Review of the safety concept for fusion reactor concepts and transferability of the nuclear fission regulation to potential fusion power plants

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Abstract

This paper summarizes the current state of the art in science and technology of the safety concept for future fusion power plants (FPPs) and examines the transferability of the current nuclear fission regulation to the concepts of future fusion power plants. At the moment there exist only conceptual designs of future fusion power plants. The most detailed concepts with regards to safety aspects were found in the European Power Plant Conceptual Study (PPCS). The plant concepts discussed in the PPCS are based on magnetic confinement of the plasma.

The safety concept of fusion power plants, which has been developed during the last decades, is based on the safety concepts of installations with radioactive inventories, especially nuclear fission power plants. It applies the concept of defence in depth. However, there are specific differences between the implementations of the safety concepts due to the physical and technological characteristics of fusion and fission.

It is analysed whether for fusion a safety concept is required comparable to the one of fission. For this the consequences of a purely hypothetical release of large amounts of the radioactive inventory of a fusion power plant and a fission power plant are compared. In such an event the evacuation criterion outside the plant is exceeded by several orders of magnitude for a fission power plant. For a fusion power plant the expected radiological consequences are of the order of the evacuation criterion. Therefore, a safety concept is also necessary for fusion to guarantee the confinement of the radioactive inventory.

The comparison between the safety concepts for fusion and fission shows that the fundamental safety function “confinement of the radioactive materials” can be transferred directly in a methodical way. For a fusion power plant this fundamental safety function is based on both, physical barriers as well as on active retention functions. After the termination of the fusion process residual heat is produced by the activated materials. Correspondingly, the fundamental safety function “cooling” is also applicable to fusion. The analyses performed so far have shown that in the case of an adequate design of a FPP the residual heat can be removed by passive heat transport. For a fission power plant the fundamental safety function “reactivity control” should prevent power excursion, guarantee that the fission process can be stopped and re-criticality is prevented. The first aspect is not transferable to fusion, because such power excursions are excluded due to the physical nature of the fusion process. The requirement for the ability

to terminate the power production can be applied in principle to a fusion power plant. It is fulfilled by the inherent features. In a FPP it is by physical nature not necessary to consider re-criticality.

As in the safety concept of fission, postulated single initiating and multiple failure events, as well as severe plant states of a fusion power plant are assigned to different levels of defence, covering the range from normal operation to very rare events. The assignment is based on probabilistic criteria and the possible radiological consequences. In a fusion power plant measures and installations are foreseen to guarantee the compliance with radiological criteria. The measures and installations are based on inherent physical principles, and passive and active safety systems. For a fusion power plant, the criteria for the measures and installations on the different levels of defence are not yet as detailed as for a fission power plant.

The safety analyses for fusion performed so far have focused on plant-internal events. For these events in an adequately designed fusion power plant, only relying on inherent and passive safety features, the analyses showed that there will be no need for an evacuation outside the plant. Together with the development of more detailed plant concepts also events resulting from external hazards, e. g. earthquakes and flooding, or very rare man-made external hazards (e. g. the crash of a large air plane) have to be discussed in more detail.

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1 Introduction

For several decades fusion power plant (FPP) concepts have been under development. However, no official regulatory framework exists so far which covers fusion power plants. E. g. in Germany, the nuclear regulatory framework for nuclear fission reactors would not be applicable formally to a fusion power plant. Nevertheless, in parallel to the plant development, safety concepts have been developed for fusion power plants based on the concept of defence in depth.

The objective of this paper is to analyse whether the safety concept of fusion power plants fits to the existing nuclear regulation, which was developed for the current commercial nuclear (fission) power reactors. For the purpose of comparing and possibly integrating the safety concept of fusion into the existing nuclear regulatory framework a literature survey was performed.

Basis for this survey were different European studies (Safety and Environmental Assessment of Fusion Power — Long Term Programme (SEAL) and Safety and Environmental Assessments of Fusion Power (SEAFP) /GUL 00/, /COO 99/, /SEA 95/; Power Plant Conceptual Study (PPCS) /MAI 05a/) as well as the licensing documentation of the ITER project /PSR 11/ which provided the most detailed description of a fusion safety concept. In addition this survey is based on the current German nuclear safety regulation, i.e. safety requirements for nuclear power plants of 2012 /BMU 13/.

As these studies for fusion power plants are based on plant concepts using a plasma of deuterium and tritium confined by magnetic fields, the corresponding fusion devices (tokamak and stellarator) are described in chapter 2 for reference.

The general safety concept of fusion power plants is described in chapter 3. The literature survey concentrated on the three aspects of the safety concept: The identification of postulated single initiating and multiple failure events as well as severe plant states (incidents and accidents, design basis and beyond design basis), the analyses of event sequences, and the investigation and description of precautionary, preventive and mitigative measures for the prevention and mitigation of consequences in case of incidents or accidents. These topics are described in chapters 4, 5 and 6.

In chapter 1 two safety concepts adopted for nuclear fission power plants (NPP) in the past and at present are compared with the safety concept of fusion power plants: The concept of a worst-case enveloping event and the concept of defence in depth. One implementation of the concept of defence in depth is realized by the German nuclear safety requirements /BMU 13/. It is reviewed if the fundamental requirements of nuclear fission like those formulated in /BMU 13/ could be applied to fusion power plants, and possible modification and extensions of the nuclear fission regulation were identified. Also, topics currently covered by the nuclear fission safety requirements are identified which were not discussed in the surveyed fusion literature (e. g. independence of systems on different levels of defence, external hazards, very rare man-made external hazards).

The results of the comparison between the fusion safety concept and /BMU 13/ are assessed in chapter 1. In chapter 8 the results are summarised and an outlook is given on topics which should be addressed in future developments of the safety concept of fusion power plants.

2 Energy from nuclear fusion

The development of fusion as a viable energy source is based on reactions between high temperature deuterium and tritium plasma ions resulting in the formation of 3.5 MeV α-particles and neutrons of 14.1 MeV kinetic energies, respectively. To achieve fusion reactions with a sufficiently high rate, the burning temperature of the D-T plasma has to be in the range 100 to 300 million K (8 to 25 keV). Here, we focus on the "Magnetic Confinement" (MC) or "Magnetic Fusion Energy" (MFE) concept. Since tritium is unstable (beta-decay with 12.26 years half-life), it has to be produced by neutron induced reactions in a lithium containing blanket around the plasma. There, in particular, neutron capture reactions on stable Li-6 and Li-7 isotopes (natural abundances of 7.4 % and 92.6 %, respectively) take place.

Since fusion is a nuclear energy source, due to tritium usage and neutron generation, the European Fusion Programme, in particular, at an early stage has already dealt with the safety implications. The evaluation of this programme (mainly based on tokamaks and stellarators; see section 2.2), has led to two "central" recommendations reported in /COL 90/:

The first recommendation reads: "It must be clearly shown that the worst possible fusion accident will constitute no major hazard to populations outside the plant perimeter that might result in evacuation".

The second recommendation reads: "Radioactive waste from the operation of a fusion plant should not require isolation from the environment for a geological timespan and therefore should not constitute a burden for future generations".

2.1 Magnetic confinement

The concept is based on the presence of closed field lines in a magnetised hot plasma which keep the hot plasma away from material walls. The most obvious approach is to bend a cylindrical magnetic field into a torus by a toroidal arrangement of the magnetic field coils. Poloidal field components have to be added to create nested magnetic flux surfaces and thus to avoid plasma energy losses due to drifts in the inhomogeneous toroidal magnetic field. The most important devices among the different magnetic confinement configurations are tokamaks and stellarators.

2.2 Tokamaks & Stellarators

In a tokamak, a toroidal plasma current is induced by an Ohmic Heating (OH) transformer generating the required poloidal field. Additional poloidal magnetic field coils provide vertical fields by which the radial position and the shape of the equilibrium magnetic surfaces are controlled (Fig. 2.1, left hand side). Since the plasma current is driven only during the discharge of the transformer, the plasma is confined in a pulsed mode. In an MFE power plant, steady state plasma operation has to be achieved by additional current drive methods. A considerable part of the total plasma current may be produced by the plasma itself due to the bootstrap current effect. The conditions for reaching a high bootstrap current fraction constitute a challenge for the control of magneto-hydrodynamic (MHD) instability modes. The plasma current provides a self-organized plasma equilibrium susceptible to a number of instabilities including kink modes, sawtooth oscillations, conventional and neoclassical tearing modes (NTM), resistive wall modes, and current disruptions. In addition, vertical position instabilities may occur in tokamaks, which lead to Halo currents into plasma facing components when the plasma comes into contact with them. The electromagnetic forces associated with disruptions and Halo currents may lead to severe damages. At present, a good physics understanding of these instabilities has been achieved and stable operation scenarios have been developed. A significant level of active feedback control, however, is necessary in a tokamak to avoid such instabilities or mitigate their consequences /ARI 08a/.

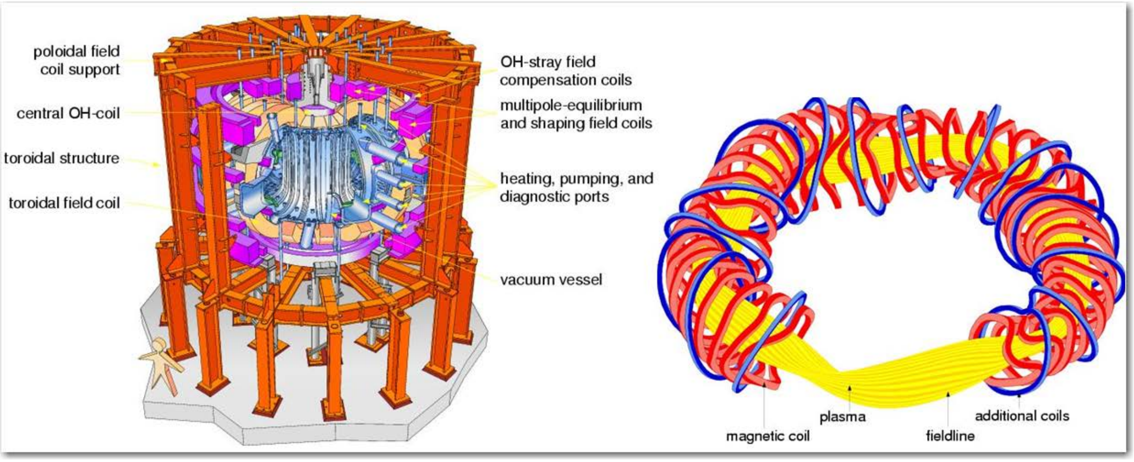


Fig. 2.1 Magnetic field coils in tokamaks and stellarators: left: components of Asdex Up-grade, right: coil system of the Wendelstein W7-X stellarator

Despite still existing challenges the tokamak is presently the most mature candidate for a fusion power plant. In particular, the JET tokamak is presently the largest MFE experiment in operation with great relevance for the planned ITER operation. In particular, in JET 16 MW of fusion power have been achieved for about 1 second /GIB 98/. The main objectives of ITER are to demonstrate a burning D-T plasma (for a duration of 300-500 seconds) with a fusion energy gain of Q~10, where Q is the ratio between total fusion power and external heating power. This means that 2/3 of the total plasma heating power consists of α-particle heating. Other objectives include the production of steady state plasmas with Q = 5 and the development and tests of fusion technology for a commercial fusion power plant /SHI 07/.

Whereas the magnetic configurations of the existing large tokamaks and of ITER are similar, stellarators including so-called torsatrons/heliotrons and similar devices feature a variety of different configurations. In all cases, the poloidal field component is generated by external coils. Therefore, such systems can be operated without any externally driven plasma current. Hence, stellarators have an inherent potential for stationary operation. In addition, current driven instabilities, in particular disruptions, do not exist in stellarators. The external coils provide a set of nested magnetic surfaces which entail, to a large extent, passive control of the plasma. The existence of an external confining magnetic field keeps the plasma always centred in the plasma vessel. In classical stellarators such as Wendelstein 7-A (Max-Planck Institute for Plasma Physics (IPP), 1975-1985) toroidal field coils in combination with additional helical windings on top of the vacuum vessel are used to generate the stellarator field. With the development of advanced stellarator configurations such as Wendelstein 7-AS (IPP, 1987-2002)

/HIR 08/ and the optimized Wendelstein 7-X stellarator (IPP, start in 2014) /GRI 98/ sets of non-planar modular field coils were introduced. They provide the full required three-dimensional (3-d) stellarator field (Fig. 2.1, right hand side). At present, the Large Helical Device (LHD) in the National Institute for Fusion Science in Japan /KOM 10/ is the largest operating heliotron-type stellarator.

In proposals of a roadmap towards the development of commercial MFE a demonstration power plant (DEMO /NEI 12/) is defined along the next step after the ITER tokamak project. Proposals and extensive safety analyses for commercial fusion reactors have been made in the framework of the European Power Plant Conceptual Study (PPCS, see section 4.1.2) /MAI 05a/, /MAI 05b/, /MAI 08/. Presently, a number of physics and technological challenges still exist for the realization of a DEMO /ZOH 13a/, /ZOH 13b/. In particular, a list of five DEMO physics issues is identified by the present EU fusion program:

1. Steady state tokamak operation
2. High density operation
3. Power exhaust
4. Disruptions
5. Reliable control with minimum sensors and actuators

Stellarators would offer solutions to at least some of these key issues. Steady state operation is inherently possible. High density operation is a favoured scenario in stellarators due to the lack of a Greenwald-like density limit /HIR 08/. Disruptions and accompanying effects by excessive forces and runaway electrons do not exist in stellarators. Also, the number of actively controlled parameters and related sensors is smaller in a stellarator due to the predominantly passive control by the external static magnetic field. The power exhaust, however, is a challenge for both concepts. The larger aspect ratio in stellarators (see Table 2.1) may help to keep the particle and energy flux densities at lower values, but the divertors in a stellarator are not yet sufficiently explored and the 3-d shape of divertor, first wall and blanket requires more elaborate solutions. A particular concern in stellarators is the density and impurity control. The High Density H-mode regime (HDH) in W7-AS /HIR 08/ is very promising in this respect. The main characteristics of tokamak and stellarator systems are summarized in Table 2.1.

Table 2.1 Main Characteristics of tokamaks and stellarators (W7-X type)

	Tokamak	**Stellarator**
Magnetic Field	toroidal field coils (planar), poloidal field coils (planar), vertical field coils (plasma position and shaping)	modular non-planar coils (combined toroidal and poloidal field)
Plasma Current	inductive, current drive (CD) systems (field line twist / rotational transform)	no current, rotational transform by external field
Aspect Ratio	small, ~ 3	generally larger, up to 10
Symmetry (field, vessel)	axisymmetric	non-axisymmetric
Divertor	“Single Null”, axis-symmetric	“Island Divertor”, 3-d shape
Stability Limits	current driven instabilities (tearing & kink modes, disruptions), vertical instabilities	pressure driven modes (interchange-like), passively stable by magnetic well
Density Limit	Greenwald (current) limit, degradation of confinement, ultimately disruptive	heating/radiation power limit (slow thermal decay)
Steady State Operation	requires steady state CD	inherent steady state capability

Therefore, in particular in Europe, alternative devices including the Wendelstein 7-X stellarator and DEMO versions on the basis of the HELIAS (W7-X like configurations) /WOL 12/ are considered in the roadmap towards an economical fusion power plant. Besides initial HELIAS reactor studies /BEI 01/, /SCH 12a/, /SCH 12b/, reactor studies have been made on the basis of Force Free Helical Reactors (LHD-like configuration) /SAG 98/, /SAG 06/, /GOT 12/, and on compact stellarators /ARI 08b/.

2.3 Components of fusion power plants and their functions

The dependence of the magnetic confinement on size parameters (such as the major torus radius) imply fusion reactor unit powers corresponding to about 1-3 GW_e (electrical power). By assuming a generic FPP delivering about 1.5 GW_e, a fusion power in the range 2.5 to about 5 GW_{fus} is required, depending mainly on the efficiency of energy conversion. These assumptions are identical with those of the PPCS. The fusion powers correspond to tritium consumption rates in the range of 0.38 to 0.76 kg T per day. These numbers illustrate the safety implications due to tritium which have to be evaluated by careful safety analyses.

An overview of the most important components and their functions in a generic FPP is given in the following. For this study, it is proposed to use a common safety analysis approach for a generic FPP, largely independent of design details. In particular, fundamental safety-relevant differences for tokamak and stellarator based reactors are not expected.

Table 2.2 Central components & systems of a FPP

Main Systems	**Basic Functions**	**Auxiliary Systems**
Plasma	source of fusion energy	central systems, such as shaping coils
Divertor	plasma energy & particle exhaust	divertor cooling system, remote handling system
First Wall	plasma-material interface	cooling system
Blanket	neutron power absorption, tritium production	cooling system tritium extraction system remote handling system
Neutron / Heat Shield for Magnets	magnet protection	cooling system
Vacuum Vessel	provision of plasma burn conditions, radioactivity barrier	vacuum pumps, diagnostics, cooling system
Magnet System	provision of magnetic confinement	cooling system, coil power supplies, quench detection
Cryostat	provision of the conditions for coil superconductivity	cryo-plant

Table 2.2 contains a schematic compilation of the systems and components of the central FPP components. They are divided into main and auxiliary systems and ordered

according to their location starting from the plasma and proceeding outward to the outer cryostat wall.

Another set of systems and components is listed in Table 2.3. It shows further important, radially penetrating components as well as peripheral components.

Table 2.3 Radially penetrating and peripheral components & systems of a FPP

Main Systems	Basic Functions	Auxiliary Systems
Heating & CD Systems	generation of hot steady-state plasmas	power supplies, cooling systems
Fuel Cycle System	fuel supply, D-T-He-ash removal	tritium storage, isotope separation plant, vacuum system, divertor pumps, tritium processing plant
Cooling Systems	temperature control of plant structures and magnets	heat exchangers, cryo-plant, heat sinks
Energy Conversion System	extraction of fusion energy (from blanket), electricity generation	blanket/divertor cooling system, steam generator/heater, turbine, heat sink
Radioactive Materials Facility	storage and temperature control of radioactive materials	storage & cooling systems, hot cells
Detritiation System	control of radioactive contaminations	cooling systems, tritium processing plant
Electrical Power Network	electricity supply / delivery	control systems, connection to external grid

Without describing the functions of the different systems in detail, Fig. 2.2 illustrates the basic components and functions of a generic FPP. It is taken from the PPCS report /MAI 05a/.

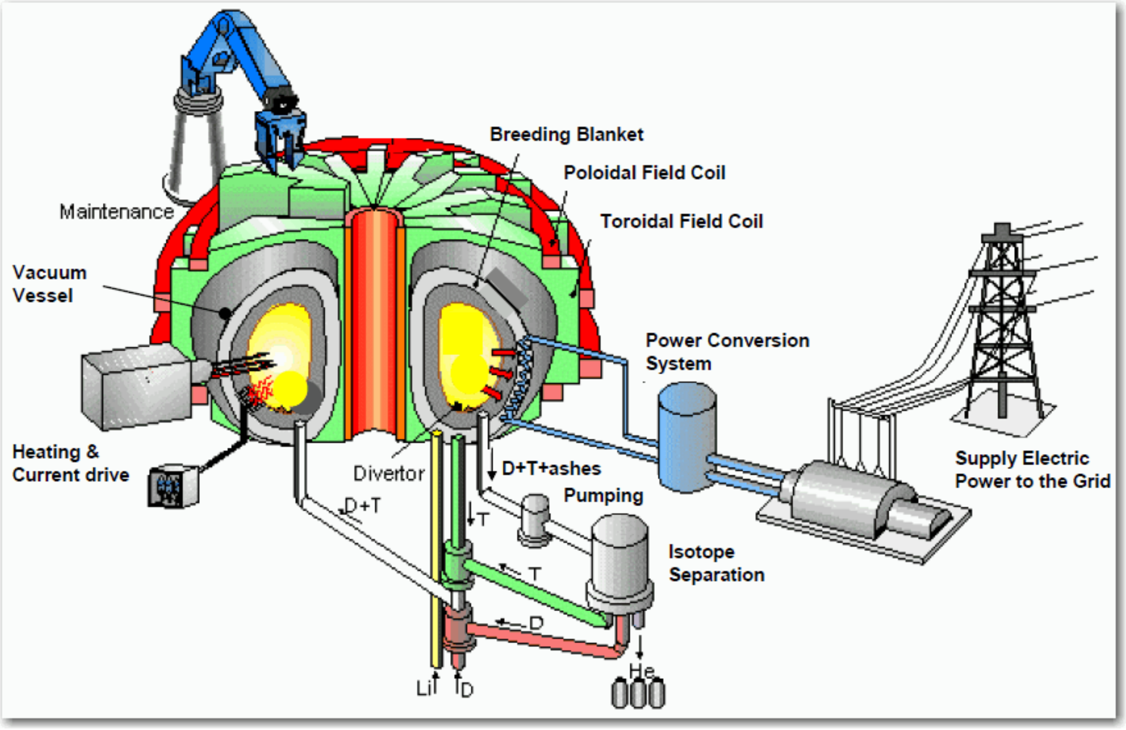


Fig. 2.2 Schematic view of main components and systems in a FPP (taken from /MAI 05a/).

2.4 Safety implications

Given the subject-matter of this study, the description of physics and of the technical components and their functions (section 2.3) is briefly put into a safety perspective. Details will be given in the next sections of the report (chapters 5 and 6), which deal with the actual safety analyses and the consequences of the release of radioactive material.

Due to the presence of similar systems and components (as shown in Table 2.2 and Table 2.3), the safety considerations for tokamak and stellarator are expected to be very similar. However, the different magnetic confinement concept makes event sequences in tokamaks more complex and serious than in stellarators. Actually, accidents which finally entail the ingress into the plasma of materials from plasma-facing components (PFCs) or from cooling systems will cause disruptions in tokamaks aggravating the event sequences, or at least requiring more demanding design solutions. This is because disruptions, accompanied by vertical displacement events, lead to enhanced energy fluxes to PFCs and large magnetic forces that may cause damages to PFCs such as coolant leaks. Further consequences could be increased dust production/mobilization along with increased risk of combined hydrogen (D, T) and dust explosions. For the following it is assumed that at this stage of conceptual design the safety assessments for stellarators can be reasonably encompassed by tokamak-related assessments.

2.4.1 Fusion Safety Characteristics

Some of the following text is based on the discussion from the SEAFP /SEA 95/ and SEIF /SEI 01/ reports. The energies stored inside the plant (see next sub-section) can hardly compromise the overall reactor assembly when released on their physics-based time scales. To maintain the burning plasma of a commercial FPP, the plasma chamber contains the fuel tritium which is produced in the power plant so that major shipments from/to off-site facilities are avoided. The nuclear power densities in the plasma are low compared to fission power reactors. The spent fuel consists of stable helium. The isotopic mix of all radioactive materials results in a radiotoxicity, lower by several orders of magnitude than in a NPP.

Continuous operation of the plant is maintained by refuelling with the D-T mixture. The fuel inventory in the plasma chamber at any time is sufficient for only about 1 to 2 minutes. So, the plasma burn can be stopped on this timescale by terminating the fuel supply. In case of an event which would impair the integrity of plasma-facing components or cause their overheating, impurities would enter the plasma causing an immediate thermal quench (on the time-scales of energy and particle confinement which are a few seconds).

Since tritium and deuterium can accumulate inside the plasma vessel one may assume, to be conservative, that tritium and deuterium outgassing from the surfaces of PFCs could sustain plasma burn. This would require the tritium and deuterium to arrange itself in a suitable way inside the magnet field configuration so that it would effectively fuel the plasma. However, such an event is considered to be implausible.

Power excursions, due to over-fuelling have in fact been considered for ITER. This scenario – whichever way it is caused – is the basis of an event studied in /PSR 11/. The conclusion is that the hazard of prolonging plasma burn does not exist, rather there exists a certain possibility of a fusion power overshoot.

Plasma burn has to be terminated reliably within about 3 seconds. This is the task of a “Fusion Power Shutdown System” (FPSS) which simply injects impurities. In addition, there exist several inherent feedback mechanisms (such as the release of material from the PFCs) which extinguish the fusion process in uncontrolled situations.

A criticality accident cannot develop. After burn termination, the residual power density due to the decay of activated structure materials is not high enough that – a proper design of the plant taken into account – melting of plant components is possible even in case of a total loss of cooling.

The plasma vessel also contains radioactive dust. These inventories are low by comparison with radioactive inventories of fission equivalents. The associated FPP radiotoxicity decays rapidly /GUL 07/.

The properties outlined above support safety and are considered to be inherent (according to the definitions used in the German rules and standards of safety requirements for nuclear power plants /BMU 13/). An inherently safe design is based on those principles of the laws of nature which by themselves have a safety-directed effect.

Even though the inherent properties may lead to the conclusion that active safety systems are not mandatory, they will be included in the design of commercial FPPs to help limiting damage and consequences at all levels of defence.

Active safety systems are also very important during normal operation to limit individual and collective doses to personnel. The active safety systems contribute to the implementation of the ALARA principle (to reach an exposure level “As Low as Reasonably Achievable”).

At present, a typical example is provided by ITER’s active detritiation systems (DS) that provide an essential element of the second confinement system. They will be highly reliable, with redundancy to cope with failures (see section 6).

If the active DS are completely lost, ITER’s confinement is still maintained by the static leak-tightness of the building. Some leakage will occur if the internal pressure rises above its normal sub-atmospheric value, so the extent of the environmental release will depend on how long it takes to bring the DS back into service.

For the reactor concepts analysed in the PPCS, a “bounding” accident analysis was performed (see section 3).

2.4.2 Energy Inventories

Stored energies may have the potential to destroy the integrity of confinement systems and to mobilise hazardous materials, thus causing their release into the environment. Therefore, the most important energy inventories in a commercial FPP are compiled and discussed in the following.

The D-T fuel content in the plasma vessel can sustain the plasma burn for only 1 to 2 minutes. The energy stored by the fuel mixture in the vessel is 325 GJ for the reactor models used for the SEAFP / SEIF studies. The complete conversion of this energy in fusion heat would require active sustainment of the burn process. It should be noted that under normal conditions, only a burn fraction of 2 % can be reached resulting in an energy release of 6.5 GJ.

The hot plasma of a FPP would typically contain a thermal energy of 1 to 2 GJ. The associated volume averaged energy density is low corresponding to plasma pressure of 3 to 6 bar.

The large energies stored in the magnetic field of a fusion reactor amount to typically 200 GJ and are a potential hazard to the first confinement barrier. Therefore, the design of ITER’s superconducting magnet system includes multiple monitoring, fault detection and protection systems. In particular, there is a safety-grade quench detection and a fast discharge system for the toroidal field coils /CIA 10/. Together with robust design, these measures aim at a minimization of the probability of magnet faults that could entail damage to the first confinement barrier.

Nonetheless, a hypothetical event sequence has been analysed to assess the bounding consequences of damage to confinement barriers /PSR 11/. Together with other conservative assumptions, large holes (1 m^2) in the plasma and cryostat vessel walls were postulated, caused by a release magnet energy. The analysis showed that even in this case the radiological consequences would be below any evacuation limit. To put the case of magnet energy implications on an even firmer basis, additional detailed analyses are performed at present.

Typical values of energy inventories are shown by Table 2.4, which is based on the SEAFP, SEIF and PPCS reports.

Table 2.4 Main energy inventories in a FPP

Energy Source	Energy	Reference
In-vessel fuel (DT)	~ 325 GJ	SEAFP, SEIF
Magnetic field	~ 200 GJ	SEAFP, SEIF
Plasma thermal energy	1 to 2 GJ	SEAFP, SEIF, PPCS
Primary coolant water enthalpy	~ 400 GJ	SEAFP, SEIF

Like in a NPP, the residual heat generation in the FPP is not stopped completely after shutdown, but will continue at a few percent level (e.g. less than 2 % level at shutdown for PPCS plant model B in /CIP 02/), and decrease exponentially for the time after. This heat generation is due to the radioactive decay of the materials that were exposed to the neutron irradiation. This heat source has to be considered in order to ensure the long term cooling of the reactor and can play an important role in aggravating the consequences of an accident if active cooling cannot be provided.

Additional energy can be released during accidents by chemical reactions; a typical example is the reaction of water (coolant) and Be (neutron multiplier) in solid breeder reactors in case of a loss of coolant accident in the Vacuum Vessel. Here, steam can have an exothermal reaction with Be at temperatures greater than 600°C.

Furthermore, chemical reaction of i.e. water at high temperature with metals (e.g. Be or W) can produce hydrogen. If this element is mixed with oxygen it can produce fire, detonation or deflagration, releasing mechanical and thermal energy.

Hence, depending on the materials used in the FPP (coolant (especially if water used), structural materials, breeder and neutron multiplier combination) potential release of chemical energy should be considered in the energy inventory.

2.4.3 Radioactivity Inventories

Tritium constitutes a major radioactive inventory in a FPP. As the tritium consumption in the vacuum vessel (VV) amounts to 153 g tritium per day for an average fusion power of 1 GW_{fus}, relevant inventories of tritium can be built up in the VV and in the fuel system due to the recirculating fuel (D-T). In addition, part of the tritium can be trapped in the surrounding structures and due to its high permeability it can escape from pipes and containing vessels. Hence, the confinement of tritium inside barriers constitutes the major safety issue, since it can be easily mobilized in the form of gas or HTO vapour in case of an accident and be release to the environment. Also large parts of the fuel that is deposited in in-vessel structures can be mobilized at high temperatures.

The burning plasma is a large neutron source that can produce large radioactive inventories by transmutating surrounding elements in radioactive isotopes. The amount and composition of these radioactive inventories varies within the structure of a FPP, corresponding to the different components and their material composition. In general these inventories are embedded in large structures (e.g. steel structures of blankets and divertors) that cannot be easily mobilised also in case of large accidents. However, part of these inventories appears in a form that can be easily mobilised; this part of the radioactive inventories is considered in the safety analysis as potential source of release.

To determine the maximum releasable inventories the following source terms (in addition to T) are taken into account:

- Dust: it is produced and accumulated in the VV: Particles of the size of µm, concretionary drops and flakes (mainly tungsten and stainless steel) coming from plasma wall interaction in normal operation, as well as in off normal and accidental events (e.g. disruption). Large fraction of the dust inventory can be easily mobilised.
- Activated Corrosion Products (ACPs): accumulated mainly in the cooling loops due to the corrosion/erosion action of the coolant (in particular water or liquid metals). The mobilisable fraction is in general only few percent of the total inventory.

Typical radioactive inventories for various PPCS models are listed in Table 2.5. They are used in the course of accident analysis /MAI 05a/.

Table 2.5 Radioactivity source terms for accident analyses used in PPCS models

Inventories	Model A	Model B	Model AB /CAP 05/	Model C
Tritium in VV [MF=1]	1 kg	1 kg	1 kg	1 kg
Dust [MF=1]	10 kg (7.6 kg of SS-dust + 2.4 kg W-dust)	10 kg (7.6 kg of SS-dust + 2.4 kg W-dust)	10 kg (W-dust)	10 kg (8.55 kg of ODS-dust + 1.45 kg W-dust
Tritium in coolant [MF=1].	15 g (per loop)	1 g (per loop)	1 g (per loop)	3 E-3 g (per loop)
ACPs total inventory [MF=0.01]	50 kg (per loop)	-	-	-
Note: MF: mobilisation fraction (ration between mobilisable part and whole inventory)				

3 Fusion safety approach

A safety approach for fusion reactor concepts was proposed in the framework of the PPCS and is reported in the work of Karditsas /KAR 04/. This approach is based on the identification of the potential hazards which are directly related to radioactive materials and which could lead to radiological consequences if no protection was defined. Although measures are taken to control those hazards, there is a finite probability that an accident may happen and it is necessary to identify these residual risks.

This approach requires that events with significant probability of occurrence have only minor or no radiological consequences, events resulting in relatively high radioactive releases are of very low probability of occurrence, and hypothetical events which could require evacuation are impossible. Fig. 3.1 from /GUL 12/ illustrates this approach. It schematizes the behaviour of public dose vs. probability (i.e. occurrence rate or frequency) on a logarithmic scale. The red line in the figure separates dose rates in the acceptable risk zone (left) and the not acceptable risk zone (right) depending on the probability. At present, there is an international fusion safety practice to discriminate between Design Basis Accidents (DBA) and Beyond Design Basis Accidents (BDBA). BDBA situations are constructed from the Design-Basis ones by postulating additional independent failures (including safety function failures) that may aggravate the sequence or by postulating events with extremely low probability of occurrence as stated in /PSR 11/. In practice, that implies considerations of probability. Therefore, as shown in Fig. 3.1, accidents events with probability of occurrence larger than 10^{-6} per year are classified as DBA; while they are considered as BDBA with a probability of occurrence lower than 10^{-6} per year (10^{-6} is given in /KAR 04/). In /IAE 00/ BDBA is superseded by Design Basis Extension conditions.

The field of acceptable accidents is further limited by a maximum admissible dose rate that is related to the so called “fusion bounding accident sequences” (green bar), which are the worst consequences of an accident driven by in-plant energies. This was assumed to be a total loss of cooling from all loops in the plant, with no active cooling, no active safety system operating, and no intervention whatever for a prolonged period. The decay heat is rejected by passive conduction and radiation only. For this kind of events a maximum limit of dose of 50 mSv was selected, which is recommended by the ICRP.

Accidents in the not acceptable risk zone can be brought back to the acceptable risk zone taking proper countermeasures, e.g. improving the confinement reduces the dose rate and can move the event down to the acceptable risk zone; adding certain safety systems, even if the dose rate is unchanged, can reduce the probability of the event moving it to the left into the acceptable risk zone.

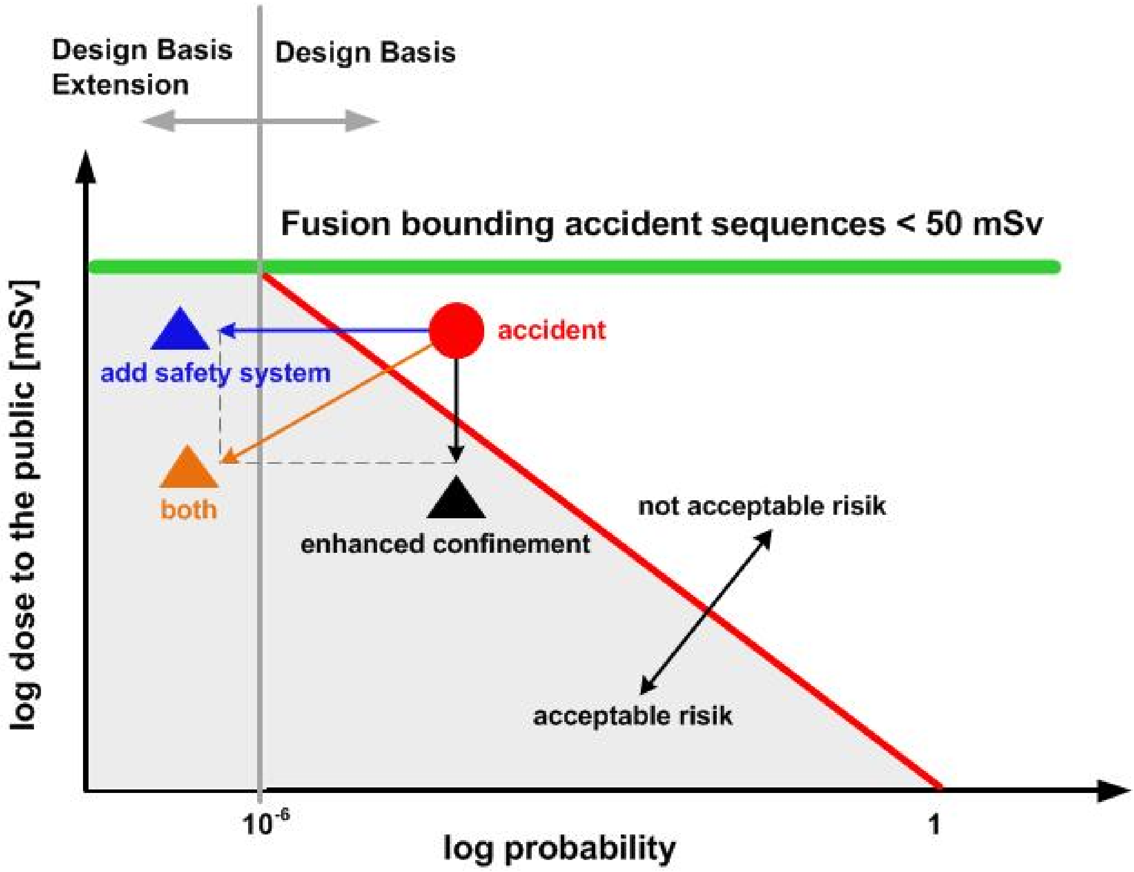


Fig. 3.1 Safety risk approach /GUL 12/

The safety requirements for a fusion facility are /KAR 04/:

- To protect the public and the environment against radiological hazards
- To protect the site workers against radiation exposure that must be maintained As Low As Reasonably Achievable (ALARA)
- To take measures to prevent accidents and to mitigate their consequences should they occur
- To avoid the need for public evacuation in any accident
- To minimize the amount of fusion facility waste

Due to the safety requirements it is necessary to establish and maintain a defence in depth against radiological hazards for DBA and BDBA. The defence in depth concept is applied to all safety activities, whether organisational, behavioural or design related, to ensure that they are subject to overlapping provisions, so that if a failure should occur, it would be detected and then compensated for or corrected by appropriate measures. Following the approach in INSAG-10 /IAE 91a/ measures relative to defence in depth are ranked in five levels of defence (LoD) in the PPCS:

Level 1 Prevention of deviations from normal operation and system failures

Level 2 Control of deviations from normal operation and detection of failures

Level 3 Control of accidents within the design basis

Level 4 Control of severe conditions

Level 5 Mitigation of radiological consequences of significant releases of radioactive materials

The first four levels are oriented towards the protection of barriers and mitigation of releases. The last relates to off-site emergency response to further protect the public. A relevant aspect of the implementation of defence in depth is the provision in the design of a series of physical barriers to confine the radioactive material at specified locations. The number of physical barriers required depends on the potential internal and external hazards, and the potential consequences of failures.

The objective of the safety approach shall be /IAE 00/: to provide adequate means to maintain the plant in a normal operational state; to ensure the proper short term response immediately following a postulated event; and to facilitate the management of the plant in and following any DBA, and in selected accident conditions beyond the design basis accidents.

Radiation protection and acceptance criteria in /IAE 00/ are adopted in /KAR 04/ for fusion reactor concepts. In order to achieve the three safety objectives (General Nuclear Safety Objective, Radiation Protection Objective and Technical Safety Objective) in the design of a nuclear installation, all actual and potential sources of radiation shall be identified and properly considered, and provision shall be made to ensure that sources are kept under strict technical and administrative control.

- General Nuclear Safety Objective: To protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defences against radiological hazards.
- Radiation Protection Objective: To ensure that in all operational states radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below prescribed limits and as low as reasonably achievable, and to ensure mitigation of the radiological consequences of any accidents.
- Technical Safety Objective: To take all reasonably practicable measures to prevent accidents in nuclear installations and to mitigate their consequences should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.

Safety objectives require that the reactor and its confinement are designed and operated so as to keep all sources of radiation exposure under strict technical and administrative control. Measures shall be provided to ensure that radiation protection and technical safety objectives are achieved, and that radiation doses to the public and to site personnel during all operational states, including maintenance and decommissioning do not exceed prescribed limits and are as low as reasonably achievable /KAR 04/.

From a review of the limits on public doses in accidents adopted in SEAFP, EUR (European Utility Requirements) and ITER, as well as IAEA guidelines, the following top-level objectives have been proposed in the PPCS:

- To ensure that, for all accidents addressed in the design of the plant, the radiological consequences, if any, would be minor.
- To ensure that, for the worst possible accident, no matter how unlikely, the consequences could not give technical grounds for public evacuation.

Based on these, quantitative public dose limits are specified for each of several categories of accident scenarios in Table 3.1. The limiting activity releases, which would satisfy these dose limits, have been evaluated based on accident scenarios studied in SEAFP-2. The limiting values can be used as guidelines for a broader range of accident scenarios and plant designs. However, for significant deviations from the SEAFP-2 assumptions and plant design concepts, new accident consequences analyses should be performed /KAR 04/.

Table 3.1 The public dose limits proposed to satisfy the safety objectives /KAR 04/

	Anticipated events	**Unlikely events**	**Extremely unlikely events**	**Hypothetical bounding events**
	LoD 1-2	LoD 3	LoD 4	BDBA
Indicative frequency (yr^{-1})	$f > 10^{-2}$	$10^{-2} > f > 10^{-4}$	$10^{-4} > f > 10^{-6}$	$10^{-6} > f$
Early dose (7 days)	Treat as normal operational events		10 mSv/event	50 mSv/event
Chronic dose (50 years, excluding ingestion)		5 mSv/event	50 mSv/event	

Table 3.2 provides a summary of the activity release limits for each class of material in each category of accident scenario considered.

Table 3.2 Guideline activity release limits to meet proposed dose limits /KAR 04/

Material class	**Activity release limit (TBq per event)**		
	Unlikely events Cat. 3	Extremely unlikely events Cat. 4	Hypothetical bounding events
Solid activation products	16	160	890
Be dust	0.6	6	440
W dust	8	80	950
ACPs	1.5	15	640
Tritium (as HTO)	2400	4800	24000

The three values for tritium are equivalent to approximately 6.7 g, 13 g and 67 g, respectively. Values for the other classes of materials relate to a mixture of nuclides, so that a constant ratio of these activity figures to mass values cannot be provided. In this context it should be noted that “Be dust” and “W dust” actually indicate a mixture of nuclides that result from the irradiation of PFC armour materials initially composed of Be and W, from which the dust is assumed to be formed.

It should be noted that these activity limits have been derived from analyses of bounding loss-of-coolant accidents in SEAFP-2 plant models. While these guidelines may be used in initial assessment of different accident scenarios and plant designs, full analyses should eventually be performed for the selected designs to confirm the doses arising from the different categories of accidents.

4 Identification of postulated events

According to the IAEA Safety Glossary /IAE 07/ a ‘postulated initiating event (PIE)’ is “an event identified during design as capable of leading to anticipated operational occurrences or accident conditions. The primary causes of postulated initiating events may be credible equipment failures and operator errors (both within and external to the facility), human-induced or natural events.”

Besides postulated (single) initiating events also selected multiple failure events including possible failure or inefficiency of safety systems and severe plant states are considered as the starting point for sequence analyses. In this paper, we refer to these starting points as ‘postulated events’. The purpose is to show that all these events induce no off-site radiological impact or only minor radiological impact (in particular, sheltering or evacuation). This explicit consideration of multiple failure events and severe plant states as a starting point for the safety analyses is comparable to the WENRA approach for the safety of new NPP designs /WEN 13/.

A postulated event itself does not directly cause a release of radioactive materials. Further failures have to occur before a sequence of additional events (e. g. failures of system, structures and components or operator actions) can entail a hazard, in particular a release of tritium or other radioactive materials. The following discussion is based on /PSR 11/.

4.1 Methods to determine the postulated events

To ensure that the selection of the postulated events is of bounding character, complementary methods are being applied.

4.1.1 Methods used by the ITER project

Most extensive and most recent work in the postulated event area has been done by the ITER project. Since the ITER work is based on a detailed technical design it is presented here as introductory demonstration of postulated event selection although ITER is not yet a ‘commercial’ FPP in all respects. It introduces also some nomenclature. Details are provided in /TAY 02/, /IZQ 05/ and /ALE 13/.

Deterministic selection

The postulated events are chosen by a deterministic process, in which event sequences are postulated that cover all main plant systems and each type of conceivable initiating events, eventually choosing events expected to have the greatest consequences in each case.

Failure Mode and Effects Analysis (FMEA)

FMEA is a systematic method for accident identification, aiming at completeness. It is an approach in the 'bottom-up' sense, i.e. it is starting at the level of individual components wherever possible.

The steps in an FMEA of a plant are:

- prepare a full and detailed list of all components in the plant;
- for each component, produce a full list of the possible failure modes, if necessary giving separate lists for each of the operational phases (such as normal operation, standby, baking of components);
- for each failure mode for each component, list the possible causes of the failure, its occurrence rates and note the actions that could prevent the failure;
- for each failure mode, list the consequences and the actions that could mitigate each consequence.

When the above steps are complete, the result is a list of elementary failures. These are subsequently put together collated into groups having similar consequences. Then the postulated event lists are allocated to the operational phases. The summation of events (for each phase) together with the expected occurrence rates gives the total occurrence rate of each postulated event.

The assignment of failure rates in the penultimate step is based on empirical failure rate data where available. A database of failure rates for fusion has been developed as part of an international collaboration. It contains data from various industrial sources for conventional equipment, such as pipework, valves, pumps, etc. It also contains data derived from extensive studies of failures in equipment of some existing tokamak facilities, for example, from the vacuum systems at JET.

For many fusion-specific systems (since they are new), however, relevant failure data are not available so that occurrence rates have to be assigned by judgment. In the early European fusion safety studies (such as SEAFP, /SEA 95/) this has been done within bands of two or more orders of magnitude width to aid categorization as an incident or accident (1 to 10^{-2}/a and 10^{-2} to 10^{-6}/a).

The international fusion safety community, as represented by the ITER project, uses a similar approach but with one important modification: ITER does not emphasize the numerical values of occurrence rates. Rather, an incident is defined as an event that is not planned, but which can nevertheless be expected to happen at least once in the lifetime of ITER. An accident is an event that is not expected to occur, but for which provisions are taken in the design. ITER uses four event categories, I, II, III, IV: “operational”, “likely”, “unlikely”, and “extremely unlikely”.

Master Logic Diagram (MLD)

Instead of starting at the component level (“bottom-up”), the MLD begins with a ‘top-down’ view. It provides a global perspective of the possible failures through a global fault tree, in which combinations of failures are elaborated through logic gates representing Boolean ‘and’/’or’ functions. It begins with the **top-level event** “excessive off-site releases” (i.e. radiological doses in excess of the limits) and breaks this down into its contributory elements:

- the release origin,
- the release pathway,
- the release species (such as tritium, activation products),
- the barriers that would have to fail to enable the pathway,
- the safety functions that protect these barriers,
- the failure events that could degrade these safety functions.

At the fourth and fifth levels in this list, and-gates appear in the logic, illustrating that there are barriers protected by multiple safety functions and, therefore, that more than one failure is required to cause a release. Because the view has a plant-level functional nature, there is not the level of detail obtained by an FMEA. Nevertheless, the list of failure event types provided by the MLD gives an alternative approach that helps to obtain completeness in the identification of the postulated events. The MLD list of failure events is used as a check to ensure that there is nothing additional to those failures identified by the FMEAs.

The reference events (this notion is sometimes used by ITER instead of postulated event) selected for ITER are included in the postulated event lists for the Design Basis and the Beyond Design Basis regimes (see Table 4.1 and Table 4.2 presented in the sections 4.2 and 4.3) and are marked there by /PSR 11/.

4.1.2 Development in the frame of European fusion plant safety studies

In the course of the European fusion plant safety studies (since about the early 1990s), the selection of postulated events has been developed.

The starting point was set by the "Safety and Environmental Assessment of Fusion Power" (SEAFP). There, systematic safety studies were done based on the method HAZOP (Hazard and Operability) and are summarised in the following on the basis of /SEA 95/, p.35 et seqq. and the references therein].

The studies concentrated on identifying potential hazards to the public and also aimed at ranking the putative event sequences in order of importance so as to guide the distribution of safety analysis effort.

The consequences of BDBAs have upper limits determined by the inventories of radioactive materials that at maximum can reside in a future commercial FPP.

Many events were considered in the past European safety studies to ensure a comprehensive listing. This was done to obtain a view of sequences which encompass many other, smaller events. A ranking scheme assigned levels of severity and likelihood. Events of higher potential severity were included even if their expected occurrence rate was expected to be extremely low. No credit was allowed for active safety measures.

It was concluded that only certain ex-plant events have a potential for breaching the ultimate radioactivity confinement barrier. It was suggested that some of them, in particular airplane impact and earthquake, be covered by the design basis.

Many events in the lower categories of severity, which may entail mobilisation within the confinement barriers of tritium or activation products, involve a loss of primary coolant. Therefore the plant models foresee the capture of escaping coolants /SEA 95/ (Sections 4.4.5, 4.4.7, 4.4.11, 4.4.12).

The potentially most severe loss-of-cooling event is judged to be caused by a total loss of all site electrical power and prolonged failure of all back-up power supplies /SEA 95/ (Section 5.3, p.36).

Events which involve an ingress of primary coolant into lower pressure loops (secondary cooling, tritium purge) which may penetrate second confinement barriers have a potential for confinement bypass. The expected occurrence rate of such event sequences led to classify them as design basis events.

The fuel cycle is fusion specific and largely interlaced with other systems. Therefore a dedicated study was made in close collaboration between fuel cycle design and safety analyses (Task Report for SEAFP, quoted as [M7-1] in /SEA 95/).

The magnet system is also fusion specific and interlaced with other components. Past work was supplemented by studies in the frame of SEAFP.

The above studies pertain to full power operation. They were supplemented by a study of events during scheduled maintenance. It was limited to severe events, ignoring minor safety hazards and those resulting in purely economic losses. Most severe is the drop of a container (enclosing an activated component) during removal. Potential consequences are rupture of pipes (cooling, fuel cycle) and break of the container proper, i.e. damage to confinement barriers. The only event during scheduled maintenance which can conceivably challenge the ultimate radioactivity confinement is load drop on the foundation raft. Preventive measures such as careful planning of the transportation routes /SEA 95/, Section 4.4.10] and provisions such as isolation valves to limit radioactive releases are suggested as design requirements.

Accident initiators eventually selected for further study are presented in the following two sections (4.2 and 4.3) dealing with postulated events and are marked there by /SEA 95/.

The European Long-Term Fusion Safety Programme was carried forward via two programmes, SEAL and SEAFP-2 /COO 99/, to update and extend /SEA 95/. In /COO 99/, a systematic ‘top-down’ accident identification was done, using MLDs and Functional Failure Modes and Effects Analysis (FFMEA).

European work proceeded and led to the report /SEI 01/ that integrates, updates and extends all previous work on fusion safety.

Safety analysis again started by systematically identifying and ranking the potential accident sequences. Already, this had been done during the SEAFP and SEAL studies but /SEI 01/ increased detail and rigour. The methods used are standard: HAZOP (Hazard and Operability), MLD, and FFMEA.

The accidents identified in /SEI 01/ that most merit analyses in detail are included in the following Sections 4.2 and 4.3 (Table 4.1 and Table 4.2) and are marked there by /SEI 01/.

Since the European “Conceptual Study of Commercial Fusion Power Plants” /MAI 05a/, /MAI 08/ comprehensively deals with FPPs, it also includes safety assessments (Section. 6.2 “Accident analyses”, p.16 and Annex A10 “Safety and environment assessment of the PPCS models”).

The selection of the accident sequences is based on the FFMEA methodology to find out representative postulated events. The FFMEA is a ‘top-down’ approach that is adequate when the level of the plant design is not so detailed to justify an FMEA at component or system level.

A plant functional breakdown was done for the main systems. Then an FFMEA followed for each lower level function of the functional breakdown. Basic system failures were grouped to postulated events, based on the expected consequences in terms of plant damage, on the mobilisation of radioactive inventory and, finally, on possible harm to workers and population.

The postulated events eventually selected are presented in Sections 4.2 and 4.3 dealing with postulated events and are marked there by /MAI 05a/.

The following two Sections (4.2 and 4.3) provide a synopsis of the postulated events determined in the course of the European power plant safety studies (identified by the references attached to the individual postulated events) and by the ITER project /PSR 11/.

In the context of the synopsis, it is important to note the following nomenclature issues:

- The postulated events inside the design basis associated with /SEA 95/ are named there “accident initiators eventually selected for further study", not postulated events.
- The postulated events inside the design basis associated with /SEI 01/ are named there “accidents that most merited analysis in detail”, not postulated events.
- The postulated events inside the design basis associated with /PSR 11/ are often named there ”reference events”, not postulated events.
- The postulated events beyond the design basis associated with /SEA 95/ are named there “beyond design basis accidents”, not postulated events
- The postulated events beyond the design basis associated with /PSR 11/ are named there “accidents that are considered beyond the design basis of ITER”, not postulated events.

Here, all those events or accidents of /SEA 95/, /PSR 11/, and /SEI 01/ are named uniformly postulated events.

4.2 Postulated Events in the Design Basis

The Design Basis postulated events are listed in Table 4.1.

Table 4.1 List of Design Basis postulated events

Design Basis Postulated Events	Reference
Loss-of-Flow (LOFA)	/SEA 95/, p.36]
Loss of the flow of the primary circuit coolant	/SEI 01/, p.20]
In-Plasma-Vessel Loss-of-Coolant (LOCA)	/SEA 95/, p.36]
Loss of primary circuit coolant inside the Plasma Vessel	/SEI 01/, p.20
Hydrogen production and potential accidental consequences	/SEI 01/, p.20
Ex-Plasma-Vessel Loss-of-Coolant	/SEA 95/, p.36
Ex-Vacuum-Vessel loss of coolant (ex-VV LOCA)	/MAI 05a/, Annex A10
In-Vacuum-Vessel loss of coolant (in-VV LOCA) due to an ex-Vacuum Vessel loss of coolant (ex-VV LOCA)	/MAI 05a/, Annex A10
Loss of primary circuit coolant outside the Plasma Vessel	/SEI 01/, p.20]
Large ex-Plasma-Vessel pipe break of the primary loop of the Divertor Heat Transport System	/PSR 11/
Large ex-Plasma-Vessel pipe break in the Plasma Vessel Primary Heat Transport System	/PSR 11/
Loss of heat rejection from the secondary cooling circuit	/SEI 01/, p.20
Break in the secondary cooling circuit with multiple Steam Generator tube rupture	/SEI 01/, p.20]
Heat Exchanger leakage	/PSR 11/
Heat Exchanger tube rupture	/PSR 11/
Coolant pipe break inside a Port Cell	/PSR 11/
Loss of Plasma Vessel vacuum (LOVA)	/SEA 95/, p.36
Breach of the Plasma Vessel	/SEI 01/, p.20
Loss of the cryogenic helium	/SEI 01/, p.20
Stuck Divertor cassette and failure of a Transport Cask	/PSR 11/
Magnet System fault	/SEA 95/, p.36
Releases of magnet energy	/SEI 01/, p.20
Toroidal Field Coil short	/PSR 11/
Electric arc near confinement barrier	ditto
Cryostat air ingress	ditto
Cryostat water ingress	ditto
Cryostat helium ingress	ditto
Fuel Cycle System fault	/SEA 95/, p.37
Tritium process line leakage	/PSR 11/

Design Basis Postulated Events	Reference
Accident with transport of hydride bed	ditto
Isotope Separation System failure	ditto
Failure of fuelling line	ditto
Leak of tritiated water from the Water Detritiation System	ditto
Loss of confinement of a Hot Cell	ditto
Fire in the Hot Cell buffer storage room	ditto

4.3 Postulated events beyond the design basis

Postulated events beyond the design basis have, in particular, been considered in the course of the safety studies by the ITER project. The analysis of these sequences is intended to demonstrate the robustness of the 'defence-in-depth' approach, to ensure, that there are no cliff-edge effects and that any counter-measures are limited within time and space /PSR 11/.

The analysis also aims at defining the measures that would need to be implemented to achieve these objectives.

Because the event sequences postulated are themselves of extremely low likelihood (i.e. “hypothetical”), assumptions that are more realistic than those for the Design Basis Accidents are applied to these situations, and 'best estimate' codes and models are used /PSR 11/.

The Beyond Design Basis postulated events are listed in Table 4.2.

Table 4.2 List Of Beyond Design Basis Postulated Events

Beyond Design Basis Postulated Events	Reference
Major in-Plasma-Vessel LOCA plus significant radioactivity mobilisation plus malfunction of the entire confinement arrangement	/SEA 95/, p. 37
Loss of Flow (LOFA) without plasma shutdown inducing an in-Vacuum Vessel loss of coolant (in-VV LOCA)	/MAI 05a/, Annex 10, p. 176
Loss of Heat Sink without plasma shutdown	ditto
Total blockage of the path to the stack	/SEA 95/, p.37
Release and vaporisation of the cryogenic helium from all loops into the Cryostat	/SEA 95/, p.47]
Hypothetical postulated event: Total loss of cooling from all loops in the plant, with no active cooling, no active safety system operating, and no intervention whatever for a prolonged period	/MAI 05a/, p. 30 and Annex 10, p. 176
Fire in the Tritium Plant with propagation to a glove box	/PSR 11/
Hydrogen and dust explosion in the Plasma Vessel	/PSR 11/
Loss of vacuum through one Plasma Vessel penetration line plus 2 hours electrical blackout and in-Plasma-Vessel First Wall coolant leak	/PSR 11/
Damage to Plasma Vessel and Cryostat resulting in large holes	/PSR 11/
Cryostat water and helium ingress	/PSR 11/
Fire in the Hot Cell waste processing area with propagation to the buffer storage room	/PSR 11/
Loss of plasma control together with multiple failure of First Wall/Blanket Primary Heat Transport System inside Plasma Vessel	/PSR 11/
Large Plasma Vessel ex-vessel coolant pipe break plus loss of flow in all intact cooling loops	/PSR 11/

4.4 Bounding events

In addition to the postulated events in /SEA 95/, p.36, a class of so-called ‘bounding events’ has been considered that is characterised as follows /SEA 95/, p.6]: extremely energetic external events (such as an earthquake of hitherto un-experienced magnitude) may cause a Beyond Design Basis Accident by directly breaching confinement barriers. The obvious countermeasure which might be suggested is design of the ultimate radioactivity confinement to withstand also these extreme external events. General achievement of such a target is, however, not credible due to the extreme nature of such postulated events.

Nonetheless, an upper bound estimate has been attempted /SEA 95/, p.58 and is described as follows: “There is an upper bound of approximately 1 kilogram for the vulnerable tritium inventory. At 1 km distance from the release point, the early dose to the most exposed member of the public from ground level release (since the radioactivity confinement barriers may be damaged) of 1 kg of tritium would be up to about 450 mSv for the most hazardous form (tritium in tritiated water i.e. HTO) and a release duration of one hour”.

This very specific dose estimate is quoted here since it has met with widespread attention during many years after the publication of /SEA 95/ and has often been conceived as being the prototype of a maximum fusion hazard.

The selection of 1 kg of vulnerable tritium is based on current estimates of the tritium inventory in the plasma vessel. This value has been assumed, for example, for a most severe accident scenario considered in the PPCS.

It is essential to note that a release of 1 kg of tritium from the plasma vessel would be retained mostly by the second confinement. This is fundamentally different from the assumed maximum possible release of 1 kg of tritium into the environment.

If the above inventory-based approach to estimating the very worst consequence is used (if absolutely all confinement functions are totally lost), the release of 1 kg of tritium to the environment is the ultimate consequence. It is emphasized that such an assumption is not scenario-based, rather it is purely hypothetical.

5 Event sequences of incidents and accidents, and impact on the plant, personnel and the environment

The reactor configurations, as well as classification of postulated events establish starting points of the study of event sequences. The implementation of the safety assessment requires the coverage of the following main aspects:

- Determination of the maximum releasable inventories
- Analysis of incidents and accident scenarios.

Event sequence analyses concern temperature transient in short / long term, chemical reaction, transport and release of radioactive source terms. The accident consequence is evaluated regarding dose rate and its influence on the public.

Since PPCS updated reactor concepts regarding structural material, source terms, inventory of radioactive products and safety, PPCS is considered as basis object for the safety analyses. In addition, representative event sequences from ITER are considered in this work as well in order to cover major accidents that were not analysed in detail during the PPCS. They involve failure of systems that would be present also in a future power plant (e.g. fire in the tritium plant). Although ITER differs in several aspects from a FPP (e.g. absence of complete fuel cycle, lower coolant temperature levels, no high activation materials, or absence of power conversion system) the experience gained in the licensing of the ITER is valuable for future fusion reactor concepts.

5.1 Event sequences selected for PPCS

The PPCS analysed in detail event sequences for five power plant models, and evaluated accident consequences with respect to the environmental source term as well as resulting dose rates.

PPCS focused at the begin on four power plant models, ranging from near-term plasma physics and materials (models A and B) to “advanced” plant models (models C and D) based on the use of the most advanced thinkable plasma physics and technology. Later a further near-term blanket concept — the Helium-Cooled Lithium-Lead blanket (HCLL) has been identified as well and it was called model AB /LIP 06/. Table 5.1 shows the main parameters of all five plant design models. From a safety point of view the most important difference among the models are coolants and material combinations used in the blanket and divertor concepts. Model A uses a Water Cooled Lithium Lead (WCLL) Blanket and an ITER-like divertor /SAR 03/. Model B is a Helium Cooled Pebble Bed (HCPB) Blanket with a helium cooled W-steel divertor /HER 03/. Both models use Eurofer as structural material. It is a reduced activation ferritic martensitic (RAFM) steel developed in EU for the fusion programme /VAN 03/. The Dual Coolant Lithium Lead Blanket (DCLL) adopted in the Model C Blanket, uses also Eurofer with He and LiPb together with a helium cooled divertor /NOR 03/. Model D consists of a Self-Cooled Lithium-Lead (SCLL) blanket and a divertor based on SiC fibre composite as structural material and LiPb as breeder /GIA 03/.

The HCPB and HCLL blanket concepts were selected for testing in ITER in the International Test Blanket Programme.

Table 5.1 PPCS parameters /MAI 06/ /MAI 05a/

Model		A (WCLL)	B (HCPB)	AB (HCLL)	C (DCLL)	D (SCLL)
Unit size (GWe)		1.55	1.33	1.46	1.45	1.53
Fusion power (GW)		5.00	3.60	4.29	3.41	2.53
Net efficiency		0.31/0.33	0.36	0.34	0.42	0.60
Plant lifetime (FPY)		25	25	25	25	25
Blanket	Structural material	Eurofer	Eurofer	Eurofer	Eurofer/ODSEurofer*	SiC_f/SiC
	PFC	W (*)	W (*)	W (*)	(W)*	W
	Coolant	Water	He	He	$LiPb_{eu}$ /He	$LiPb_{eu}$
	Coolant T_in/out (°C)	285 / 325	300 / 500	300 / 500	480 / 700 300 / 480	700 / 1100
	Breeder	$LiPb_{eu}$	Li_4SiO_4	$LiPb_{eu}$	$LiPb_{eu}$	$LiPb_{eu}$
	Neutron multiplier	$LiPb_{eu}$	Be	$LiPb_{eu}$	$LiPb_{eu}$	$LiPb_{eu}$
	TBR	1.06	1.12	1.13	1.15	1.12
	Service lifetime (FPY)	5	5	5	5	5
Divertor	Peak load (MW/m^2)	15	10	10	10	5
	Structural material	CuCrZr	W alloy	W alloy	W alloy	SiC_f/SiC
	Armour material	W	W	W	W	W
	Coolant	Water	He	He	He	$LiPb_{eu}$
	Coolant T_in/out (°C)	140 / 167	540 / 717	540 / 717	540 / 717	600 / 990
	Service lifetime (FPY)	2.5	2.5	2.5	2.5	2.5

ODS: Oxide dispersion-strengthened alloy
$LiPb_{eu}$: eutectic at the composition of ~15.8 mol% of Li.
(*) According to common understanding W could be present as protection layer on the FW [43], but not in every PPCS analysis it has been considered.

From the overall accident scenarios a limited set of them can be chosen as the most representative for a safety assessment in terms of containments challenging, radioactive products mobilisation and possible radioactive release towards the environment. Since models C and D are the “advanced” plant models for future, their accident sequences are not discussed below.

The selected DBA and BDBA for models A and B are listed in Table 5.2. The bounding accident is selected as hypothetical event, which leads to the worst consequences of an accident driven by in-plant energies. Its consequence demonstrates that the dose rate of the radiological releases is far below the limits internationally accepted.

Table 5.2 Event sequences selected for models A and B in PPCS /MAI 05a/

Model	DBA	BDBA	Hypothetical event
A	ex-vessel LOCA	LOFA + in-vessel LOCA	Bounding accident
	ex-vessel LOCA + in-vessel LOCA	Loss of heat sink (Loss of condenser)	
B	ex-vessel LOCA	LOFA + in-vessel LOCA	Bounding accident
	ex-vessel LOCA + in-vessel LOCA	Loss of heat sink (Loss of condenser)	

For plant model AB six different accident sequences from FFMEA have been analysed (see Table 5.3). Tritium and dust inventories of model AB have been shown in Table **2.5**. Releases for LOFA + in-vessel LOCA are shown in Table 5.4 and the corresponding 7 days dose in Table 5.5.

Table 5.3 Accident sequences analysed in plant model AB /CAP 05/

Accident	Description
LOFA + in-vessel LOCA	In-vessel LOCA due to a break of 5 (C1) or 10 (C2) FW cooling channels when FW temperature reaches 1073 K after the LOFA
Generalised loss of heat sink	As in the previous cases but affecting all the 9 loops of the cooling helium. The rupture of 5 channels has been considered.
Ex-vessel LOCA	Double guillotine break of a main piping inside the TCHS vault (58,000 m^3) and rupture disk intervention at 0.14 MPa towards the other vaults (59,600 m^3). The aim is to define the rupture disc area between TCHS and the expansion vaults and to verify the volumes available to limit the pressure ≤ 0.16 MPa in the Expansion Volume (EV) itself
Interface LOCA between FW and Breeding Blanket	In-vessel LOCA. Double guillotine break of a helium manifold of a module (D_i = 220 mm). The aim is to define the rupture disc area between VV and the expansion volume to limit the pressure inside VV ≤ 0.2 MPa
Steam generator tube rupture	Preliminarily analysis of a steam generator tube rupture (10 tubes affected, D_i = 20 mm) to verify the pressurization of one helium loop.

All the Environment source terms (EST) assessed for the selected accident sequences and for the bounding temperature accident sequences are summarised in Table 5.4. In terms of environmental release the most challenging accident scenario is a LOFA followed by an in-vessel LOCA for the plant model B. It was conservatively assumed a mobilisation fraction of 100 % for the dust at the beginning of the accident sequence. Similar results were obtained for LOFA + in-VV LOCA for plant model C. Anyhow, the results obtained in terms of environmental source terms confirmed the full validity of the design as far as the confinement design is concerned.

Table 5.4 EST assessed for accident sequences in PPCS /MAI 05a/, /SAR 06/

Model	Accident cases	Release time [h]	Tritium (g)	ACP (g)	Dust (g)	FW (g)	divertor (g)
A	Ex-vessel LOCA	24	0.0024[a]	0.00072	-	-	-
	Ex-vessel LOCA + in-vessel LOCA	24	0.17[a]	0.0095	1.63	-	-
	LOFA + in-vessel LOCA	N.A.	N.A.	N.A.	N.A.	-	-
	Loss of heat sink	14	N.A.	N.A.	N.A.	-	-
	Bounding temperature sequence	168	13.6[a]	1.78	35.3	0.24	0.27
B	Ex-vessel LOCA	30	~ 0	-	-	-	-
	Ex-vessel LOCA + in-vessel LOCA	30	0.6[b]	-	0.17	-	-
	LOFA + in-vessel LOCA	24	3.5[b]	-	19.1	-	-
	Loss of heat sink	0.55	N.A.	-	N.A.	-	-
	Bounding temperature sequence	168	~ 8.1[b]	-	18.2	1570	177
C	LOFA + in-vessel LOCA	24	4.7[b]	-	24.6	-	-
AB	LOFA + in-vessel LOCA	24	3.0	-	4.2	-	-

[a] as HTO, [b] as HT

It is important to assess the consequences of potential source terms to the public in terms of doses. As early emergency actions such as evacuation of the population are most disruptive for the normal live, it must be assured that they will never occur or are at least very limited following potential accidental releases of radionuclides to the environment.

As the evacuation dose differs from country to country, the German regulation was applied as reference for the calculations in /MAI 05a/ (an effective dose of 100 mSv by external exposure and inhalation within 7 days). Dose conversion factors according to International Commission on Radiological Protection (ICRP-60) were applied in the public dose calculations.

As the weather conditions play an important role in the transport and dispersion of the radionuclides released into the atmosphere, one of the most effective approaches is the use of so called ‘probabilistic’ weather samples. This comprises 144 different weather sequences within one year with respect to turbulence, rain and travel time. Results of such calculations are doses with a certain probability of occurrence. In particular the 95 % percentile of the distribution is often used as criterion in national regulations for licensing assessments.

As the investigations in /MAI 05a/ were carried out for a generic site, a standard set of weather data representing the area around Karlsruhe, Germany were taken. It was assumed that the release takes place over a 24-hour period, as the characteristic release time ranges typically from 1 - 7 days. In agreement with national licensing arrangements, the upper 95 %-percentile of the results was taken as the reference dose criteria. Another important parameter is the release height, which was set to 10 m as these results in the highest dose in the vicinity of the plant. Besides for 1 km distance, the evacuation criteria for the most exposed individual were calculated at various distance bands. However, this summary concentrates on the results for the MEI (Most Exposed Individual) at 1000 m distance for which the dose values are given in Table 5.5. Sequences not evaluated for the corresponding plant model are marked with “-“ in Table 5.5.

Table 5.5 Worst case values for the 7-day dose to MEI at 1000 m distance (24 h release, 95 % fractile) /MAI 05a/, /SAR 06/

Plant model	**Dose (mSv)**			
	Bounding temperature sequence	**Ex-vessel LOCA**	**Ex-vessel LOCA + in-vessel LOCA**	**LOFA + in-vessel LOCA**
A	1.16	1.71E-3	0.16	-
B	18.1	-	-	0.42
AB	-	-	-	0.4

Doses from all the release scenarios are far below 50 mSv denoted in Fig. 3.1 and the dose for the bounding temperature sequence is below this dose limit as well. The contribution of the activation products and tritium to the overall dose is mostly of similar magnitude, thus there is no particular fraction of the source term dominating the results.

Based on the results of the activation analysis and the temperature transient calculations of plant models C and D, together with other features of the design, it is justified to assume that the consequences of the bounding accident scenario comparable to those of model A and B will likely be not higher than for models A and B. A potential assessment for model C is expected to be similar to that of model B, whereas for model D the consequences would be considerably lower, mainly due to the extremely low decay heat and negligible temperature rise in that model.

5.2 Event sequences selected for ITER and two EU Test Blanket Modules (TBM) concepts

For ITER facility nine of all 25 reference events (DBA) and eight of 12 BDBA have been identified in /PSR 11/ as the most challenging events concerning radiological consequences, and the design of the main safety important components implementing safety important functions (Table 5.6).

The event selection criteria of the DBA analyses are the following /PSR 11/:

- the event sequences should be the most challenging in terms of expected radiological consequences,
- Analysis results of sequences inform the design with respect to the main safety important components implementing safety important functions (i.e. confinement systems including VVPSS, isolation valves, detritiation systems, residual heat removal).
- Consequently, the DBA analysis addressed in this context considers mainly event impact on structural integrity and limited radiological consequences.

The BDBA event selection for ITER analysed in this context is limited to the following event sequences:

- Confinement loss events in the tritium plant, which are enveloped by a hypothetical fire subsequently leading to a glove box confinement failure within the tritium building.

- A hypothetical failure of the prevention system leading to hydrogen and dust explosion in the VV, which incorporates various pathways of hydrogen production and a potential dust explosion.

- A loss of vacuum through one vacuum vessel penetration line accompanied by a 2 h station black-out and an additional in-vessel FW coolant leak event. This sequence corresponds to a design basis event with additional aggravating events.

- A hypothetical damage of a confinement barrier resulting in large holes in the vacuum vessel and cryostat, which covers events equal to a damage of the confinement systems within the tokamak building.

- Potential cryostat water and helium ingress scenarios, which are similar to overpressurization sequences of the cryostat.

- A hypothetical large-scale fire in the waste processing area of red radiological zone with a subsequent propagation to the buffer storage room, which covers the event family of a confinement loss of the hot cell facility.

- Anticipated uncontrolled fusion power excursion, caused either by a control loss or an “over fuelling” of the plasma, complemented by a simultaneous failure of the FPTS (Fusion Power Termination System). This event type represents a bounding case related to all types of possible plasma control loss transients. Additionally this class of incidents in conjunction with a failure of a FW/BLK PHTS ends up in a BDBA scenario.

- Anticipated VV cooling loop failure along with a loss of cooling flow in all other (intact) loops, which demonstrates the design capability to remove the decay heat without loss of the retention function.

Table 5.6 List of the most challenging events selected for ITER for sequence analyses /PSR 11/

DBAs	BDBAs
Multiple FW/BLK (Blanket) PHTS pipe break inside the VV	Fire in the tritium plant along with the propagation to a glove box
Loss of vacuum through one VV penetration line	Hydrogen and dust explosion in the VV
Large VV PHTS ex-vessel pipe break	Loss of vacuum through one VV penetration line complemented by a 2 h electrical blackout and a simultaneous in-vessel FW coolant leak
Large divertor PHTS ex-vessel pipe break	Damage to VV and cryostat causing in large holes
Isotope separation system (ISS) failure	Cryostat water and helium ingress
Failure of fuelling line	Fire in the hot cell waste processing area in conjunction with a propagation to the buffer storage room
Loss of confinement in hot cell	Loss of plasma control and a simultaneous multiple failure of FW/BLK PHTS inside the VV together
Leak of tritiated water in the WDS (Water Detritiation System)	Large VV ex-vessel coolant pipe break and an additional loss of flow in all remaining cooling loops
A stuck divertor cassette and failure of cask during maintenance	-

For estimation of doses the following hypothetical, but reasonable release data have been used in ITER:

- a short term of 48 h adult exposure dose, 200 m from the release point and
- a long term dose at 2.5 km abroad the site border for both:
  - 50 years exposure of an adult and
  - 70 years exposure time of a child or a 1 year old baby.
- The release point considered occurs in 58 m height above mean ground level, corresponding to the elevated release point above the roof of the Tokamak Complex. The assessment comprises the wake effect induced by the building, leading to an effective height half of the actual elevation, of about 30 m. In certain accident scenarios, some of the release may occur through building leaks, which is represented by a ground level release point (0 m above ground).

The computed environmental release transport concerns three meteorological scenarios:

- DF2 class meteorological conditions (low diffusion with 2 m/s wind velocity and without rain)
- DN5 class meteorological conditions (normal diffusion with 5 m/s wind velocity and without rain),
- DN5P class meteorological conditions (normal diffusion with 5 m/s wind velocity and with rain at 5 mm/h).

The simulation results of the overall early MEI doses, which incorporates tritium, dust and ACPs, yield for 200 m from the release point a dose of less than 9.9E-2 mSv for DBAs and 0.6 mSv for BDBAs, while for the long term MEI doses in 2.5 km distance the values fall below 18 µSv for DBAs and 320 µSv for BDBAs.

In additional, total loss of cooling is considered in ITER as well /TAY 12/. A site power loss causes a coolant pumps coast-down of the primary cooling circuits of the in-vessel components (FW, blankets, divertor) and of the VV. Due to the active control, however, the plasma will also terminate and the remaining heat source constituted by the decay heat of activated material only. Only a low-flow pump powered by an emergency diesel generator provides a minimal cooling of the VV cooling circuit. The analysis of this scenario reveals that the residual heat in the in-vessel components can be transferred to the vessel and removed by this slowly circulating water rate.

The emergency electrical power supplies are designed as fully redundant, but postulated a hypothetical failure of all diesel generators, the analysis yields that the ITER facility still remains in a safe condition. Moreover, the computations yield that a safe state can be maintained even for 10 days without any interference after the loss of power and still the temperatures remain considerably below the level of any structural degradation or even failure.

Deterministic assessments have been done for the two EU breeding blanket concepts (HCPB and HCLL /BOC 11/) tested in ITER in form of TBM as well. Deterministic sequence analyses have been performed for ex-vessel LOCA (/JIN 12/, /JIN 08/, /BOC 07/, /SPO 07/, /POR 07/), in-box LOCA (/BOC 07/, /GIR 08/) and in-vessel LOCA (/BOC 07/), which demonstrate that ITER limits for source terms were not exceeded and cliff-edge effects were absent. The analyses of typical accidents of a TBM may exhibit events being relevant for a FPP concept.

6 Investigation and description of precautionary, preventive and mitigative measures

In the present chapter safety functions and systems for PPCS and ITER are listed. The near-term concepts in the PPCS, namely plant model A, B and AB are taken into account. ITER is considered as well, since the methodology and experience gained in the licensing of ITER is conceived to be valuable for future fusion reactor concept. A proposal for a future FPP has been sketched by us on the ITER basis and is summarized in Table 6.4.

6.1 PPCS safety functions and systems

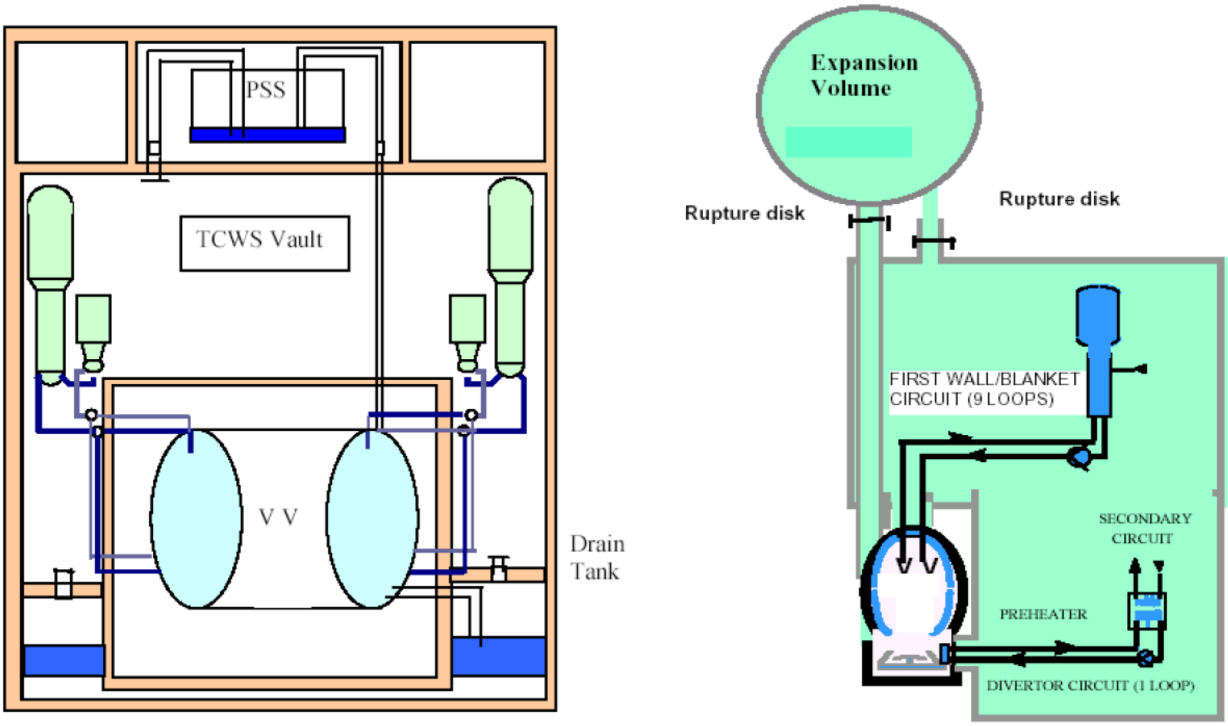
In the PPCS safety functions are divided into primary safety functions and secondary safety functions, whose purpose is to support primary safety functions. They are listed in Table 6.1.

Table 6.1 Safety functions as described in the PPCS /KAR 04/

Primary safety functions	– Confinement of radioactivity – Control of operational release – Limitation of accidental release
Secondary safety functions	– Protection of confinement during abnormal scenario – Ensure plasma shutdown – Ensure decay heat removal – Control the effects of coolant energy – Control chemically stored energy – Limit the impact of release of magnetic stored energy – Limit airborne and liquid radioactive releases to the environment

Radiation protection and acceptance criteria already mentioned in chapter 3 are adopted.

The confinement systems have been outlined in /MAI 05a/ on the basis of pressure suppression systems, which depend on the coolant selected for the in-vessel components. For the Model A (water cooled), the overall containments (VV, TCWS vault, cryostat, rooms surrounding the VV) have been assumed to be connected to a pressure suppression system (PSS) with dry-well and to a drain tank (DT), by passive pressure relief devices (i.e.: rupture disks). The containments of Model B (helium cooled) reactor, instead, are connected to an expansion volume (EV), always by the way of passive pressure relief devices.



(a) Model A (b) Model B

Fig. 6.1 PPCS confinement options /MAI 05a/

Accident sequence analysis includes an evaluation of the parameters that may affect the performance of barriers preventing or limiting the transport of radioactive material from the reactor to the environment:

- First barrier: the VV, its ducts, its penetrations and the in-vessel components of the primary heat transfer systems,
- Second barrier: the heat transfer system vaults, the cryostat and its penetrations, and heat transfer system guard-pipes outside of the cryostat,
- Third barrier: the wall and roof of the reactor building.

An emergency detritiation system (EDS) was applied for the plant model B concerning the radioactive releases to the external environment. The parametric study /MAI 05a/ has shown that using the combined adoption of an EDS and the increase of the EV tightness the external total releases can be reduced strongly.

Emergency cooling and inventory control have not been treated in the PPCS. H_2/dust explosion has been investigated in ITER /DEN 10/, /XIA 10/, where a number of possible solutions and design options as well as technical provisions are given, which are not discussed in this context.

6.2 ITER safety functions and systems

ITER relies mainly on the fulfilment of two fundamental safety functions /CIA 10/:

- radioactive material confinement: ensuring the personnel, public and the environment are protected against radioactive material releases, which is achieved by establishing confinement barriers and provision of associated confinement systems;

- limitation of exposure to ionizing radiation.

The main systems, structures and components (SSCs) performing safety functions in the ITER facility are the VV, VV Pressure Suppression System (VVPSS), Primary Heat Transfer Systems (PHTSs), Fusion Power Shutdown System (FPSS), Emergency cooling Systems, Hydrogen Risk Mitigation Systems, Relief Quench Valves, and emergency power supply (EPS).

Table 6.2 lists these main safety functions and those functions supporting the main ones.

Table 6.2 ITER safety functions /CIA 10/

Confinement of radioactivity	Process confinement barriers (first confinement barrier: VV and extentions process piping, etc.)
	Building confinement barriers including systems for maintaining depression and filtering/detritiating effluents
Limitation of exposure	Shielding to limit exposure and meet ALARA principle
	Access control
Protection of systems for confinement and limiting exposure	Management of pressure
	Management of chemical energy
	Management of magnetic energy
	Management of heat removal and long term temperatures
	Fire detection/mitigation
	Handling of mechanical impact (including seismic, dropped load, etc.)
	Management of mobilizable radioactive inventory
Supporting functions	Management of activated and contaminated material
	Control of safety protection and mitigation systems
	Providing auxiliaries essential for implementing safety functions
	Monitoring plant status: safety functions, radiation monitoring, etc.
	Providing protection of important-to-safety systems (e.g. earthing, lightning)
	Provide transport/lifting of radioactive components/materials
	Providing support to operator intervention (lighting, communications, etc.)

The introduction of a Safety Importance Class (SIC) classification scheme describes for ITER all SSCs that perform a safety function and their contribution to it so that the conformance to meet the General Safety Objectives during any anticipated scenario is met. Hereby the following criteria to allocate a SSC to the corresponding SIC category have been adopted for ITER:

A. A failure of a SSC can directly initiate an event, leading to significant risks of exposure or contamination.

B. The necessity of a SSC operation is required to limit the consequences of an incident leading to significant risks of exposure or contamination.

C. A SSC operation is mandatory to ensure the functioning of SIC components.

Two classes of SIC (SIC-1 and SIC-2) were defined in order to graduate the SIC components to the criteria A, B and C. Hence, SIC-1 are SSCs required to transfer to and to maintain ITER into a safe state. SIC-2 are SSCs to prevent, detect or mitigate incidents/accidents, but are not classified as SIC-1, since they are not required for ITER to attain a safe state. All remaining other components are termed as ''non-SIC'', but a further category, Safety Relevant (SR), is assigned to those which have some safety role to play, even though not credited in safety analyses .

In this context it is essential to define a single failure criterion /FER 13/: an assembly of equipment satisfies the single failure criterion if it is able to meet its purpose despite a single random failure assumed to occur anywhere in the assembly. The single failure criterion must be met at the system level for each SIC-1 class system (for example, a SIC-1 system has to provide redundancy of grade n+1, where n is the number of necessary systems) (see Table 6.3). IEC 61226 defines safety categories for nuclear safety I&C functions (category A, B, C) [56].

The single failure criterion must be taken into account in the design of the SIC-1 and SIC-2 Safety I&C systems, by using adequate solutions. Hence, the SIC assemblies must provide redundancy, independence, physical separation, and electrical isolation.

Table 6.3 Relation of SIC levels and function category /CIA 10/ /FER 13/

SSC	Single failure criterion	Emergency Power Supply (EPS)	Function safety level IEC 61226 - category
SIC-1	Yes	Yes	Category A
SIC-2	Yes	Yes	Category B / C
SR	No	No, normally under IP[1]	Category C / non safety

[1] The Investment Protection (IP) at ITER is provided by the Interlock Control Systems. These are the systems in charge of implementing all the instrumented protection functions of the tokamak and its associated plant systems.

The confinement of radioactive and hazardous materials constitutes the fundamental safety function which shall limit the mobilization and dispersion of tritium and activation products in the event of an accident /PSR 11/. Confinement refers to all types of physical and functional barriers which provide protection against the spread and release of hazardous materials. Two types of confinement are provided in ITER: a first and a second confinement. A schematic example of a confinement systems of a fusion facility (resembling the design of ITER) is illustrated in Fig. **6.2** /COR 12/. In the figure systems marked with "N" are in permanent operation to serve for the filtration and detritiation of areas potentially contaminated during normal operation. Systems marked with "S" are in stand-by and activated on demand. Fig. 6.2 shows the confinement concept during normal, non-maintenance operation.

The first confinement prevents the dispersion of radioactive or hazardous material within the facility during normal facility conditions, e.g. operation, testing and maintenance. The second confinement system limits environmental releases in events during which the first confinement system fails to completely contain the inventory. Each confinement system includes one or more static barriers and dynamic components such that sequential barriers are provided for each inventory at risk.

- The integrity of the static confinement barriers must be maintained to remain in the authorized operating range of the facility. Leak rate requirements are established for all the static containment barriers in order to allow a potential use of dynamic systems. Examples of static barriers used in the confinement systems are /COR 12/: VV and extensions (first confinement system), process piping (first confinement system), glove boxes (second barrier of the first confinement system), process room walls (second confinement system), and external walls of the nuclear buildings equipped with a detritiation system (second confinement system).

The static confinement is supplemented by the dynamic confinement ensured by the detritiation systems (DS). Dynamic confinement requires moving parts in order to fulfil their confinement function. The dynamic systems are /COR 12/:

- For the first confinement system with permanent tritium contamination (e.g. the VV), an Atmosphere Detritiation System (ADS) under permanent operation for the glove boxes and for the VV.
- For the second confinement systems, two type of systems are implemented: a detritiation system (called DS) and a ventilation system (called HVAC).

The rooms with permanent tritium contamination (e.g. hot cells) are permanently served by ADS. The rooms with potential tritium contamination only in accidental conditions are ventilated permanently by the HVAC system in normal situations, and the DS is triggered in case of tritium contamination. If both these systems fail, the static confinement of the building prevents significant releases. Initially the volume inside the building is sub-atmospheric, but if HVAC and DS are lost for a long time, the pressure would gradually rise and some small leakage to the environment could be foreseen (through cracks, door seals, etc.).

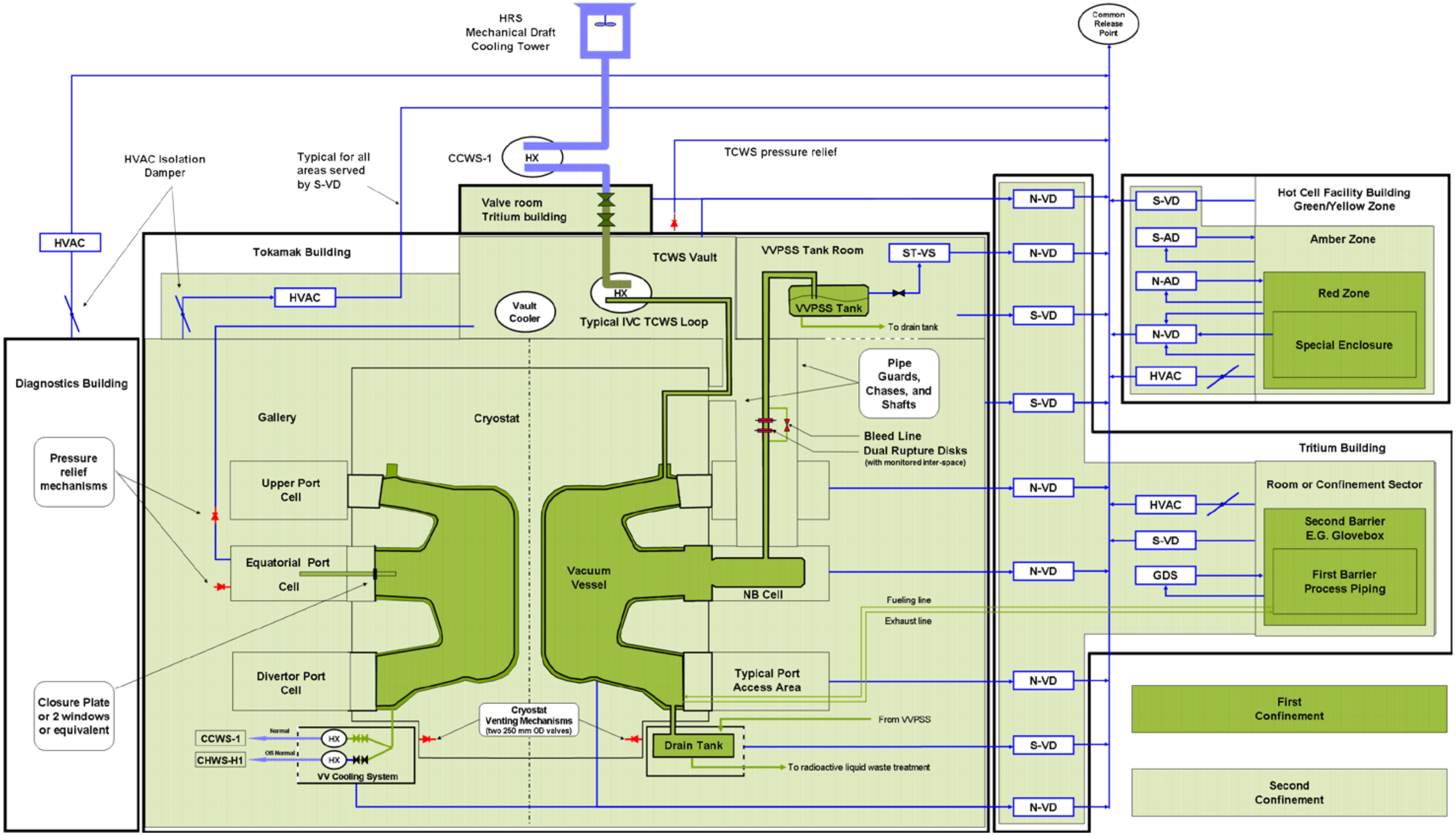


Fig. 6.2 Example for primary and secondary confinement systems /COR 12/

In a FPP the safety objectives will be also focused on the mitigation of any outside emergency responses irrespective of any postulated event of internal or external nature. Since a FPP comprises in terms of nuclear inventories and components a larger bandwidth, correspondingly the safety analyses have to be complemented compared to ITER either in dimensioning, parameters and performance of the safety functions but also by additional safety relevant systems.

6.3 The proposed fusion safety systems based on ITER

A new concept for a future FPP safety system has been sketched by us on the basis of the ITER approach. Besides the maturity of the design and the on-going licensing activities in ITER, additional or different systems may be required for a FPP. Based on ITER reference documentation, Table 6.4 illustrates the systems that are credited to provide major safety functions that could be applied in a future FPP without claiming completeness. For each system the following information is given:

- intended function of the system
- Logic to perform the safety function (active/passive), according to the definition in /IAE 07/
- Under which conditions the system is required
- What happens if the system fails

Hereafter is reported the definition of passive components from /IAE 07/, which has been used as reference for Table 19.

Passive component: component, whose functioning does not depend on an external input such as actuation, mechanical movement or supply of power. A passive component has no moving part, and for example, only experiences a change in pressure, in temperature or in fluid flow in performing its functions. In addition, certain components that function with very high reliability based on irreversible action or change may be assigned to this category (e.g. rupture disks).

Examples of passive components are heat exchangers, pipes, vessels, electrical cables and structures. It is emphasized that this definition is necessarily general in nature, as is the corresponding definition of active component. Certain components, such as rupture discs, check valves, safety valves, injectors and some solid state electronic devices, have characteristics which require special consideration before designation as an active or passive component. Any component that is not a passive component is an active component.

The barriers shown in Table 6.4 are defined for the systems as follows:

- The first barrier is the VV and the PHTSs.
- The second barrier is the VVPSS and its connection to the VV. Together with the first barrier they are part of the first confinement.
- The third barrier is the Tokamak Building and the active systems. They provide the second confinement.

Table 6.4 Summary of major SSCs important to safety for future FPPs

System	Safety function	Level of confinement	Typology referred to /IAE 91b/, /IAE 07/	Call on service	Consequence if it fails
VV and its extension	Confinement	First barrier, first confinement	Passive	Always	Loss of first confinement
VVPSS	Confinement	Second barrier, first confinement	Passive	In-Vessel LOCA	Loss of first confinement, second barrier; Release into second confinement
Tokamak and tritium building	Confinement	Third barrier, second confinement	Passive	Always	Loss of second confinement, release to the environment
FPSS	Plasma Termination		Active	E.g. ex-vessel LOCA, in-vessel LOCA, loss of heat sink etc.	Potential partial failure of PFC
Emergency cooling	Decay Heat Removal	Second barrier, first confinement	Active	Unavailability of the VV PHTS	Failure of active heat removal
HVAC	Room air conditioning/ maintaining depressurized atmosphere in second confinement	Third barrier, second confinement	Active	Normal operation	Rise in pressure of the tokamak building, need of common release point
Normal Detritiation System (NDS)	Collect tritium released during normal operation	third barrier, second confinement	Active	Normal operation	SDS starts
Stand-by Detritiation System	Collect tritium re-	third barrier, second	Active	High level of radioac-	Rise in pressure of

System	Safety function	Level of confinement	Typology referred to /IAE 91b/, /IAE 07/	Call on service	Consequence if it fails
(SDS)	leased during abnormal scenario, pressure control	confinement		tivity inside the second confinement	the tokamak building, possible release of radiological inventory
Common release point	Ensure the second confinement pressure not exceeding the maximum design pressure by release through the stack	Third barrier, second confinement	Active	Second confinement overpressure signal	Second confinement overpressure
Nitrogen Injection/PAR	Avoid hydrogen explosion		Passive	hydrogen generation	Possible hydrogen explosion
Coil fast discharge system	Avoid arc in magnets, avoid short in magnets		Passive	Temperature increase in magnets	Quench of the magnets, possible damage to the confinement barrier
EPS	Supply emergency safety systems		Active	LOSP	No power supply to the safety systems
Fire barriers/suppression	Prevent propagation of fire		Passive/Active	Fire	Propagation of fire and possible release

7 Comparison of the nuclear fission safety concepts (enveloping event, defence in depth) with the fusion safety concept

In the following the safety concept of a FPP is compared with the concept of an enveloping event, as well as the concept of defence in depth as used in a nuclear fission power plant (NPP). The fundamental requirements of the German nuclear law ("Atomgesetz") are substantiated in the safety requirements for NPPs ("Sicherheitsanforderungen an Kernkraftwerke", /BMU 13/). Germany has released this new detailed regulatory framework in 2012. The current German regulation for the safety of NPPs /BMU 13/ is used as basis for this comparison, as it can be seen as the state of the art in science and technology implementation of nuclear fission safety concepts in national regulation.

In section 7.1 the possible consequences of an enveloping event both for NPPs as well as FPPs are identified. On this basis it is discussed whether a detailed safety concept for the prevention of such consequences is necessary for FPPs. In section 7.2 the safety concept of NPPs is compared with the current safety concept of FPPs.

7.1 Enveloping event

The need for a detailed safety concept for NPPs largely relates to its radioactive inventory. The release of relevant fractions of this inventory can lead to significant effective doses for members of the public.

Because the source terms are substantially different between fission and fusion, a comparison between the radiological inventories and the possible release fractions will give an indication about the necessary level of safety.

7.1.1 Radiological risk of NPPs

The major risk of NPPs is related to the possibility of the release of parts of the radioactive inventory. The inventory of the reactor core of a typical NPP is given in Table 7.1.

Table 7.1 Inventory of important radionuclides of a pressurised water reactor with 3733 MW thermal power at end of cycle, 6 hours after shutdown according to /SSK 04/

Nuclide	Activity in Bq
Iodine	$1{,}9\times10^{19}$
Cs-137	$3{,}0\times10^{17}$
Noble Gases	1,2×1019
Aerosols	$1{,}7\times10^{20}$
Total	$2{,}5\times10^{20}$

For design basis accidents in NPPs, the German radiation protection ordinance requires to show that releases do not lead to an effective dose of more than 50 mSv (lifetime dose including ingestion).

For beyond design basis accidents, large releases are possible. Maximum values for release fractions are resulting from early and large containment failures and amount to 100 % of noble gases, 50-90 % for Iodine, Caesium and Tellurium, 40 % for Strontium and about 4 % of the actinide inventory /SSK 04/.

For example, a release of about 1 % of the total Caesium inventory of a NPP e. g. about 3×10^{15} Bq of Cs-137 may result in the need to evacuate people (based on an effective dose of 100 mSv by external exposure and inhalation within 7 days) in 1 km distance to the source /BMU 99/. Thus, a release of large parts of the reactor core may result in early doses to the public of the order of several Sieverts or more.

7.1.2 Radiological risk of FPPs

The radioactive inventory of a FPP consists mainly of the tritium inventory, radioactive dust, and activated material within the cooling system. The typical inventory for a FPP is given in Table 2.5.

Under the assumptions discussed in detail in section 5.1, bounding sequences will lead to doses for the most exposed individual (early dose[2] within 7 days) of 1.16 mSv for Model A and 18.1 mSv for Model B.

These are the highest doses calculated for events driven by in-plant energies analysed so far for PPCS. The doses for the most exposed individual are not directly comparable to the requirements of the German radiation protection ordinance, which relates to the lifetime dose including ingestion, but they are comparable to the early dose rates underlying the need to evacuate people. In any case, these doses are relatively low and would not necessitate early radiation protection measures.

In these events, only about 1 % of the radiologic inventory of a FPP is released. This is due to the small fraction of mobilized radioactive inventory and the high retention factors due to the integrity of at least one confinement structure.

In /SEA 95/ the consequences of complete destruction of the confinement were analysed. Such destruction could be the result of external events like a very large earthquake or an airplane crash. In /SEA 95/ an upper limit for release of tritium of 1 kg was assumed. Under worst-case assumptions this would result in a dose to a member of the public of up to about 0.4 Sieverts, in a small area close to the plant. The same result is cited in /SEI 01/.

Based on these numbers, the theoretically possible maximum releases from FPPs may thus result in doses to the public of the order of one Sievert or below. These doses are several orders of magnitude lower than those for hypothetical worst-case scenarios of NPPs /GUL 93/. Nevertheless, without a safety concept to exclude such releases, these doses would necessitate radiation protection measures outside of the FPP.

To prevent such releases, a fusion specific safety concept is necessary, which was discussed in chapters 3 to 6.

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[2] "Here, the dose criterion is defined as committed effective dose equivalent for the first 7 days exposure. This includes the exposure pathways external irradiation from the passing cloud and the first week external irradiation from the ground, the internal exposure from inhalation + skin absorption and the internal exposure from inhalation + skin absorption from the reemitted radionuclides during the first week." /MAI 05a/

7.2 Comparison of fundamental aspects of the fission safety concept with the fusion safety concept

A direct application of the current nuclear regulations to a FPP will not be possible without fusion specific adaptions due to the differences in underlying physics and technologies.

For a comparison between the safety concept of fusion and fission, in the following sections fundamental aspects of the current fission safety concept are discussed.

Then, the similarities and differences between fission and fusion safety concepts are identified. The differences may arise either from the fundamentally different physics or from different technical approaches to guarantee safety.

It will be discussed

- whether the regulatory requirements could be adapted analogously to future FPPs,
- whether these requirements would already be covered by the current safety concept of fusion,
- whether there are differences between fission and fusion, so that requirements could be completely omitted or
- whether fusion specific requirements would need to be added.

7.2.1 Reactivity control, fuel and inventory

In a NPP by far the largest part of the inventory is stored inside the fuel rods. The fuel rods are either located inside the reactor core or in the spent fuel pool.

The decay of the fission products and actinides produces residual heat which has to be removed to avoid melting of the fuel. Also special care has to be taken to avoid an unwanted (re-)criticality of the fuel which would result in the (additional) production of heat and the renewed production of fission products.

Therefore, /BMU 13/ explicitly specifies requirements for the properties of the cladding of the fuel rods as the first barrier[3], for cooling the fuel in the reactor core[4], for the handling and the storage of the fuel in the spent fuel pool [5], and for the control of reactivity and the prevention of re-criticality[6].

A FPP is based on a thermonuclear burn process in a tenuous plasma, which is fundamentally different in its reactivity behaviour from that of a chain reaction. The high temperature needed can only be maintained if the magnetic configuration forms perfect nested shells, and impurities of the plasma, caused e. g. by reactions with the first wall, are kept at a very small level. Any ingress of particles in case of an arbitrary postulated event would lead to a termination of power production. In addition, the amount of fuel contained in the plasma can maintain the reaction only for a couple of minutes (see chapter 2.4.2). Also an over-fuelling of tritium and deuterium would lead to plasma instabilities and termination of power production. Excursions of the reaction rate can therefore be excluded, and any abnormal event will lead to a termination of the fusion reaction.

The neutrons generated in fusion do not actively take part in the reaction. Hence, a reactivity in the sense of a feedback as in a NPP is not given by physical means. The fusion process is a self-limiting nuclear reaction, which does not involve a chain reaction.

Due to the intrinsic heat conductivity of the plasma, a FPP has to have a certain size. This translates in terms of the volumetric power density in the plasma during the plasma burn to values which are by orders of magnitude lower than in the core of a NPP. As a consequence, the energy stored in the plasma is not sufficient for a large scale destruction of the enclosing structure.

However, secondary effects like plasma instabilities (i.e. disruptions) may affect the integrity of the first barrier. Therefore, dedicated measures in terms of a disruption mitigation system are foreseen to mitigate plasma instabilities and prevent major damages

[3] /BMU 13/, 2.2 (3)

[4] /BMU 13/, 3.3

[5] /BMU 13/, 3.10

[6] /BMU 13/, 3.2

to the first wall. The disruption mitigation system is in that sense not a measure to prevent an unintended power excursion but rather a tool to preserve the first barrier.

To which extent a fusion power shutdown system is necessary is not clear at present in the absence of operational experience either in tokamaks or stellarators. However, even without FPSS no power excursions are possible, but rather a plasma disruption may take place potentially leading to a damage of in-vessel components.

The fusion process can be stopped either by operational measures (e.g. stopping the re-fuelling), a safety system like FPSS or it will stop inherently.

Thus, requirements to fulfil the fundamental safety function for the control of reactivity in /BMU 13/ cannot be applied to FPPs. Especially there is no need for reactivity control for the spent fuel (He) in FPPs. The activated material in fusion arises from interaction of the neutrons with plasma facing materials in both the blanket and the divertor. However, neither the spent fuel nor the activated material is capable to attain a critical mass, initiating a nuclear chain reaction as it may occur in a NPP. The fundamental safety function for the control of reactivity in /BMU 13/ also includes the requirement to be able to shut down the facility under any circumstances. For a FPP this requirement is fulfilled, because all physical processes potentially affecting the fusion reaction are of passive fail-safe nature in the sense that they will terminate the fusion reaction

Most of the activated material in FPPs is distributed in a large volume with a low volumetric power density and moreover is immobile, as long as no other energy sources are available to mobilize it.

The design and the choice of materials of the plasma facing components significantly determine the potential for activation. An additional source for activated products is the coolant. It can be activated itself (e. g. PbLi) or can contribute to the transport of radioactive materials (tritium in water or tritium in He). In principal, it is possible to limit the inventory of activation products, by an appropriate choice of materials and coolants. Table 2.5 shows that for all plant models in /MAI 05a/ radioactive dust has to be considered. If water is used as coolant, then ACPs have to be considered, too.

7.2.2 Barriers

According to /BMU 13/ the technical safety concept of NPPs is based on the safe confinement of the radioactive materials. To realise this, multiple barriers, supported by retention functions and the protection of those by measures and installations on several consecutive levels of defence are implemented. Besides confining the radioactive inventory, the integrity of the barriers is furthermore important to ensure the coolability of the fuel.

The barriers in nuclear fission consist of the fuel rod, the reactor coolant pressure boundary and the containment.[7] Additionally, the containment has to be enclosed by a reactor building. The reactor building has the fundamental function to protect the containment against loads from internal or external events, including very rare man-made external hazards.[8]

Besides the static barriers, retention functions are implemented that comprise e.g. of ventilation systems to produce driving pressure differences and to collect leakages out of the containment. Additionally, the closure of containment penetrations has to be ensured.[9]

The barriers and retention functions shall be designed in such a way that the respective safety-related acceptance targets and acceptance criteria as well as the radiological safety objectives are met on the different levels of defence for all events or plant states.[10]

On levels of defence 1 and 2, retention functions of all three barriers have to remain intact, while on level of defence 3 and 4a at least the retention functions of the containment has to stay intact. The integrity of the fuel and the reactor coolant pressure boundary may be compromised by loss of coolant events on level of defence 3.[11] In NPPs severe fuel damage may result in the melting of the fuel and the mobilization of

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[7] /BMU 13/, 2.2 (1) and 3.6 (1)

[8] /BMU 13/, 3.6 (4)-(6)

[9] /BMU 13/, 3.6 (3)

[10] /BMU 13/, 2.2 (1)

[11] /BMU 13/, 2.2 (3) – 2.2 (6)

the radioactive inventory. Therefore, the level of defence 4c aims to mitigate the consequences of severe fuel damage[12].

Possible sources for the release of radioactivity to the environment in FPPs are tritium and activated materials.

The heat source in a FPP is provided by a DT reaction occurring statically in the vacuum vessel domain if the ignition criterion is matched. Hence, the reaction is not confined to a prescribed geometry as e.g. a fuel rod. Therefore, the vacuum vessel with the corresponding components acts as a first confinement barrier. In order to ensure the integrity of this barrier at accidental conditions, passive measures to reduce a possibly evolving overpressure are foreseen, such as pressure suppression pool or expansion volumes.

A second confinement barrier is provided by the reactor building itself.

The radioactive inventory of a FPP is not concentrated in the fuel. Thus different potential sources have to be taken into account. For all of these, different barriers are to be implemented in a FPP.

With respect to inventories other than the fuel, other first barriers do exist, like the piping of the heat transfer systems, tritium process lines or the hot cells. A second confinement is typically provided by the surrounding building, too.

Sufficient confinement integrity in FPPs necessitates the use of the following retention functions:

Operational systems collecting exhausts from torus pump-down systems demand filtering and detritiation to ensure a safe release of gaseous products to the environment. Also the atmosphere of the different buildings needs the presence of systems with a detritiation function on an operational basis.

Under DBA conditions the retention function must be fulfilled by a standby detritiation system. Additionally, a closure of all penetrations of the secondary confinement is fore-

[12] /BMU 13/, 4.4

seen, especially the operation of the Heating, Ventilation and Air Conditioning (HVAC) system has to be terminated.

Depending on the radiological criteria of the different levels of defence (see chapter 3), a sufficient confinement of the radioactive inventory has to be demonstrated.

Furthermore, technical acceptance criteria have been formulated, like maximum allowable temperatures for the different internal structural components so that the structural integrity of these barriers is ensured. Similarly upper bounds of the leak tightness of the second confinement barrier are set.

The analysis of all DBA and BDBA scenarios in /MAI 05a/, which are initiated by the release of plant-internal energies, shows that the second confinement remains intact.

Thus, the safety concept of FPPs is comparable to the concept of NPPs relying on a set of physical barriers and retention systems. The basic principle is to prove the integrity of at least one barrier, so that the radiological limits are fulfilled. However, due to the entirely different inventory and progression paths, as well as energy sources and fluxes, the technical implementation of the barrier concept and the establishment of active countermeasures require adapted defence strategies, which are substantially different compared to NPP.

7.2.3 Defence in depth and independence of levels of defence (and corresponding safety functions)

The fission safety concept is based on the concept of defence in depth (see Table 7.2 and /BMU 13/ 2.1). Thereby, both initiating events as well as plant states are assigned to levels of defence. In

- level of defence 1 the normal operating conditions, including testing conditions,
- level of defence 2 events anticipated to occur during the operating lifetime of the plant,
- level of defence 3 a spectrum of events not to be expected to occur during the operating lifetime of the plant, but which have to be assumed to demonstrate the safety of the plant

have to be considered for the design of the measures and installations.[13] For the level of defence 4, very rare events, events with multiple failure of safety installations, and accidents involving severe core damage have to be considered.[14]

According to /BMU 13/, on levels of defence 2 and 3, measures as well as installations shall be provided that are arranged in such a way that upon the failure of measures and installations on levels of defence 1 and 2, the measures and installations on the subsequent level re-establish the required safety-related condition independent of measures and installations of other levels of defence. Additionally, measures and installations that have to be effective on all or on several of these levels of defence are designed such that they can withstand the impacts associated with these levels in accordance with the criteria that apply to these levels. [15]

Table 7.2 Levels of defence as defined in /BMU 13/

Level of defence	Description	Objectives
1	Normal operation	Prevention of abnormal operation
2	Abnormal operation	Control of anticipated operational occurrences, prevent design basis accidents
3	Design basis accidents	Control design basis accidents prevent the onset of events involving the multiple failure of safety installations
4a	Very rare events	Control the effects of very rare events
4b	Events with multiple failure of safety installations	Prevent severe core damage (preventive accident management measures)
4c	Accidents involving severe fuel damage	Limit the release of radioactive materials into the environment as far as possible (mitigative accident management measures)
5		Support of off-site radiation protection measures

[13] /BMU 13/, 4.1

[14] /BMU 13/, 2.1

[15] /BMU 13/, 2.1 (6)

With respect to the independence of the different levels of defence, it shall be ensured that a single technical failure or erroneous human action on one of the levels of defence 1 to 3 will not jeopardize the effectiveness of the measures and installations on the next level.[16]

Thus, in NPPs, several safety functions are ensured by multiple installations related to the different levels of defence. Important examples in pressurized water reactors are the control of reactivity (by the volume control system, an additional borating system, control rods and inherent features of the core), the pressure control and the heat removal from the secondary side (by the steam bypass to the condenser, pressure relieve valves and safety valves), and the auxiliary function of electricity supply (by the main generator, the external electrical grid, emergency diesel generators or steam driven systems).

The safety concept of fusion is also based on the concept of levels of defence (see chapter 3 and Table 7.2). The assignment of initiating events to different levels of defence is part of the safety concept (see chapter 4).

In principle it is possible to assign the safety functions of a FPP to certain level(s) of defence according to the level of defence of the postulated event (design basis accident, beyond design basis accident), where a function is used during the event sequences, or the corresponding plant state. However, this mapping necessitates a detailed plant design, which is at present not available. Hence, assignments are conducted on a functionality basis and reasonable order of magnitude assessments.

While operational systems are used on the first levels of defence, specific safety systems are introduced to deal with design basis events. For several beyond design basis accident scenarios, active as well as passive features of FPPs are credited.

Based on the current level of FPP designs the implementation of the concept of defence in depth can be identified for different measures and installations e.g. for shutdown (see discussion in section 7.2.1) and heat removal (see discussion in section 7.2.7).

[16] /BMU 13/, 2.1 (7)

Also, the fundamental implementation of the defence in depth principle can be shown today for some of the necessary measures and installations to fulfil the safety functions of a FPP. A detailed discussion of the defence in depth, especially for the independence of the different measures and installations for all safety functions is currently not possible, because detailed design information is not yet available.

For example, a fast coil discharge system is necessary to deal with the possible quench of the superconducting magnets. Several fusion facilities have fast magnet discharge systems, which are designed for a controlled shut-down of the energized magnets. To guarantee the safety of a FPP a quench of the magnets should not compromise the integrity of safety relevant components. It is still subject to analysis whether and how the effects of a quench and of consequential damage can be controlled by independent installations on different levels of defence.

7.2.4 Level of defence 4

As discussed for example in /WEN 09/, Annex 2, the original defence in depth approach for NPP covered three levels of defence, which constitute the design basis. As consequences of lessons learned from the development of probabilistic safety assessment (PSA) and from NPP plant accidents, this original approach was adapted and two additional levels of defence were added. Because these were not initially within the design, these levels and the corresponding measures were considered to be beyond design. For new NPPs, these levels and the corresponding measures will be part of the design. This covers especially measures and equipment to deal with selected multiple failure events and postulated core melt accidents.[17]

According to /BMU 13/, even in the current regulatory framework, the fourth level of defence is integral part of the safety concept. This fourth level of defence in Germany covers two distinct groups of events.

On level 4a dedicated very rare events with the postulated failure of the shutdown system have to be assumed. While of high relevance for fission reactors, anticipated transients without scram as covered in the level of defence 4a will probably not be as important for FPPs (see the discussion in section 7.2.1). Based on the physical character-

[17] /WEN 09/ p. 21ff.

istics of fusion the requirements of level of defence 4a of /BMU 13/ are likely not to be applicable to fusion.

Furthermore, preventive and mitigative accident management measures have to be introduced to deal with events with multiple failures of safety installations or accidents involving severe core damage on levels of defence 4b and 4c. These measures have to ensure, that at least one confinement barrier remains intact.[18]

While appropriate measures and installations of the levels of defence 1 to 3 may also be used on level of defence 4 depending on their availability, other measures and installations are provided specially for level of defence 4. These specific measures and installations may not be credited on the former levels of defence 1-3. The aim of this level of defence is to prevent early or large releases and to ensure, that radiological consequences outside of the plant necessitate only measures that are limited in time and space.[19]

Level of defence 4c covers plant states with damage to the reactor core or the spent fuel. During such events, phenomena different from those in DBAs will occur. Measures on levels of defence 4 include a diverse heat sink in case of loss of the ultimate heat sink, installations for depressurisation of the reactor cooling system, measures to prevent a long-term temperature or pressure increase in the containment (venting) and the prevention of combustion processes of gases (H_2, CO) endangering the integrity of the containment (inertisation of containment atmosphere and/or passive hydrogen recombiners).[20]

To ensure that at least one barrier remains intact, the current safety concept of FPPs covers five levels of defence (see chapter 3). The fourth level of defence in fusion (control of severe conditions) and the fifth level (mitigation of radiological releases) are consistent in their objective with levels of defence 4b and 4c in /BMU 13/. The fact that BDBA are already covered in the safety concept of FPPs is also in line with the current state of the art in science and technology of international developments like /WEN 09/ and /WEN 13/.

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[18] /BMU 13/, 2.1(1), 2.1 (3), 2.2 (5), 4.3(4).

[19] /BMU 13/ 2.1(10), 2.1(11) and 2.5 (1).

[20] /BMU 13/ 3.3(5), 3.4 (5b), 3.6(7), 3.6 (8), 3.8(2).

While the fundamental phenomena that may arise in FPPs during severe accident conditions will differ from those of NPPs and may thus justify differences in the safety concept, some similarities do nevertheless exist.

Under the assumption of a total loss of cooling, the temperature in the vacuum vessel due to residual heat of activated materials will increase. Correspondingly, a long term increase of both pressure and temperature in the second confinement may take place, see Table 6.4. A common release point was identified as a safety function, which would be used during such BDBA (equivalent to a venting system of a NPP).

If the primary heat transfer system in a FPP would rely on water, an interaction of hot internal structures of the plasma vessel (first wall, blanket structures) with water could be assumed in case of in vessel loss of coolant events. This in turn may lead to the production of hydrogen and potentially associated with this to hydrogen deflagrations or detonations. Thus, measures to address this phenomenon like a fast inertisation of the vacuum vessel and/or the installation of hydrogen recombiners have to be integrated in the design (see Table 6.4). Independent of the coolant, the existence of certain dust quantities within the vacuum vessel opens the possibility of dust explosions after vacuum collapse along with oxygen ingress in the vacuum vessel, which requires a safety analysis.

In /MAI 05a/ it was shown that even without an active cooling of the vacuum vessel, melting or structural damages of the components of the vacuum vessel cannot be expected. If this can be confirmed for a detailed FPP design, requirements with respect to cooling systems can be anticipated to be less restrictive for FPPs (e. g. no need for a diverse heat sink).

The current fission safety requirements can be applied correspondingly to FPPs with respect to the necessity to consider special accident sequences, accident phenomena, and the need for specific accident management measures. Fusion specific sequences and phenomena are addressed in the current safety concept of fusion. The required measures and installations can only be concretised based on a specific plant design.

7.2.5 External events and very rare man-made external hazards

A complete fission reactor safety analysis incorporates an analysis of the impact of external events on the plant. /BMU 13/ requires that all measures and equipment needed to safely shutdown the reactor, to remove decay heat and to confine the radioactive inventory have to keep their function even under the assumption of external events.[21]

Specifically, the safety systems and measures and equipment to deal with very rare man-made external hazards (examples see below) shall be available during external events.[22] Especially the integrity of the confinement has to be ensured under the conditions of very rare man-made external hazards as well as other external events.[23]

Depending on the site characteristics, the worst external events possible including earthquakes, flooding and extreme meteorological condition (storm, lightning …) as well as combinations thereof have to be considered.[24] As very rare human induced external events at least an airplane crash, external explosions and the impact of dangerous goods have to be taken into account.[25]

These requirements /BMU 13/ are applicable to fusion with respect to the event spectrum to be considered. External events and very rare man-made external hazards are not yet explicitly covered in the safety concept of FPPs. Still, it was concluded in earlier safety studies, that only certain ex-plant events have a potential for breaching the ultimate confinement barrier. Thus, it was suggested that some external events, in particular airplane crash and earthquakes, are to be covered by the design basis (see chapter 4.1.2).

Provided that the measures and installations for the shutdown of the reactor and the residual heat removal of a FPP are based on inherent features of the plant (see chapter 7.2.1 and 7.2.7) the implementation of these safety functions will be easier also in the case of external events and very rare man-made external hazards compared to NPP.

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[21] /BMU 13/ 2.4 (1) and 2.4 (4).

[22] /BMU 13/ 2.1 (5).

[23] /BMU 13/ 3.6(1)

[24] /BMU 13/ 4.2 (1). Typically, external hazards with a return period of no less than 10^4 years are considered.

[25] /BMU 13/, Appendix 3.

Nevertheless, external events will influence the site selection process for future FPPs. Depending on the selected site, different impacts on the facility will have to be covered or sites with unfavourable characteristics will have to be excluded.

In any case, man-made external hazards like the crash of a large airplane will pose demands especially on the reactor building (second confinement) of FPPs. Because of the fundamental role of the second confinement, especially the capacity to withstand external events will have to be clearly demonstrated.

7.2.6 First of its kind

The current regulatory framework requires the use of proven technologies and qualified materials as well as validated calculation methods for the safety demonstration of a NPP based on this operational experience.

Specifically, as an overall technical criterion, in /BMU 13/ the use of qualified materials as well as production and maintenance technologies and of equipment that have been proven by operating experience or which have been sufficiently tested is required on levels of defence 1 to 4a.[26]

Furthermore, it shall be possible to inspect and maintain all safety-relevant installations to a sufficient degree. Otherwise, measures have to be provided to control the possible consequences of failures of such installations.[27]

With respect to the safety demonstration of NPPs, calculation methods that are validated for the respective scope of application shall be used.[28]

For FPPs, in comparison only minor operational experience is available up to now. Since the 1960s, several experimental fusion facilities are in operation to test and verify the fundamental aspects of fusion physics and technology. Still, no experimental facility or prototype reactor exists today that is operating under reactor like conditions of a future FPP.

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[26]/BMU 13/, 3.1 (2)

[27] /BMU 13/, 3.1 (12)

[28] /BMU 13/, 5 (4)

Several aspects of a future FPP are beyond current operational experience. They mainly address aspects as:

- material properties (at high neutron fluences and large temperature gradients)
- fuel cycle performance at full scale and tritium management. Especially, the fuel cycle and tritium breeding in test blanket modules are to be tested in ITER.

Also specific technologies required for a FPP, are out of NPP engineering scope such as:

- very large vacuum vessels (several thousand m³),
- high magnetic fields (> 5 Tesla) enfolding large volumes
- very high heat fluxes through first wall and divertor (peak divertor wall loads can attain ~ 10 MW/m² compared to fission typical data of about 1 MW/m² for the fuel cladding heat flux),
- high energy neutrons (about 14 MeV) interacting with the atoms in the wall structures and high resulting material damage
- liquid metals as breeder material.

While the requirements of /BMU 13/ on the high quality of measures and installations will also apply for FPPs, the detailed requirements with respect to the evaluation of the operation experience will not be directly applicable to future FPPs.

7.2.7 Cooling

In NPPs the decay heat from used fuel elements has to be removed by active systems to avoid eventual fuel element damage. Thus, the continuous cooling of the reactor core and the spent fuel pool is an important safety function. The current regulatory framework specifies requirements for all levels of defence to ensure the cooling of the fuel.[29]

In FPPs the activated structures of the blanket, the divertor and other in-vessel components produce decay heat which can reach significant levels. For PPCS plant model B,

[29] /BMU 13/, 3.3

the decay heat is about 0.6 % of nominal thermal power (3180 MW) one hour after shutdown. So a residual heat production of about 20 MW to 40 MW has to be expected at that time, a level comparable with that of a fission power plant (about 1 % of nominal thermal power). However, in the blanket of a FPP the decay heat per volume is orders of magnitude lower than in the fuel rods of a fission reactor. For a detailed design, analyses are necessary to show that any local decay heat production does not endanger the integrity of the first barrier.

Accident analyses in /MAI 05a/ showed (under simplified design assumptions) a significant temperature increase has to be expected in the components of the vacuum vessel if all active cooling systems fail. Nevertheless the maximum temperatures achieved are below melting points of components or structural failure limits of the first barrier. If this can be confirmed for the detailed design of a future FPP, then the fundamental safety function “cooling” (of the activated components) could be guaranteed without active systems under BDBA conditions.

In principle, there is the requirement for cooling the activated components of a FPP, because the decay heat in a FPP is of the same order of magnitude as in a NPP. If it can be proven that the heat can be removed passively, reduced requirements would apply for the detailed implementation of the measures and installations to meet this fundamental safety function.

7.2.8 Leak before break

/BMU 13/ requires that for certain parts of the piping the component integrity is guaranteed by applying the “leak-before-break concept” (LBB).[30] The underlying safety importance for NPPs is related to the fact, that in case of a fast opening of a large break in the cooling system, dynamic forces may occur, such that the geometry of the core components cannot be guaranteed anymore. However, the integrity of the core geometry is required to prove the coolability of the fuel. This effect is important if water (or other liquids) with high pressure are used for cooling.

The fulfilment of the LBB concept would have to be considered already in the design of a plant, e. g. for the selection of appropriate materials.

[30] /BMU 13/, 3.4 (3)

No information was found in literature whether the LBB concept will be a requirement for the coolant piping and cooled blanket structures in a FPP. Only one reference is given in /USD 99/. For a FPP, especially if water under high pressure is used for cooling like in PPCS model A, it would either be mandatory to demonstrate, that pressure waves induced by a fast large break LOCA would not challenge the integrity of barriers and the coolability of the reactor, or the LBB concept would have to be implemented.

The analogous applicability of the LBB concept on FPPs cannot be assessed based on the currently available literature.

8 Summary and outlook

The analysis of the radioactive inventory of a FPP shows that it is necessary to confine the inventory safely. A safety concept is required to guarantee the confinement. The safety concept of FPP is based on the concept of defence in depth. It emphasises the use of inherent characteristics or passive safety mechanisms. Each postulated event, the resulting plant states and the measures and installations which are foreseen to control DBA and BDBA can be assigned to the different levels of defence.

The potential release of tritium has been identified as a major potential hazard.

The safety properties of a FPP, however, appear to be much more advantageous compared with those of a fission reactor:

- Due to the nature of the fusion reaction only a small amount of tritium is available to sustain the burning fusion plasma, sufficient only for a few minutes of plasma burn. The power densities are comparably small (few MW/m^3).
- The power generation can be stopped reliably by switching off the fuel supply. During failures of systems (magnets, cooling, vacuum leaks etc.) a plasma quench will occur by inherent processes, and hence the energy production is stopped.
- Also, potential power excursions would lead to the termination of power production. Criticality accidents are principally excluded.

Furthermore it was discussed whether the NPP safety concept can be transferred to FPPs. It was shown that the safety concept of NPPs can be in principle applied to FPPs. Aspects were discussed which imply differences to be taken into account in the safety concept due to differences in the underlying physics and in fusion specific technologies.

The detailed requirements for the measures and installations of a FPP on the different levels of defence have not been established yet. Ultimately, the requirements for the safety concept depend on the hazard potential of a plant. /IAE 12/ states in paragraph 2.14, that “the number of barriers that will be necessary will depend upon the initial source term in terms of amount and isotopic composition of radionuclides, the effectiveness of the individual barriers, the possible internal and external hazards, and the potential consequences of failures.” Due to the lower potential releases it is plausible that future requirements on measures and installations in a FPP will be different from those in a NPP.

The aim of the safety concept of fusion is to avoid the necessity for any off-site emergency response or disaster control measures. So far fusion safety analyses focused on plant-internal events. The analyses for plant-internal events based on stored physical, chemical and magnetic energies show that this goal can be achieved in principle. The safety concept of fission will evolve in the future. Events like very rare man-made external hazards, events with multiple failures of safety systems, and core melt accidents should be covered by DBA /WEN 09/. In principle, it has to be expected that the requirements with respect to the event spectrum for FPPs will be comparable to those of new NPPs with regard to natural external hazards and very rare man-made external hazards. Together with an increased level of detail of the plant designs of future FPPs, external events have to be analysed in detail, e. g. external hazards like earthquakes and flooding, as well as very rare man-made external hazards like the crash of a large air plane.

Especially the question if plant-external disaster control measures (corresponding to level of defence 5 of the fission regulation) will be necessary, and how detailed the plans for those measures have to be, depends on the detailed plant concept and the resulting hazard potential. At the moment no information is available on plant-external disaster control measures for FPPs in terms of level of defence 5 of the fission regulations. In the future parameters like the radioactive inventory, possible releases, and resulting doses in the vicinity of the plant have to be considered when specifying the requirements for this level of defence.

Based on the evaluation of the currently available literature, it is concluded that an elaboration of more detailed FPP concepts should go hand-in-hand with the development of more detailed safety concepts.

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Abbreviations

ACP	Activated Corrosion Product
ALARA	As Low As Reasonably Achievable
BDBA	Beyond Design Basis Accidents
BLK	Blanket
CD	Current Drive
DBA	Design Basis Accidents
DCLL	Dual Coolant Lithium Lead
DEMO	DEMOnstration Power Plant
DS	Detritiation System
DT	Drain Tank
EDS	Emergency Detritiation System
EPS	Emergency Power Supply
EST	Environmental Source Term
EV	Expansion Volume
FFMEA	Functional Failure Modes and Effects Analysis
FMEA	Failure Modes and Effects Analysis
FPP	Fusion Power Plant
FPSS	Fusion Power Shutdown System
FPTS	Fusion Power Termination System
FPY	Full Power Years
FW	First Wall
HAZOP	Hazard and Operability
HCLL	Helium-Cooled Lithium-Lead
HCPB	Helium-Cooled Pebble Bed
HDH	High Density H-mode regime
HTO	Tritiated Water
HVAC	Heating, Ventilating, Air Conditioning System
ICRP	International Commission on Radiological Protection
INSAG	International Nuclear Safety Group
IP	Investment Protection
I&C	Instrumentation and Control
ISS	Isotope Separation System
ITER	International Thermonuclear Experimental Reactor
JET	Joint European Torus

LHD	Large Helical Device
LoD	Level of Defence
LOCA	LOss of Coolant Accident
LOFA	LOss of Flow Accident
LOSP	Loss of Off-Site electric Power
LOVA	Loss of Plasma Vessel Vacuum
MC	Magnetic Confinement
MEI	Most Exposed Individual
MFE	Magnetic Fusion Energy
MHD	Magneto-Hydrodynamics
MLD	Master Logic Diagram
NDS	Normal Detritiation System
NPP	Nuclear Power Plant
NTM	Neoclassical Tearing Modes
OH	Ohmic Heating
PAR	Passive Autocatalytic Recombiner
PFC	Plasma Facing Component
PHTS	Primary Heat Transport System
PIE	Postulated Initiating Event
PPCS	Power Plant Conceptual Study
PSA	Probabilistic Safety Analysis
PSS	Pressure Suppression System
RAFM	Reduced Activation Ferritic Martensitic
SCLL	Single Coolant, Lithium Lead
SDS	Stand-by Detritiation System
SEAFP	Safety and Environmental Assessments of Fusion Power
SEAL	Safety and Environmental Assessment of Fusion Power — Long Term Programme
SEIF	Safety and Environmental Impact of Fusion
SIC	Safety Importance Class
SR	Safety Relevant
SS	Stainless Steel
SSC	Systems, Structures and Components
TBM	Test Blanket Modules
TBR	Tritium Breeding Ratio
TCHS	Tokamak Cooling Helium System

TCWS	Tokamak Cooling Water System
TES	Tritium Extraction System
VV	Vacuum Vessel
VVPSS	VV Pressure Suppression System
W	Tungsten
WCLL	Water Cooled Lithium Lead
WDS	Water Detritiation System
WENRA	Western European Nuclear Regulators Association

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