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G. Caruso(a), S. Ciattaglia(b), B. Colling(c), L. Di Pace(d), D.N. Dongiovanni(e), M. D'Onorio(a), M. Garcia(f), X. Z. Jin(g), J. Johnston(c), D. Leichtle(g), T. Pinna(e), M. T. Porfiri(e), W. Raskob(g), N. Taylor(c), N. Terranova(e), R Vale(c), all contributors to the WPSAE

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DEMO – Main Achievements in the Generic Site Safety Report

G. Caruso[(a)], S. Ciattaglia[(b)], B. Colling[(c)], L. Di Pace[(d)], D.N. Dongiovanni[(e)], M. D'Onorio[(a)], M. Garcia[(f)], X. Z. Jin[(g)], J. Johnston[(c)], D. Leichtle[(g)], T. Pinna[(e)], M. T. Porfiri[(e)], W. Raskob[(g)], N. Taylor[(c)], N. Terranova[(e)], R Vale[(c)] and all contributors to the WPSAE

a) Sapienza University of Rome, Dep. of Astronautical Electrical and Energy Engineering (DIAEE), C.so Vittorio Emanuele II 244, Rome, Italy
b) EUROfusion Consortium, PPPT Department, Boltzmannstrasse 2, Garching, Germany
c) CCFE, Culham Science Centre, Abingdon OX14 3EB, United Kingdom
d) CSM-RINA, Via di Castel Romano, 100, 00128 Roma, Italy
e) ENEA, Fusion Technical Unit, Nuclear Technologies Laboratory, Via Enrico Fermi 45, 00044 Frascati, Rome, Italy
f) UNED, E.T.S. Ingenieros Industriales. Dept. Ingeniería Energética, Calle Juan Del Rosal 12, Madrid, Spain
g) Karlsruhe Institute of Technology (KIT), Herrmann-von-Helmholtz-Platz 1, 76344 Eggenstein-Leopoldshafen, Germany

The Safety and Environmental Work Package (WPSAE) has the scope to progress the safety studies for the future EU DEMO reactor, in the frame of the Eurofusion programme. A Generic Site Safety Report (GSSR) has been designed to include all the steps necessary to cover the safety issues of the nuclear fusion plant, from the definition of principles and requirements, through the detection of the source terms at risk, selecting the postulated initiating events, analyzing the accidents, quantifying the doses to the population and investigating waste production and its management. Eleven GSSR volumes collate the studies performed. The final goal is to prepare safety documentation, as complete as possible, to initiate a preliminary safety report when the plant site is selected.
In parallel, the safety studies are supporting the design of DEMO, providing a feedback on the technical choices for the machine, the selection of materials, the use of space, the equipment necessary to correctly manage the safety risks by means of a continuous collaboration with the design teams of other DEMO WPs.
The main achievements of the pre-conceptual phase of the DEMO program FP8 are presented in this paper, by summarizing the contents of GSSR volumes. Completion of the work is foreseen during FP9.

Keywords: DEMO safety, Source terms, ACP, DBA, BDBA, MELCOR, FFMEA, ORE, Activation.

1 DEMO Safety Approach

Safety is one of the main challenges for DEMO in the development of the reactor design and operation, being the first of a kind fusion machine. The ITER experience in dealing with the particular safety issues of a fusion machine provides acquired knowledge supporting DEMO. However, for DEMO additional critical aspects have to be considered, related to the safety of the tritium breeder units, the Balance of Plant (BoP) and the supply of pulsed energy to the electricity grid, and radioactive waste issues.
A DEMO safety approach, outlined in Figure 1-1, has been developed to explore and detail the phases of the process that accompany the project from the early design phase to the preliminary safety analysis report (PSAR), necessary to require the construction license for the reactor. An intermediate step in the safety studies is the compilation of GSSR that collates all the documentation necessary to prove that the plant complies with the safety requirements of a nuclear fusion facility, independently from the location in which it will be built.
Currently EU DEMO is dealing with the activities foreseen to finalize the GSSR. The safety approach originates from the definition and quantification of the key radiological source terms (RST) such as tritium, dust, activation corrosion products (ACPs) (see § 4) that represent the major risks if released. By means of a functional failure modes and effects analysis (FFMEA) the postulated initiating events (PIEs) are selected and analysed (see § 5) to evaluate when, where and how much structures, systems and components (SSC) could be impaired during abnormal transients. The outcomes of the accident analyses are the maximum doses to the workers and to the population that have to be minimized and maintained below the plant safety limits (see §2). The impact of accidents on the design needs to be developed following the safety requirements, established in accordance with the standards and best practice of the nuclear fusion plants. The calculated accident consequences supply the input for the definition of the safety important class (SIC) of the SSCs of the plant (see §10). On that basis the safety operational limits (source terms inventories, fusion power, pressure, temperature, magnetic field, etc.) can be fixed to avoid abnormal transients that can result in unauthorized radioactive releases.

author's email: mariateresa.porfiri@enea.it

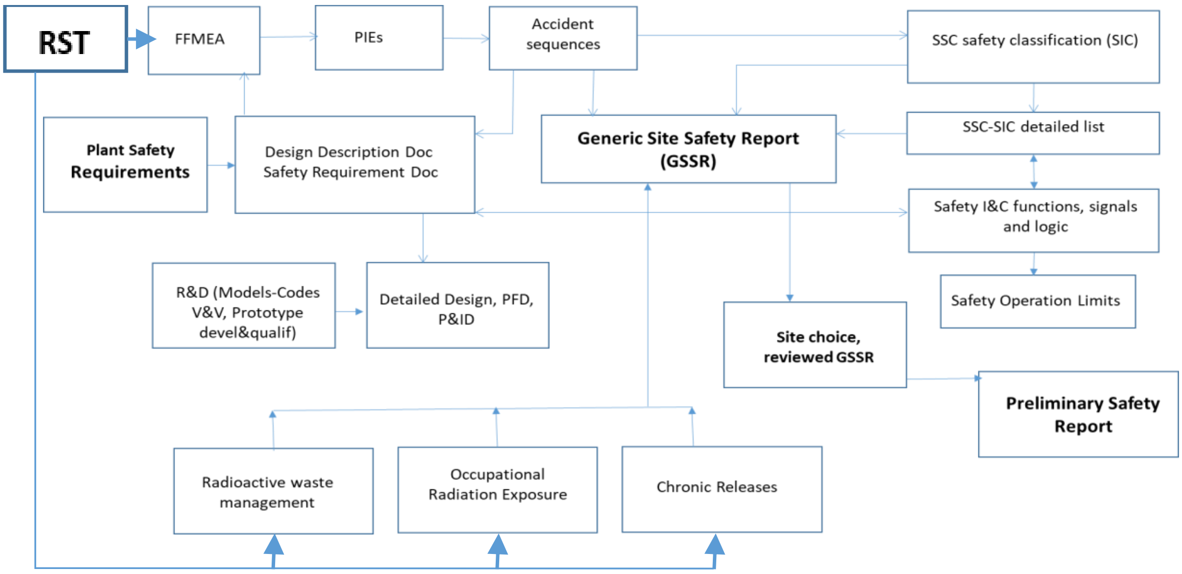

Figure 1-1
DEMO safety approach – Main paths

The RST are also essential input for the occupational dose (see § 8), for the waste management (see § 7) and the chronic releases (see § 6). Their behaviour has to be verified not only during accidental transients, but also in normal operation and during maintenance.
Continuous feedback from the safety analyses to the design is required to ensure that all necessary provisions are included to prevent or/and mitigate an accident and to confine the radiological inventory.
An ALARA (As Low As Reasonable Achievable) process is required to lower the radiological risks whilst maintaining the ability of the plant to be safe and economically acceptable at the same time.
The GSSR deals with all the issues described above.
The evolution of the GSSR will be a PSAR, when a site for the construction is selected. With the PSAR the licensing procedure of DEMO reactor will have its beginning.

2 Safety requirements, safety functions

2.1 Safety Functions

A Safety Function is a specific purpose that must be accomplished for safety for a facility or activity to prevent or to mitigate radiological consequences of normal operation, anticipated operational occurrences and accident conditions [2-1]. A set of four Fundamental Safety Functions have been defined to meet the top-level safety objectives:

- Confinement of radioactive and hazardous materials;
- Limitation of exposure to ionizing and electromagnetic radiation;
- Limitation of the non-radiological consequences of conventional hazards;
- Limitation of Environmental Legacy.

The last bullet point refers to the limitation of environmental releases, the effective management of waste and of plant decommissioning at the end of life.

These Fundamental Safety Functions apply under all normal and accident conditions and for the full lifetime of the facility. To ensure their achievement, a number of supporting functions are identified:

- Control of plasma energy (e.g. ensure safe plasma shutdown if/when needed);
- Control of thermal energy (e.g. ensure decay heat removal to prevent damage to first confinement barriers);
- Control of confinement pressure;
- Control of chemical energy;
- Control of magnetic energy;
- Control of electrical energy;
- Control of coolant energy;
- Limitation of radiation and toxic material exposure to workers;
- Limitation of airborne and liquid operating releases to the environment;
- Limitation of electromagnetic field exposure to workers;
- Limitation of other industrial hazards;
- Limitation of waste volume and hazard level;
- Facilitation of clean-up and removal of components.

Each of these functions is achieved by the incorporation in the design of specific features to provide the protection or take the actions that are needed. This leads to specific requirements on those SSCs that perform these functions. Any SSC that performs a safety function is classified as important to safety (see § 10).

2.2 Confinement strategy

The confinement of radioactive and hazardous material is the most important of the fundamental safety functions. It is achieved by implementing in the design a succession of physical barriers. Multiple provisions are made so that, in accordance with a defence-in-depth approach, the failure of one barrier does not result in a release to the environment or to rooms in which personnel could be exposed. Ventilation systems maintain a pressure cascade between rooms so that air flow is always towards the more contaminated volume.

Provision of the confinement function by passive barriers is preferred over active systems, and the number of barriers, their leak tightness, and reliability will be specified to achieve the required performance for the safety function. Where active components such as isolation valves are required, multiple components may be required to achieve the required overall reliability.

Protecting confinement barriers is the purpose of many of the supporting functions listed in section 2.1. For example, for the in-vessel inventory of retained tritium and activated erosion dust, the first confinement barrier is the vacuum vessel (VV) itself, including its many extensions. Thus, any over-pressurization of the vessel, e.g. in an accidental leak of coolant, must be avoided by the provision of a pressure relief system.

2.3 Requirements for inventory controls

There is a requirement to define a tritium inventory limit for every volume in which a significant inventory is liable to arise. Tritium inventories will be optimised to be ALARA. A maximum inventory of 1.0 kg tritium within the VV at any time is set as a design target. Separate tritium inventories will also be defined for the breeding blankets (BBs). A maximum activated dust inventory of 1000 kg within the VV at any time is set as a design target.

For in-vessel inventories of both tritium and dust, the targets will be refined by further analysis as the design matures. Appropriate monitoring and control strategies will be required to demonstrate compliance with an appropriate safety margin within both limits.

Within rooms, the airborne tritium inventory limit will be minimised and controlled in line with access requirements and a ventilation zoning system. In potentially contaminated areas where routine personnel access is permitted without respiratory protection, atmospheric concentrations of tritium will be controlled to below 1 Derived Air Concentration (DAC) to ensure compliance with the occupational effective dose limit. Ventilation zoning will be applied, defining confinement classes, in accordance with the ISO17873 standard [2-2], and with associated depression values.

Maximum leak rates of 1 volume % per day (VV, extension and cryostat) and 100 volume % per day (rooms of the Tokamak building) are set as design constraints for the confinement barrier with respect to the pressure difference between adjacent regions. Maximum leak rates and minimum efficiency targets will also be specified for components that are identified as SIC for confinement, e.g. building walls, penetrations and barriers. A detritiation efficiency of greater than 99% is set as a design constraint for the detritiation systems and a filter efficiency of greater than 99.9% for HEPA filters.

2.4 Requirements in Normal Operation

2.4.1 Occupational doses and ALARA

Radiological doses to plant personnel will be maintained ALARA. This includes doses incurred during planned and unplanned maintenance activities, which must be optimised to minimize exposure of workers to direct radiation or to internal exposure. Where necessary remote handling will be implemented to reduce human exposure to radiological doses.

Individual personnel doses will be ALARA and in any case subject to an effective dose limit of 100 mSv averaged over 5 years, with a maximum of 50 mSv in one year. A design objective will be to maintain all individual effective doses below 5 mSv per year. Furthermore, a collective dose target for the entire facility of 700 person-mSv/year is adopted.

2.4.2 Radiological Zoning

To assist in achieving the occupational dose targets and limits noted above, a radiological zoning scheme will be employed which, together with access control measures, will control dose exposures during maintenance activities. The adopted zoning scheme is presented in Table 2-1.

Table 2-1
Radiological zoning scheme adopted for DEMO

<table>
<tr><th colspan="2">Zone type</th><th>Zone identification</th><th>Maximum total effective dose (external plus internal)</th><th>Maximum external dose to hands, forearms, ankles and feet</th></tr>
<tr><td colspan="2">Unregulated</td><td>White</td><td>80 µSv/month</td><td></td></tr>
<tr><td colspan="2">Supervised</td><td>Blue</td><td>7.5 µSv/hr</td><td>200 µSv/hr</td></tr>
<tr><td rowspan="4">Controlled</td><td>Limited</td><td>Green</td><td>25 µSv/hr</td><td>650 µSv/hr</td></tr>
<tr><td>Specially regulated</td><td>Yellow</td><td>2 mSv/hr</td><td>50 mSv/hr</td></tr>
<tr><td rowspan="2">Forbidden without specific authorization</td><td>Orange</td><td>100 mSv/hr</td><td>2.5 Sv/hr</td></tr>
<tr><td>Red</td><td>above 100 mSv/hr</td><td>above 2.5 Sv//hr</td></tr>
</table>

2.4.3 *Environmental releases*

Gaseous and liquid releases of radioactive and other hazardous materials to the environment during normal operation and maintenance will be minimized to be ALARA. Radioactive effluents will be limited so that the effective dose to the most exposed individual (MEI) outside the site boundary does not exceed 1 mSv/year. As a design objective a target of 100 μSv/year is adopted with a constraint of 300 μSv/year as an upper value for the off-site dose.

The requirement to minimize environmental releases will be achieved by minimizing inventories in the plant design, particularly of tritium, providing robust confinement systems and restricting possible pathways to the environment. All atmospheric releases will be controlled, monitored and directed via a ventilation system that employs detritiation and filtering to minimize vented releases.

2.5 Abnormalities and accidents

Design basis accidents (DBAs) are a set of postulated accident scenarios considered in the design. They include the failures of SSCs that may initiate an accident sequence. For DBAs, the radiological consequences for the public must be minor. Beyond design basis accidents (BDBAs) are hypothetical bounding event sequences. These enable demonstration of the ultimate safety margin of the design. Their analysis must demonstrate the absence of “cliff-edge” effects, i.e. to show that as an event sequence is made successively more unlikely (e.g. by postulating more independent failures), a point is not reached where there is a sudden increase in the event consequences.

DBAs and BDBAs are categorized according to their expected frequency, or likelihood. This can be based on engineering judgement of the perceived likelihood of their occurrence unless reliability data is available and systematic methods can be used.

A top level safety objective is to limit the hazards from off-normal events such that, in any event, there is no need for public evacuation on technical grounds. This leads to maximum off-site dose limit based on the IAEA recommendation that evacuation should be carried out if a dose of 100 mSv or more in 7 days of exposure may be averted [2-3]. To provide a margin, 50 mSv is adopted as the limit on the early dose to the MEI in any event, no matter how unlikely.

Off-site Consequence Limits/targets for accidents DBAs, including postulated multiple failure events, and BDBAs are presented in Table 2-2.

Table 2-2
Proposed DEMO Off-Site Consequence Limits/Targets for Off-Normal Events

	Anticipated events[1]	**Unlikely events**	**Extremely unlikely events**	**Hypothetical bounding events**
Event Category	1-2	3	4	BDBE
Event Frequency	f>1E-02	1E-02> f>1E-04	1E-04 >f>1E-06	f<1E-06
Early Dose			10mSv/event	50mSv/event
Chronic Dose	Treat as normal operations	5mSv/event	50mSv/event	

[1] Category 1 refers to Operational events and Category 2 refers to likely event sequences

In order to ensure accidental doses are kept below the annual dose limit for normal operation, an on-site individual effective dose of 20 mSv/event is adopted as a Design Objective for events expected to occur much less frequently than once per year.

2.6 Limitation of Environmental Legacy

2.6.1 *Minimization of activation and volumes of waste*

The volume of material that becomes radioactive due to neutron activation or contamination with tritium will be minimized in the design, e.g. by restricting the number of replacements of activated components. The level of activation of such material will also be minimized, by careful choice of materials compositions and provision of adequate neutron shielding. Particular attention is to be paid to restricting the levels of impurities in materials that may transmute to long-lived activation products.

2.6.2 *Waste classification*

Radioactive material will be classified into one of three classes of waste [2-4]:
1. Non Active Waste (NAW) for material which can be cleared from regulatory control i.e. classed as non-active, using the criteria for clearance recommended by IAEA [2-3];
2. Low Level Waste (LLW) is above clearance levels but has limited amounts of long-lived radionuclides. The waste can be buried near to the surface (up to 30m depth). This class includes Very Low Level Waste (VLLW);
3. Intermediate Level Waste (ILW) that contains a greater quantity of long-lived radionuclides or alpha activity. The waste is required to be buried at greater depths than near surface.
No High Level Waste is expected to be produced.

2.6.3 *Waste management*

In addition to the clearance of material as non-active waste, efforts will be made to recycle the maximum quantity of material. Material in the LLW and ILW classes will be further categorized as Potentially Recyclable were possible. Materials choices at the design stage will be made to maximize this recycling potential.

Material that will eventually require disposal in a radioactive waste repository will be processed, treated and packaged to meet the requirements of the host country regulations and the specific requirements of the destination waste repository. Storage for an interim period to allow the decay of short-lived nuclides may be required as part of the process. Tritium that has permeated into the bulk of waste materials will be recovered by appropriate detritiation processes to the maximum extent possible and returned to the fuel cycle.

2.7 Non-radiological risks

There are a number of conventional operating hazards such as those associated with the large scale use of High Voltage (HV) electrical systems and cryogenic systems. In addition to these there are non-radiological hazards associated with flammable gases, laser diagnostics, toxic materials such as beryllium and mercury, and magnetic fields and radio frequencies. Such hazards may be subject to regulations of the host country, and international standards are also available for guidance.

Electrical, laser and cryogen hazards are immediate and are protected against using engineered preventative systems. For toxic materials, and magnetic and radiofrequency (RF) hazards where exposure thresholds apply, the hazard can be limited by controlling exposures. For RF fields, within design requirements, the exposure hazard can be limited by design.

Regulatory occupational exposure limits will be applied for beryllium and mercury. Evaluation of the potential for off-site releases will be carried out but it is not anticipated that these will be significant. Regulatory limits will also apply to exposure to magnetic fields, and an occupational maximum exposure of 200 mT over an 8-hour non-stop working day has been set as a design objective. Until the design has matured, it is not considered appropriate to set further limits, e.g. for maintenance where residual magnetic field may have an impact. For exposure to RF fields, a limit of 5 $mW.cm^{-2}$ is set for ECRH waves and 1 $mW.cm^{-2}$ for ICRH plane waves, expressed as equivalent power density.

Component replacement activities may introduce potential dropped load and collision hazards which can be managed with regulatory lifting requirements.

References Chapter 2

[2-1] IAEA Safety Glossary, Terminology Used in Nuclear Safety and Radiation Protection 2018 Edition, IAEA, Vienna, 2019.

[2-2] ISO 17873:2004 Nuclear facilities — Criteria for the design and operation of ventilation systems for nuclear installations other than nuclear reactors.

[2-3] IAEA GSR Part 3 Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards General Safety Requirements, IAEA, Vienna, 2014.

[2-4] IAEA. (2009). Classification of Radioactive Waste. IAEA. Vienna: IAEA

3 Activation calculations, decay heat and shutdown dose rate

Within the strategic approach for neutronics in the European fusion programme, [3-1], neutronics support for design integration of safety aspects plays a key role. This effort is related primarily to the limitation of exposure to ionizing radiation. This link is most obvious in the implementation of the ALARA principle for radiological protection, where nuclear analysis and shielding/maintenance design are strongly correlated.

During operation of the DEMO tokamak, all SSCs are subjected to certain levels of neutron irradiation which lead to transmutation and activation of materials and the generation of radioactive nuclides. Their subsequent decay gives rise to radiological hazards, which need to be controlled and minimized. Activation and coupled radiation transport-activation analyses aim to provide all relevant radiological quantities, such as specific and total activity, decay heat, contact dose rate and shutdown dose rates (SDDR) covering time scales from immediately after shutdown of plasma operation, periods of various maintenance activities in the timeframe of days and weeks, treatment in hot cell or radwaste facilities, up to decommissioning and long-term storage and disposal.

After cessation of plasma operation various interventions are planned or considered for the DEMO tokamak maintenance and inspection needs. In all cases, the configurations are quite different to the plasma operation with respect to the radiation transport description. This involves drainage of coolants, opening of bioshield plugs, cutting or removal of in-vessel components (IVCs) and other equipment and preparation of appropriate work station protections. This leads to changes in the activation source term distributions as well as in the resulting radiation fields. Global and generic shutdown dose rate maps have been produced as a guideline as to the principal issues of harsh radiation environments after relevant operation and cooling times.

3.1 Methodological approach

To calculate reliable nuclear parameters for activation inventory, decay heat and shutdown dose rate, the computational approach encompasses static and dynamic methodologies. The static approach addresses the activation of permanent structures under neutron irradiation whereas the dynamic approach simulates the cycling of fluids (coolants and liquid breeder material) passing through spatially varying irradiation zones.

The static approach is based on particle transport simulation using the MCNP code to provide neutron spectra throughout the relevant locations. Geometry and material data is obtained from current DEMO baseline with up-to-date design information. Up-to-date breeder blanket design models with a high level of heterogenous representations and detailed material specifications with impurities are adopted. Inventory calculations using FISPACT-II and ACAB codes adopt those neutron spectra to irradiate respective material compositions and to compute transmutation and activation inventories, which in turn provide decay heat and contact dose rate estimates. Additionally, the resulting decay photon sources can be transported to obtain distributions of SDDR around the activated components. Coupled tools for this purpose (e.g. Common-R2S and Advanced-D1S-dynamic) have been developed for standard use in DEMO neutronics.

Dynamic modelling for PbLi and water loops have been introduced within the codes ACABLoop, [8-2], ActiFlow and GammaFlow, [3-2], to simulate various relevant effects. This includes the passage through blankets with various irradiation fluxes, tritium extraction and fluid purification, as well as the accumulation of ACPs during cycling operation.

3.2 Activation of tokamak systems

Due to the high level of neutron fluence within the plasma chamber the activation inventories of IVC (breeder blanket and divertor) and of the VV (mainly the inner shell) deserve special attention as to the feasibility of maintenance, repair and recycling as well as considerations on the radiological waste classification and storage. The starter blankets will be replaced after the first phase of operation, i.e. 5.2 years with 1.57 FPY of plasma operation. This holds also for the first set of divertor cassettes, which has been used as the envelope condition of all following sets.

Activation analyses, see e.g. [7-12] have been performed accordingly on IVC and VV for the provision of safety relevant parameters for full system and individual components/materials and for cooling times up to 100 years after shutdown. Those are fed into the DEMO Safety Data List (SDL) for consistent use in safety analyses.

Decay heat in the Helium Cooled Pebble Bed (HCPB) blanket amounts to 18.4 MW at 1 s after shutdown (ca. 1% of operational heat) and decreases to 0.18 MW after 1 month cooling. The decay heat generation in the Eurofer steel parts dominates the total results for the blanket. Up to 1 hour after the shutdown the heat due to the decay of ^{56}Mn is the most significant contribution. The decays of ^{182}Ta and ^{187}W coming from the irradiated W armour layer dominate the total results up to 1 year after the shutdown.

In the Water-Cooled Lithium Lead (WCLL) blanket the total decay heat at 1 s after shutdown is ca. 24 MW (ca. 2% of operational heat), of which 10 MW is due to the activated PbLi. After 1 month it reduces to 0.7 MW. The dynamic modelling of PbLi activation does not introduce significant effects at very short cooling times (up to tens of seconds), but up to 50% reduction at longer times. PbLi decay heat within the blanket is significant again at very short cooling times; the bulk is provided by tungsten and Eurofer, which dominates after several days.

Divertor activation responses have been obtained in both HCPB and WCLL blanket environment, with higher values for WCLL (by ca. 10-20% except at longest cooling times). At shutdown the respective decay heats (Table 3-1) of all 48 cassettes are 2.6 MW and 3.0 MW for HCPB and WCLL, respectively; at 1 month cooling the values are 0.1 and 0.11 MW. Similarly, decay heat and activity of the VV behind HCPB and WCLL blankets reflects the shielding characteristics of the breeder blankets. At shutdown the VV in HCPB DEMO has a total decay heat of 1.2 MW; in the WCLL case it is one order of magnitude lower at 96 kW.

Table 3-1
Total decay heat [MW] per component after cooling time of 1 s to 1 y.

Component decay heat [MW]	1 s	1 h	1 w	1 m	1 y
HCPB Breeding Blanket	18.4	10.1	0.83	0.18	6.6E-03
WCLL Breeding Blanket	24.1	10.2	0.91	0.73	0.18
Divertor (in HCPB DEMO)	2.57	1.80	0.14	0.10	0.023
Divertor (in WCLL DEMO)	3.01	2.17	0.17	0.11	0.025
VV (in HCPB DEMO)	1.20	0.91	0.059	0.052	0.035
VV (in WCLL DEMO)	0.095	0.071	5.6E-03	4.9E-03	2.8E-03

3.3 Shutdown dose rate mapping

Radiation levels of ionizing radiation during maintenance periods are essential input to Occupational Radiation Exposure (ORE) assessments. As conceptual designs have still to be developed, the respective maintenance design requirements and corresponding detailed maintenance plans are still under development. To assess the feasibility of hands-on assisted maintenance, radiation mapping for biological SDDR is provided based on current assumptions and design both inside the bioshield as well as at important locations within the nuclear building for access times of typically 1 or 12 days after shutdown.

As estimates for SDDR (Figure 3-1) inside the cryostat are typically 1 to 3 orders of magnitude above the stipulated target for frequent access of 500 µSv/h, recent scoping efforts looked at thickening VV and its in-wall shield, filling of inter-coil shield blocks and prototypical port system shielding studies. The results obtained indicate a promising reduction of 1 to 2 orders of magnitude particularly in port interspace areas, which are candidate maintenance corridors.

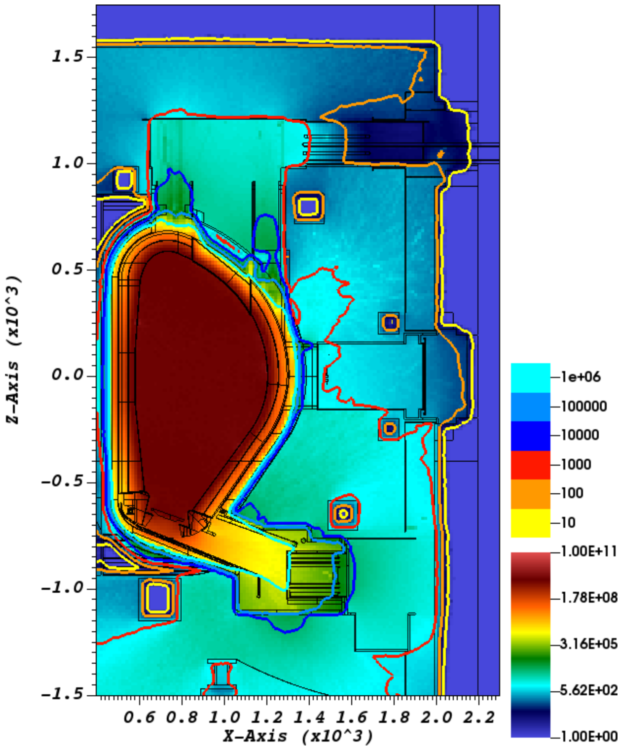

Figure 3-1
Provisional biological SDDR [µSv/h] in HCPB DEMO after end of life (EOL) operation and 12 days cooling with prospective shielding improvements.

SDDR mapping inside the nuclear building has been assessed for ACP in water and activated PbLi as well as activation due to ^{17}N-decay around water pipework. ACP in water loops can contribute very significantly to dose fields. Typically, the target of 10 µSv/h after 1 day cooling is not fulfilled inside rooms with active loop components. In the BB primary heat transfer system (PHTS) area, due to further decay and improved shielding, some free volumes are better protected. Excessive dose rates above 100 mSv/h emerge from activated PbLi (if not drained and completely cleaned), which does not allow for workers' access. Protection by concrete slabs to neighbouring rooms is sufficient for access, however, optimization by using heavy concrete is favorable. Activation of primarily pipe steel by ^{17}N-neutrons is a significant SDDR source, which generates fields of ca. 1 mSv/h immediate to the pipework. Accordingly, the target of 10 µSv/h is exceeded in large volumes of building rooms containing weakly shielded pipework.

Further work is ongoing on various aspects of the dose rate assessments as the design progresses. Current results and assessments are used to steer efforts on configuration layout and shielding as well as on refinement and reduced margins of the relevant source terms.

References Chapter 3

[3-1] D. Leichtle, U. Fischer & C. Bachmann (2021): Strategical Approach for the Neutronics in the European Fusion Programme, Fusion Science and Technology, DOI: 10.1080/15361055.2021.1887716.

[3-2] C.R. Nobs, et al., Computational evaluation of N-16 measurements for a 14 MeV neutron irradiation of an ITER first wall component with water circuit, Fusion Engineering and Design, 159, 2020.

4 Inventories of radioactive source terms and energies

4.1 Radiological source terms

Radioactive materials within DEMO include tritium and neutron activated products. The localisation of radiological inventories is important for understanding the occupational hazard and potential releases during normal operation, maintenance, and accident scenarios. According to the current DEMO plant layout the tokamak building, the tritium building, the active maintenance facility (AMF) and the radioactive waste building will be the most significant for potential radiological releases. The current focus is on the tokamak building inventories due to lack of design maturity and available data for the other buildings.

4.1.1 Tritium

As an isotope of hydrogen, tritium is the most mobile of the key radioactive sources that will be present in DEMO and requires special handling and confinement procedures to minimise release. Tritium inventories in DEMO will result from:

- Unspent tritium fuel retained within the VV.
- Tritium bred in lithium and beryllium containing materials within the VV by nuclear transmutation reactions.
- Tritium will be present throughout the fuel cycle systems, the AMF and waste treatment systems.

4.1.2 Activated products

Most of the activation products are tightly bound to metal structures and considered immobile. Smaller inventories will be found in dust, structures outside the VV or circulating in coolant streams. Some activated products may become mobilised through mechanical or chemical mechanisms. Activated material will contribute to both the activated product source term and the stored energy source term due to the associated decay heat. Activated material mainly contributes to occupational safety issues; only a small fraction has potential for environmental release, even in accident scenarios.

4.1.3 Tokamak building source terms

The key inventories within the tokamak building are:

- Dust produced within the VV.
- Unspent tritium fuel retained within the VV.
- Tritium in the breeder blankets.
- Activated products in cooling systems.

Dust quantities within the VV will build up over the plant operation. Dust, described as particles smaller than 100 µm in diameter, could be produced through a number of mechanisms, both in routine operation and accident scenarios. During routine operation there will be dust produced through the 'normal' erosion of the plasma facing materials and as a result of abnormal plasma events. Dust could also be produced through plasma events that may be considered accident scenarios, such as disruptions. Additionally, dust particles can arise during maintenance activities due to mechanical abrasion on component replacement or due to the presence of air. Dust inventories may need to be controlled through dust cleaning mechanisms – though some dust will also be removed during divertor replacement as it is trapped within the components.

Recent safety analysis (see §5.2) uses a conservative dust inventory of 1034 kg based on scaling from ITER limits [4-1] and with the same ITER pessimistic assumption [4-2] that all the 'dust' can be mobilized. However, ongoing source term assessments show lower dust estimations taking into consideration:

- the different pathways to dust production, including the important erosion to dust conversion ratio – the majority of eroded particles form deposit [4-3],
- the tungsten first wall (FW) material is expected to have lower erosion rates than beryllium [4-3], [4-4],
- in DEMO there should be no unmitigated major plasma disruptions during routine operation.

Preliminary investigations into the dust produced from normal erosion pathways have shown the amount of impurities within the tungsten to be an important factor with high-Z impurities (Ne, Ar, Kr, Xe) dominating the gross erosion. Future work will look to refine the dust source term based on updated erosion - deposition simulations for a full-size DEMO geometry.

Unspent tritium fuel within the components of the VV will also build up over the plant operation phases due to implantation/diffusion into the plasma facing materials, dusts and deposits.

Much of the tritium will become trapped within the materials of the plasma facing components (PFCs), but some could be mobilised in the event of a loss of vacuum accident or opening of vacuum for maintenance. The tritium could be mobilised through contaminated dust or degassing from the plasma facing materials and/or deposit. A conservative estimate, derived through scaling from ITER limits, of 2673 g in-vessel tritium inventory has been used in recent safety analysis (see §5.2). However, source term assessments considering different methodology show lower values when considering both tritium retention in tungsten and tritium co-deposition (which is expected to be lower in tungsten [4-4], [4-5]). To refine the mobilizable tritium inventory, future works will look to assess the fraction of tritium that could be mobilised from the plasma facing materials in an accident scenario considering the temperatures within the DEMO vacuum vessel.

As mentioned in §2.3, a safety target of 1 kg tritium within the VV is achievable for DEMO, potentially controlled through detritiation measures such as baking (although normal operating temperature in DEMO is higher than the baking temperature in ITER); additionally, some tritium will be removed from the vessel during blanket/divertor replacement.

Tritium from unspent fuel and breeding in the blanket materials, can permeate into the coolant of the FW and breeder blanket and circulate within the PHTS. It will be important to minimise this inventory to reduce potential releases from permeation and coolant leaks etc. Key inventories of tritium within the blanket are given in Table 4-1. Additionally, taking into account the tritium that permeates from the plasma, several analyses were carried out using TMAP7 [4-6] and TESSIM-X [4-7]. The calculated total tritium permeation rates and retained tritium amounts at the end of life of the second breeder blanket (5 fpy) are reported in Table 4-2. Although these studies are still preliminary, a correct assessment of trapping of tritium within the breeder blanket FW is particularly important. It (i) affects the starting inventory (more tritium is required to saturate the breeder blanket structures at start-up), (ii) poses a serious concern for the confinement of tritium during remote maintenance operations, (iii) makes waste management more complex and (iv) increases the inventory mobilizable within vacuum vessel.

Table 4-1
Tritium breeder blanket inventories for the 2017 DEMO baseline from a sensitivity analysis

Tritium inventory	HCPB		WCLL	
	Min	Max	Min	Max
T in coolant [g]	0.001	0.01	5	55
T in Eurofer structure [g]*	0.003	0.04	3.0	3.5
T in breeder [g]	25	25	30	36
T in purge gas [g]	0.03	0.1	N/A	N/A
T in multiplier [g/FPY]	72	72	N/A	N/A

* Not including the tritium inventories due to permeation from the plasma.

Table 4-2
Tritium permeation from plasma to the breeder blanket coolant and retention in the FW assuming a 2 mm tungsten layer with no cracking

Tritium inventory	HCPB	WCLL
T permeation to coolant [mg day-1]	0 - 300	0
T retention in FW [g]	700	800 - 1350

The current analyses of tritium permeation and retention are based on some assumptions that are not confirmed yet. In particular:

- It is assumed that there are no interface barriers or traps between tungsten and EUROFER layers.
- The tungsten layer is assumed dense with a smooth surface.

Further investigations supported by experiments are, therefore, necessary in the future.

Activated products may find their way into coolants through corrosion; the dominant mechanism in the WCLL concept. Only small fractions of the corrosion products are as ions in solution or as small particles (crud) in suspension, the majority are deposits and fixed oxides on surfaces of the pipes. Currently 10 kg of ACPs per cooling loop have been assumed in safety analysis as a conservative value for deposit mass, and 20 g in ions and cruds [4-8]. In accident scenarios, such as a loss of cooling accident (LOCA), the mobilizable quantity of ACPs is to be reassessed to consider that only a portion of the total ACP deposit inventory (plus ion and crud) will be mobilised by the accident flow rate, i.e. that which is in proximity to the rupture. ACP inventories located on surfaces far from the pipe rupture should not be considered for mobilisation.

4.1.4 Other buildings

There will be radiological inventories in other buildings and systems, such as the tritium building, the AMF and the radioactive waste building – however, the design of these is less mature. The design of the fuel cycle architecture will have significant impact on the inventories of tritium in the fuel cycle and tritium building.
The tritium fuel and vacuum (TFV) systems are mainly located in the tritium building with some connecting piping and smaller equipment, such as pellet injector, within the tokamak building. The layout of the tritium building and the tokamak building interface is not yet maturely defined. Additionally, there will be tritium storage and buffer tanks within the tritium building to consider. Estimated operation tritium inventory for the major fuel cycle systems is shown in Table 4-3.
The removal of IVCs from the VV will move activated products and tritium inventory to the AMF and the radioactive waste building. The radioactive waste management will aim to minimise radioactive waste through component reduction/dismantling and detritiation of components after removal from the vessel. Source terms relating to some of the proposed radioactive waste processing methods form part of future work.

4.2 Stored Energy

DEMO will contain a number of energy sources that could potentially drive a release of radioactivity or hazardous material in the event of an accident. The main energy sources are given in Table 4-4. Safety issues related to the stored energy sources for a DEMO plant are discussed in [4-4], with identified key stored energy safety issues: decay heat in-vessel loss of coolant accident (LOCA), loss of plasma control, magnet stored energy, and potential for hydrogen explosion. The chemical reaction energy sources between water/air and tungsten are based on the equations of [4-9] – [4-10].

Table 4-3
Operational tritium inventory for major fuel cycle systems

Fuel cycle system	**Tritium inventory (g)**
Matter injection	12
Vacuum pumping	75
Exhaust purification	10
Isotope rebalancing	100
Exhaust detritiation	60
Water detritiation	190
Isotope separation	360

Table 4-4
Energy source terms

Energy source	Amount of energy	
Plasma power	1998 MW	
Plasma thermal energy	1.3 GJ	
Magnetic (toroidal field coils)	120 – 147 GJ	
Magnetic (poloidal field coils)	20 GJ	
Decay heat	~1% of total nuclear power generation	
Thermal energy (GJ)	HCPB	WCLL
Divertor	30	30
First wall	*	154
Breeder blanket	3	545
Vacuum vessel	62	62

*Separate cooling circuits in WCLL for first wall and breeder zone, whereas presented as single breeder blanket loop for HCPB.

References Chapter 4

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5 Dominant accident sequences and environmental releases

5.1 Introduction

Accident analysis needs to be performed for each reference PIE to evaluate safety consequences in compliance with safety limits, and to support the selection of the reference DEMO concept. Both DBAs and BDBAs have been investigated for DEMO (see §2).

Based on the FFMEA 21 most representative PIEs have been identified for DEMO in 0. The FFMEA is a suitable methodology to define possible accident initiators when insufficient design detail is available to allow for more specific evaluation at component level. The FFMEA has been performed for the main systems of DEMO such as HCPB BB system, WCLL BB, divertor (DIV), PHTSs, BOP, Coolant Purification System (CPS), tritium extraction system, fuel cycle, magnet system, VV system, thermal shields, cryostat system, etc. For each reference PIE deterministic assessment is required.

During the pre-conceptual design phase, a series of the representative events have been investigated with respect to:

a) the different IVCs HCPB BB, WCLL BB and divertor, i.e. LOCAs in the VV (in-VV LOCA),

b) outside the VV (ex-vessel LOCA),

c) in the BB (in-BB LOCA) due to failure of related channels / pipes,

d) Loss of Flow (LOFA) due to pump / blower trip,

e) loss of heat sink (LHS) due to loss of condenser vacuum, etc.

The accident analysis for each event has been performed and documented in the following steps:

1. identification of causes, accident description, and assumptions for different scenarios;
2. generation of analysis model with proper computer code;
3. implementation of the initial conditions, assumptions and control methods to the model;

4. simulation of scenarios and evaluation of transient results;
5. analysis of radiological releases;
6. indication of uncertainties in the modelling;
7. recommendations for model improvement and to the designers;
8. summary for different scenarios.

MELCOR 1.8.6 for fusion 0 is the qualified code for the required accident analysis selected for the simulation.

In sections 5.2, 5.3 and 5.4 the most representative events are described for the HCPB concept, the WCLL concept, and the divertor respectively complying with the scopes above. Despite different event initiations and accident sequences, some common definitions and assumptions are listed in the following:

- All reference designs are due to DEMO baseline 2017. The tokamak is divided in 16 sectors and each sector contains 5 blanket segments: 3 outboard (OB) and 2 inboard (IB) segments.
- For the DBA, a fast plasma shutdown (FPS) is activated 3.0 s after the detection of the selected PIE.
- The FPS is followed by a mitigated plasma disruption with 0.75 MJ/m² for 10 ms which affects a certain FW surface area.
- In case of an in-VV LOCA, the fluid ingress into the VV is followed by an unmitigated plasma disruption with 3 MJ/m² for 1 ms in rise and 7 MJ/m² for 3 ms in decay which affects a certain FW surface.
- A loss of off-site power (LOSP) for 32 h is assumed to coincide with the plasma disruption such that all pumps / blowers in the BB-PHTS and DIV-PHTS stop. Only the VV-PHTS is supported by the Decay Heat Removal System (DHRS) to transfer the decay heat removed from all IVCs during emergency conditions.
- For the BDBA, the FPS fails to be triggered so that the plasma burns continuously during the event until the FW temperature reaches 1000 °C (T_{EF1}). It is assumed that at T_{EF1} the structure integrity fails due to EUROFER97 yield strength behavior.
- Structural material of the BB is EUROFER97, for which design temperature is limited at 550 °C (T_{EF}), and melting temperatures are in the range of 1325 – 1530 °C (T_{EF2}).
- The functional material of the IVCs is tungsten (W), which is the PFC with a thickness of 2 mm on the FW of the BB.
- The maximum pressure of the VV is limited at 200 kPa (p_{VV}).
- VVPSS-He (VV pressure suppression system) and VVPSS-H2O are designed for the HCPB and WCLL respectively. The pressure set point is at 90 kPa (p_{BL}) for the bleed line (BL), and at 150 kPa (p_{RD}) for the rupture disk (RD).
- For transport of mobilizable source terms MELCOR radionuclide package is applied.

The release data to the environment (tritium, dust, ACP), which had been determined by MELCOR simulations for four events, have been applied as time integrated values for dose calculation. The dose results will be assessed in § 5.5.

5.2 The selected event for the HCPB concept

The reference event in-VV LOCA (DBA, LFV1 in 0) is selected to be described here for the HCPB concept. It is defined as loss of coolant in the PHTS inside the VV due to large rupture of the FW structure during normal operation. Two main cases are considered due to the assumed FW failure sizes and locations. CaseI is for the failure of 30 FW channels with a double break size of 8.64E-03 m² in one OB sector of one of three modelled loops. Three scenarios are considered for CaseI to check the behavior of the VV pressurization: CaseIa with the wet expansion volume (EV) of the VVPSS-He as reference scenario, CaseIb with 30% enlarged wet EV (3900 m³), and CaseIc with the wet EV connecting a dry EV. The wet EV enhances heat exchange between water and He to keep the volume temperature at a low level. To estimate the maximum possible VV pressurization in worse case, the FW failure in all OBs is assumed in CaseII such that four channels with the break size of 1.152E-03 m² fail in each of 16 OB sectors in all eight loops. Both wet and dry EVs are needed for pressure suppression. Transport of dust and tritium for the HCPB is studied in CaseIa and CaseII with the proposed inventories detailed in the DEMO SDL, an internal project document. It is a living data base for the information necessary in the accident analyses, continuously updated. The total W-dust inventory in the VV is 1034 kg, and 5 kg dust due to the plasma disruption. The averaged tritium inventory in the VV is 2673 g. Tritium mass in the BB coolant is 4.18E-02 g and 5.83E-02 g in the BB-PHTS. The mobilization fraction is 1.0. All scenarios are simulated for 33 h (t_{end}), one hour after the LOSP.

The reference designs are the HCPB2018 0, the associated PHTS 0 , and the VVPSS-He including wet and dry EVs. The PHTS consists of eight independent loops where each loop serves two sectors. The averaged mass flow rate of each loop is 222.2 kg/s. The BB inlet and outlet temperatures are 300 °C and 520 °C respectively, and the inlet pressure is 8 MPa. In the VVPSS, six BLs and three RDs are installed in the pipe connection between the VV and the wet EV with a cross section of 0.05 m² for the BL and 1.0 m² for the RD. The pressure is 4.5 kPa for the VVPSS-He and 98 kPa for the Tokamak Cooling Room (TCR), while their temperature is 30 °C. The free volume of the TCR is 6.07E+04 m³. A delay time of 2 s (t_{delay}) is assumed for the BL opening. The volume of the wet EV is 3000 m³ including 5% water, and the dry EV of 13500 m³. The VV consists of the plasma chamber (PC) of 2900 m³, the upper port (UP) and lower port (LP) volumes of 1500 m³ and 2000 m³ respectively. The VV temperature is 300 °C and the structure temperature of 40 °C due to water cooling of the VV-PHTS. Leakage occurs from the VV to the TCR when its pressure exceeds TCR pressure. The divertor (DIV) is considered as a heat structure inside the VV.

The main consequences of the in-VV LOCA in terms of the pressurization of the VV (Figure 5-1) and radiological releases are evaluated. The FW failure affecting one sector in CaseIa results in He ingress into the VV and thus the opening of the BLs and RDs within 7.5 s and 37.2 s respectively. For the FW failure affecting all sectors in CaseII, the fast RD opening at 5.322 s is 0.0285 s earlier than the BL opening due to t_{delay} of 2 s. Thus, the BL function fails in CaseII. In CaseI, the VV pressure is suppressed below p_{VV} by enlarging the wet EV of additional 30% (CaseIb), or using the dry EV (CaseIc). In the last case the VV pressure is below p_{RD} such that the RD is not activated. In CaseII, the PC pressure exceeds p_{VV} at 31.7 s, reaches the maximum of 8.1444E+5 Pa at 363.0 s, and drops to 3.8134E+5 Pa at t_{end} such that the VV pressure exceeds p_{VV} all the time. T_{EF1} is not reached on the affected FW by the unmitigated plasma disruption and decay heat condition such that no aggravating event occurs. In CaseIa, the VV system removes the heat from the blanket of ~1.383 MW till t_{end}, which corresponds to 49.35% of the decay heat for 6 sectors at shutdown, while the heat removal is ~2.598 MW in CaseII, which corresponds to 34.75 % of the decay heat for all 16 sectors at shutdown.

The leak rates lead to pressurization of the TCR over the atmospheric pressure and thus the releases into the environment. In CaseIa, the largest releases are found in the wet EV (459.73 kg dust, 1.184 kg tritium), and they start to release to the environment at 32.23 h approaching t_{end}. At t_{end}, 1.20E-04 g dust and 6.08E-07 g tritium are released to the environment, which are minimal. In CaseII, the largest releases are found in the dry EV (525.77 kg dust, 1.347 kg tritium), and they start to release to the environment at 1.69 h. At t_{end}, 1.767 g dust and 8.62E-03 g tritium are released to the environment, which are only 11.2% of dust and 13.6% of tritium in S2 due to the LHS BDBA (§ 5.5).

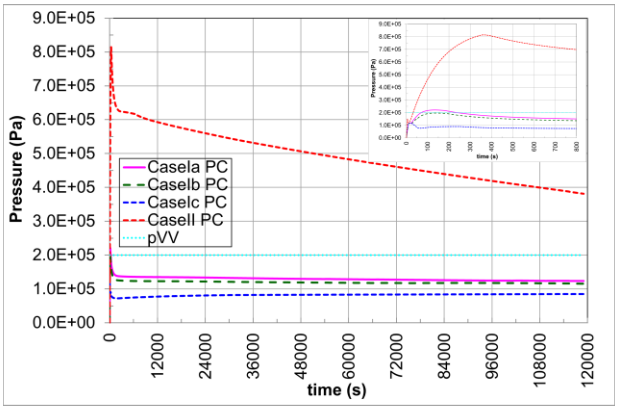

Figure 5-1
Pressure behavior in the plasma chamber

5.3 The selected event for the WCLL concept

The reference event described for the WCLL concept is an ex-vessel LOCA (DBA, LBO1 in 0). The PIE is a large rupture (0.4926 m^2) of the breeder zone (BZ)-PHTS distributor ring during the plasma burn, causing a loss of coolant inside the TCR. Because the primary cooling system involves a large amount of energy due to the pressurized water coolant (15.5 MPa), the large amount of fluids spilled into the tokamak building may damage the TCR internal structures causing a loss of confinement function.

The reference breeding cell adopted for modelling purposes is the WCLL 2018 V0.6 Central OB equatorial cell, described in detail in 0. Data for the WCLL DEMO tokamak building have been taken from the SDL. In the TCR of the pre-conceptual design phase only Upper Pipe Chase (UPC), Vertical Shafts (VS), and Lower Pipe Chase (LPC) are connected, forming a total free volume available for expansion of 17543 m^3; the PHTS area and the Top Maintenance Hall are designed as stand-alone compartments.

Starting from these design assumptions, four different case scenarios have been studied by connecting different TCR compartments:

- CASE 1: UPC, VS and LPC are the only volumes available for steam expansion;
- CASE 2: PHTS volume is made available by opening a connection with the UPC. The added PHTS volume is 49975 m^3;
- CASE 3: TCR configuration is like that of CASE 2, however in order to further increase the volume available for steam expansion, the HCPB TCR design is used. For this case the PHTS area volume is extended for the entire length of the tokamak building (about 96.0 m), for a total volume of 120000 m^3;
- CASE 4: Top maintenance hall is made available (together with PHTS area) by opening a connection with the PHTS area.

The double guillotine break is assumed to occur at 0.0 s. Coolant is discharged at a huge rate into the UPC. Hot and cold legs of both FW- and BZ-PHTS are equipped with trip valves, to limit the amount of water entering the TCR. The signal for the detection of the abnormal event has a delay of 3 s and 7 additional seconds closure time of the valve is assumed. The trip valves begin to close when the pressure in the pressurizer is below 13.0 MPa. This set-point is reached at 3.65 s after the PIE and the fully closed state is reached after 13.65 s, when the pressure in the BZ pressurizer is 3.53 MPa. Because of the position of the break, upstream isolation valves have no effect on the depressurization of the BZ in-vessel volumes which reach the equilibrium with the TCR pipe chase volume about 95.0 s after the PIE. The decay heat in the BZ loop, which is not affected by any rupture, led to the activation of the safety relief valve (SRV) which set point has been chosen to be 1.88 MPa. The first opening of SRV occurs 1160 s after the PIE. The pressure difference between the PHTS and the TCR volume was so large that a substantial amount of coolant was ejected into the TCR. The initial mass flow rate of coolant at the break was extremely large (maximum flow rate of 34917 kg/s) and was then followed by gradual decrease due to the progressively depleted primary system coolant inventory and to the intervention of trip valves. Around 171 tons of water are released inside the TCR.

The large amounts of steam and water released from the FW-PHTS cause an excursion of pressure and temperature into the containment volume. The TCR pressure rapidly increases to a maximum of about 543.1 kPa. Then, both pressure and temperature decrease because of the presence of concrete heat structures, having a large surface and facing the external environment simulated as a large control volume containing air and maintained at 30 °C. Since no active or passive system is provided for the cooling of TCR volumes currently, pressure remains higher than the atmospheric pressure. The fusion power termination system will actuate on a signal from a pressure sensor in the vault or primary cooling system and terminate plasma burn in three seconds. The plasma is terminated through a mitigated disruption. Overpressure detection in the TCR occurs in 0.025 s. The plasma control system triggers plasma shutdown 3 s after the signal. Thus 3.025 s after the PIE plasma facing surfaces are affected by a higher heat flux due to a mitigated plasma disruption. The FW temperature reaches a maximum value of 512.04 °C and then decreases because of DHRS operation. In this scenario, the FW temperature never reaches values that would result in failure by melting of EUROFER.

In Figure 5-2 the pressure in the UPC is reported for a LOCA in the BZ-PHTS distributor ring for the four different cases. The pressure limit imposed of 200 kPa for the TCR structure is exceeded with a wide margin in Case 1 and Case 2, but also in Case 3 and 4 the problem exists.

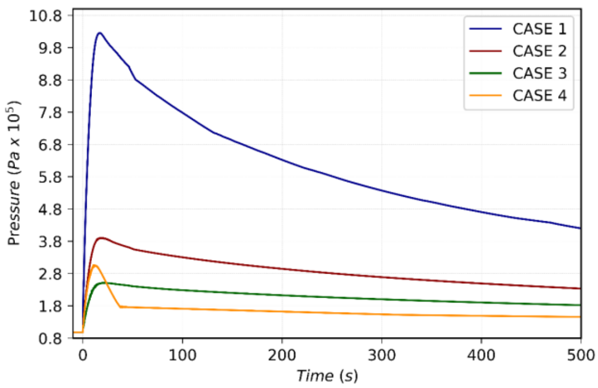

Figure 5-2

Pressure in the TCR after a LOCA from the BZ-PHTS

In the present analysis, the mobilizable radioactive materials are: activated dust and tritium (as tritiated water - HTO) from the VV, and HTO and activated corrosion products (ACP) from the failed BZ PHTS cooling loop. Since the integrity of in-vessel structure is maintained, there are no tritium and tungsten dust releases inside the TCR volumes or VVPSS. Only the HTO diluted in the FW-PHTS and loop related activated corrosion products are mobilized toward the TCR after the initiating event.

The tritium concentration in the primary cooling system is 0.015 g-T/m^3 water (SDL), for a total amount of tritiated water of 32.0 g in the BZ-PHTS. The quantity of ACP in one BZ-PHTS loop is 10 kg, a mobilization fraction of 7% has been used for ACP. 99.1% of the total mobilizable fraction of the FW-PHTS ACP inventory is released into the TCR. 99.3% of the HTO mobilizable inventory is moved into the TCR.

Because the TCR structure has failed, while the pressure inside remains higher than the atmospheric pressure, all the coolant spilled onto the TCR presents a direct environmental release. A flow path connecting the TCR volume with the volume simulating the external environment has been used to simulate the leakage of radioactive inventory, with a constant volumetric flow rate while the TCR pressure is greater than atmospheric. In Table 5-1 the total mass of ACP and tritium (in the form of HTO) released into the environment is reported.

Table 5-1

Mass leaked from DEMO containment building.

Scenario	ACP [g]	Tritium [g]
CASE 1	6.0709E-09	3.26E-06
CASE 2	3.3916E-06	1.22E-03
CASE 3	3.5336E-06	1.80E-03
CASE 4	6.934E-06	3.17E-03

5.4 The selected event for the divertor

The main reference design basis events 0 identified for the divertor system include in-VV LOCA and ex-vessel LOCA due to pipe or manifold breach and a LOFA in the primary cooling loop of the divertor due to pump seizure. LOFA was selected since it has the highest occurrence frequency. The divertor loop is divided into two main heat transfer systems (HTS), each serving 24 of the 48 divertor cassette and plasma facing unit (PFU). Both LOFA and LOCA event analyses have been performed on the PFU heat transfer system. Ex-vessel PFU HTS loop 00 includes a pressurizer (10.6 m^3 partially filled in water (5.3 m^3) operating at 3.82 MPa, including 3 SRVs set point 4.57 MPa and a relief tank); a heat exchanger (HX) (T in/out 136 °C / 129.7 °C); a pump providing 1.82 MPa to the loop and a distributor and collector rings serving the 8 sectors. Total water coolant inventory is 57 m^3 including in-vessel PFUs volume (2.33 m^3) of which about 0.2 m^3 for vertical targets. Pressure and temperature working point at PFU inlet is set at 5 MPa and 129 °C respectively. In-vessel heat structures include: BB system FW, BZ, divertor cassette and liner, VV and connected volumes (UPs and LPs) inner walls. HX provides 68 MW of heat sink power compensating during plasma pulse for surface and volumetric nuclear heating for the PFU of the 8 considered sectors. A total of about 460 kW shutdown decay heat power is also applied to the 24 PFU in-vessel divertor components. PFU heat structures (HS) are radiatively coupled 0 to the underlying cassette structure which is in turn coupled to VV walls. VV Walls are assumed to be kept at 200 °C, cooled by decay heat removal system.

The LOFA analysis foresees a pump stop and related flow path closure within 5 s, the detection of loss of flow (<80% nominal mass flow rate) triggers the FPS. Conservatively, adiabatic conditions were applied to HTS ex-vessel structures. In a baseline CASE 1, the pressure transient (**Error! Reference source not found.**-upper box-CASE 1) in the PFU hit by disruption shows a gradual convergence to pressurizer pressure after initial oscillations due to pump seizure. Due to PFU HS and liquid temperature rise until water trapped within the PFU (at a higher elevation than the loop) vaporizes causing a pressure rise (though not reaching the pressurizer SRV set point), this results in a large insurge of colder water into the pressurizer, ultimately lowering system pressure. Also, a sensitivity case with no LOSP after the PIE event was studied as CASE 2 (Figure 5 3-upper box-CASE 2). This case was studied to verify that accident consequences are bounded when considering an aggravating impact on loop pressurization from non-safety active equipment (i.e. pressurizer heater on 44 kW), Figure 5-3-upper box-CASE 2. In both cases melting temperature for the PFU CuCrZr cooling channels (assumed in 1085° C for Cu) is not reached, so that a LOFA accident does not evolve into an in-VV LOCA. Future analysis shall foresee an integrated analysis of both cassette and PFU HTS.

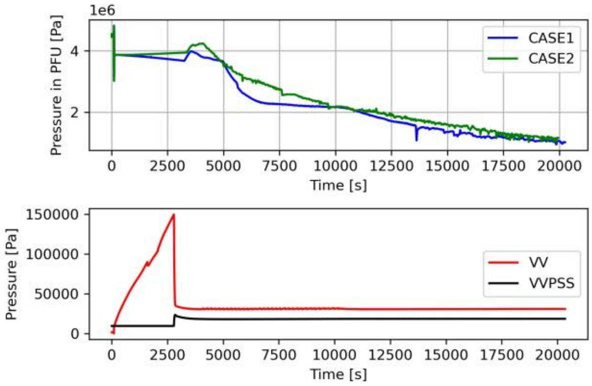

Figure 5-3
Pressure transient in LOFA (upper box) and In-VV LOCA (lower box) events

5.5 Dose results

Dose calculations have been performed for four event scenarios (S1 – S4). The computer programs UFOTRI for assessing the consequences of accidental tritium releases and COSYMA for the activation products (W-dust, ACP) are used for the dose assessments. Historic weather conditions from Cadarache (ITER) in 1991 are applied for a probabilistic assessment 0.

- S1: WCLL, ex-vessel LOCA DBA described in section 4.3. Tritium and ACP in BZ-PHTS are mobilized into the TCR volume where overpressure causes leaks from the tokamak buildings to the environment.
- S2: HCPB, LHS BDBA leads to an aggravating in-VV LOCA with the failure of 15 FW channels assumed in each of 6 OB sectors from three modelled loops. Tritium and W-dust are transported from the VV to the TCR and released to the environment due to the leak rates and resultant overpressure.
- S3: WCLL, FW-PHTS ex-vessel LOCA BDBA leads to an aggravating in-VV LOCA with the failure of 1 FW channel in all the OB1 blanket modules along the toroidal circumference. An unmitigated plasma disruption produces an additional in-vessel failure of the OB4 modules.
- S4: DIV PFU, in-VV LOCA DBA is characterized by a relatively slow pressurization pattern (**Error! Reference source not found.**-lower box) resulting in VV pressure being higher than external containment volume for about 500s leading to cumulated releases by means of a VV leak of in-vessel Tritium (5.1E-02 g) and W (2.2E-02 g). Results in Table 5-2 conservatively consider such inventory released directly to external environment as no detritiation system is accounted for in current analyses.

The dose results are shown in Table 5-2. At all distances in all scenarios, except at 500 m for the EDE (Effective Dose Equivalent) in S2, the dose is far below 1 mSv, which is the dose limit adopted in DEMO for normal operation or anticipated events / incidents in one year.

5.6 Accident sequences summary

Three representative events for the HCPB concept, the WCLL concept and the divertor have been described in this paper. The main identified issues for each event are summarized below.

The main concern of the in-VV LOCA for the HCPB concept is the pressurization of the VV. Both wet and dry EVs with adequate volumes are required to suppress the VV pressure below the defined limit (p_{VV}). To reduce releases from the wet EV, the effectiveness of pool scrubbing needs to be investigated.

Ex-vessel LOCA analysis performed for the WCLL concept highlighted that efforts should be made to reduce the pressure peak inside the TCR. Solutions could be: the segmentation of the WCLL PHTS loops to reduce the inventory discharged; a modification of the TCR volumes to provide additional volume for steam expansion in the TCR.

The divertor HTS loops coolant inventory partition have shown efficacy of the VVPSS-H2O system to manage pressurization > p_{RD}. The slow pressurization pattern highlighted the possible need for improving management of small inventories leakage so as to limit the occurrence of VV pressurizations between 100 kPa and p_{RD} leading to releases according to the adopted VV leak model. A HTS layout possibly reducing trapping of steam within IVCs volume emerged as a recommendation from LOFA assessment.

The main uncertainties in the performed analyses for the IVCs are: reference design data and the level of MELCOR geometric and phenomenological modelling details.

A preliminary sensitivity analysis to evaluate the effects of DEMO uncertainty parameters on an ex-vessel LOCA transient is reported in 0. The uncertainty band has been evaluated through sensitivity analyses programmed, collected, and statistically manipulated through the RAVEN software tool. Results showed that the FW temperature at which plasma in-vessel breach occurs is a parameter that affects the overall VV pressurization and thus the VVPSS response. Also, the mass of hydrogen produced is strongly affected by the maximum temperature limit of FW structure.

During the conceptual design phase accident analyses will be further investigated for the IVCs with respect to the identified issues based on the performed analyses, coupled systems (e.g. the WCLL and PbLi loop, BB-, DIV- and VV-PHTS, which are directly/indirectly connected to the power conversion system (PCS)), the design improvement, updated boundary conditions such as plasma behavior, confinement, leak and pressure suppression conditions, etc., and relevant PIEs which have not yet been studied. Moreover, accident analyses for tritium process systems, blanket system connecting to the tritium extraction removal system, release of cryogenic fluid, seismic safety, etc. will be investigated as well. Dose assessment for the radiological impact will be continued for further event scenarios.

Table 5-2
Dose (95%percentile) in mSv at selected distance

Scenario	95%percentile	0.5 km	1.0 km	5.0 km	10.0 km
S1 (WCLL)	Early dose	1.6E-03	8.7E-04	6.0E-05	1.2E-05
	EDE with ingestion	6.8E-03	3.6E-03	2.8E-04	7.9E-05
S2 (HCPB)	Early dose	1.0E-01	3.9E-02	7.1E-03	3.9E-03
	EDE with ingestion	1.3E-00	4.8E-01	9.3E-02	6.0E-02
S3 (WCLL)	Early dose	1.1E-02	6.3E-03	4.0E-04	1.4E-04
	EDE with ingestion	5.4E-02	3.5E-02	2.8E-03	1.4E-03
S4 (DIV)	Early dose	6.0E-02	3.4E-02	1.7E-03	7.2E-04
	EDE with ingestion	2.8E-01	1.6E-01	1.3E-02	9.5E-03

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6 Chronic releases

The tritium content in the DEMO cooling systems due to the permeation from the plasma chamber and/or the breeder blanket represents a safety concern. Indeed the chronic leakages of helium and water affect the zones in the plant in which human presence is foreseen for normal maintenance and inspection.
The chronic leakages are a function of the coolant pressure, viscosity, dimensions and numbers of the cooling pipes, in accordance with the equation (6-1):

$$Qm = \frac{1}{8} * \left(\frac{d}{2}\right) exp4 * \left(\frac{Pi-P0}{\mu L}\right) * \rho l \qquad (6\text{-}1)$$

Qm mass leak rate (m^3/s)
d leak path diameter (m)
L leak path length (m)
Pi internal pressure (Pa)
Po external pressure (Pa)
μ viscosity (Pa*s)
ρl fluid density (kg/m^3)

The chronic releases based on CANDU reactor experience [6-1], [6-2], [6-3] evaluated over ten years of operation, were of the order of 296 TBq/a. The assessment made for ITER FEAT for a tritium concentration of 1 Ci/kg of water gave about 50% of the CANDU figures, due to the absence in ITER of the heat exchanger, in which the releases are more frequent.
The analysis for releases in the DEMO WCLL reactor is not available currently but due to the larger size of the cooling loops and the higher pressure in the pipes they are expected to exceed the ITER chronic releases.
In the HCPB concept the most relevant concerns regarding leakages are:

- Seals of He circulators
- Isolation valves
- Flanges or threads connecting component or instruments.

By employing the best available technologies it seems possible to reduce helium leaks below the limit of 0.1% of the He coolant per day.

The leakages through these components in HCPB have been evaluated in the order of about 8.2 Ci/day.
In addition, the leakages through the steam generator blowdown in HCPB concept are not negligible. They need to be quantified for a complete picture of the leakages in the helium fusion reactor and are planned in FP9.

References Chapter 6

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7 Radioactive waste

7.1 Assessment

The current data available on expected DEMO waste arisings relies primarily on neutronics calculations. The materials used will partly depend on the DEMO design selection: HCPB or WCLL. Waste common to both designs include Eurofer blanket structures, the SS316 in the VV, plus a layer of W on the FW. The operation of DEMO is expected to create a large volume of radioactive metallic material through neutron activation and/or tritium contamination. The storage of blanket and divertor components will also create secondary waste. The removal of tritium from metal components is necessary to minimize gas leakage and prevent cross-contamination of affected areas. Tritium release through the stack is inevitable but it will be ALARA thanks to HEPA filtration combined with an atmosphere detritiation system

Operational and analytical activities with tritium will involve production of low-level tritiated water. Other sources of tritiated water may come from detritiation of components and storage ponds for PFC's decay heat removal.
Housekeeping waste will include a mixture of hard and soft material (filters, clothes, redundant plant, and equipment). Related waste processing options available, aimed at volume reduction, include incineration and super compaction.

7.2 Classification

The radwaste classification proposed for DEMO is in accordance with the IAEA [2-4] (see § 2.6.2).

These may be applied using the regulations of the DEMO hosting country and Waste Acceptance Criteria of specific repositories. The waste management strategy aims to enable classification of the majority of the solid radioactive waste arisings as LLW.
Acceptance criteria for LLWs vary from country to country and are based on specific activity of individual or a class of radionuclides considering half-life, emission and mobility in the geosphere and biosphere. Other criteria are based on limits for the total repository inventory. The guiding criterion adopted in the safety analysis of a repository is that of the committed dose at MEI <10 µSv/y.

7.3 Solid Waste Levels Expected

Error! Reference source not found. shows the expected masses (in kg) used in DEMO components for the two main design options.

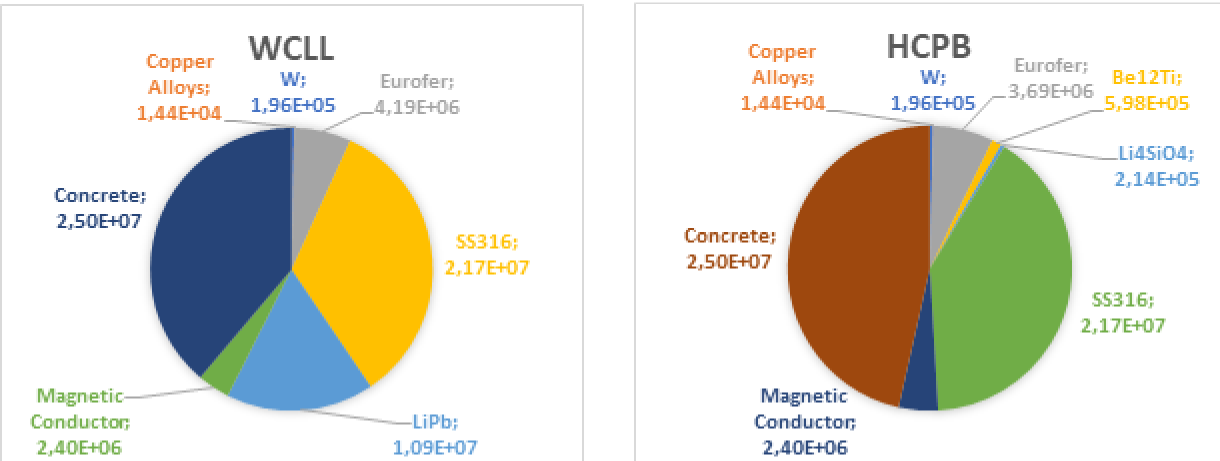

Figure 7-1

Approximate material quantities (in kg) used in the WCLL and HCPB designs. The scheduled component replacement masses are not included. The VV SS316 figure includes steel from the Cryostat.

Modelling of neutron activation has been undertaken using a method consisting of a two-step calculation of neutron transport and inventory described in [7-1].

The activity is mass averaged for a given material within a given component. This mass averaged specific activity A_M, is calculated for with the relation:

$$\bar{A}_M = \frac{\sum_{0}^{N_{cells}} A_i}{\sum_{0}^{N_{cells}} M_i}$$

Equation 1: Specific Activity over multiple cells

.

Where A_i is the activity in cell i and M_i is the mass of cell i.

Using current disposal facilities acceptance limits and the activation results from neutronic calculations (**Error! Reference source not found.**) shows the acceptability of mass averaged DEMO waste for relevant facilities in Europe.

Table 7-1

Performance of mass averaged activity DEMO components against current repository acceptance criteria after 50 years post DEMO life. Specific radionuclides identified preventing consignment or overall beta, gamma or alpha levels exceeded for cells in red. Amber cells identify radionuclides that are very close to limit and may be a problem in the future.

Material Type	Repository Acceptance Criteria Performance			
	Konrad (DE)	LLWR (UK)	CSA (FR)	El Cabril (ES)
WCLL				
Tungsten	C14	β+γ		
Eurofer	C14	β+γ	Nb94	Nb94
Stainless Steel	C14			
LiPb	H3	β+γ	H3	H3
HCPB				
Tungsten	C14	β+γ		
Eurofer	C14	β+γ	Nb94	Nb94
Stainless Steel	C14			
Be12Ti	H3, C14	α,β+γ	H3, α	H3, α

7.4 Management

The proposed management of Eurofer and SS316 consists of smelting as soon as this option becomes technically available – with several anticipated benefits:

- Corrosion reduction, which is the main activity release to biosphere [7-2]
- Self-shielding by homogenizing radionuclides
- Detritiation factor of 24600 and 38720 achieved for argon and hydrogen atmosphere respectively [7-6]
- Decarburization factor of 100 in preliminary trials [7-7]
- Volume reduction
- Easier consignment

Pre-consignment management techniques will need to cope with significant decay heat from the DEMO components [7-8] (see **Error! Reference source not found.**). Previous reports have concluded that the limit for remote handling feasibility would be 2 kW m^{-3} [7-9].

For Be and W the processes foreseen are chlorination followed by thermal reduction to remove harmful long-lived radionuclides, [7-3] and [7-4]. Chlorination with chlorine gas results in the production of metal chlorides, significantly more volatile than pure metal counterparts. W chlorination is more complicated than Be, as the expected metal chloride impurities have melting points above and below that of WCl_6.

Thermal reduction would then allow recovery of base metal from metal chloride, by exposing the metal chlorides to a very hot (>1400 °C) heat source to decompose into base elements.

Recycling is important for LiPb given the high cost of lithium enrichment and the complex manufacturing of LiPb eutectic. There are however, limited uses for recycled lead [7-5] because of the longhalf live of Pb-205 which would require further expensive purification. Hence, only lithium is planned to be recycled by separation from lead. Pyro-metallurgical and hydrometallurgical processes have been identified relying on the large solubility variation between lithium and lead compounds. It may also be possible to directly reuse the eutectic in a different fusion reactor without separation if the Li-6 levels can be replenished.

7.5 Final route of waste

The waste generated at the DEMO site would fall into either the LLW or ILW category depending on the regulations of the country that it is generated within. After an interim storage period the preferable scenarios for the material is for it to be either recycled or to be consigned for disposal in a near surface facility. Clearance will not be a realistic possibility for the reactor steel components (IVCs and VV) although clearance of parts of the bio-shield may be achieved.

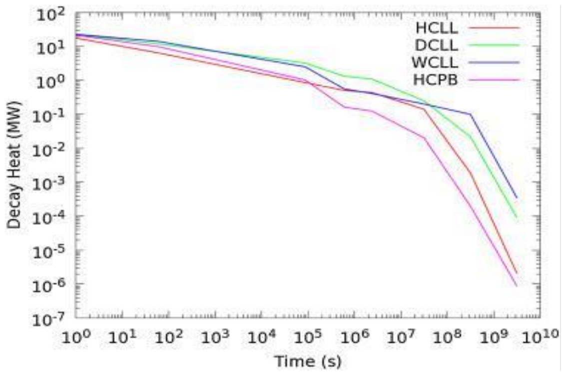

Figure 7-2
Decay heat for the different blanket concepts over cooling time from [7-8]

Recycling of the material will require the development of remote handling techniques that allow for the majority of the steel to be refabricated safely and economically and the identification of suitable uses for the material.
Disposal in near surface facilities will require the problem radionuclides identified in **Error! Reference source not found.** to be removed through treatment methods. The thermal treatment techniques identified should be sufficient to remove ^{3}H and ^{14}C and allow for the majority of the material to be consigned to near surface facilities. ^{94}Nb in the steel is based on neutronics calculations which assumed a 50 wpm level in the Eurofer steel. It is technically feasible to reduce this to 20 wpm which should have a beneficial impact on the waste categorisation.

7.6 Recycling Pathways

Steel recycling, restoring its original composition and properties, is possible with current methods.
A simple process may use an electric arc furnace for melting, followed by a vacuum oxygen decarburisation or argon-oxygen decarburisation step. The ingot could then be rolled/forged into semi-finished or finished products [7-10]. It might be recommended that a conservative safety factor is applied by moving recycled steel into less challenging environments than it was originally exposed to .
Most of the Be long-term dose rate is dominated by radionuclides arising from impurities like U and Co [7-11]. For pure Be, its low dose rate coming from Be-10 would already permit hands-on handling (HOH) only one day after shutdown (asd) [7-12].
The main problematic radionuclides for irradiated tungsten arise both from W itself (Re-186, W-181) and from its impurities (Co60, Nb94). The W-radionuclides activity falls rapidly due to short half-lives, reaching clearance within 10 years asd [7-12]. In both cases chlorination may be sufficient to enable recycling of the material.
The lead content coming from lead lithium is not expected to be recyclable according to current methods. This will mean that irradiated lead will need to be disposed of either in near-surface or in deep geological deposits. During a planned DEMO breeder blanket lifetime of 5 FPY, ~9050 tonnes of lead will require disposal. Reuse of the eutectic in another fusion facility should be explored or the remaining 1850 tonnes of Li should be recycled for future use.

7.7 Radioactive waste summary

DEMO is expected to create a large amount of metallic waste from blanket changes during operations and the VV at the plant end of life. Efforts at planning a strategy for dealing with the waste arisings has focused on melting the SS316 and Eurofer steel waste into ingots. The neutronics assessment has shown tritium and C14 to be key radionuclides preventing disposal at several sites (see **Error! Reference source not found.**). Techniques of detritiation and decarburization that can be realised with metal in the molten state should enable DEMO waste to be acceptable in the majority of the European near surface repositories considered.

A further consideration is that most reference radionuclides in DEMO materials such as in Eurofer Co60, C14, Mn54, Fe55 and Nb94 closely approach the activity per weight limits in pre-existing repositories. A melting treatment step will remove volatile radionuclides whilst mitigating against hot spots of activity affecting sampling accuracy and preventing consignment. This will reduce the environmental impact and enable more confidence in waste characterization.

Techniques involving chemical separation of the waste from its transmutation products have been suggested as potentially viable processes for other DEMO waste (LiPb, W and $Be_{12}Ti$).

Overall, a preferable scenario has been identified of maximising the recycling of the waste for future use in fusion reactors. This will depend on advances in remote handling technology with material changes due to irradiation expected to be minimal.

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8 Occupational radiation exposure

Minimization of ORE, understood as the exposure of personnel to ionizing radiation, is key for safety. To illustrate this, we here focus on the ex-vessel PbLi systems in a WCLL reference configuration. We identify the major contributors to ORE as the basis for its minimization by predicting the residual doses associated with hands-on maintenance activities.

Because of the PbLi circulation through the loop systems and components, an ionizing radiation field is expected to be around systems of the loops where the activated alloy is present. During non-operation phases several manual maintenance activities will be done for components of the loops, the most important being: i) Storage tank, ii) Pump system, iii) Heat exchanger and iv) Buffer tank. For these and other components we first calculate the residual dose rates around them and then collective dose rates are predicted.

Regarding the applied methodology, we manage two categories: on one hand the method for ORE assessment [8-1], and on the other hand the specific methods for residual doses prediction [8-2].

Regarding the first category, the first step is to use the reference Plant Breakdown Structure (PBS) to list the main equipment/components needing hands-on operations. The second step is to identify, on the basis of an engineering judgment, for each one of the low level component/sub-systems, information on specific maintenance activities in terms of: i) type of hands-on operations, ii) room where maintenance activities are performed, and iii) radiation zones of the maintenance areas. Then, for each maintenance activity, we estimate: expected dose rate, foreseeable time required for the hands-on activity, expected number of workers involved, expected yearly hands-on frequency, aggravating factor for the use of protective suits/masks (vs tritium, dust), N° of elements/Unit and N° operating Units in the system. According to the above values, a first appraisal of the collective dose (in p-mSv/y) can be inferred for the systems in DEMO.

Regarding the second category, it has been demonstrated that the methodology used for prediction of residual doses on the PbLi alloy needs to be improved due to the complexity of the loops, where we have pulsed scenario with extraction (tritium at the tritium extraction system, TES; impurities at the purification extraction system, PES) and incorporation (corrosion products around the circuits) of materials. We use the ACABLoop tool [8-2] which reflects most realistically those essential features of the loops.

To determine the cumulative collective dose two maintenance scenarios were considered:

- Empty PbLi loops, with the alloy all concentrated in the storage tank (scenario A).
- Components of the loops fully filled with activated PbLi. The storage tank has been envisaged with a 50 cm-thick shield of concrete. An operation cycle of 5 years has been assumed as a reference with a cooling time of 1 day (scenario B).

For scenario A, determining the residual dose rates around individual components and traps or filters of the purification systems is not straightforward. Activated deposits, cluster and films present in the individual components, pipes and valves must be realistically assessed after PbLi drainage and loop purging. Average dose rate values for workers access have been then chosen according to the radiation zone classification (between 25 and 100 µSv/h).

Table 8-1

Collective dose rates for scenario A with average dose rates around components.

Average Dose Rate (µSv/h)	All maintenance interventions (conservative) (p-mSv/y)	No PM on valves and no HEX sludge lancing (p-mSv/y)
25	104	63
50	207	125

100	412	250

Table 8-1 reports the collective doses in p-mSv/y assuming different values of average dose rates around the individual components. Since details on the maintenance plan were not available yet, two sets of results are presented. One includes all possible preventive maintenance (PM) interventions on valves, heat exchangers (HEX), pumps and tanks leading to more than 400 p-mSv/y, in the worst scenario. The other excludes maintenance interventions on the valves and the most hazardous operations on the HEXs, yielding values which are 40% lower.

For scenario B, dose rates calculated using the ACABLoop tool have been used. Values around 1000 μSv/h are found for 1 – 5 years of DEMO operating and 1 month cooling close to several components of the PbLi loops. It is worth noting that impurities are the major contributors to dose for such long operating times; such that reduction of alloy impurities as well as a very high performance of the purification system can reduce these dose rate levels. Due to the presence of large amounts of activated PbLi in some of the system components, many time-consuming hands-on maintenance actions have been considered to be not applicable, keeping only external inspections integrated by remote means. This assumption led to an overall collective dose of 243 p-mSv/y.

References Chapter 8

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9 Safety Room Book

The Safety Room Book is a basic living document for DEMO. It provides information about the environmental conditions in which the Safety Important Class - Structures, Systems, and Components (SIC-SSC) of the fusion reactor operate during the lifetime of the facility.

Those SSCs classified as SIC need to be qualified for those conditions concerning the safety function they are asked to provide; therefore, the Room Book is an important reference for SSCs design and procurement.

In the current approach, for the various DEMO normal operation (NO) modes and accident conditions, it includes specific information for rooms/zones of the tokamak building such as:

- Dimensions
- Volumes
- Magnetic field B
- Radiation dose rate
- Pressure, temperature and humidity in NO
- Fire zoning
- Radiation zoning
- Max pressure in DBA
- Max temperature in DBA
- Max temperature in BDBA
- Max pressure in BDBA
- Heating and Ventilation System (HVAC) system and settings (if any)
- Ventilation and Detritiation System (VDS) system and settings (if any).

The main parameters are referred to the different DEMO modes of operation, that are:

- Mode 0 : Plasma Operation State
- Mode 1 : Shutdown phase
- Mode 2 : Equipment transfer phase
- Mode 3 : Test and Conditioning phase.

In Table 9-1 the data relating to the drain tank area are shown, as an example of the Room Book contents.

Currently 12 rooms/areas of the tokamak building are characterized in the Room Book. Specifically the galleries, the cubicle room area, the lithium lead components room, the dome, the PHTS area, the upper pipe chase area, the cryogenic auxiliary cold box area, the generic port cell area, the lower pipe chase area, the drain tank area, the LiPb tank area and the rupture disk room for the WCLL concept.

The completion of the Room Book will include all the buildings in the nuclear area in which safety classified components and systems are allocated.

Table 9-1

Room Book data – Drain tank zone

roomTag	txt_Room	ype of roo	Level	Surface	Height	Volume	otal Volum	Expansion volume	lagnetic fiel	Dose to silicon		Dose rate		emperatur	Pressure in Op. Mode 0	Humidity in Op. Mode 0
			(m)	(m^2)	(m)	(m^3)	(m^3)	(m^3)	(mT)	(µGy/h)	Error	(µSv/h)	Error	Temperature in NO (ºC)	(kPa)	%
11-B3&B4-Drain tank	Drain tank area	area	B4 (-14.50)	620	8,6	5332	5332	2832	10 < B < 36	n.a.	n.a.	201	n.a.	n.a.	Pressure difference vs P atm -120÷-140 Pa	n.a.

roomTag	Fire zoning		Radiation zoning			Max. pressure in DBA	Max temperature in DBA	Max. pressure in BDBA	Max temperature in BDBA	HVAC system and settings	VDS system and settings
	Classification	Justification	Op. mode 0 Op. mode 3c	op. mode 1b	op. mode 2 op. mode 3a&3b	(kPa)	(°C)	(kPa)	(°C)		
11-B3&B4-Drain tank	Fire Sector	• Presence of ignition source, • Presence of flammable materials, • Presence of dispersible radioactive materials, • SIC to be protected	Taking into account constraints linked to the magnetic zoning and to the radiological conditions, access is not allowed **Access prohibited**	risk of external exposure (yellow) **Limited for all personnel**	risk of internal exposure (Orange) **Limited for all personnel**	*after incidental analysis*	*after incidental analysis*	*after incidental analysis*	*after incidental analysis*	HVAC equipment operating during maintenance only with a depression -0.120 <DP<-140 kPa to atmospheric pressure.	The N-VD system maintains a depression -0.120 <DP<-140 kPa to atmospheric pressure.

10 Safety classification and relevant implications

DEMO, as a nuclear installation, shall demonstrate that its safety objectives are met all over the licensed period considering all initiating events, incidents and accidents. The first important step is the definition of the SSCs implementing such functions. Another step is the definition of internal and external events, and environmental conditions such that the SIC components will perform the requested safety function in the most pessimistic conditions (e.g. DBA at the end of plant operating life). The final step is the identification of the associated implications for the design of SSCs, as well as for fabrication, commissioning, operation, maintenance, inspections and tests. This is important in the early design stage for the correct definition of system requirements and design criteria, and for the preliminary layout and integration of the different systems. It may also assist in a comparison of the alternative blanket concepts.

Those SSCs assigned as a SIC will receive adequate attention during the design, fabrication, installation, commissioning and operational stages. The objective is to ensure and demonstrate that they will meet the minimum performance and reliability requirements throughout their intended lifecycle so that the safety function is provided when required.

A gradation of safety important SSCs is adopted for DEMO [10-01] on the basis of the IAEA Guide No. SSG-30 [10-02]. The higher the risk defined in terms of consequence and likelihood, the higher should be the safety protection and the SIC grade required for SSCs.

10.1 Safety classification process

The safety classification process starts with a basic understanding of the plant design, its safety analysis and how the main safety functions will be achieved. All the plant states of the machine shall be considered in the process.

In this context, the following steps are pursued to define if an SSC shall be classified of safety relevance:

- Identification of the system to classify;
- Identification of the safety functions to be provided by the SSCs;
- Identification of the SSC operating modes that are relevant for the safety classification;
- Identification of the possible failures of the SSCs leading to lose the safety functions;
- Identification of the abnormal operating condition or accident event significant in defining severity of the events induced by the loss of safety function;
- Identification of the failure event probability or failure event category (categories defined in Table 2-2);
- Identification of the main criteria applicable for the safety classification;
- Identification of the safety function category;
- Identification of the safety class.

All the steps are described in detail in [10-03] and here summarized:

1. The SSCs to classify are identified by the PBS, where systems are detailed up to a level enabling component classification. As an example, the hierarchical breakdown could start at system level as the PHTS of BB and close at component level as pumps, HXs, pressurizers.
2. The fundamental and supporting safety functions apply under all normal and accident conditions and for the full lifetime of the facility.
3. The loss of a safety function by an SSC can cause different levels of consequences. Then, the relevance of the safety function associated to the SSC is defined by categorizing the functions associated to the SSC on the basis of the radiological consequence induced by failure of the SSC, the frequency of occurrence of the IE (Initiating Event) for which the function can be lost or can be called upon, the significance of the contribution of the function in achieving either a controlled state or a safe state. Three safety function

categories are defined in [10-01]. The dose to public limits fixed to define the severity classes for events of category 3 and 4 are reported in Table 10-1.

4. Three top level criteria are used to check eligibility of SSC as SIC:
   Criterion A the SSC failure can directly initiate an incident or accident, leading to risks of exposure or contamination,
   Criterion B: the SSC operation is required to limit the consequences of an incident or accident that would lead to risks of exposure or contamination, or
   Criterion C: the SSC operation is required to ensure the functioning of other SIC components.
   If one or more of the above criterions is applicable to the SSC, the SSC shall be classified as SIC.
5. The grading of SIC for DEMO SSCs is defined in [10-01] by means of three classes, SIC-1, SIC-2 or SIC-3:
   SIC-1. If SSC failure could lead to an event with consequences exceeding one-tenth of the limits set out in plant safety requirements (see Table 2-2), or the SSC needs to prevent, detect or mitigate an accident from resulting in consequences exceeding the above limits, or the SSC is needed to bring and maintain the plant into a safe state.
   SIC-2. If SSC failure could lead to an event with consequences exceeding a dose of 100 μSv to the most exposed individual member of the public, or the SSC is needed to prevent, detect or mitigate an incident or accident, although not required to reach a safe state, or the SSC is needed to ensure adequate shielding from radiation during normal operation.
   SIC-3. If SSC failure could lead to an event with consequences exceeding a dose of 10 μSv to the MEI of the public, or although not needed to prevent, detect or mitigate an incident or accident, the SSC helps to further reduce the consequences of such an event.
   All other components are defined as "non-SIC". They shall not impair SIC functions in any condition.

Table 10-1
Limits for severity classes of categ. 3 and 4 events

Severity class	Limit of dose to public for events of category 3	Limit of dose to public for events of category 4
S1	≥ 0.5 mSv	≥ 1 mSv
S2	100 μSv - 0.5 mSv	100 μSv - 1 mSv
S3	10 μSv - 100 μSv	10 μSv - 100 μSv

10.2 Safety classification of PHTSs

The process described in the previous section has been applied for the safety classification of the PHTS of the HCPB [10-04] and the WCLL reactor models [10-05], as reported in Reference [10-03]. Normal operation, incident and accident conditions were considered. The first step was to identify the overall failure events related to the SSCs under investigation [10-06]. The second step was the estimation of the likelihood of the failure events. The third step was to define the radiological consequences of the events.

Unfortunately, outcomes from dedicated deterministic analyses to quantify the possible consequences related to accident events were not yet available. Therefore, very simplified, but conservative, calculations were carried out to estimate the order of magnitude of the possible dose to public due to SSCs failures. An example of the data used for the safety classification of the two PHTSs are reported in Table 10-2. The accident represented as reference is a large ex-vessel LOCA from the PHTS into the TCR.

Table 10-2
Data obtained for large ex-vessel LOCA

BB model	Worst category for large LOCA from PHTS components	Tritium released in TCR		Release of HTO to environment	Dose to public	Severity class for Cat. 3 events
		g of T	[Bq]	[Bq]	[μSv]	
HCPB	3	10	3.57E+15	5.35E+13	39	S3
WCLL	3	200	7.14E+16	1.07E+15	779	S2

The main outcomes obtained from the study were the following:

- The main safety function for both PHTSs of HCPB and WCLL is the confinement of radioactive products at the level of process barrier;
- SSCs of HCPB PHTS could be classified as SIC-3;
- SSCs of WCLL PHTS could be classified as SIC-2.

Clearly, this study has to be considered as very preliminary and it will be redone once further deterministic and probabilistic assessments are available. In fact, a correct SIC designation requires precise estimations of the likelihood of the failure events and of the related consequences.

10.3 Identification of safety relevance for the SSCs

Since with the current state of the design and of the safety analyses, both the probabilistic and deterministic assessments are not yet available at the level required for a correct SIC designation, the complete SIC designation is not yet possible.

Nonetheless, in the design development, having a picture of the relevance of the different systems from the safety point of view is very useful for the designers. Such relevance has been outlined for the various SSCs on the basis of the FMEA carried out for the various systems, which qualitatively analysed in detail the possible failures and the expected consequences [10-07].

The outcomes of FMEA performed both at functional level and component level carried out for safety and RAMI purpose have been taken as basis for the assessment of the safety relevance of the SSCs.

The following data has been treated for the significant elements of the DEMO PBS:

- Safety function performed by the SSC.
- Possible failures of the SSC.
- Significant event.
- Safety Classification Criteria used to check eligibility of SSC as SIC.
- Safety relevance in terms of high (H), medium (M), low (L) relevance or non (N) safety relevance,
- Specification of the safety action required to the SSC, e.g. Trigger a fast discharge in the event of a quench;

1st confinement barrier for LiPb containing tritium and ACPs, Limit releasable radioactive inventory.

Given the criticality in checking and testing the components inside the vessel once they are activated, in-vessel SSCs are not designated as relevant for safety. However, they will be designed, manufactured, assembled and commissioned according to the highest quality standards and the greatest strength constraints. The same will apply for the major part of the SSCs in performing their process functions distinct from safety functions. Failures in performing process functions are identified as N (Non-safety relevance), but even if not for safety purposes, stringent requirements will be adopted for investment protection.

The indications about the safety relevance of the SSCs are also reported in the GSSR.

References Chapter 10

[10-01] T. Pinna, D.N. Dongiovanni, S. Ciattaglia, L. Barucca, "Safety important classification of EU DEMO components", Fusion Engineering and Design 146 (2019) 631–636, https://doi.org/10.1016/j.fusengdes.2019.01.040

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[10-04] A. Tassone et al., "Recent Progress in the WCLL Breeding Blanket Design for the DEMO Fusion Reactor," in IEEE Transactions on Plasma Science, May 2018, doi: 10.1109/TPS.2017.2786046.

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11 Main DEMO safety issues and possible mitigations

In the previous chapters, a picture of the main achievements in the DEMO safety assessment has been presented, highlighting the need for compliance with the safety requirements . The progress pursued in the frame of the Eurofusion DEMO program FP8 from 2014 to 2020 is significant but not sufficient to state that all the safety goals have been reached and all the issues solved.

The inventories of the radiological source terms are among the open safety issues for DEMO. As discussed in §4 large uncertainties still exist due to the lack of a fusion device that can be related to a reactor. The experimental data on tritium trapping and diffusion and on the plasma erosion occurring in a vacuum chamber during the burning phase and supplied by the small fusion devices like JET, ASDEX etc. have a weak relevance for DEMO. They have been used as a first approach but they need to be complemented with physical modeling and additional dedicated experiments. The simulation work is progressing with the development of a 3D simulation with the ERO code [11-1] for data on dust production, and experimental campaigns to study tritium diffusion in Eurofer in Q-Pete facility in KIT [11-2].

The accident analyses performed do not yet cover the complete list of PIEs selected in the FFMEA [11-3], only the design relevant accidents such as LOCA, LOFA and loss of heat sink events have been analysed. Some of them are demonstrated in §5. The plan for the next FP9 work programme is to address the missing PIEs and to complement them with internal fire and explosion accident analyses as well as with magnets and cryogenic ones.

The risk of explosion is one of the critical issues for DEMO safety: the tritium and deuterium presence in the plant, the hydrogen generation in accidents such as loss of water into the VV or loss of water with simultaneous loss of liquid lithium lead and consequent chemical reaction in the plasma chamber, and loss of vacuum are events that can lead to triggering an explosion due to H_2 concentration in hot spot zones.

To mitigate the risk of explosion the use of hydrogen recombiners connected with the VV is a solution adopted in the fission plants. The conversion and adaptation to fusion environmental conditions (sub-atmospheric pressure and saturated conditions) is under investigation [11-4] and it seems to be promising for the reduction of up to 60% of the H_2 gas generated during an in VV LOCA in the WCLL concept.

A further safety concern is the design pressure assumed for the DEMO VV. The VV is the primary confinement barrier to avoid the release of the radiological source terms. It is designed to stand up to 200 kPa of absolute pressure because of the diamond windows, its weakest components.

But in the case of in-VV WCLL LOCA [11-5], to maintain the pressure well below the VV design pressure, the intervention of rupture disks (5 m^2) that open towards a VVPSS is requested and in addition the actuation of safety valves aimed at the limitation of the water discharge in the plasma chamber. Both the technical solutions, large sections of rupture disks and isolation valves, are not straightforward to implement due to the limited space available inside the VV for the rupture disk and, in the ex-VV zone, for the safety valves (each of them has to have a redundant valve).

On the other hand, splitting of the FW (or BB) water cooling loop into smaller cooling loops requests additional space in the PHTS vault.

An ALARA process is requested to optimize the design and safety performances.

The occupational safety, dealt in § 8, is another key point to be solved. From the first provisional results, the plant target of the collective dose (700 p-mSv/y) is expected to be reached with a few maintenance activities. That means a heavy plan for remote maintenance has to be designed accurately to minimize the impact of the hands-on activities and to reduce the risk of rescue operations in case of failure of the remote maintenance equipment. An extensive study of all the maintenance requested for the operation and the refurbishment of the plant has to be planned in the early phase of the conceptual study.

The last open issue treated in this inexhaustive list is related to the waste management. As detailed in § 7 the two main problems are the research and the selection of materials that could guarantee together with the resistance to the harsh environment of the combined radiation, magnetic and high temperature fields, as well as a low activation.
The combination of these features would result in a lower amount of radioactive waste at low or medium level of activity.
In the meantime, the investigation of the possible treatment of waste by means of detritiation, decarburization and smelting processes are opening ways for the reuse and recycling of the materials as well as the possibility of separating highly activated parts of massive components from less activated parts.

References Chapter 11

[11-1] A. Eksaeva et al., ERO modelling of tungsten erosion in the linear plasma device PSI-2, Nuclear Materials and Energy 12 (2017) 253–260

[11-2] D. Klimenko et al., Definition of the Q-PETE experiment for investigation of hydrogen isotopes permeation through the metal structures of a DEMO HCPB breeder zone, https://doi.org/10.1016/j.fusengdes.2018.03.024

[11-3] T. Pinna et al., Identification of accident sequences for the DEMO plant, Fusion Engineering and Design 124 (2017) 1277–1280

[11-4] G. Mazzini et al., Hydrogen Explosion Mitigation in DEMO Vacuum Vessel Pressure Suppression System using Passive Recombiners, https://doi.org/10.1016/j.fusengdes.2021.112713

[11-5] M. D'Onorio et al., Preliminary safety analysis of an in-vessel LOCA for the EU-DEMO WCLL Blanket concept, https://doi.org/10.1016/j.fusengdes.2020.111560

12 Lesson learned from ITER

As pointed out in the previous chapters, ITER experience and lessons learned are considered at the maximum extent for the DEMO safety approach and relevant implementation.
The licensing process, the commitments associated to the construction license, and the post Fukushima 'Stress Test', represent a fundamental reference for DEMO, with ITER being the first fusion plant to be licensed as a nuclear facility. Associated with that are the standards adopted for SIC SSC, and the extrapolation from NPP standards.
The R&D done or ongoing in ITER to validate the assumptions made in the safety analyses, as, e.g., the RST assumed, the relevant diagnostics, the validation of fusion nuclear codes such as MELCOR or OSCAR, and the validation of the VVPSS.
Of particular importance are the lessons learned on the layout of the nuclear building, particularly of the tokamak building, for the complex layout and the challenging environmental conditions: increasing of space, improving shielding, facing the issues of high radiation dose from N16 and N17 in the water PHTS, remote handling and ORE assessment, leak tightness of the tokamak building and relevant critical penetrations not present in NPP, and contamination control and zoning.
Finally, the management of radioactive wastes, e.g. the design of Hot Cell, the component detritiation, the sampling of wastes, the experience with ANDRA also represent useful references for DEMO.

13 Conclusions

An extensive excursus on the DEMO safety has been done to depict the state of art of the studies. As highlighted in §11 the work is in progress, several issues have been faced, several remain to be solved, others are linked tightly to the evolution of the design and will change in the next years, during FP9.
The most important achievements relate to the definition of the basic principles and requirements for the construction of the Generic Site Safety Report, as detailed in §2. The activation studies (§3) demonstrated that the optimization of the material compositions have to be enhanced together with the use of the shields to lower the dose rates in the zones in which SIC equipment are located or the human access is foreseen for the maintenance of the components. The selection of the PIEs and the analyses of some important abnormal events are described in §5. The outcomes of the accidents studied showed that, in spite of the pessimistic and conservative assumptions for the source terms (§4), the radiological releases and the doses to the population are far below the limits established for the plant. On the other hand, the need to review the design of the cooling systems, reducing the inventory of the single loop, is confirmed to avoid the over-pressurization of the tokamak building.
The evaluation of the masses that will contribute to the waste inventories in the plant (§7) are approached. The radionuclides that can represent a problem for the future storage put in evidence in relation to the different rules applied in European countries. The detritiation issue represents the main problem in this field. The treatment for the reduction of waste, both in terms of mass and activation, is under evaluation with the proposal of advanced techniques (smelting, decarburization, etc.).
The full assessment of the occupational dose is yet to be achieved (§8) due to the patchy definition of the maintenance activities. An effort is requested to optimize durations and procedures applying a rigorous ALARA process.
A room book to collect the technical information on the zones (§9) in the main buildings that can affect the safety management of the plant is designed to allow direct access to the plant basic data.
The SIC classification of the DEMO components is in progress (§10). They have to be verified after completion of the accident analyses, while the criteria for classification have been fully established.
To overcome all the pending issues a robust R&D program has been launched in the frame of FP9 dealing with the explosion issues, enhancement of the waste treatment, enhancement of diagnostics for the in-vessel

source terms detection and control, and development of fusion codes for the accident analyses.

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