing System	Anticipated event
Leak of T process line inside Glove Box confinement of Tokamak Exhaust Processing System	Anticipated event
Break of Tritium Storage and Delivery System process line inside Glove Box confinement	Anticipated event
Leak of Tritium Storage and Delivery System process line inside Glove Box confinement	Anticipated event
Break of T process line inside ISS Hard Shell confinement	Anticipated event
Leak of T process line inside ISS Hard Shell confinement	Anticipated event
Release of tritiated effluents to environment due to pressure and temperature fluctuations on ISS CD1	Anticipated event
Break of T process line inside ISS Cold Box confinement	Unlikely event
Leak of T process line inside ISS Cold Box confinement	Unlikely event
Cryoplant and cryodistribution	
The FMEA on Cryogenic systems instead, pointed out that no elementary failures initiate safety relevant events in terms of mobilization, spreading and release of radioactive material. Therefore, no PIEs have been set in the FMEA tables. This conclusion might be challenged in the next revision of the PIEs list.	
Coil power supplies	
TF coil over-voltage due to circuit opening	Unlikely event
CS/PF over-voltage due to circuit opening	Unlikely event
Electrical power network	
Loss of off-site power for short duration (< 1 hr.)	Anticipated event
Loss of off-site power for long duration (1 < t < 32 hr.)	Unlikely event
Loss of all AC power for up to one hour	Extremely unlikely event
Neutral beam heating and current drive system	
Small ingress of gas into NB chamber and VV	Anticipated event
Small ingress of cryogenic coolant in NB chamber and VV	Anticipated event
Partial loss of coolant flow in a NB component due to plugs in the cooling channels.	Anticipated event
Rupture of a refrigerated NB component causing large ingress of water in NB chamber	Anticipated event
Water leakage in a refrigerated NB component causing small ingress of water in NB chamber	Anticipated event
Loss of flow in the NB coolant circuit because of pump trip	Unlikely event

Large ingress of gas into NB chamber and VV	Unlikely event
Ex-vessel coolant leakage due to small rupture of NB cooling circuit pipe inside TWCS vault	Unlikely event
Ex-vessel coolant leakage due to rupture of tubes in a primary NB heat exchanger	Unlikely event
Large ingress of cryogenic coolant in NB chamber and VV	Unlikely event
Ex-vessel coolant leakage due to large rupture of NB cooling circuit pipe inside TWCS vault.	Extremely unlikely event
Hot Cell	
Radioactive products release (fraction of dust and tritium) into hot cell building green areas because of breach in containment boundary or slight cell pressurization	Anticipated event
Cell pressurization and loss of dynamic containment because of sudden release of material (gas) and/or energy	Unlikely event
Fall of stored basket or waste containers	Unlikely event
Water Detritiation System	
Deactivation of exchange catalyst inducing tritium release to the environment through WDS effluents	Normal operation
Leak of high level water tank inducing small release of tritiated water into the plant areas	Anticipated event
Leak of process fluids into service water	Anticipated event
Break of exchange tower double containment inducing release of tritiated hydrogen to the operating area	Unlikely event
Break of high level water tank inducing large loss of tritiated water into the plant areas	Unlikely event
ICH&CD, ECH&CD and Diagnostics	
Ex-vessel LOCA due to small rupture of a PFW/BLK cooling circuit pipe inside port cell	Unlikely event

A list of incidents and accidents has been selected from the table above. They are representative of their category and their family of reference events. This list of incidents, accidents and their consequences is intended to be revised.

Incidents or accidents are categorized by severity levels:

- High (for Designed Basis Accident, DBA);
- Medium (Anticipated Operational Occurrence, AOO);
- Low (worker's dose above limits).

Four additional scenarios were added: 'Coolant leak in port cell', 'Leak of tritiated water from the Water Detritiation System', 'Cryostat helium ingress' and 'Fire in hot cell'. Finally, the event 'Loss of plasma control' was re-categorized as normal operation in accordance with the safety provisions.

Normal operation and plant conditions are those planned and needed for normal operation, including some faults and events, which can occur as a result of the experimental nature of ITER.

Incident sequences are not planned but are likely to occur one or more times during the life of the plant and do not include normal operation. The aim of incident event analysis is to show that no significant radiological releases result from these events and no serious consequential failures are expected.

Accident event sequences are not likely to occur during the life of the plant but are postulated in order to assess the safety of the facility. Accident sequences are nevertheless taken into account in the design and that is why we consider these events within the design basis accidents (DBAs).

Postulated events that are extremely unlikely to occur, for example because they involve two completely independent aggravating failures each of low likelihood, are called beyond design basis accidents (BDBAs). The ITER Preliminary Safety Report [II–1] mentions as one of the worst case scenarios (BDBA) a fire in an electrical cupboard with propagation to a glove box in the Tritium building. This fire may induce confinement damage leading to all of the tritium in the glove box being released within the building. Another BDBA is to consider multiple ruptures of cooling loops in the VV combined with some bypass issues.

The depth of the incident/accident analyses performed on the selected ITER events, as part of the DBA, have the objective to systematically identify, model and analyse the selected event sequences and to demonstrate that the ITER design has sufficient provisions to comply with its General Safety Objectives. To assess the potential for public radiation exposures and the effectiveness of implementing design requirements and fulfilling safety functions in the ITER design, a comprehensive analysis of reference events was carried out. They are presented in the ITER Preliminary Safety Report [II–1] and lists of DBA and BDBA for ITER are also summarized in Table II–3 and Table II–4, separately.

TABLE II–3. LIST OF DBA FOR ITER

DBA 1	Loss of Power	Loss of off-site power for 32 hours	accident
DBA 2	In-vessel events	In-vessel FW/BLK PHTS pipe leakage	incident
DBA 3		Multiple FW/BLK PHTS pipe break	accident
DBA 4		Loss of vacuum through one VV/cryostat penetration line	accident
DBA 5	Ex-vessel PHTS events	Loss of divertor heat sink and impact on the divertor	incident
DBA 6		Pump trip in the primary cooling loop of the divertor PHTS	incident
DBA 7		Pump seizure in the primary cooling loop of the divertor PHTS	accident
DBA 8		Large ex-vessel pipe break in the VV PHTS	accident
DBA 9		Large ex-vessel pipe break of the primary loop of the DV PHTS	accident
DBA 10		Heat exchanger leakage	incident
DBA 11		Heat exchanger tube rupture	accident
DBA 12		Coolant pipe break inside Port Cell	accident
DBA 13	Maintenance event	Stuck divertor cassette and failure of cask	accident
DBA 14	Tritium plant and fuel cycle events	Tritium process line leakage	incident
DBA 15		Accident with transport hydride bed	accident
DBA 16		Isotope separation system failure	accident
DBA 17		Failure of fuelling line	accident
DBA 18		Leak of tritiated water from Water Detritiation System	accident
DBA 19	Magnets	Toroidal field coil short	accident
DBA 20		Arc near confinement barrier	accident
DBA 21	Cryostat events	Cryostat air ingress	accident
DBA 22		Cryostat water ingress	accident

DBA 23		Helium ingress	accident
DBA 24	Hot Cell Events	Loss of confinement of a Hot Cell	accident
DBA 25		Fire in the Hot Cell buffer storage room	accident

TABLE II–4. LIST OF BDBA FOR ITER

1-BDBA	Loss of vacuum through VV/cryostat penetration line, plus 2-hour electrical blackout and in-vessel FW coolant leak
2-BDBA	Multiple failure of first wall cooling loops inside vacuum vessel together with failure of both windows in an RF heating line (‘wet bypass’)
3-BDBA	Loss of plasma control together with multiple failure of FW/BLK PHTS inside vacuum vessel
4-BDBA	Ex-vessel rupture of FW/Blanket primary cooling circuit, with failure of Fusion Power Termination System
5-BDBA	Large leak of air into the vacuum vessel followed by hydrogen deflagration or detonation, possible dust explosion
6-BDBA	Damage to vacuum vessel and cryostat resulting in large holes
7-BDBA	Large VV ex-vessel coolant pipe break plus loss of flow in all intact cooling loops
8-BDBA	Cryostat water and helium ingress
9- BDBA	Leakage from tritium process line in tritium plant together with failure of second confinement barrier or failure of DS
10- BDBA	Fire in the tritium plant with propagation to a glove box
11-BDBA	Hydrogen deflagration or detonation in the tritium plant
12-BDBA	Fire in the waste processing area with propagation to the buffer storage room in the Hot Cell facility

II–3. CONFINEMENT

The ITER fusion process uses tritium. During plasma pulses, only a very small amount of tritium is used during the process—only a few grams at any one time. Each pulse or plasma discharge will involve approximately 100 g of tritium of which approximately 0.3 g will be burnt [II–2]. Therefore, the burn-up fraction in the plasma is not expected to exceed 0.3%. In addition, several grams will remain trapped in the plasma-facing components and the vacuum vessel internal components such as the torus cryopumps. Thus, the most important safety objective at ITER is the confinement of tritium within the fuel cycle. A multiple-layer barrier system is to be designed to protect the workers, the public and the environment against the spread or release of tritium.

For ITER, the Licence[3] states the following:

> “Control of confinement
>
> “The design, construction and operation of the installation ensure the control of the risk of dissemination of hazardous substances, such as radioactive substances or beryllium, inside the installation or in its environment, in normal, incidental or accidental operation.

[3] Décret n° 2012-1248 du 9 novembre 2012 autorisant l'Organisation internationale ITER à créer une installation nucléaire de base dénommée « ITER » sur la commune de Saint-Paul-lez-Durance (Bouches-du-Rhône).

"The containment of hazardous materials consists of two systems, based on two modes of containment, static and dynamic:

(1) A first system located as close as possible to the materials and ensured in particular by the tokamak vacuum chamber and its extensions, the tritium processes, the maintenance cells and the associated ventilation systems as required;
(2) A second system aimed at limiting releases into the environment, consisting of the walls of the premises and buildings and the associated ventilation systems as required.

"Dynamic containment consists of :

- a nuclear or conventional ventilation system, called 'HVAC', used in normal operation ;
- a detritiation system, called 'DS', used in normal, incidental and accidental operation."

An extensive list of components important to safety contributing to the confinement safety function is presented in the preliminary safety studies, in accordance with the definition of SIC: systems, structures and components that perform a safety function and contribute towards the General Safety Objectives at ITER during incident/accident.

The first confinement barrier is the vacuum vessel while the Tokamak cooling water system vault including the pipe shafts and pipe chases, as well as the buildings themselves are part of the second confinement barrier.

The SIC classification applies on a system and a subsystem level. The SIC classification is given below:

SIC-1: Those SIC components required to bring and maintain ITER in a safe state,

SIC-2: Those SIC components used to prevent, detect or mitigate incidents or accidents, but not required for ITER to reach and maintain a safe state.

II–4. EQUIPMENT QUALIFICATION

ITER mechanical, electrical as well as instrumentation and control components classified as SIC fulfil their safety function during their life, in normal operation and accident conditions. The qualification includes, when applicable, integrated radiation exposure, humidity, pressure, seismic load and ageing simulating the cumulative operational environment. They will be qualified accordingly through analysis supported by testing when necessary.

Inspection, examination, testing and maintenance, will be performed in accordance with the ITER planned periodic maintenance and test schedule. This ensures that any item that fulfils a safety function is subject to suitable testing and maintenance to ensure its capability to continue to provide the safety function. Thus, re-qualification might be necessary to confirm safety performance capability prior to a full return to service.

REFERENCES TO ANNEX II

[II–1] ITER, ITER Preliminary Safety Report (RPrS), version 3.0, 12 Dec 2011.

[II–2] IRSN, Nuclear Fusion Reactors – safety and radiation protection considerations for demonstration reactors that follow the ITER facility, 17 Nov 2017.

ANNEX III. Plasma control

III–1. FUSION POWER PRODUCTION BY MAGNETIC CONFINEMENT AND ITS CONTROL – EXPERIENCE FROM ITER

One way to produce fusion energy is through the reaction of two heavy forms (isotopes) of hydrogen: deuterium and tritium. The nuclear reaction of deuterium (D) and tritium (T) produces helium (He) and one neutron.

$$D + T \rightarrow He + n + \text{Kinetic Energy } (\sim 3 \times 10^{-12} \text{ Joules})$$

Because the neutron is four times lighter than the helium nuclei, it carries most of the energy produced in the reaction (75 % of the total) while the helium nucleus carries 25%. Unlike conventional fission reactions, the D-T fusion reaction is not a chain reaction since the neutron produced does not lead to subsequent D-T reactions.

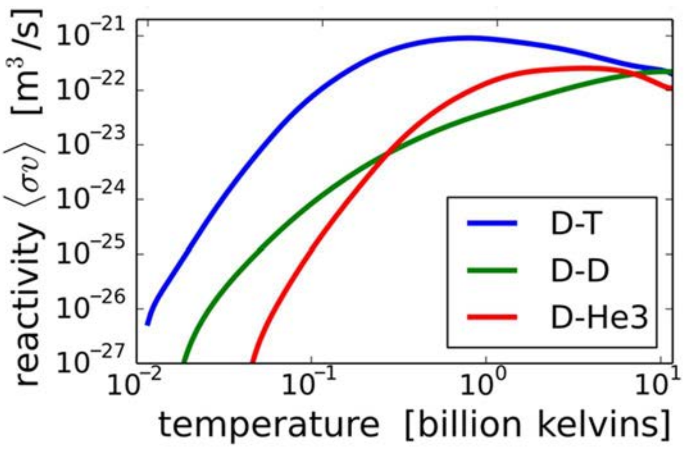

FIG. III–1. Reactivity of three fusion reactions versus temperature showing the higher reactivity of the deuterium-tritium reaction compared to the others and that it reaches its maximum value at temperatures of few hundred million kelvin (source https://en.wikipedia.org/wiki/nuclear_fusion).

To get the deuterium and tritium nuclei to fuse it is necessary to overcome the electrostatic repulsion force, since both nuclei are positively charged and repel each other. When the nuclei approach sufficiently close to each other the nuclear fusion process can effectively take place and this ensures a high production of fusion energy. Such close approach requires deuterium and tritium nuclei to collide at high velocities and this is realized by heating the deuterium and tritium mix to very high temperatures (hundreds of millions of Kelvins) (see Fig. III–1). At these high temperatures, the thermal agitation of deuterium and tritium ensures that a sufficient fraction of them have enough velocity for the fusion reaction to take place. In addition, the collisions between deuterium and tritium nuclei needs to be frequent enough for copious power production to take place; this implies that the density of deuterium and tritium is sufficiently high.

The fusion power produced in a fusion reactor is given by Eq. (III–1) below, where n_D and n_T are the deuterium and tritium densities, $<\sigma V>$ is the reactivity of the fusion reaction

(Figure III.1), E_{DT} is the energy produced in the D-T reaction and V is the volume of the hot deuterium-tritium gas in the reactor.

$$P_{fusion} = n_D \, n_T \, <sV> \, E_{DT} \, V \qquad \text{(III–1)}$$

In these high temperature conditions, the hydrogenic gas is ionized and electrons are not bound anymore by electrostatic forces to the hydrogenic nuclei; this state of matter is called plasma. Such hot plasmas lose heat by conduction and convection and they naturally expand because of their pressure in a similar way to hot air in a balloon. To sustain fusion power production it is necessary to keep the plasmas hot and with sufficient pressure since expansion and heat losses would decrease the plasma temperature and density and stop the fusion reactions.

Stars achieve these goals thanks to their huge dimensions, which slow down heat losses, and to their mass, which provides the gravitational force to compensate for the expansion of the plasma. Magnetic fields are one way to confine the plasma and achieve fusion power production on earth. This is what is called magnetic confinement fusion, with the most common reactor design types to demonstrate fusion using this method being tokamaks and stellarators; ITER is a tokamak design.

The long term goal for the developers of most magnetic confinement fusion reactors is to demonstrate net production of electric power by exploiting the D-T fusion reaction. In a magnetic fusion reactor, the energy carried by neutrons is to be captured and used to generate electricity, often via a conventional steam cycle. The energy of the helium nuclei is transferred to D and T ions in the plasma by collisions. This increases their temperature and enables the fusion reaction to self-sustain. Control of fusion power production is, therefore, carried out through the control of the density of fuel, of the proportion of D and T in the fuel mix and of the temperature of the plasma.

In a tokamak fusion reactor, D and T fuel needs to be continuously injected to replace the burnt fuel and to exhaust the ashes (helium). Without continuous provision of fuel, the D and T densities decrease (typically in timescales of seconds), the helium ashes accumulate and fusion power production is reduced and eventually stops. Similarly, the ratio of D and T in the plasma needs to be maintained near its optimum (1:1) to maintain fusion power levels. D and T density and D-T mix control are performed by the injection of neutral D-T atoms that become ionized in the plasma. This can be done by injecting gas at the edge of the reactor or by the injection of small frozen deuterium or tritium pellets at high speed that penetrate further into the reactor plasma.

The plasma temperature can also vary on timescales of several seconds in fusion reactors. Plasma temperature needs to be controlled to maintain the plasma within the range that provides maximum reactivity. Temperatures that are too low or high lead to ineffective fusion power production, as shown in Fig. III–1. Control of the plasma temperature can be done in a direct way by external schemes to heat the plasma. This is performed by injecting radiofrequency waves that transmit their energy to the plasma electrons and ions or through the injection of high-energy deuterium atoms (with velocities similar to that of the helium nuclei produced in the D-T fusion reaction. Indirect control of the plasma temperature can be done through the control of the fusion power production (e.g. through control of the D and T densities or of the mix of D-T fuel), since the helium nuclei constitute the dominant heat source that sustains the plasma temperature in fusion reactions.

In general, the control of fusion power production in tokamak reactors, also called burn control, can be effectively implemented through the two approaches above, i.e. D-T density and fuel mix control by fuelling schemes and temperature control by external heating schemes, but other approaches are possible. For example, the plasma density and temperature can be decreased by ‘worsening’ in a controlled way the confinement provided by the magnetic fields and increasing plasma losses or by the injection of ‘impurities’, such as noble gases (Ne, Ar, etc.), that radiate power away by emission of electromagnetic radiation and cool the plasma down.

In the event of loss of control, for instance of heating and fuelling schemes, the most likely event is that such schemes decrease the amount of injected fuel and heating power into the plasma. This reduces plasma temperature and D-T density and this reduces fusion power production, as a consequence. A hypothetical extremely unlikely event would be that control of both fuelling and heating schemes is lost and heat losses from the plasma evolve in such a way that plasma density and temperature rise out of control, while maintaining the temperature in the region of maximum fusion power production, and together with them fusion power increases. An uncontrolled temperature rise alone eventually leads to a decrease of the fusion power since the fusion reactivity decreases at high temperature (see Fig. III–1). It is further assumed that no other reactor control and protection scheme acts to prevent this rise, which is extremely conservative.

TO HERE In such a case, the fusion power increase in the reactor would be ultimately limited through the maximum achievable pressure that can be achieved for the magnetic fields applied. Above this maximum pressure, the balance between the expansion forces in the plasma due to its pressure and the compression forces by the magnetic fields becomes unstable and plasma confinement is lost, spontaneously terminating fusion power production. This is the so-called magneto-hydro-dynamic beta (β_N) limit and it is an intrinsic limit of magnetic fusion plasmas. The modelled fusion power achievable in ITER versus normalized plasma pressure to the magnetic field (β_N) up to this limit is shown in Fig. III2. The parameter $\beta_N = 2$ corresponds to nominal ITER operation with plasma current of 15 MA and fusion power production of 500 MW. The maximum achievable and self-terminating fusion power production in the unlikely event described above is in the range of 750-860 MW for ITER.

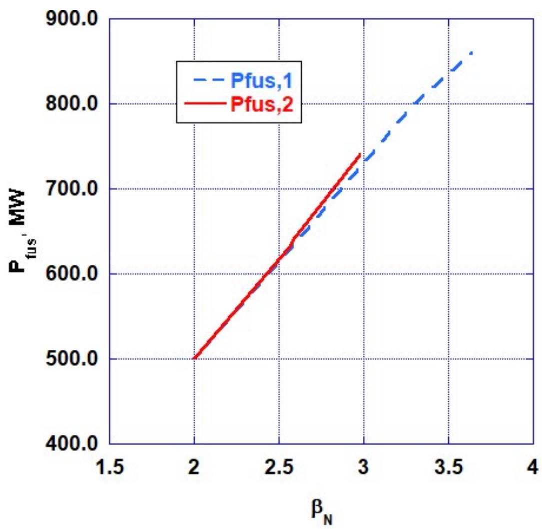

FIG. III–2. Maximum achievable fusion power in ITER versus normalized plasma pressure following loss of fusion power control assuming: (1) that core plasma heat losses and (2) access to second magneto-hydro-dynamic stability allow an increase of the plasma pressure up to the ideal magneto-hydro-dynamic b_n limit.

III–2. LOSS OF PLASMA CONTROL – EXPERIENCE FROM THE JET SAFETY CASE [III–1]

The plasma in JET is magnetically confined within the vacuum vessel by a combination of toroidal and poloidal magnetic fields. The plasma position is controlled to within a few centimetres of the inside surfaces of the vacuum vessel plasma-facing components. However, there are conditions under which the plasma may interact with the inner components in a magneto-hydrodynamic (MHD) event called a disruption. A disruption will deposit very large plasma energy into plasma-facing components over a very short time. This produces large currents which flow in the structural elements (halo currents) causing large, localized forces and in a worst case scenario, melting of in-vessel components.

All plasma configurations are carefully assessed for the maximum predicted forces which could arise in the event of a disruption. Configurations using high current (>2MA) and toroidal field (>3T) that could lead to substantial forces are limited and approval from the Chief Engineer is needed prior to pulsing. The maximum expected stress of 165 MPa is expected to be in the area of the root of the Main Vertical Port on the JET device.

A runaway electron discharge, usually at relativistic energies, may accompany the disruption event. The electron energy is dumped on interposing structures such as the poloidal limiters, thus protecting the vacuum vessel wall. A result of the electron impact would be localized melting and high energy X rays and photoneutrons. The radiation produced is insignificant compared to that generated during a pulse.

The vacuum vessel mechanical supports, or brakes, are locked in place using compressed air and serve to prevent vessel displacement due to disruptive loading. A separated manifold exists for the upper and lower brakes. A pressure switch on each manifold provides a signal to a PLC which inhibits pulsing if the manifold pressure reaches a pre-set limit. A loss of air, combined with switch or monitoring failure, would result in unsafe mechanical support during a disruption. This could possibly lead to damage to the vacuum vessel.

Failure of any in-vessel components which form part of the vacuum boundary may lead to an air ingress (LOVA). Failure of cooling or cryogenic systems due to a disruption may cause a fluid spill inside the vacuum vessel (LOCA).

The vacuum vessel support structure and the in-vessel components are designed to accept the largest estimated disruptive forces. Design basis accidents therefore encompass failure of the protection systems and a disruptive force occurring. A reduction in the frequency of disruptions may occur as a result of a computer-controlled system, such as the Plasma Fault Protection System, which can act rapidly on the magnets and heating systems via the Pulse Termination Network to allow a ‘soft stop’ to occur in the event of a plasma fault that could otherwise lead to a disruption. However, these systems are not credited towards meeting any particular safety targets during operations. The consequences of a loss of plasma control are limited to either a LOVA or LOCA and, as such, the specific safety measures defined in those fault sequences apply. For JET, provided that the Machine Protection operating limits and restrictions (identified above) that define the Operating Envelope are met, no further safety measures are necessary for loss of plasma control.

REFERENCES TO ANNEX III

[III–1] CCFE/DTE2/007, JET Torus DTE2 Safety Case, D Perry, J Johnston, R Wong, T Xu, N Wang, J Deane, Issue 1, 10/08/2020.

ANNEX IV. Questionnaire on Design Approaches for Fusion Power Plant (FPP) and Existing Relevant Experience Questionnaire

The text in this Annex is a reproduction of the questionnaire that was sent out to fusion operating organizations and designers, regulatory bodies and TSOs.

IV–1. BACKGROUND

Since the 1970s, the safety of fusion devices has been on the IAEA's agenda of safety activities. Over the past four decades, the fusion community has accumulated a substantial amount of knowledge and experience in fusion through the construction and operation of the fusion experimental devices around the world. This includes the design, construction and operation of TFTR and JET as well as the design and construction of ITER and CFTER. Also, a substantial private sector effort has emerged across multiple countries, and private funded companies are engaged in the design and construction of prototypes demonstrating a diversity of Fusion Power Plant (FPP) design approaches. The experience accumulated provides useful insights to develop design principles for future FPPs.

The information collected by the questionnaire will be used to inform and develop an understanding of safety aspects and principles regarding FPP design that will be captured by an IAEA Technical Document. The parties responding to the questionnaire will be acknowledged in the IAEA publication.

This questionnaire consists of two parts.

- The first part aims to gather public sector and private sector experts' views on the general safety design approach for FPPs in view of their safety characteristics.
- The second part aims to collect relevant technical information on specific design safety aspects and safety analysis relevant to FPPs from projects planned and under development (public and private). The questionnaire also aims to gather experience from existing and shutdown experimental fusion devices that can be of relevance to FPPs. For these cases, the questionnaire answers in the second part should focus on the specificities of the device that are relevant to FPPs.

All questions in this questionnaire focus on FPPs, used to produce net electrical power or heat from the energy released in a fusion reaction. A technical definition of FPP has been included in Annex I.

This questionnaire does not target specific FPP design options. The questions have been formulated generally to be applicable to the different FPP designs, to the extent practicable. There may be a limited number of instances where some questions do not fully apply to some designs or may need interpretation. Questions do not need to be addressed when not applicable or can be interpreted as needed to be adapted to specific FPP designs. If a respondent feels that a specific interpretation is necessary to answer a question, please provide an explanation of any such interpretation.

The information received in the questionnaire will be treated as confidential unless the contributor provides permission for disclosure. When using this information, references to specific design features will be generalised and anonymised. Comprehensive responses are important to ensure that the work of the IAEA is as technology inclusive as possible and considers a wide range of technological solutions, so any gaps or areas that need further work by the IAEA can be identified in a comprehensive manner.

IV–2. PART I- QUESTIONS ON GENERAL DESIGN SAFETY APPROACH

IV–2.1. Fundamental Safety Objective and Fundamental Safety Principles

The IAEA Fundamental Safety Principles (Safety Fundamentals No. SF-1) provides safety objectives and fundamental safety principles for all facilities and activities with radiation risks. Please indicate your views on any missing or not applicable aspects of these principles as applied to future FPPs.

IV–2.2. Quantitative Acceptance Criteria

Please provide your views on whether it will be useful to develop quantitative acceptance criteria for specific safety aspects of FPPs, including radiological acceptance criteria and probabilistic criteria on the occurrence of accidental conditions.

If this is considered useful, please indicate:

(1) What metrics do you recommend (for example, dose targets in normal operation, frequency and dose targets for accidents, etc.) and why?

(2) Please give your suggested approach for formulating quantitative acceptance

(3) What numerical values do you suggest for quantitative acceptance criteria for FPP?

(4) Your views on how these targets could be used in the evaluation of the device safety (e.g. using probabilistic analysis? deterministic analysis? simplified and conservative deterministic analysis?).

(5) Issues that you may envisage in the application of probabilistic analysis for FPP and suggested resolution (For example, non-existent data on failure rates of some novel component, need for such analyses if deterministic approaches may be suitable).

IV–2.3. Please identify the Fundamental and main subordinated FPP Safety Functions and your views on the approach to identify FPP safety functions.

Definition and examples of safety functions for FPPs are provided in Annex I.

IV–2.4. What are your views on the categorization of plant states to be considered in the design (as defined in IAEA Safety Glossary under entry 'plant states') of an FPP?

What plant states do you think should be considered for the design of an FPP (e.g. normal operation, anticipated operational occurrences, accidental conditions)?

Please indicate:

(1) Categorization of plant states;

(2) What criteria should be associated to each plant state in terms of impact on barriers and/or releases?

(3) What is necessary to ensure a 'safe state' in practice after accident conditions (when all fundamental safety functions can be ensured and maintained stable for a long time)

(4) Please provide examples of what you consider needs to be included as part of each plant state.

(5) How the different plant states need to be considered in the design?

(6) Please highlight any other issue of discrepancy.

Please, if possible, mention specific design features for each of the applicable plant states.

IV–2.5. What are the FPP features that are inherent to the fusion reaction and contribute to the effectiveness of physical barriers in confining radioactive material at specified locations?

For example, please identify FPP inherent features that would terminate the fusion reaction without external intervention, such as automatic or manual interventions to turn off power, control pollution of the plasma or stop tritium supply?

IV–2.6. Elimination by design of conditions that can lead to early radioactive release or a large radioactive release.

(1) Please identify what events may lead to early and/or large radioactive release for an FPP and should be eliminated by design (practical elimination).

(2) For each of these, please indicate what provisions are in place to ensure practical elimination.

For example, for a Tokamak design, please provide a detailed explanation of the consequences of plasma events and what provisions need to be in place for these events.

IV–2.7. Defence in Depth

The primary means of preventing and mitigating the consequences of accidents is 'defence in depth'. Defence in depth is implemented primarily through the combination of a number of consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people or to the environment. If one level of protection or barrier were to fail, the subsequent level or barrier would be available. When properly implemented, defence in depth ensures that no single technical, human or organizational failure could lead to harmful effects, and that the combinations of failures that could give rise to significant harmful effects are of very low probability. The independent effectiveness of the different levels of defence is a necessary element of defence in depth.

Defence in Depth (DiD) has been fundamental to preventing and mitigating the consequences of accidents for nuclear fission facilities.

Fundamental Safety Principle 8 (SF-1) indicates "Defence in depth is provided by an appropriate combination of:

— An effective management system with a strong management commitment to safety and a strong safety culture.

— Adequate site selection and the incorporation of good design and engineering features providing safety margins, diversity and redundancy, mainly by the use of:

o Design, technology and materials of high quality and reliability;

o Control, limiting and protection systems and surveillance features;

o An appropriate combination of inherent and engineered safety features."

In the fission NPPs DiD relies on five levels of DiD.

Please provide your views how the DiD concept may apply to FPP; If not, why not?

In your response, please include the consideration of whether the implementation of DiD may need to be graded to the FPP hazard potential, and your views on what levels of DiD may be relevant to FPPs.

IV–2.8. Your views on what should be key general design requirements for safety aspects of FPPs

The following requirements have been extensively used in the design of nuclear fission installations and may not apply to FPPs. Please indicate your views on their applicability to FPPs.

- Physical separation and independence of safety systems

Interference between safety systems or between redundant elements of a system is prevented by means such as physical separation, electrical isolation, functional independence, and independence of communication (data transfer), as appropriate.

- Common cause failures

The design of equipment takes due account of the potential for common cause failures of items important to safety, to determine how the concepts of diversity, redundancy, physical separation and functional independence have to be applied to achieve the necessary reliability.

- Single failure criterion

The single failure criterion is applied to each safety group incorporated in the plant design. Design is developed in a way that no single failure could result in a loss of the capability of a system to perform as intended, unless the time available from onset of the accident would be sufficient for operator actions.

- Fail-safe design

The concept of fail-safe design is incorporated, as appropriate, into the design of systems and components important to safety.

Systems and components important to safety are often designed for fail-safe behaviour, as appropriate, so that their failure or the failure of a support feature does not prevent the performance of the intended safety function.

- Operational limits and conditions for safe operation are established, including:

(a) Safety limits;

(b) Limiting settings for safety systems;

(c) Limits and conditions for normal operation;

(d) Control system constraints and procedural constraints on process variables and other important parameters;

(e) Requirements for surveillance, maintenance, testing and inspection of the plant to ensure that structures, systems and components function as intended in the design, to comply with the requirement for optimization by keeping radiation risks as low as reasonably achievable;

(f) Specified operational configurations, including operational restrictions in the event of the unavailability of safety systems or safety related systems;

(g) Action statements, including completion times for actions in response to deviations from the operational limits and conditions.

IV–2.9. Qualification of items important to safety

What are the expectations for the qualification of the novel components in FPP, such as vacuum vessel, blanket, fast discharge units, etc.? Do FPP components require qualification and why? Can FPPs have different qualification requirements compared to fission systems?

IV–2.10. Consideration of external hazards in the FPP design

What are your general views regarding the need to consider external hazards in the FPP design and what external hazards should be considered in an FPP design and why?

IV–3. PART II- REQUESTS FOR EXAMPLES FROM SPECIFIC ORGANIZATIONS

IV–3.1. Name and description of the fusion device:

- The envisaged fusion power plant (if relevant);
- The existing and shutdown fusion device (if relevant);
- Future fusion experimental and demonstration projects (if relevant).

Describe the similarities/differences of the fusion device with an FPP- what are the key aspects of the device that can be relevant for the design of a future FPP (e.g. JET for handling tritium).

Please focus the response to the following questions to only the areas that are relevant to FPP. Annex 1 describes key characteristics of an FPP from the safety point of view.

IV–3.2. Radioactive Inventories and Source Term

What is the inventory of your fusion device and how do you determine it (tritium, dust and activated corrosion products, activated water, systems that produce tritiated water, etc.). Further information and definitions in Annex I.

How is the source term derived from inventory (including identification of the release pathways)?

The amount and isotopic composition of radioactive material released or postulated to be released from a nuclear facility is called the source term.

(1) Source term during normal operation, if any;

(2) Source term for accident conditions.

Fraction of the inventory that is mobilized and then released following an accident through identification of the transfer functions up to the release point. Inherent contribution to release fractions.

IV–3.3. Physical Barriers

Please identify confinement barriers/systems which prevent the release of radioactive material and the key principles that you have used for their design (including the tritium management systems if applicable).

IV–3.4. Postulated Initiating Event (PIE)

(1) What approach have you used or plan to use to determine the PIE and scenarios of PIE?

(2) Please provide your proposed PIE table for each relevant plant state considered in the design.

IV–3.5. Systems and Components of your design – general concepts – complete as much as possible

Please identify the main structures, systems, and components (SSCs) for your fusion facility and related functions envisaged, if any. A system diagram is particularly welcomed. Please also describe key design requirements.

IV–3.6. Classification of SSCs

Please indicate your approach to identify and classify the safety importance of structures, systems, and components. If not applicable, please explain why.

(for more information please see IAEA-TECDOC-1851)

IV–3.7. Internal hazards and external hazards

Please identify the internal hazards and external hazards that you have considered or plan to consider in your design and why (e.g. considering information provided in effects and remarks column). If none, please explain your response.

IV–3.8. Radiation Protection Requirements

Please indicate the dose limits taken into account to design the radiation shielding.

Please indicate the arrangements related to facilitate maintenance, fuelling, operation, etc., in view of the radiation protection implications of the features in the FPP design.

Organizations who responded to the questionnaire:

The following organizations responded to the questionnaire used in the production of this publication:

- EUROFusion – DEMO, EU
- General Atomics, United States of America
- Helion Energy Inc., United States of America
- IBRAE RAN, SRC Kurchatov Institute, Russian Federation
- Institute of Nuclear Energy Safety Technology, International Academy of Neutron Science, China
- Korea Institute of Fusion Energy (KFE), Republic of Korea
- LPPFusion Inc., United States of America
- National Institutes for Quantum Science and Technology (QST), Japan
- Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Germany
- JSC 'N.A.Dollezhal Research and Development Institute of Power Engineering' (JSC RDIPE), Russian Federation
- Southwestern Institute of Physics, China
- China Nuclear Power Engineering Company, China
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